ORAU Team **Dose Reconstruction Project for NIOSH** Technical Basis Document for Atomic Energy Operations at the Iowa Army Ammunition Plant (IAAP)	Document Number: ORAUT-TKBS-0018 Effective Date: 03/14/2005 Revision No.: 01 Controlled Copy No.: Page 1 of 79
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ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	11/04/2003	00-A	New document to establish the technical basis for the development of a radiation exposure matrix for IAAP. Completed by Don Bihl, filling in for John A Leonowich while he was on travel.
Draft	11/12/2003	00-B	Resolved security issues and revised section on tritium and added an additional table to the external dosimetry section.
Draft	01/30/2004	00-C	Incorporated changes in response to OCAS review and comments on 00-B. Changes affected all sections.
Draft	03/03/2004	00-D	Incorporated changes in response to internal reviews. Changes affected all sections. Added Attachments A, B, and C.
Draft	03/19/2004	00-E	Incorporated changes in response to OCAS comments. New intake scenario in section 2.6. Numerous changes in section 3 and Attachment B incorporating statistical analyses of external doses.
Draft	04/14/2004	00-F	Improvements to section 3 incorporating statistical analyses of all the years measured doses and minor change to section 6 to make radon instructions more explicit. External skin doses prior to 1958 reserved.
Draft	04/16/2004	00-G	Adjustments made to DU ingestion intakes to be consistent with OTIB-009 per OCAS comments.
04/16/2004	04/16/2004	00	First approved issue. Initiated by Donald Bihl.
Draft	01/20/2005	01-A	Revision to incorporate public comments, and to expand time period for dose reconstructions to include early unmonitored workers. Changed document section order and formatting for conformance with typical site profiles. This was a major revision with significant changes throughout the document in methodology, assumptions based on new data and other information.
Draft	03/14/2005	01-B	Incorporated comments from OCAS Associate Director for Science.
03/14/2005	03/14/2005	01	Approved issue of Revision 01.

ACRONYMS AND ABBREVIATIONS

AEC	US Atomic Energy Commission
AMAD	Activity Median Aerodynamic Diameter
AP	Anterior-posterior (or front-to-back) irradiation of the body
BAECP	Burlington Atomic Energy Commission Plant
CF	conversion factor
CV	coefficient of variation
DCF	dose conversion factor
DOE	U. S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
DU	depleted uranium
EDA	Explosives Disposal Area
EU	enriched uranium
FS	Firing Site
GSD	geometric standard deviation
HVL	half value layer
Hp(d)	personnel dose equivalent at depth d in tissue
IAAP	Iowa Army Ammunition Plant (also sometimes IAAAP)
IARC	International Agency for Research on Cancer
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
ISO	International Standards Organization
ISO	isotropic
keV	kilo (thousand) electron volts, a unit of energy
LAT	lateral view X-ray
MED	Manhattan Engineer District
MeV	million electron volts, a unit of energy
MPC	maximum permissible concentration
MDL	minimum detectable level
NCRP	National Commission on Radiological Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NTA	Eastman Kodak Nuclear Track Emulsion type A

OCAS	Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
PBX	plastic-bonded explosive
PNL	Pacific Northwest Laboratory [also PNNL – Pacific Northwest National Laboratory]
PA	posterior-anterior
PAEC	potential alpha energy concentration
R	roentgen, unit of exposure to ionizing photons in air
REF	radiation effectiveness factor
RGD	radiation generating device
ROT	rotational
SRS	Savannah River Site
TBD	technical basis document
TLD	thermoluminescent dosimeter
WLM	working level month

1.0 INTRODUCTION

Technical basis 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 of 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 IAAP. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy facility" as defined in the Energy Employee Occupational Illness Compensation Program Act of 2000 (42 U.S.C. § 7384l (5) and (12)).

This document provides information on interpretation of dosimetry records and exposure matrices to estimate internal and external doses for workers at the Iowa Army Ammunition Plant in Burlington, Iowa. The Iowa Army Ammunition Plant was responsible for high explosive (HE) fabrication, assembly of non-nuclear and nuclear components, retrofits, modifications, surveillance, and disassembly of nuclear weapons.

2.0 SITE DESCRIPTION, OPERATIONAL HISTORY, AND PROCESS

The Iowa Army Ammunition Plant (IAAP) is a load, assemble, and pack munitions facility that began production in 1941 and continues to operate as a Government-owned, contractor-operated installation. IAAP is in the southeastern part of Iowa, near the town of Middletown in Des Moines County. It is about 10 miles west of the Mississippi River and the town of Burlington). Less than a third of the IAAP's 19,015-acre (30-square-mile) property is occupied by active or formerly active production or storage facilities. The remaining land is evenly divided between leased agricultural acreage and woodlands.

Since operations begin in 1941, IAAP has used explosives and lead-based initiating compounds to produce a wide variety of ordnance items. The Line 1 area, portions of the Firing Site (FS) area, the Explosive Disposal Area (EDA) sites, and Yards C, G, and L came under the jurisdiction of the Atomic Energy Commission [AEC; now the U. S. Department of Energy (DOE)]. In addition, the Security Command Center, the Emergency Response Command Post, the Deactivation furnace, Line 3 Warehouse 301, and the North Burn Pads Landfill might have been utilized. This area, totaling around 1,630 acres, became known as the Burlington Atomic Energy Commission Plant (BAECP). The site was officially renamed the Iowa Army Ammunition Plant in 1965. The site has also been referred to as the Iowa Ordnance Plant. Specific buildings and the functions of various radiological buildings can be found in Appendix A of this report.

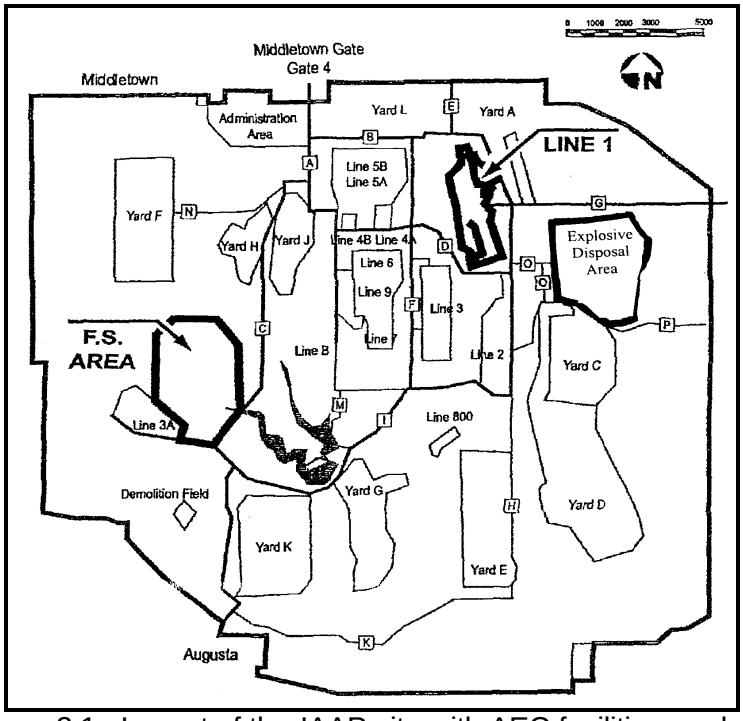


Figure 2.1. Layout of the IAAP site with AEC facilities marked.

2.1 OPERATIONAL HISTORY

During the summer of 1947, Silas Mason Company entered into a contract with the Ordnance Department to assist in the design and engineering to perform the construction and to operate a facility for the purpose of supplying the Atomic Energy Commission (AEC) with explosive components nuclear weapons. In May 1948, Silas and Mason Company supervisor personnel entered a training program at the Naval Ordnance Test Station (China Lake, California). Upon returning from training, these supervisors trained other IAAP personnel. By the spring of 1949, IAAP was at full production for HE fabrication (Mitchell 2003). In March 1949, it was decided that certain weapon assembly operations (non-nuclear components) would also be conducted at IAAP (Poole and Harrison, 1954, Mitchell, 2003). Until March 1949, all of the initial work at IAAP focused solely on HE explosive manufacturing.

The initial Iowa Army Ammunition Plant (IAAP) site profile reserved the time period from 1947-1958 due to a lack of information about the early time period operations. Based on a review of IAAP Project History reports (Poole and Harrison, 1954), NIOSH has concluded that the first nuclear weapon assembly operations began in 1949 with the Mark IV.

Throughout the remaining years until plant closure, IAAP workers tested, assembled, conducted surveillance and disassembled a wide variety of nuclear weapons. Through a review of the IAAP project history reports, records sent to Pantex from the Burlington Plant, and other records stored at the Pantex plant, NIOSH has compiled a list of the weapons programs worked on at IAAP (Appendix B). With the assistance of the Department of Energy (DOE), the first assembly and disassembly years are also provided.

2.2 FISSILE MATERIALS ONSITE

The early time period in the initial site profile (ORAU, 2004a) was reserved due to uncertainty as to when fissile materials (plutonium or uranium pits) were onsite. Based on the review conducted during the development of the initial site profile, NIOSH felt certain that fissile materials were onsite at least in 1958 forward. The concern for these materials is that they are generally the most radioactive component of a nuclear weapon and result in the largest external dose and if unsealed represent the greatest potential for internal dose. NIOSH has uncovered considerable evidence that indicated fissile materials were not onsite at IAAP until 1955. This evidence considers IAAP's mission, early weapon design and the development of a radiation safety program.

2.2.1 Early Mission and Responsibilities

According to documentation reviewed, the Burlington Plant was responsible for High Explosive (HE) fabrication, assembly of non-nuclear components, retrofits, modifications, and disassembly for weapon retirements (Mitchell, 2003).

Further documentation indicates that 1956 was the first year that Mason & Hanger company handled fissionable material (Lemert, 1979).

> *Pantex opened in 1953 with the Procter & Gamble Company as operating Contractor. Three years later Mason & Hanger – first to manufacture the explosive components – recorded two more firsts at the Burlington AEC Plant. The company began to assemble nuclear weapons for guided missiles, and in doing so it handled the fissionable material for the first time. Beginning with the introduction of plutonium and Uranium 235 into the configuration of the Genie air-to-air missile, the list of nuclear weapons assembled at Burlington lengthened to include artillery fired projectiles of various sizes and warheads for other air-to-air, air-to-ground, and ground-to-ground missiles (Lemert, 1979).*

2.2.2 Early Weapons Design

Early weapons (Mark 4, 5, 6, 7, and 12) assembled at IAAP were called In-Flight-Insertable (IFI) weapons. These weapons were of ball (tamper) and capsule (pit or special nuclear material (SNM)) design (Mitchell 2003, Loeber 2002). The main purpose of this design was to prevent an accidental nuclear detonation during transport and delivery. In the IFI weapon design, the nuclear capsules (pits) were kept physically separated (outside) the tamper and the surrounding high explosives. Prior to use of the weapon, the nuclear capsule (pit) would be inserted into the ball (tamper). From a review of the IAAP History of Line 1 Operations (Ahlstrand, 1955), non-radioactive "mock-up" pits were used during assembly. This was likely done to ensure an exact fit (i.e., tolerances were met) with the tamper once a real nuclear capsule (pit) was combined with the completed (assembled) explosives packages of the weapons.

In the mid 1950s, a new "sealed pit" was designed, tested and produced. The following excerpt was found in DOE documentation (Mitchell 2003):

> *In this design, the non-nuclear mechanical assembly was built around the nuclear capsule, or pit; and the weapon, in the case of a bomb, was ready for use when it left*

> *the assembly plant. In the case of a warhead, it was the military's responsibility to mate the AEC-delivered warhead to the military's delivery system, usually a missile. Working two 12-hour shifts, (seven days a week) at the Burlington Plant, this new design entered the stockpile in December 1956, in an "Emergency Capabilities" status as the W-25 warhead.*

This encapsulated pit encompassed a "fail-safe" design such that accidental detonation was highly improbable. With this new design, nuclear components (fissile pits) were assembled with the explosive package such that a complete weapon was the final product from IAAP. As a result, the first IAAP assembly workers would have been exposed to fissile materials in late 1955 or early 1956 with the receipt of the first sealed pits. This is also evident in other information obtained from Mitchell (2003).

> *The sealed pit design weapons precipitated several fundamental changes in the nuclear weapons complex. New facility designs were required and constructed at Burlington and Pantex to accommodate production work involving encapsulated SNM for the first time at either site.*

According to the History of Line 1 Operations (Ahlstrand, 1957b), the first "Gravel Gertie" assembly buildings at IAAP were placed in operation in October 1957.

2.2.3 Radiation Safety Program

According to the IAAP History of Line 1 Operations (Ahlstrand, 1956a, 1956b, 1957b), with the planned assembly of these new weapons designs, IAAP instituted a radiological monitoring program. This program primarily consisted of routine external dose monitoring for some workers using film badges, but also included a radiological survey program, continuous air monitoring, and training of the workforce during safety meetings. Some of the training included specialized training for urinalysis. A chronology of significant events can be found in Table 2.1.

2.2.4 Fissile Material Assumption

Although the records indicate that fissile materials were not routinely handled at IAAP prior to the W-25 assembly in 1956, the possibility that nuclear weapons taken out of the stockpile for retrofit or disassembly were returned to IAAP with the fissile material capsules cannot be ruled out.

The first assembly operations began with the Mark IV in March 1949. Following a two year production schedule, the plant was shutdown in October 1951 for construction to enlarge the assembly operation for the next generation of weapon assembly (Poole and Harrision 1954, Lemert 1979). Operations were restarted approximately 8 months later in May 1952. As a result the first logical point when retrofits could have been conducted on the Mark IV would likely have been after May 1952.

Since NIOSH cannot definitively state when fissile materials first appeared onsite, to error in favor of the claimant, NIOSH assumes that fissile materials were onsite since the beginning of assembly operations in 1949. Since workers were not routinely monitored onsite prior to 1962, NIOSH has scaled the external dose distributions based on differences in dose rate

from later pits designs to a "generic" pit. This methodology is discussed in section 6.5 of this site profile.

Table 2.1 Significant Radiation Safety Program Events

Time	Event	Reference
November 1955	Selected workers issued film dosimetry on an intermittent basis. Badge exchange frequency was weekly.	TracerLab Reports (1955)
March 1956	Safety Manager attends Rad-Safe training at LASL.	Ahlstrand (1956a)
July 1956	Formal establishment of a Rad-Safe program Training of all Division B Personnel Acquirement of Rad-Safe Equipment Established Continuous Air Monitoring	Ahlstrand (1956b)
July 1957	The following excerpt is from the IAAP Project History Report in 1957. *Continued emphasis on Rad-Safe Training for the safe handling of radioactive materials in order to prepare all workers for the new phases of work in division "B". Even though this material is received at this installation in sealed containers and is low energy radiation, employees are trained in all phases of safe handling of these materials.*	Ahlstrand (1957b)
September 1957	Establishment of a team for Off-site Emergency Radiological Monitoring	Ahlstrand (1957b)
January – June 1958	Extensive Rad-Safe training conducted including training on urinalysis. 72 Employees attended Primer Course 2T 7 employees spent a week in April at the Nevada Test Site for special Rad-safe training The Plant's Registered nurses and certain members of the Safety, Mechanical and Chemical Laboratory departments spent approximately 200 hours special training on urinalysis procedures and the operation and maintenance of monitoring equipment.	Ahlstrand (1958a)
July 1959	Radiation Safety inspector visits Pantex and LASL to discuss radiological and health problems.	Ahlstrand (1959b)
1962	First instance of AEC Form 191 documenting no internal exposures at IAAP during the year.	AEC Report 191 (1962-1974)
September 1962	Routine issuance of film dosimeter badges to personnel. Badge exchange frequency was weekly. First routine area dosimeters in buildings 1-11 and 1-77.	Landauer Reports (1962)
January 1963	Badge exchange frequency changed to bi-weekly.	Landauer Reports (1963)
January 1964	Badge exchange frequency changed to 4 weeks. A total of 13 cycles per year. Some highly exposed workers continued on bi-weekly basis.	
1968	Significant increase (doubling) in the issuance of dosimeters to personnel.	Landauer Reports (1968)
October 1969	IAAP establishes hand monitoring program (extremity dosimeters) following an AEC recommendation in 1969: *Hand exposure studies have been made in the past, with results that indicate only nominal exposure. However, no hand monitoring has been done within the past year. With new items that are coming into production, and the strong possibility of a reduced exposure guide for extremities, a program of hand monitoring should be provided for those employees who work directly with the radioactive units that have significant surface dose rates.*	Davis (1969)

In summary, the assumption that fissile materials were onsite since March 1949 is considered claimant favorable for the following reasons.

- Evidence that prior to the assembly of the W-25 in 1957, non-radioactive model pits (mock-ups) were used for assembly (Ahlstrand 1955).

- From NIOSH's review of the semi-annual reports, the only potential contact with fissile materials would be during retrofit or during disassembly operations, whereas the dominant process was assembly.

- Other documentation reviewed (Mitchell 2003) indicates that the nation's nuclear capsules (pits) were initially stored at Sandia and Los Alamos. The list of storage areas greatly expanded through 1957 to encompass military bases in Kentucky, New Mexico, Louisiana, Maine, South Dakota, Washington, California, Virginia, Nevada, Massachusetts, Texas, and New York. From the documentation it is apparent that, during assembly and transport, nuclear capsules were initially kept and stored separately from the high explosives manufactured by IAAP.

- IAAP History of Line 1 Operations Reports on the production of the W-25 by the end of 1956 correlates with initial personnel monitoring beginning in November 1955 and the institution of a radiation safety program.

3.0 OCCUPATIONALLY RELATED MEDICAL X-RAYS

Medical examinations at the IAAP were required as a condition of employment. Generally, an annual chest X-ray was given to each IAAP employee. During the worker outreach meeting in July 2004, some workers indicated that they received chest X-rays on a semi-annual basis, while others indicated that it was more frequent on a quarterly basis. Assuming medical monitoring practices were similar between IAAP and Pantex, there is evidence that male employees in certain job categories (i.e., heavy lifters) received lumbar spine examinations, the frequency of which was not available. At Pantex, lumbar spine examinations were given to men when they were hired to check for pre-existing back conditions. As a result a single lumbar spine x-ray should be assumed for occupations that involve heavy lifting such as (assembly workers, transportation personnel, crafts, etc…)

Background information on X-ray doses can be found in *Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures* (ORAU 2003a). Only limited data was found within the 1973 IAAP dosimetry records concerning occupational medical X-ray exposures at the hospital. These records indicate an exposure study was conducted in which several Landauer film badges were exposed to select X-ray machine settings. Table 3.1 below summarizes the X-ray settings and measured film badge dose for this special study.

Table 3.1 Results of X-ray study

#	Amperage (mA)	Voltage (kVp)	Duration (sec)	Dosimeter Reading (mR)
1	200 mA	85 kVp	0.1 s	20
2	100 mA	95 kVp	0.6 s	690
3	100 mA	70 kVp	0.4 s	300
4	100 mA	45 kVp	0.05 s	0

A chest exposure of 20 mAs (200mA x 0.1s) is fairly common for typical chest exposures as indicated in exposure #1. Since this exposure was first in the series, it is expected that these were the default values for the common chest X-ray. As indicated in ORAUT-OTIB-0006, X-ray measurements with films results in large uncertainty, and great care is needed to properly evaluate the dose. This study likely did not consider the calibration of the commercial film badge nor the effects that filtration has on the resulting spectral energy. As a result, the 20 mR reading could be either an underestimate or an overestimate depending on a multitude of variables. Because of this, the claimant favorable default values provided in ORAUT-OTIB-0006 should be used for chest X-rays (Tables 3.3-1 and 4.0-1).

The dose reconstructor should assume an annual posterior-anterior (PA) chest X-ray for all employees applicable from 1947 through 1975. Semiannual chest X-rays should be assumed for all assembly workers, and quarterly chest X-rays should be assumed for workers involved in radiography. If the job description is unknown, the claimant favorable default assumption should be semi-annual examinations. Do not apply X-ray dose for years other than 1947 through 1975, regardless if the worker was employed at IAAP during other years.

ORAU (2003a) does not provide default values for lumbar spine examinations. The Occupational Medical Dose TBD for the Rocky Flats site (ORAU 2004b) provides a method for calculating organ doses from lumbar spine examinations. Estimated median entrance skin exposures were 1.79 R for the AP view and 5.79 R for the lateral view based on information in Lincoln and Gupton (1958). Distributions of entrance skin exposures were created using the Crystal Ball® computer program (Decisioneering Inc. 2000) for different filtration half-value layers (HVL) as presented in Lincoln and Gupton. Tables A2 through A8 in ICRP Publication 34 provide organ doses in units of mGy per Gray entrance skin exposure for the thyroid, ovaries, testes, lungs, female breast, uterus, and active bone marrow (ICRP 1982). For practical purposes, the units can be considered mrem per R. Multiplying the entrance skin exposure to the kerma-to-organ dose factors in ICRP Publication 34 results in the organ doses and geometric standard deviations listed in Table 3.2, varying the HVL from 1.5 to 3.0 mm AL. ORAU (2003a) provides guidance for dose to organs not provided in ICRP 34 by relating their proximity to organs that are listed in ICRP 34. To account for a field of direct exposure larger than that used by ICRP 34 (as might have happened in the 1940s and 50s), additional organs were added to the ovary category, including stomach, kidneys, adrenals, and pancreas. Organs not listed by ICRP 34 added to the modeled organ based on proximity to the ICRP 34 organ (Table 3.2).

For skin and testes, consistent with the approach taken in the Rocky Flats Occupational Medical Dose TBD (ORAU 2003b), the values measured in the Lincoln and Gupton (1958) paper were used directly to account for the difference in collimation in the Lincoln and Gupton (1958) measurements versus ICRP 34 factors.

The AP view should be used unless person-specific information is found indicating otherwise. The lumbar spine organ dose should be assigned to males in the year of hire from 1947 through July 1975. The dose is assigned in the year of hire and is assumed to be lognormally distributed.

Table 3.2 Organ doses from lumbar spine X-rays at IAAP.

Organ	View	Organ Dose Geometric Mean mrem Geo. Std. Dev. (GSD)
Thyroid, eye, brain	AP LAT	0.50 (3.0) 0.10 (1.9)
Ovaries, liver, gall bladder, stomach, intestines, colon, rectum, kidneys, adrenals, pancreas, spleen	AP LAT	330 (2.6) 230 (2.1)
Lungs, thymus, esophagus, bone surfaces	AP LAT	120 (2.5) 64 (2.0)
Active bone marrow	AP LAT	58 (2.6) 110 (2.1)
Testes[a]	AP LAT	26 (3.1) 40 (1.5)
Skin[a]	AP LAT	1,800 (2.4) 5,800 (1.8)

a. Calculated from information in Lincoln and Gupton (1958) as presented in the Rocky Flats site TBD (ORAU 2004b).

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

4.1 INTRODUCTION

The occupational environmental dose refers to the dose received by workers outside of the normal production facilities. These doses can be internal and external depending on the characteristics of the individual radionuclides. Radionuclides present at IAAP include tritium, uranium, plutonium, and thorium. Although there are some enhanced exposures to naturally occurring radon from certain structures, no noble gases are used or released at the IAAP site. While most radionuclides when inhaled would give a dose to particular organs in the body, tritium gas would give a dose to the whole body. These radionuclides are addressed in the following sections.

4.2 INTAKES FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

Intakes to workers outside facilities are determined from air concentrations that resulted from individual facility releases and ground-level releases (e.g. burning activities). Unmonitored workers could have received internal or external occupational doses (or both) from any or all of these sources.

4.2.1 Intakes from Tritium Releases

The University of Iowa Needs Assessment for IAAP (Fuortes 2001) quotes an annual release of 0.006 Ci (6,000 µCi) of tritium from the site; however, an effluent summary document lists a total of 0.13 Ci for the period December 1965 through December 1970, for an average of 26,000 µCi per year. To be claimant-favorable, the latter value was used. In comparison to other sites that handled tritium, this release level is very small and provides some indication that not much tritium escaped containment. It is likely that most of the tritium released from stacks or vents was tritium gas, whereas the worker dose would come from tritiated water.

To estimate the intakes to workers outside facilities when little or no atmospheric information is available, the NCRP has suggested screening techniques. These techniques were originally published as NCRP Commentary No. 3 (NCRP 1989) and updated in NCRP Report 123 (NCRP 1996). The NCRP recommends a graded approach, with three screening levels. Level 1 is the most conservative and requires the least amount of input information. The Level 1 method for determining a conservative upper bound air concentration can be written as:

$$X(pCi/m^3) = \frac{f \times Q(pCi/\sec)}{V(m^3/\sec)} \qquad \textit{Equation 4.1}$$

Where:
- X = annual average upper-bound air concentration,
- f = assumed fraction of time the wind blows in the direction of the subject, assumed to be 0.25,
- Q = release rate of the radionuclide from the source, and
- V = volumetric flow rate of the vent. The default value is 0.3 m^3/sec, type of hood ventilation rates.

The model essentially assumes that the subject breathes undiluted effluent from the vent or stack, slightly modified by the fraction of time the wind blows in his/her direction (the factor of 0.25).

Supporting documentation for Federal Guidance Report 13 (Eckerman et al 1999) shows that dose per unit intake of tritium in the form of water vapor (HTO) is 10,000 times larger than the dose per unit intake for elemental tritium gas (HT). Because tritium (as water) can be absorbed through the skin, the tritium inhalation intake is multiplied by a factor of 1.5 to obtain the total intake of tritium water.

With the NCRP assumptions, the annual average air concentration near the release point should be less than:

$$26{,}000\ \mu Ci/yr * 1 \times 10^6\ pCi/\mu Ci * 0.25\ */ (3.15 \times 10^7\ sec/yr * 0.3\ m^3\ /sec) = 688\ pCi/m^3$$

Assuming a breathing rate of 2400 m^3/yr and assuming all the tritium is in the form of water results in an estimated annual intake of

$$688\ pCi/m^3\ * 1.5 * 2{,}400\ m^3\ = 2.5 \times 10^6\ pCi$$

This intake is claimant-favorable because it uses the highest value for tritium releases, assumes minimal dilution between the release point and the worker, and much, perhaps most, of the inhaled tritium would have been tritium gas. The dose reconstructor should assign a 2,500,000-pCi intake per year as HTO to all workers (chronic, 6,800 pCi/D). The annual doses from these intakes are constant upper bound distribution types for input into IREP.

4.2.2 Intakes from Release of DU from Burning Sites

High explosives contaminated with DU were routinely burned in the Explosive Disposal Area, which is an irregularly shaped region of slightly less than 1 square mile just north of the "C" Yard and about 1 km south southwest of Line 1.

The source term has been determined to be about 2,000 g/year of DU (TN & Associates 2001). This is an estimate of the material handled in the burn yard. Burning of DU-contaminated high explosives can be assumed to create aerosolized particles of DU. The most likely form of uranium released in the air from the burning would be in the form of an oxide, although the TN & Associates report suggested that much of the metal might not have been oxidized because the temperature was too low and the burning of explosives too rapid. Because uranium metal and some uranium oxides can exist in a chemical form associated with absorption type M, dose reconstructors should assume exposure to either type S or type M to maximize the dose to the organ/tissue of concern.

The burning was sufficiently frequent that modeling might consider it a continuous source during normal working hours. Thus, the 2,000 g/yr can be estimated as 1 g/work-hr.
The NCRP screening models for atmospheric releases provide a generic and conservative approach for estimating atmospheric dispersion (NCRP 1996). This approach is depicted in equation 4.2.

$$C = \frac{f\,Q\,P}{u} \qquad \textit{Equation 4.2}$$

Where: C = the annual average air concentration, g/m^3,
f = the fraction of time the wind blows in the direction of the subject, assumed to be 0.25
Q = the release rate, g/sec
u = the average wind speed, assumed to be 2 m/sec.
P = the Gaussian diffusion factor appropriate for down wind distances x

Values of P are provided in NCRP (1996, Figure 2.2). For ground-level releases for distances of 100 m, 500 m, and 1000 m, the values are 3.5×10^{-3}, 2.0×10^{-4}, and 5×10^{-5}, respectively.

Individuals not directly involved in the burning operations could have been anywhere on the IAAP Site. The nearest portions of the C Yard are about 500 m, and the nearest portions of Line 1 are about 1,000 m away. The average calculated air concentration at 500 m is about 6.9×10^{-9} g/m³; and at 1,000 m C is about 1.7×10^{-9} g/m³, assuming that all the DU becomes airborne, certainly a conservative upper bound. This approach neglects lofting of the plume caused by heat from the fire, which would reduce the calculated concentrations.

Assuming a breathing rate of 2400 m³/yr, the amount inhaled at a location in or near the C Yard would be:

$$6.9 \times 10^{-9}\ g/m^3 * 2{,}400\ m^3/yr = 1.7 \times 10^{-5}\ g/yr \text{ of } 6.3\ pCi/yr\ DU$$

Similarly, for locations at or beyond Line 1:

$$1.7 \times 10^{-9}\ g/m^3 * 2{,}400\ m^3/yr = 4.1 \times 10^{-6}\ g/yr \text{ or } 1.5\ pCi/yr\ DU$$

If there is no information regarding location of worker activities, use the claimant-favorable C yard value.

4.2.3 DU Intakes from Potentially Contaminated Drinking Water

Drinking water for the Iowa Army Ammunition Plant (IAAP) during the period of operation by the Atomic Energy Commission was obtained from Mathes Lake (also referred to as Long Creek Lake) which is within the IAAP site perimeter. Mathes Lake was fed in part by Long Creek which drained the hydroshot firing site designated as FS-12. FS-12 was the principal firing site for hydroshots containing depleted uranium (DU) from 1965 through 1973. Hence, there was potential for DU contamination in Long Creek, Mathes Lake, and drinking water for the site.

Results of uranium measurements from the water sampling program that have been discovered to date are shown in Table 4.1. The reported results are for total uranium with no distinction made between DU and natural uranium. Fluorimetry was the analysis technique used for the uranium measurement. The apparent detection limit for this technique was 1.0 µg/L

Table 4.1 Total uranium results from water samples at IAAP[a]

Sample date	Location	Measured U concentration (µg/L)	U concentration in pCi/L assuming 100% DU
12/1968	Long Creek between FS-12	Trace	<0.37
12/1968	SW of FS-12	1.2	0.44
12/1968	Reservoir	0	<0.37
12/1968	Drinking water I-04 Bldg.	0	<0.37
11/1969	Long Creek between FS-12 and lake	Trace	<0.37
11/1969	SW of FS-12	1.2	0.44
11/1969	Reservoir	0	<0.37
9/1970	Long Creek between FS-12 and lake	<1.0	<0.37
9/1970	SW of FS-12	1.0	0.37
9/1970	Boat dock in lake	<1.0	<0.37
9/1970	Reservoir	<1.0	<0.37
3/1971	Long Creek between FS-12 and lake	2.0	0.74
3/1971	SW of FS-12	<0.4	<0.15
3/1972	Long Creek influent	2.20	0.82
3/1972	Long Creek influent	1.40	0.52
4/1973	Long Creek at FS Bridge	0	<0.37
4/1973	Drinking water in I-04 Bldg.	0	<0.37

a. From Shahan 1970, Shahan 1971, Meek and Shahan 1972, Holmberg 1974.

The Technical Report No. 181, Part 5 (Meek and Shahan 1972), from which the 1972 data were obtained, stated, "Experience at this installation indicates that the typical or background concentration of uranium in this area is as follows: Small Streams 1-3 µg/L…" The report also concludes, "Only background amounts of uranium (D-38) concentrations were detected in the various plant streams…" It is hard to judge for sure whether the concentrations of uranium in Long Creek were due to DU transported from the firing site or from natural uranium, although the site safety staff were of the opinion that the material was natural background uranium.

Even if there was DU in the intake to the water treatment plant from Mathes Lake, some of it would have been removed in the treatment plant. Water treatment was described as,

> *"Water is treated to control odors and tastes by pre-exposure to activated carbon and chlorine. It is then subject to coagulation with ferric sulfate, lime, and soda ash. Following suspended solids removal and the softening process, it is clarified through rapid anthrafilt filters" (Shaykin 1969).*

The two measured concentrations in the drinking water (1968 and 1969) were less than detectable, as were all lake samples. With additional concentration reduction in the water treatment system, it is reasonable to assume that earlier and subsequent concentrations were also less than detectable.

Assuming that the upper bound drinking water concentration was just at the detection level, that all the activity was DU, that a worker might consume 2 L/day of water onsite, and adjusting for 250 working days per year to convert to intake per calendar day, the upper bound daily intake might have been:

intake (pCi/d) = (0.37 pCi/L) (2 L/work day) (250 work day/yr) / (365 cal. day per yr)

intake = 0.51 pCi/cal. day.

This ingestion intake rate was assigned to the entire period of the hydroshots, 1965 through 1973, and organ doses were determined using IMBA assuming all the activity was ^{234}U (f = 0.02) which maximizes the dose. The organs with the highest annual doses were the bone surface and kidneys. The doses to those organs are shown in Table 4.2. The highest annual organ dose is 0.3 mrem to the bone surface.

Maximizing assumptions included in the calculations were

- all of the uranium was DU from site operations
- the concentration in Mathes Lake was just at the detection limit for the analysis
- this concentration existed for the entire period 1965 through 1974
- no credit was taken for removal of uranium by the water treatment system
- all of the DU activity was in the soluble form, despite the source term originating from explosions
- workers consumed 2 L of the site drinking water each work day, year round
- ^{234}U was used for calculating organ doses.

Based on the maximizing assumptions and the calculations shown above, it can be concluded that the upper bound annual dose to any organ was less than 0.3 mrem. Annual doses to most organs were less than 0.1 mrem.

Table 4.2 Upper bound equivalent doses (mrem) from ingestion of DU contaminated drinking water

Calendar Year	Bone Surface Dose (mrem)	Kidney Dose (mrem)		Calendar Year	Bone Surface Dose (mrem)	Kidney Dose (mrem)
1965	9.80E-02	9.27E-02		1986	7.20E-02	1.40E-02
1966	1.46E-01	1.10E-01		1987	6.80E-02	1.22E-02
1967	1.78E-01	1.21E-01		1988	6.46E-02	1.08E-02
1968	2.06E-01	1.31E-01		1989	6.12E-02	9.44E-03
1969	2.29E-01	1.39E-01		1990	5.83E-02	8.30E-03
1970	2.51E-01	1.46E-01		1991	5.57E-02	7.30E-03
1971	2.70E-01	1.52E-01		1992	5.35E-02	6.44E-03
1972	2.88E-01	1.58E-01		1993	5.12E-02	5.66E-03
1973	3.03E-01	1.63E-01		1994	4.92E-02	5.00E-03
1974	2.19E-01	7.40E-02		1995	4.73E-02	4.41E-03
1975	1.83E-01	6.05E-02		1996	4.58E-02	3.91E-03
1976	1.64E-01	5.30E-02		1997	4.41E-02	3.46E-03
1977	1.48E-01	4.62E-02		1998	4.26E-02	3.07E-03
1978	1.33E-01	4.04E-02		1999	4.12E-02	2.72E-03
1979	1.21E-01	3.53E-02		2000	4.00E-02	2.43E-03
1980	1.11E-01	3.10E-02		2001	3.86E-02	2.16E-03
1981	1.02E-01	2.70E-02		2002	3.74E-02	1.93E-03
1982	9.43E-02	2.37E-02		2003	3.63E-02	1.73E-03
1983	8.76E-02	2.07E-02		2004	3.52E-02	1.55E-03
1984	8.19E-02	1.82E-02				
1985	7.65E-02	1.59E-02				

4.3 EXTERNAL DOSE

Dosimetry records for IAAP indicate that radiation workers were the only employees monitored for radiation exposure. These personnel worked primarily in facilities in Line 1, Yard C storage area, the Explosive Disposal Area, and the Firing Site. Radiation workers accounted for a small fraction of the workers on the site. Employees working in other areas of the site were not monitored; however, there was a small potential for external dose from occupational environmental sources.

Workers at IAAP were subjected to external doses from the ambient radiation levels on the site. From 1962 through 1974, bi-weekly film badges were used for area radiation monitoring. These area badges were placed in various assembly buildings, Gravel Gerties, and Storage Igloos. Most of this monitoring data (~ 70%) was below the detection limit of 10 mR (Figure 4.1). Assuming a 2080 hour work year and using the dose distribution data in figure 4.1, an onsite ambient dose is estimated to be 37 mR per year with a geometric standard deviation of 4.6.

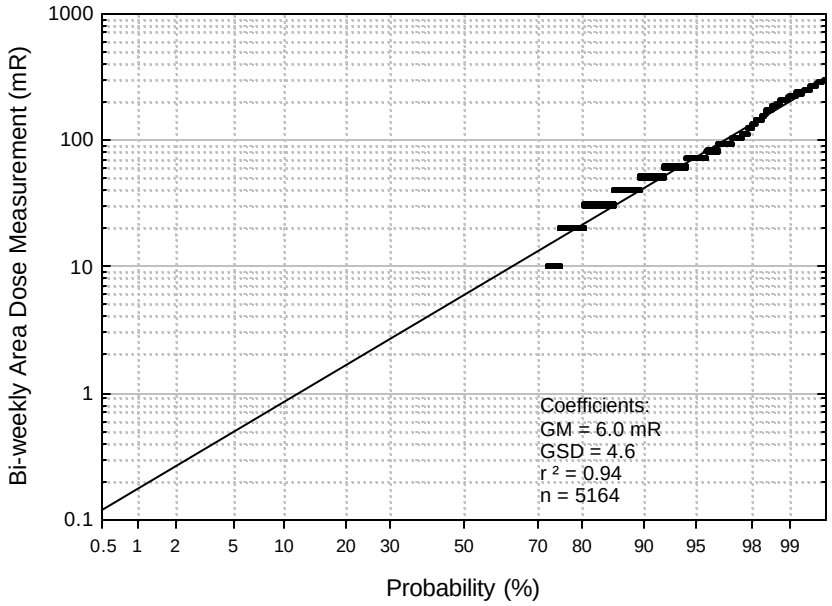


Figure 4.1 Probability plot of bi-weekly non-storage area dosimeters from 1969 - 1974

Considering the placement of the area dosimeters (i.e. near fissile material handling operations), the onsite ambient dose throughout the remainder of the plant was near background levels. Instead of using the distribution illustrated in Figure 4.1, the dose reconstructor can use an upper bound of 260 mrem/year to estimate the onsite ambient dose for the unexposed (non-line 1) workers. This estimate assumes a non-line 1 worker was exposed at the detection level of the area dosimeters for an entire year.

Although area dosimetry data in process areas was not abundantly available prior to 1969, extrapolation to the early time periods is considered claimant favorable. Area dosimeter data is available for the storage areas as shown in figure 4.2 from 1962 -1974. Also depicted in this figure is the upper 95th% of the non storage area distributions which are observed to be relatively stable and follow the general dose rates in the storage areas. As a result, the onsite dose distribution for non-line 1 workers is expected to be relatively stable throughout the operations.

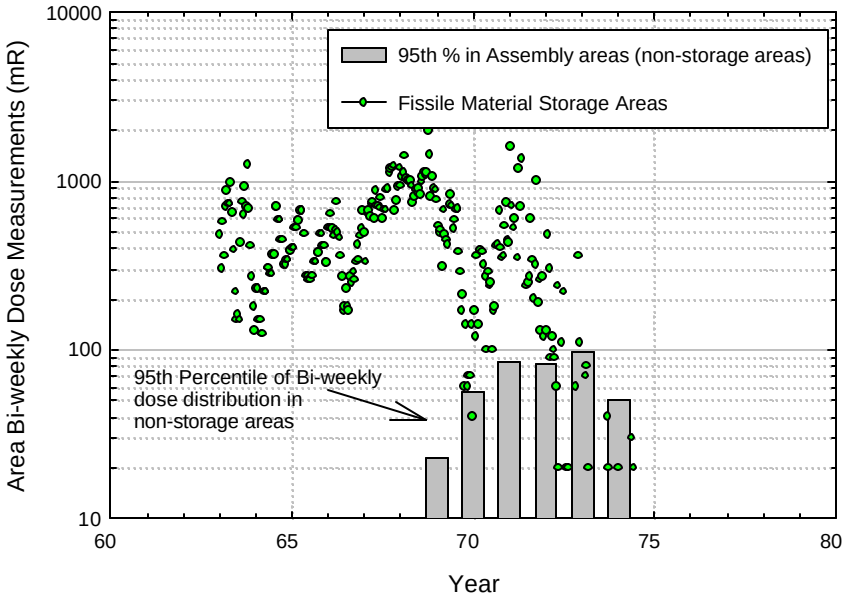


Figure 4.2 Plot of Area Monitoring data from 1962 – 1974

4.4 UNCERTAINTY

As discussed above, estimates of annual occupational environmental doses were based on accepted screening techniques (NCRP 1989, 1996). These techniques in themselves rely on considerable conservatism based on nominal values and uncertainties of known parametric values. In addition, the analyses made additional claimant-favorable assumptions, as stated. Because of the scarcity of available environmental data and the use of multiple claimant-favorable assumptions, the environmental intakes should be considered a constant upper bound.

5.0 ESTIMATION OF INTERNAL EXPOSURE

A 1969 Health Protection Appraisal (Shaykin 1969) indicated that swipe counting and direct surveys routinely monitored removable and non-removable radioactive contamination at IAAP. A personal interview with Joe Shannan, long-time (1958-1985) employee in the Safety Department and eventually Radiation Safety Manager, confirmed that shipments of radiological materials were swipe tested on entry and before being sent to the assembly facilities (Fix and Bihl 2003). The contents of the containers were swipe tested when the containers were opened. Mr. Shannan said that during his employment at the BAECP, contamination outside or inside the incoming containers was rare.

Based on materials used to assemble weapons and considering that the fissile material was encapsulated (sealed), the radionuclides most likely to result in an intake at IAAP were depleted uranium (DU, D38, or tuballoy) and tritium (^{3}H). A radiological survey of the plant

conducted in 2001, years after radiological work had ceased, found only DU and ^{137}Cs. The document providing the results of the survey also states, "Radioactive materials used at the line were received in a sealed configuration and were swipe tested before use. Known radioactive materials include depleted uranium (DU), enriched uranium (EU), plutonium, tritium gas, and polonium (^{210}Po)." The source of the ^{137}Cs on the swipes is unknown but seems inconsistent with the nature of the work. When questioned specifically on the source of the ^{137}Cs, Mr. Shannan said he was not aware of any ^{137}Cs used at IAAP other than small sealed sources at µCi levels as instrument check sources.

Selected employees were given bioassays to detect intakes of radioactive material. However, no bioassay records were discovered during the records search. Tritium gas reservoirs, which arrived sealed in metal containers, were purged under a ventilation hood prior to assembly into weapons. The actual emplacement of the reservoirs into the weapons did not involve breaking seals or the release of tritium gas. The plutonium and EU were also encapsulated and not available for intakes, even during disassembly. Pantex did some plutonium bioassay, but no intakes were ever recorded, with the exception of one well-documented accident. No reports of similar accidents at IAAP were found in the literature. If early IFI style weapons were returned from the field for retrofit and disassembly with the nuclear capsule, then ^{210}Po might be present. Polonium-210 was also encapsulated and was not further disassembled at IAAP.

Note: Intakes discussed below and in Sections 5 are summarized by year and work task in Table C-1 in Attachment C.

5.1 TRITIUM EXPOSURE

Tritium intakes could have occurred and probably did occur to a certain extent during weapons assembly and disassembly procedures. The tritium reservoirs came from the Savannah River Site (SRS), so the earliest possible date for tritium exposure at IAAP would be 1954. Based on a review of classified documentation, first weapon assembled that used tritium was after 1954, thus it is a claimant-favorable overestimate to assume that tritium intakes occurred each year that a worker was involved in assembly, disassembly, inspection, stores, shipping and receiving (and other duties with elevated tritium exposure potential) from 1954 to 1975.

The technology of tritium usage in nuclear weapons is classified. No tritium bioassay results for IAAP workers have been located. However, material and procedures at IAAP were almost certainly the same as those at Pantex because the same company operated both plants, and the materials and tasks were the same. This assumption was confirmed in a telephone interview with Mr. Herman Phillips, who was a safety engineer at Pantex but also worked for some time at IAAP. Hundreds of tritium bioassay results were obtained at Pantex in the 1970s and 1980s. The largest internal dose at Pantex from tritium recorded during any year in this period, with the exception of a major accident, was 122 mrem. Using the standard calculation for tritium in the 1970s, which used a quality factor of 1.7 [based on International Commission on Radiological Protection (ICRP) Publication 10 (ICRP 1968) and explained in NUREG-0938 (NRC 1983)][1.5 mCi = 425 mrem], 122 mrem was indicative of a chronic annual uptake of 430 µCi of tritium. The uptake as calculated using ICRP 10 methodology accounts for tritium in body fluids from any and all intake modes.

Routine IAAP tritium air sampling results from 1959 through 1964 were located and used to estimate the potential upper bound intakes for workers involved in tritium operations. Tritium reservoirs were shipped to the Iowa Ordnance Plant in pressurized metal vessels known as JP containers. Prior to the opening of the JP container, which contained the tritium reservoir, the pressurized air was vented in a controlled manner under a ventilation hood. The air purged from the container was surveyed with a T-289 tritium monitor to ensure that the tritium air concentrations were less than 90 µCi/m^3 before the container was released to production (Shaykin 1969).

Print out charts from the tritium monitors were used to estimate upper bound tritium intakes. It was assumed that 2 JP containers containing tritium air concentrations at the release level of 90 µCi/m^3 were opened each day in the relatively small enclosed space of a gravel gertie. The JP container volume was estimated to be 0.0136 m^3, based on an inner radius of 15.5 cm and a container height of 18 cm (DOE EIS-0225), thus the tritium activity present in the headspace would be about 1.22 µCi per container. Gravel gertie dimensions (DOE NV-710, 2001) were used to estimate a volume of 437 m^3 into which the tritium was dispersed. Thus, for intake and dose calculation purposes, it was assumed that 2.445 µCi of tritium was vented into the cell volume of 437 cubic meters for 365 days each year. The resulting upper bound air concentrations to which workers were potentially exposed was estimated to be 2.043 µCi/m^3. For intake calculations, it was assumed that an assembly/disassembly worker was involved in tritium operations for 8 hours per day of a 2000 hour work year, with a breathing rate of 1.2 m^3/hour. The resulting upper bound chronic tritium intake was estimated to be 4902 µCi/year, or 13.4 µCi/day, which corresponds to an upper bound tritium dose of 0.331 rem/year in IMBA Expert OCAS Edition.

In all AEC Form 191s generated at IAAP from 1962 to 1973 (AEC 1962-1973a), the site reported "no internal deposition during the period." In a health protection appraisal and report in September 1969, C. E. Davis, from the AEC Albuquerque Office, stated:

> *"Routine internal exposure monitoring is provided only for tritium. Two urine samples are analyzed biweekly for various selected individuals who work in areas where there is potential for exposure. To date, there has never been a positive result. If air samples, radiation survey results, or unusual conditions should indicate the possibility of internal exposure to any radioisotope, special bioassays would be necessary. There has been no occasion for such tests to date." (Davis 1969).*

Although the criteria for IAAP tritium monitoring, reporting levels and positive results have not been found, historical reviews of detection capabilities at other AEC programs indicate that the IAAP annual uptake of 4902 µCi would have been readily detectable and, therefore, this value is considered an upper estimate.

Another approach for estimating intakes of tritium uses tritium effluent values. There was some information regarding the release of tritium from the site in the 1970s. The University of Iowa Needs Assessment for IAAP (Fuortes 2001) quotes an annual release of 0.006 curie (6,000 µCi) of tritium from the site. But another 2001 report (TN & Associates 2001) quotes an effluent summary that indicates annual releases of about 0.026 Ci in the latter half of the 1960s. Either way, in comparison to other sites that handled tritium, this release level is very small and provides some indication that not much tritium ever escaped containment. Most of the tritium released from the stack was probably tritium gas, whereas the worker

dose comes from tritiated water. The 4902 µCi annual intakes estimated in the paragraphs above are almost 19% of the total airborne effluent for a typical year in the 1960s and are approximately 80% of the reported effluent for other years. It is improbable that any single worker had a chronic annual intake of the magnitude assumed for dose calculation purposes, especially not of tritiated water.

As discussed in the preceding paragraphs, the 4902 µCi intake is an overestimate, especially when applied for all years of employment; therefore, for IREP input the distribution type is constant. Because of the way IMBA handles tritium intakes, uptake equals intake (personal communication Anthony James 2003), so the 4902 µCi should be modeled as an 4902 µCi injection.

5.2 DEPLETED URANIUM

Intakes of DU might have occurred during disassembly of old, oxidized DU bomb parts, during hydrotesting, or during machining of baratols (explosive charges that surround the ball). DU emits less radiation per gram than natural uranium. By weight DU is essentially pure ^{238}U. Isotopic abundances of ^{234}U and ^{235}U in DU can vary, but those isotopes generally contribute less than 10% of the alpha radioactivity. Typical weight percents and activity fractions of uranium isotopes are listed in Table 1. These activity fractions are slightly different from the default values in IMBA, compiled for the NIOSH's Office of Compensation Analysis and Support (OCAS), but are within the variability of batches of DU. Either the values in Table 5.1 or the default IMBA values can be used.

Table 5.1 Mass and radiological characteristics of depleted uranium.

Weight percentage[a]	**DU**
^{234}U	0.0005
^{235}U	0.2500
^{236}U	Negligible
^{238}U	99.7500
Specific constituent activity in mixture (µCi/g, nCi/mg, or pCi/µg[b]	
^{234}U	0.0313
^{235}U	0.0054
^{236}U	Negligible
^{238}U	0.3352
Total	0.3718
Specific constituent activity in mixture (dpm/µg)[b]	
^{234}U	0.0694
^{235}U	0.0120
^{236}U	Negligible
^{238}U	0.7441
Total	0.8254

a. From Carbaugh 2003.
b. Can be used to represent specific alpha activity as well.

The chemical and radiological risks of DU were acknowledged at IAAP, but they were generally considered insignificant in relation to other chemical hazards (such as beryllium); therefore, only limited air sampling data were taken during DU operations. Concerning the

inhalation absorption type, the *Pantex Internal Dosimetry Technical Basis and Quality Assurance Document* states that "the compounds of uranium at Pantex are pure metal or air-oxides; it is assumed that all forms encountered will exhibit class Y aerosol behavior" (Pantex 2001, Section 13.2.1). The same document lists in Table 7.3 that ^{238}U should be considered 20% class D and 80% class Y. Uranium contamination at IAAP would be similar to that at Pantex with the exception of material detonated in hydroshots. The most likely form of uranium at IAAP would be very insoluble, associated with lung absorption type S and a gastrointestinal-tract-to-blood uptake factor, f1, of 0.002. However, uranium oxides can exist in many states, and it might be too simplistic to assume a pure absorption type when the chemical form is not known for certain. The dose reconstructor should assume either type M or type S to maximize the dose to the organ of concern. Exposure to type F uranium at IAAP is not considered credible.

5.2.1 DU Intakes from Hydroshots

From 1965 to 1973, there was a potential for workers to be exposed to DU oxide-bearing dust in proximity to the North Firing Site 12 (FS-12) immediately following the detonation of a hydroshot (hydrodynamic test)). In addition, there might have been a few DU tests at the South Firing Site 6 (FS-6), though the number there was small in comparison to FS-12. A hydroshot was a diagnostic operation that used DU as a surrogate for weapons-grade material, and was a quality control technique for measuring the performance of plastic-bonded explosives (PBX) produced at IAAP. In a personal interview, Mr. Shannan described the manner in which the outdoor hydroshot operations were conducted (Fix and Bihl 2003). One or two persons occupied the test fire control bunker, which was next to ground zero. All other site employees were kept outside a fenced area with the closest proximity about 1 mile from ground zero. A cable tunnel ran underground from the test fire control bunker to ground zero. A driver was at the fence gate. Within minutes of the explosion, the driver would enter the restricted area, pick up the workers in the bunker, and drive to the blast area to retrieve instruments. Then the workers would leave the fenced area. Neither the driver nor the control bunker operators wore respirators. According to Mr. Shannan, exposure to a plume would have been for a few minutes at most. Records indicate that 701 hydroshots occurred between 1965 and 1973 at FS-12, reportedly involving approximately 4,000 kg of DU (ATSDR 2003). Mr. Shannan said that shots were infrequent but could have been bunched, including more than one on a given day. Records list 530 hydroshots between December 2, 1965, and March 3, 1969, 3 hydroshots under a different program presumably between March 4 and July 14, 1969, and 168 hydroshots between July 15, 1969, and December 31, 1973.

The limited amount of air sampling performed after hydroshots in 1971 and 1972 (Meek and Shahan 1972) is summarized in Table 5.2. The data were plotted on log-probability paper to determine the geometric mean (or median) and geometric standard deviation (GSD) air concentration values. Only a few samples were taken at the control bunker, and the results were less than the 100-yard concentrations. Nevertheless, the data from the FS-12 tunnel were more robust, were more claimant-favorable, and compensate for the trip to the blast area after the dust had settled. Assuming an exposure of about 30 minutes for the operators and driver for each shot, the intake of DU for each of the periods listed in the above paragraph would be:

(air concentration $\mu g/m^3$) (1.2 m^3/hr breathing rate)(0.5 hr) (no. of shots in the period)

Table 5.2. Measured DU air concentrations from hydroshots.

Location	DU concentrations ($\mu g/m^3$)[a]	Geometric mean air concentration ($\mu g/m^3$)[b]
FS-12 tunnel	0.0 – 21.82	2.8
100 yards from shot[c]	0.0 – 9.12	0.9
1 mile from shot[d]	0.0 – 2.47	0.24

a. Air sampling data from Meek and Shahan (1972).
b. Taken directly from log-probability plot of air concentrations
c. Includes three data at 150 yards.
d. Includes one datum at 0.75 mile.

Based on a field investigation at Los Alamos Scientific Laboratories in November 1974, it was determined that approximately 10% of the total mass of uranium was aerosolized in a hydroshot. Data show that uranium particle sizes were lognormally distributed with an aerodynamic mass median diameter of 0.1 to 1 micron and a standard deviation of 8 for three separate experiments (Dahl and Johnson 1977). For FS-12 workers, an activity median aerodynamic diameter (AMAD) of 1 micron for uranium particles will be used in lieu of the IMBA default value of 5 micron AMAD. The DU intakes from hydroshots were assumed to be chronic, and lognormally distributed. For the drivers and test fire operators, the geometric mean of 2.8 $\mu g/m^3$ was used and the GSD was 4.6. Intakes for the three periods are provided in Tables 5.3 and C-1 (Table C-1 also has results in pCi/d). For input into IMBA, the intake per calendar day is needed, which was determined by dividing the total intake for the period by the calendar days in the period.

Table 5.3. Chronic DU intakes from hydroshots for test fire operators and drivers.

Period of exposure	Total intake (mg)	Calendar days	Daily chronic intake (mg)
December 2, 1965 through March 3, 1969	0.890	1187	7.5E-4
March 4, 1969 through July 14, 1969	0.00504	132	3.8E-5
July 15, 1969 through December 31, 1973	0.282	1630	1.7E-4

Everyone at the site might have been exposed to a small degree to the plumes from hydroshots. It is unlikely that the plumes always drifted in the same direction, and the decrease in air concentration as the plume moved across the site is not known. But because the nearest AEC facilities were about equally distant from the fence as was ground zero to the fence, an assumption of a factor-of-4 decrease from the mean of the 1-mile air concentration is claimant-favorable for an annual intake, especially considering the high density of DU in relation to dust or nearly all other types of airborne effluents and variable plume directions when averaged over a year. The assumed period of exposure at this distance was 2 hr (assumes turbulence type A at 4000 m and 1 m/s drift speed then doubled to ensure conservativism (claimant favorability)); other assumptions are the same as for the operators.

$$Intake = (0.24\ \mu g/m^3)(1.2\ m^3/hr\ breathing\ rate)(2.0\ hr)(no.\ of\ shots\ in\ the\ period)/4$$

Intakes for all other personnel from the hydroshots are provided in Table 5.4 and C-1. The intakes are chronic, lognormally distributed, with a GSD of 4.0, obtained from the probability plot.

Table 5.4 Chronic DU intakes from hydroshots for other AEC personnel.

Period of exposure	Total intake (mg)	Calendar days	Daily chronic intake (mg)
December 2, 1965 through March 3, 1969	7.63E-2	1187	6.4E-5
March 4, 1969 through July 14, 1969	4.32E-4	132	3.3E-6
July 15, 1969 through December 31, 1973	2.42E-2	1630	1.5E-5

An unsigned, undated record in the IAAP files (file locator 000916) indicates that FS-12 employees picked up pieces of D-38 metal lying around ground zero by hand without gloves to bag them as waste. Therefore, it is possible that a potential for ingestion of DU existed. Assuming 76 mg of DU contamination on hands, of which about 10% is ingested (assuming hands are not washed before eating), results in an ingestion of about 7.6 mg of DU per cleanup task. (Details of the calculation are provided in Appendix C.) There is no record of how many times cleanup of uranium pieces occurred; a claimant-favorable assumption is after each hydroshot. Results of the calculations are provided in Tables 5.5 and C-1. This should be modeled as chronic ingestion, insoluble material, constant upper bound.

Table 5.5 Chronic DU ingestion intakes from hydroshots for cleanup crew.

Period of Exposure	Total intake (mg)	Calendar days	Daily chronic intake (mg)
December 2, 1965 through March 3, 1969	4.0E3	1187	3.3
March 4, 1969 through July 14, 1969	2.3E1	132	0.17
July 15, 1969 through December 31, 1973	1.3E3	1630	0.78

In addition to the potential for intakes of DU during the hydroshots, there was also determined to be a potential for DU intake from the resuspension of contaminated soil following routine high explosives detonations at FS-12. Air monitoring data were located in a document titled: "Health Protection Survey of BAECP – 1974," (Holmberg, 1974) which indicated that the DU deposited in the soil immediately surrounding ground zero of the firing site was being resuspended during routine detonation of high explosives (non-hydroshots).

Several weekly continuous air monitoring results were reported in 1974 for the resuspension of DU during non-hydroshot detonations at FS-12. The highest air sample result located was reported to be 101.4 E-13 µCi/cm^3 for an 8 minute sample time. As the air sampling volume and time after the HE detonation increased, the airborne DU activity quickly dropped. The potential for DU resuspension was linked to the number of high explosive tests at FS-12 in any given day. Given that the exact number of high explosive tests could not be located, for the purpose of intake estimation and dose calculations, the second highest of the reported DU air concentrations (1.96 E-12 microcuries/cm^3) was assumed to have been continuous at FS-12 for the period of 1965 through 1975. Using the air monitoring data from 1974 to estimate intakes from 1965 – 1974 is claimant-favorable, since no additional depleted uranium was being deposited at FS-12 following the last hydroshot in 1973. It was assumed that an FS-12 employee would have been exposed to this level of airborne DU for an entire 40 hours per week, even though additional air monitoring data indicates that this continuous exposure potential did not exist. In addition, it was assumed that employees were exposed for 10 hours per week to the highest recorded DU air concentration of 101.3 E-13 µCi/cm^3. This approach overestimates the actual intake

potential and the subsequent dose resulting from the intake of DU at FS-12. The calculations for this exposure potential are provided below.

Constant DU air concentration = (1.96 E-12 µCi/cm^3) (1 E +6 pCi/µCi) (1 E + 6 cm^3/m^3)/(DU specific activity of 0.402 pci/µg) = 4.87 µg/m^3

Intermittent DU air concentration = (101.4 E – 13 µCi/cm^3)(1 E + 6 pCi/µCi)(1 E +6 cm^3/m^3)/(DU specific activity of 0.402 pCi/µg) = 25.22 µg/m^3

Intake = (4.87 µg/m^3)(1.2 m^3/hr)(40 hrs/week)(52 weeks/year) = 12156 µg/year

= (25.22 µg/m^3)(1.2 m^3/hr)(10 hrs/week)(52 weeks/year) = 15737 µg/year

Total Annual DU Intake from resuspension = 27893 µg/year or 11217 pCi DU / year

For input into IMBA Expert, a chronic daily upper bound inhalation intake of 30.7 pCi/day for depleted uranium should be assigned as either absorption Type M or S.

5.2.2 DU Intakes from Machining Baratols

On Line 1, from 1948 through about 1962 (TN & Associates 2001), the first step of the production process was the casting of baratols. Machining or grinding these components might have released small quantities of DU if conducted during assembly. Although DU was not machined directly, unintentional "nicking" of the DU occasionally occurred during machining on the explosive charges. DU-contaminated explosive waste was reportedly taken to the Explosives Disposal Area burn pads for burning. Beginning in about 1962, the process of casting baratols was replaced by a new process that involved pressing explosives in a plastic state into molds. Thus, the need for machining was eliminated (ATSDR 2003).

Mr. Shannan confirmed that machining directly on DU was not done because it produces hot, smoldering filings that would have been extremely hazardous because of the intimate proximity to explosives (Fix and Bihl 2003). Machining on contaminated metals might have produced some airborne contamination and cleanup operations around the machines and have created low, temporary airborne concentrations of DU. An assumption of some intake of DU by the machinists is reasonable, although not comparable to sites where actual machining on uranium occurred. Airborne DU contamination was probably intermittent and did not exist for the full 40 work hours every week, and it is unlikely that any worker was exposed for the full 40 hours each week. An exposure at 2% of the maximum permissible air concentration (MPC) for 20 hours per week is assumed as an upper bound for machining or cleaning around the machines. This assumes concentration of airborne contamination is consistent with values measured at the Hanford Site from machining of uranium and cleanup of machinery at a fuel fabrication plant (Wilson 1958), and is believed to be claimant-favorable because the source term at IAAP was much smaller. The MPC for insoluble ^{238}U (which would apply to DU as well) established by the National Commission on Radiological Protection and Measurements (NCRP) in 1959 was 1 x 10^{-10} µCi/cm^3 (NBS 1959). Assuming a breathing rate of 9.6 m^3 per workday (light work) results in a chronic intake of DU of about 2.4 x 10^{-3} µCi per year (6.6 pCi/day for input into IMBA).

In addition, the machinists might have ingested DU by transfer from contaminated hands to food or cigarettes. For estimating ingestion resulting from contamination inside buildings, ORAU recommends a daily ingestion of 0.2 times the air activity per m^3 (ORAU 2004c). This approach includes ingestion from transfer from hands and settling of contamination onto open sources of drink, such as a coffee mug. The approach assumes continuous settling of material from the air onto surfaces for 24 hours per day, 7 days per week. The air concentration discussed in the preceding paragraph (2% MPC) was assumed to apply only about half the time, so the average continuous air concentration would have been 1% MPC or 1×10^{-12} µCi/cm^3. The daily ingestion intake would have been

$$\textit{Ingestion Intake} = (0.2)(1 \times 10^{-12}\ \mu Ci/cm^3)(10^6\ pCi/\mu Ci)(10^6\ cm^3/m^3) = 0.2\ pCi/d.$$

0.2 pCi/d should be the mode of a triangular distribution with the minimum at 0.1 pCi/d (no open drinks) and the maximum at 0.4 pCi/d (to account for the possibility that some contaminants on the hands may have come from handling a baratol that was contaminated on the surface as opposed to touching general work surfaces (infrequent but possible when the DU was "nicked").

5.2.3 DU Intakes by Operators at Burning Yard

About 2,000 g/yr of DU as contamination on scrap explosive components was burned at the Explosive Disposal Area (TN & Associates 2001). Section 5.2.2 addresses inhalation by general plant workers from the effluent of the burning. The ash was bagged and shipped off the site. Intakes might have occurred during the bagging of the ash. Probably more than 99% of the DU remained in the ash, so 100% was assumed (airborne release fractions from burning DU are generally 10^{-3} or 10^{-4} [DOE 1994]). This means about 10 g/workday was bagged. Even when mixed with non radioactive ash from the explosives, the total amount of ash bagged per day was small and should have taken only a few minutes to sweep up and dispose of in a bag or drum. Airborne release fractions and respirable fractions of radioactive materials, including uranium, under many different scenarios have been compiled by Doe (DOE 1994, pages 4-9). The scenario considered most appropriate for bagging ash was described as "free-fall spill of cohesionless powders: free-fall <3 m, air velocity normal to powder flow, general forced enclosure ventilation or low-wind outside conditions." The median airborne release fraction was 3×10^{-4} and the median respirable fraction was 0.5; the upper bound values for the same parameters were 2×10^{-3} and 0.3. The upper bound values might apply to an acute event, but for daily intakes the median values were considered more appropriate. It was assumed that the dust produced from this process was dispersed in 1 m^3 of air. Because of the small amount of ash, 5 minutes was assumed as the time for gathering the ash and disposing of it in a bag or drum. The DU air concentration from this activity was:

$$\textit{(source µg/d)(airborne release fraction)(respirable fraction)/(air vol. } m^3)$$

$$\textit{or } (10{,}000{,}000\ \mu g/d)(3 \times 10^{-4})(0.5)/1\ m^3 = 1{,}500\ \mu g/m^3$$

The DU inhaled was: *(airborne concentration)(breathing rate)(time)*

$$(1{,}500\ \mu g/m^3)(1.2\ m^3/hr)(0.0833\ hr) = 150\ \mu g/workday.$$

The inhalation per workday is equal to 100 µg/calendar day or 38 pCi/calendar day. This is assumed to be the median value of a lognormal distribution with a GSD of 3.

This intake would apply to the period from 1949 through 1975. Exposure to the plume from burning was considered for burning yard workers using the same approach applied to workers outside the burning yard, except 100 m was assumed for the source-to-worker distance. Section 5.2.2 describes the calculation of intakes from the plume. This source of intake was calculated to be < 1 µg/work day, and was considered negligible compared to the intake from the cleanup of the ash, which would apply to the same workers.

5.2.4 DU Intakes from Disassembly of Weapons

Disassembly of nuclear weapons might be another source of intake of DU. Evidence gathered at Pantex indicated that the DU material in the disassembled weapons was generally clean metal with minimal potential of airborne contamination. Mr. Shannan indicated that contaminated internal parts were rare, and, when necessary, decontamination was performed before work on the weapons was started (Fix and Bihl 2003). However, there is a possibility that disassembly activities were different in the early years at IAAP prior to when Mr. Shannan was employed and routine low level DU contamination was more common. Although the primary mission of IAAP was assembly of new weapons, there was some early disassembly conducted in the 1950s and surveillance activities through the 1960s and early 1970s.

Experience at Pantex indicated that "about a half of a cup" (118 cm^3) of oxidized DU was available for resuspension. The density of UO_2 is 11 g/cm^3, thus the mass of UO_2 is estimated to be approximately 1300 g, of which about 1140 g is DU [(238/270)(1,300g) = 1,140g.]

The airborne release fraction and respirable fraction were obtained from the DOE handbook (1994) using the scenario described as the free fall spill of UO_2 powder from a height of 1 meter. The median value for the airborne release fraction was 0.00008, and the respirable fraction was 0.5. The volume of air into which the contamination was suspended was assumed to be 27 m^3, and the exposure time was 1 hour/day. These latter two assumptions assume quick work and no dilution by ventilation. Conversely, it could have been assumed that the work pace was slower but the concentration was decreased by ventilation, which would have produced about the same result. The estimated intake of DU per disassembly is then:

$$(1{,}140g)(0.00008)(.5)(1.2\ m^3/hr)(1\ hr)(10^6\ \mu g/g)/27m^3 = 2027\ \mu g \text{ or } 815\ pCi$$

Based on conversations and interviews with former employees, it was determined that surveillance, testing, and disassembly occurred more frequently than initially credited, thus providing the potential for exposures to DU. Although it is unlikely that the amount of oxidized DU estimated above was present in all cases, the claimant-favorable assumption was made that the potential for intake of this quantity did exist throughout the covered time period at IAAP (1949 – 1974). For the purpose of dose reconstruction, it was assumed that on average 100 surveillance/disassembly operations might be conducted by a single worker per year with the potential for internal exposures to the quantity of DU estimated above.

The intake should be modeled as a chronic annual intake of 81500 pCi of DU per year, as a constant upper bound.

It is important to note that not all weapons programs exhibited oxidized DU components. There is evidence from sampling records that some systems used a stainless form of uranium in some components called Mulberry. As a result, the oxidation was greatly reduced such that the above calculation would overestimate any potential uptake. Since NIOSH cannot publicly identify the programs that had significant DU oxidation, the claimant favorable assumptions are applied to all disassembly workers over time.

5.3 RADON

Weapons assembly/disassembly was conducted in bays and special cells called Gravel Gerties that were at ground level but had an overlay of earth on the roof and part-way up the sides. Three hundred forty-two radon measurements were taken in various buildings at IAAP by the Army from December 1989 through January 1991 (not including a few outlier values that had been scratched off the dataset) (Tec/Ops Landauer, Inc. 1991). To date, the authors have gained access to the results of the measurements, but not the link between results and specific buildings; hence, the information is of marginal value. Nevertheless, the average, standard deviation and geometric mean of the data are less than the corresponding values from the Pantex data discussed below. Without additional information, it was deemed claimant-favorable to use the Pantex data.

A DOE complex-wide survey of radon levels was performed in 1990 (UNC Geotech 1990). Most of the Pantex measurements were made over a 2-month period during the winter, normally expected to be the time with the highest radon concentrations because buildings are closed and heated most of the time. There were 137 locations sampled at Pantex including in bays and Gravel Gerties of similar design to those at IAAP. The data for the Pantex Plant were listed in their entirety in Table 5-11 in the Pantex Internal Dosimetry TBD (ORAU 2004d) and are summarized in Table 5.6. Pantex radon measurements and dose calculations were assumed to be the best indicators of radon exposure at IAAP, and were used as discussed below.

Table 5.6. Summary statistics of 1990 radon measurements at Pantex.

Parameter	All buildings	Underground buildings	Above-ground buildings
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

As listed in Table 5.6, the geometric mean (median) for all buildings at Pantex was 1.4 pCi/L with a GSD of 1.7. Values ranged from 0.8 to 8.1 pCi/L. Underground buildings had a slightly higher median concentration than above-ground buildings. Gravel Gerties and bays were considered “underground,” albeit not below “grade.”

The Pantex measured radon concentrations were converted to equilibrium equivalent concentrations by multiplying the radon concentration by the equilibrium factor *F* using an assumed *F* 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 per working level (WL) to arrive at a potential alpha energy concentration (PAEC). These operations were combined to create

$$PAEC = CxF/100\ pCi/L/WL$$

where C is the radon concentration in pCi/L and PAEC is in working levels. The PAEC is multiplied by the months per year the worker is exposed to determine the exposure in working level months (WLM) for input into IREP.

Because knowledge of whether a worker spent most of his/her time in a facility with an earthen cover will probably not be obtainable, dose reconstructors should use the Pantex median value for underground buildings, 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

$$(1.5\ pCi/L)(0.4)(12\ M)/100\ [pCi/L]/WL = 0.072\ WLM\ per\ year$$

Radon exposure is only assigned when lung is selected as the cancer model in IREP. The exposure distribution is lognormal. Parameter 1 is the median value in working level months. Parameter 2 is the GSD. Use a GSD of 3 to allow for uncertainties in the application of the 1990 radon measurements to the full-time period 1948-1975 and possible differences between Pantex and IAAP.

5.4 OTHER SOURCES

Enriched uranium (EU), plutonium, thorium, and perhaps Po^{210} were present at various times during assembly or disassembly of nuclear weapons. All of these sources were encapsulated (sealed), and with the careful control of contamination before release of components to production was allowed, it is unlikely these radioelements would have been available for intake.

6.0 ESTIMATION OF EXTERNAL RADIATION EXPOSURE

6.1 INTRODUCTION

Archived external dosimetry information for IAAP workers has been collected by University of Iowa College of Public Health researchers under funding provided by DOE to support medical screening of former IAAP workers.[1] Additional records were obtained from the Pantex site who currently is the custodian for most of historical IAAP records. This information has been examined as part of the effort to develop this technical basis document (TBD). Workers involved in nuclear weapon assembly activities for the AEC from about 1949 through 1975 at the IAAP were associated with a facility known as Line 1 or Division B. The primary work activity involving external radiation exposure involved testing nuclear components using DU, handling sealed nuclear components called pits containing enriched uranium or plutonium (Brinck and Jacobson 1977), and industrial radiography operations.

6.2 WORKPLACE EXTERNAL RADIATION FIELDS

The nuclides in the sealed nuclear weapon component pits emit beta, X- and gamma rays, and neutron radiation. However, radiation exposure to the workers depended significantly on processes used in the preparation, design and construction of the respective weapons. The main contribution to external radiation at IAAP was due to processes involving the handling or working around the nuclear components (pits) which contained plutonium and/or highly enriched uranium (HEU). In addition, radiography of explosive components at various stages of assembly also results in some potential for external exposure. Some early nuclear weapons may also have contained ^{210}PoBe initiators with the capsule or what we would now call a pit, which is a solid, hermetically sealed object (DOE 1997).

6.2.1 Photon Radiation

Photon (x-ray and gamma) radiation was associated with several IAAP work activities. Sources of external ionizing radiation at IAAP are dominated by fissile materials used in pits; however, there is also some low level exposure to low-activity radioactive sources, such as those used to check or calibrate radiation detectors, as well as analytical devices employing X-rays produced by a radiation generating device (RGD). These sources could have included alpha, beta, photon, and neutron emitters and were of the types and source strengths typically used by mainstream industrial or process-related users. Although some inventory records exist, a complete historical inventory of small sources has not been found in the archival material reviewed. Doses associated with the proper and widespread use of small check sources is generally negligible compared to the fissile materials. In addition to the small sources, there were at least two larger ^{60}Co sources with original activities of 5 and 50 curies, respectively (Shaykin 1969). These larger ^{60}Co sources, as well as the RGDs,

[1] In 1993, Congress passed Public Law 102-484. Section 3162 of this law required DOE to screen for occupational health conditions among former employees who might be at risk.

had the potential for producing significant exposure to workers if not used properly. Gamma radiation of 2.2 MeV resulted from ^{1}H (neutron, gamma) ^{2}H interactions caused by neutron radiation scattering (i.e., moderation) and absorption in the hydrogen-rich materials in the nuclear components and building materials (concrete) (Shleien, Slayback, and Birky 1998).

Weapons assembly at IAAP was performed with sealed nuclear components of purified radioactive metals. The purification process separates natural progeny radionuclides from their parent radionuclide. This process provides some insight into potential sources of radiation. Plutonium is purged of progeny radionuclides when it is purified. However, ^{241}Am starts growing in as its parent radionuclide ^{241}Pu decays with a half-life of 14.4 years. The ^{241}Am reaches a maximum activity after about 80 years, but it reaches about 85% of this maximum in 40 years. Figure 6.1 depicts the MCNP spectra from a generic pit of weapons grade plutonium aged 15 years (Traub et al 2005)). The details and specifications for the generic pit can be found in Appendix D of this report.

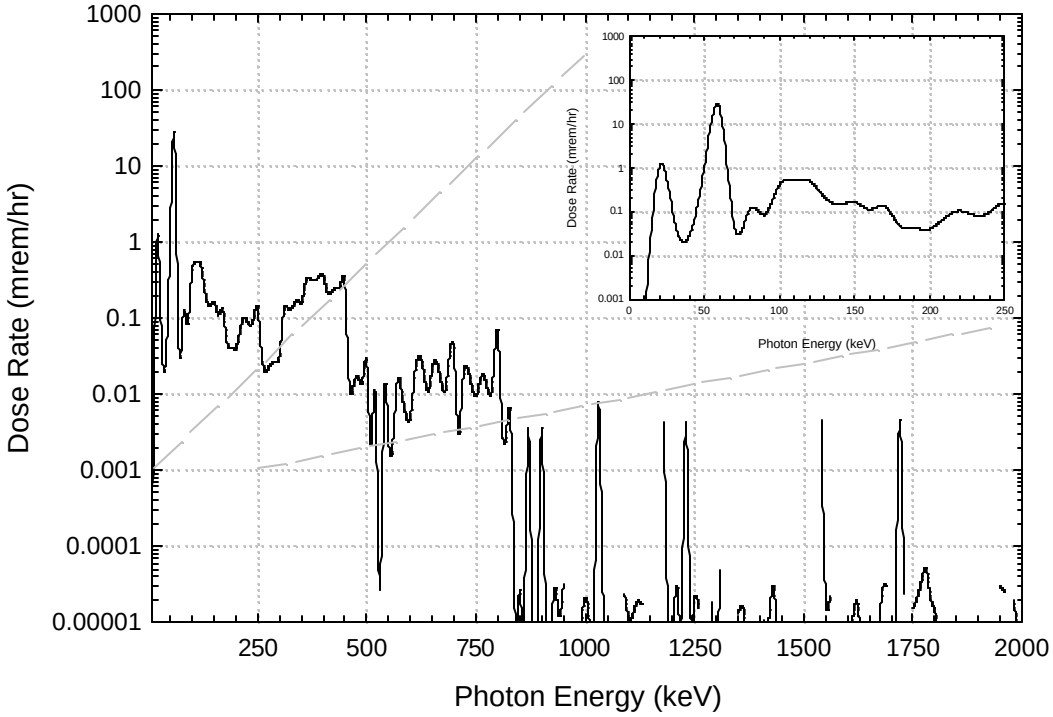


Figure 6.1 MCNP calculated spectra from generic pit
(Bare unshielded mass of weapons grade plutonium)

The generic pit spectrum clearly indicates that the external dose contribution is predominately from 30-250 keV photons, with approximately 70% of the total dose rate coming from the 60 keV photopeak from Am-241. This high magnitude is primarily due to the claimant favorable assumption of 15 year aged weapons grade plutonium. During assembly operations, the plutonium is not expected to be more than 3-4 years old; however, considering disassembly operations, an exposure to 15-year-old plutonium is credible. The average age of the weapon's components at IAAP was less than 10 years. Considering that the plutonium was a few years old during assembly, an average of 15 years at disassembly is reasonable and claimant favorable.

Although the age of the plutonium greatly affects the magnitude of the low energy photon dose, this component is also dependent upon the shielding or cladding materials surrounding the fissile materials. The type of cladding material is particularly significant for lower energy photons (<70 keV). High-Z materials such as uranium greatly reduce the photon dose rate and almost completely shield the worker from low energy photon emissions. Conversely, low-Z materials such as beryllium, do not provide much shielding and allow a relatively large quantity of lower energy photons to pass. Through discussions with DOE, NIOSH has learned that exact information on the cladding material and thickness for each weapon design assembled and disassembled at IAAP remains classified.

The initial IAAP site profile (ORAU, 2004a) assumed a high Z material of sufficient thickness such that the low energy photon dose was negligible. Through further research, NIOSH has learned that while this may be correct for some components, it is not globally accurate, and in some instances could result in a claimant unfavorable dose estimate. As a result of this finding, NIOSH has significantly revised this site profile to include a low energy photon dose for all workers handling pits.

It is important to note that not all components had a significant low energy photon dose. There are three basic types of pits used in assembly and disassembly at IAAP: 1) enriched uranium pits; 2) plutonium pits; and 3) composite pits (combination plutonium and enriched uranium). In the composite pits, the plutonium always had an outer shell of enriched uranium. Since the low energy photon dose from enriched uranium is negligible, only the plutonium pits had the potential for significant low energy photon dose.

However, due to security considerations, NIOSH cannot publicly identify which programs contained plutonium only pits versus those for which low energy photon dose was negligible. As a result, a claimant favorable approach is adopted in which NIOSH assumes, for low energy photons dose calculations only, that plutonium pits were exclusively handled and that none of the pits were clad. The low energy photon dose is, therefore, calculated based on the ratio between the low energy photons and higher energy photons.

This assumption maximizes the possible low energy photon dose component. If actual data were used in the dose reconstruction, the low energy photon dose would be reduced. As a result, does reconstructions using this methodology report doses that will exceed the actual dose that the workers received. While these assumptions will overestimate the external dose, they are considered reasonable and necessary in order to meet the intent of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), and still preserve and protect national security interests. (See Appendix E for additional discussion about pit cladding).

Using the generic pit described in appendix D, the photon dose breakdown for each IREP photon energy interval is provided in Table 6.1.

Table 6.1 Energy Distribution from Generic Pit

Photon Energy Interval	Dose Rate (mrem/hr)	Percentage of Dose
< 30 keV	1.18	4%
30-250 keV	27.76	83%
> 250 keV	4.34	13%
Total	33.28 mrem/hr	100%

6.2.2 Neutron Radiation

Uranium (^{234}U, ^{235}U, ^{236}U, and ^{238}U) and plutonium (primarily ^{239}Pu, but also ^{240}Pu and ^{242}Pu) are alpha-emitting nuclides with the expectation of (alpha, neutron) interactions with light elements in addition to spontaneous fission. Only a limited number of dosimeter badges (27 of 215 per exchange cycle) at IAAP contained NTA film for neutron monitoring (Davis, 1969). The significance of neutron radiation exposure to IAAP nuclear weapon component assembly workers cannot be directly assessed due to limitations in the monitoring technology. Although Mr. Shannan considered the neutron dose to IAAP workers from pits to be very low (Fix and Bihl 2003), a review of the area badge (building) data indicates that a few areas had some significant neutron exposures. In these areas when the neutron dose was detectable, the photon doses were also fairly high, thus indicating a neutron to photon ratio is feasible and appropriate. A neutron to photon ratio was calculated using paired non-zero area monitoring data (Figure 6.2). To ensure that the ratio was not dependent on the magnitude of the photon exposure, a plot was developed of the measured neutron—photon ratio as a function of the measured photon exposure (Figure 6.2).

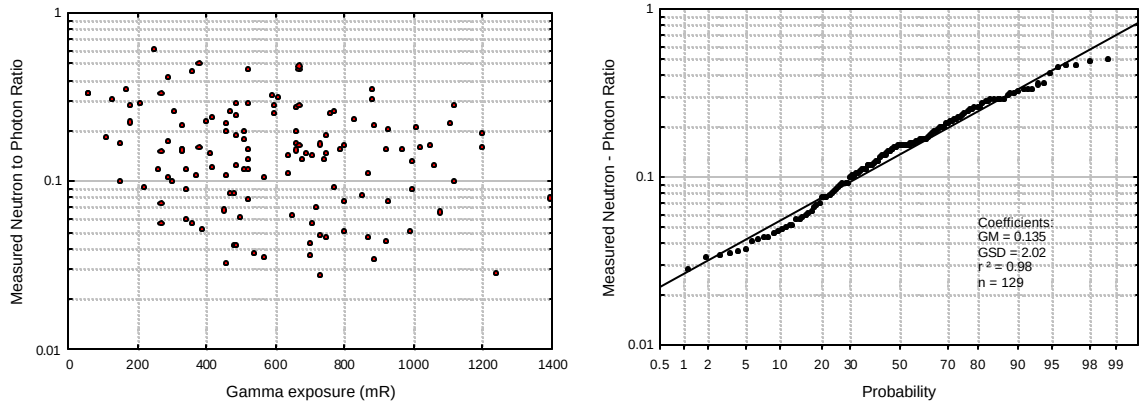


Figure 6.2 Measured neutron to photon ratio in IAAP area badges and a function of gamma exposure and the resulting fitted lognormal distribution.

It is important to note that the measured neutron to photon ratio in Figure 6.2 is an underestimate of the true neutron to photon ratio. This underestimate results due to the inability of the NTA film to accurately measure neutrons below about 800 keV. This underestimate can be corrected using MCNP to calculate the portion of the neutron dose that would not have been detected by the NTA film. Although the neutron to photon ratio varies significantly depending on pit design, MCNP calculations indicate that no more than 40% of the typical neutron spectra was below 800 keV. As a result, the measured ratio was adjusted in a claimant favorable nature (doubled) to correct this technological shortfall.

Thus for IAAP, a claimant favorable neutron to photon ratio of 0.27 with a GSD of 2.0 was calculated.

Although an evaluation of the NTA film was conducted and indications are that the neutron to photon ratio was typically less than 30%, The recommended approach to estimate potential neutron dose for IAAP monitored workers is to utilize the distribution of neutron-to-photon dose ratio from measured Pantex dosimeters during the period of 1993 to 2003. This data resulted in a higher neutron to photon ratio and is thus more claimant favorable. This higher ratio is mostly due to the practice of wearing lead aprons at Pantex. This practice results in a lower photon dose than would be experienced by Iowa workers; however, the neutron dose is not greatly affected by the lead apron. Since the ratio is developed by dividing the neutron dose by the photon dose, a low biased photon dose would result in a higher neutron to photon dose ratio and is therefore more claimant favorable. Although it is usually better to use site specific information over surrogate facility data, the uncertainty in neutron monitoring and calculated correction factors justify using the more accurate measurements from surrogate source term data.

It should also be noted, however, that a detectable neutron dose was not always found with high photon doses. The presence and magnitude of the neutron dose associated with nuclear weapons components greatly depends on the weapon design. As a result, estimating the neutron to photon ratio based solely on corrected paired (neutron-photon) measurements will overestimate the total neutron dose for the work force when applied to all photon doses.

6.2.3 Depleted Uranium

Another component of photon dose comes from depleted uranium. The important progeny nuclide for this external exposure is the decay of ^{238}U decay to ^{234m}Pa with a half-life of 24 days. Thus in a matter of a few months, DU components have ^{234m}Pa activities nearly equal to that of ^{238}U. The radionuclide ^{234m}Pa emits beta radiation 98.6% of the time when it transitions to its ground state with a maximum energy of 2.28 MeV and an average energy of 0.825 MeV (Shleien, Slayback and Birky 1998). While this beta particle does not result in a significant deep dose to most organs, the bremsstrahlung radiation produced can be significant for some workers who dd not handle pits, but worked with depleted uranium. MCNP was used to model spectral characteristics of bremsstrahlung photons from 1-cm and 30-cm diameter ^{238}U spheres. The results were similar for both spheres. Figure 6.3 shows the MCNP calculated photon spectrum emitted from ^{238}U as excited by the ^{234m}Pa beta spectrum shown on a logarithmic vertical axis. Note the smooth bremsstrahlung spectrum and the uranium characteristic K X-rays at 90-109 keV and the L X-rays in the range of 13-19 keV. As can be observed from the figure, the dose contribution from low energy photons (<30 keV) is relatively insignificant.

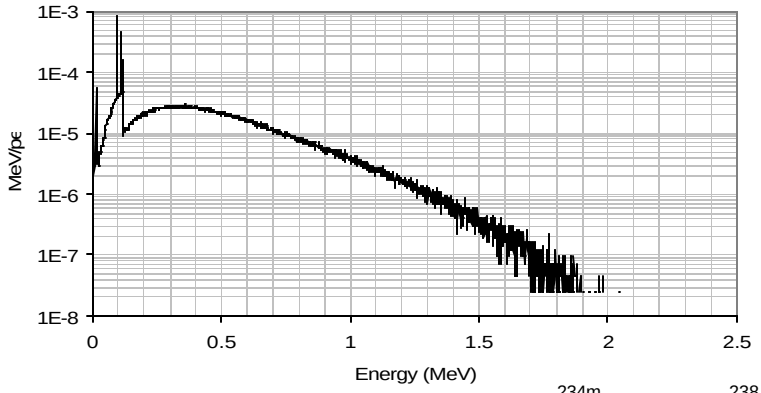


Figure 6.3 MCNP calculated photon spectra emitted from ^{234m}Pa beta in ^{238}U spheres.

As noted in section 5, workers handled DU during disassembly of bomb components containing DU, during and following hydrotesting, and during machining. DU fragments were collected by workers from the test area after each non-nuclear detonation (Archive 010000914 "BAECP Former Worker Program Needs Assessment"). The DU could contribute a significant extremity and skin dose to workers unless precautions were taken to protect workers from the beta radiation. A bare slab source of DU contributes an $H_p(0.07)$ dose of approximately 230 mrad/h compared to an $H_p(10)$ dose of approximately 2 mrad/h (ORAU 2003b).

6.3 EXTERNAL RADIATION DOSE RECORDS

A review of IAAP dosimetry data was conducted and only intermittent dosimetry data has been located between 1955 through 1962. The majority of the dosimetry data is from R. S. Landauer Company (1955-1974), and only a few results from TracerLab have been located to date. Table 6.2 indicates the time periods where dosimetry records have been located. As indicated in the table, starting in 1963 the full 52 weeks/year of monitoring data is available. Since prior to 1963, only partial year data is available; the dose reconstructor should not assume the absence of dosimetry data indicates an unmonitored worker. During these time periods, the dose should be assigned either based on an individual worker's available data multiplied by the appropriate number of weeks exposed or based on the methodology described in section 6.5.

Based on these records and the number of dosimeters issued, the monitoring practices at IAAP can be subdivided into four monitoring eras. These eras are described in Table 6.3.

Additional dosimetry records indicate that the IAAP Rad-Safe department routinely assigned and evaluated pocket ionization chamber (PIC) measured doses from at least June 18, 1965, through November 7, 1974. The content of these records is consistent with information from Shannan that monitoring was done sporadically depending on need (Fix and Bihl 2003). Most of the pocket dosimeter results appear to be related to using high level radioactive source for radiography.

Table 6.2 Summary of Dosimetry Data Available

Year	Exchange Frequency	# Weeks of Data	First Badge Date	End Date for last cycle	Minimum Reported Dose
1955	Weekly	7	November 14, 1955	January 1, 1956	20
1956	Weekly	6	January 2, 1956	February 13, 1956	20
1957	Weekly	13	May 6, 1957	August 4, 1957	10[a]
1958	Weekly	14	July 28, 1958	November 2, 1958	5[a]
1959	Weekly	14	June 29, 1958	October 4 1958	5[a]
1960	Weekly	15	March 7, 1960	June 19, 1960	5[a]
1961					
1962	Weekly	12	September 3, 1962	December 3, 1962	10
1963	Bi-Weekly	52	January 7, 1963	January 5, 1964	10
1964	4 Week	52	January 6, 1964	January 3, 1965	10
1965	4 Week	52	January 4, 1965	January 2, 1966	10
1966	4 Week	52	January 3, 1966	January 1, 1967	10
1967	4 Week	52	January 2, 1967	December 31, 1967	10
1968	4 Week	52	January 1, 1968	December 29, 1968	10
1969	4 Week	52	December 30, 1968	December 28, 1969	10
1970	4 Week	52	December 29, 1969	December 27, 1970	10
1971	4 Week	56	December 28, 1970	January 23, 1972	10
1972	4 Week	52	January 24, 1972	January 21, 1973	10
1973	4 Week	52	January 22, 1973	January 20, 1974	10
1974	4 Week	52	January 21, 1974	January 19, 1975	10

[a]Given the dosimeter designs of this time period, it is highly unlikely that the film dosimeter could measure this level. For this early time period, a detection level of 20 mR is more credible.

Table 6.3 Personnel Dosimeter Monitoring Eras

Era	Time Period	Description	Dose Reconstruction Method
1	1949-1954	No personnel monitoring conducted.	Exposure Matrix (Section 6.5)
2	1955-1962	Limited personnel monitoring data available; however, time periods within the year are intermittent.	Exposure Matrix (Section 6.5)
3	1963-1967	Routine personnel monitoring and continuous monitoring records are available.	Annual Dose Distributions or personal monitoring data, whichever is greatest.
4	1968-1974	Extensive personnel monitoring, most radiological workers should have some dosimetry records.	Annual Dose Distributions or personal monitoring data, whichever is greatest.

6.3.1 IAAP Annual Dose Distributions

The IAAP annual dose distributions were developed by evaluating individual dosimeter results from 1955 through 1974. This analysis also considered missed dose that was not reported on the dosimetry reports. Typically there are two scenarios that result in a zero dosimeter reading. The first considers that a worker was exposed to low level radiation that resulted in a dose that was below the limit of detection or an administrative reporting limit. The second is that a worker wore a dosimeter badge but was not exposed to radiation.

During worker outreach meetings in July 2004, a new issue with missed dose was identified. The issue concerned the radiological monitoring practices at the site. Through discussions with former IAAP workers who conducted both assembly and disassembly, NIOSH discovered that film badge dosimeters may not have been worn all of the time. One worker indicated that he always wore his film badge, while another indicated that he would only wear it when one was given to him. Through this discussion, it became apparent that in general workers were supposed to wear their film badges, but strict adherence was not necessarily enforced. As a result, a third scenario occurred in which a worker was issued a dosimeter badge but did not wear it during exposure to radiation. When this badge would be processed, this could also result in a zero reading. The effect of these three scenarios is that there is likely some missed or unrecorded dose (Figure 6.4). As shown in Figure 6.4, in 1966 approximately half of the dosimeter readings were below the detectable or reporting level.

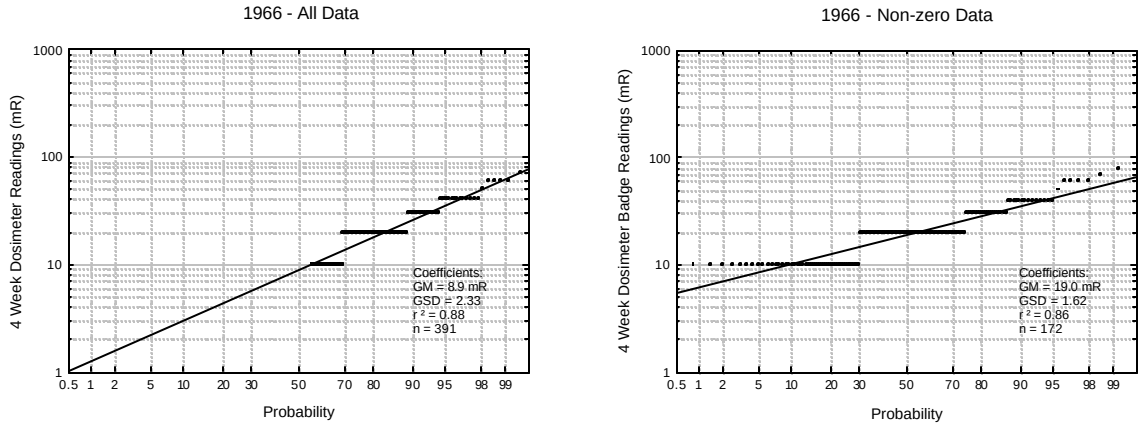


Figure 6.4 Comparison of analysis using all dosimeter data versus including only non-zero data.

In Rev 0 of this site profile (ORAU, 2004a), only the first two scenarios were considered in the development of Table 7. The same data were analyzed using standard regression methodology to account for readings that were below the limit of detection or reporting criteria. The effect of the third scenario is that too many zero readings are considered in the development of the initial annual dose distributions. To assess the true dose, those badges that were issued but not worn must be removed from the dataset. When these zero measurements are removed, the regression line shifts to the left, thus the geometric mean of the distribution is effectively increased. NIOSH considered estimating a percentage of the zero readings that resulted from workers not wearing their badges; however, any estimate of the percentage would be somewhat arbitrary and subjective. Since it is not possible to either remove the erroneous zero data or to estimate the percentage of zero readings at this time, NIOSH reanalyzed the entire dataset discarding all zero readings. This analysis assumes that all of the non-zero dosimeter readings are a representative sample of all the

dosimeters in a given year. This type of analysis introduces significant bias in that it is known that some radiation exposures are below a limit of detection and that in some instance workers would have worn their dosimeters during non-radiological activities. The true minimum dose is not 10 mR per cycle, but zero for workers who were not exposed during the monitoring period.

In summary, the annual dose distributions provided in Table 6.4 were developed by fitting a lognormal distribution to all of the non-zero dosimetry data for a given year. This analysis overestimates the true dose; however, given the data limitations, is considered necessary in order to give the benefit of the uncertainty to the claimant. The geometric mean and geometric standard deviations listed in Table 6.4 were calculated based on the fitted distributions depicted in the figures of Appendix F. A summary plot of the annual dosimetry data is depicted in Figure 6.5.

Table 6.4 Summary of Annual Dose Distributions

Year	Total # of Dosimeter Readings	# Dosimeter Readings ≥ 10 mR used in the analysis	Geometric Mean (mR)	Geometric Standard Deviation (GSD)	Upper 95th Percentile (mR)
1955	64	7	1664	1.26	2434
1956	48	10	1347	1.30	2074
1957	104	33	910	1.36	1509
1958	275	99	988	1.85	2718
1959	182	59	520	1.54	1058
1960	311	92	582	1.52	1160
1961					
1962	450	33	619	1.32	975
1963	783	295	468	1.67	1088
1964	693	156	176	1.44	320
1965	334	133	226	1.70	541
1966	391	172	247	1.62	546
1967	444	140	261	1.78	675
1968	900	222	359	2.03	1150
1969	2107	292	296	1.88	837
1970	2682	848	514	2.93	3009
1971	3890	1527	586	2.75	3094
1972	4236	1086	400	2.47	1772
1973	3108	627	553	2.36	2268
1974	1490	342	527	2.40	2222
Total	22492	6173			

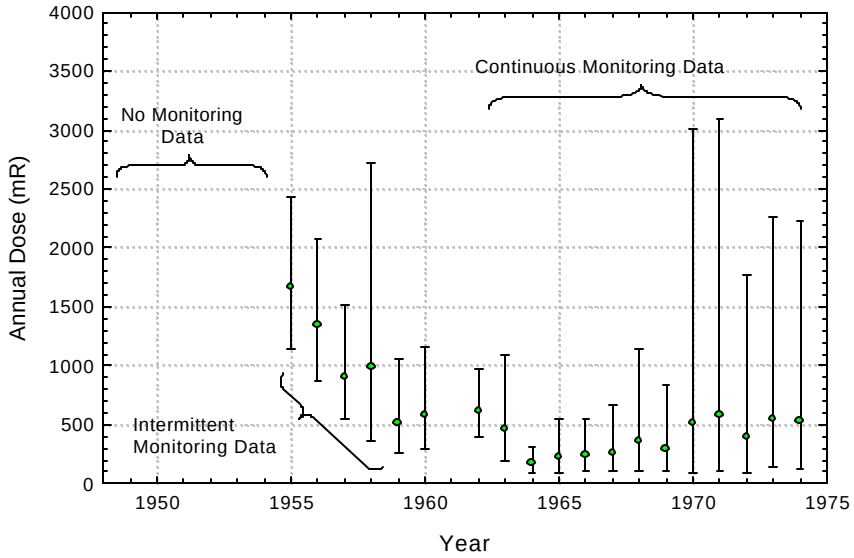


Figure 6.5 Annual dose distributions with the associated uncertainty. The error bars are not one standard deviation, but represent the range from the 5th percentile to the 95th percentile of the log-normal distribution.

6.4 DOSIMETER TECHNOLOGY AND PERFORMANCE STUDIES

6.4.1 Photon Radiation

Specific designs of the Landauer film dosimeters used at IAAP have not been located, and no records of the mentioned earlier use of TracerLab dosimeters have been found. However, from the content of IAAP-submitted AEC termination reports and personal testimony (Fix and Bihl 2003), it is likely that the film dosimeter was, at least, a two-region design (i.e. non-penetrating dose calculated from film response to open window or generally unfiltered region of the film and penetrating dose calculated from film response under a selected, usually metallic, filter). Table 6.2 and 6.3 have summarized the monitoring technique and exchange frequency for the IAAP dosimeters contracted from a commercial service and the minimum reported dose. An important addition to the exchange frequencies noted in Table 6.2 is that some of the more highly exposed workers continued on a bi-weekly exchange frequency through 1974. As a result, the dose reconstructors should consider this additional frequency when calculating missed dose. A quick review of the individual's dosimetry records identify the higher exchange frequency.

The AEC conducted performance testing of several commercial and in-house film dosimeter services during 1954 with exposures provided by the National Bureau of Standards (AEC 1955). Specific dosimeter design specifications are included in the documentation. The testing included 40-, 70-, and 210-keV narrow spectral beam X-ray techniques, ^{60}Co gamma radiation, and selected mixtures of these beams. Measured response data are provided in the report for each of the respective dosimeter open-window and filtered regions of the film. This information exhibits the significant overresponse of the open-window and lightly filtered regions of the film at lower (i.e. 40 and 70 keV) photon energies and an underresponse of the heavily filtered portions of the film dosimeters to photon energies less than 70 keV.

There is considerable uncertainty as to whether the IAAP dosimeter badge could reliably measure a penetrating dose from the 60 keV photons from Am-241. As illustrated in figure 6.1, the Am-241 60 keV photopeak contributes approximately 70% of the total photon dose. According to the R. S. Landauer Company, the IAAP film badge had a lower energy response threshold of approximately 30 keV (Landauer, 1965). However, dosimetry measurements conducted at Hanford (Larson and Roesch, 1955) using a two element film badge similar to the early IAAP dosimeter, photons less than 70 keV were severely attenuated. Based on the Hanford report, at 60 keV, only 37% of the penetrating dose would be measured. At IAAP, several film badges were exposed to low energy photons at the medical facility. In an X-ray exposure at 70 kVp read less than half a similar exposure to 95 kVp X-rays although the intensity of the beam was only decreased by 67%. Furthermore, the exposure at 45 kVp was not detectable. Since the level of filtration greatly affects the average photon energy, exact energy threshold information cannot be estimated. If the beam was relatively heavily filtered, the kVp would correlate to near the peak energy; however, with little filtration, the 95 kVp would result in an average photon energy of approximately 32 keV.

Based on the Hanford information and professional judgment, a claimant favorable fraction of 30% of the Am-241 photopeak is assumed to have been measured by the IAAP film badge. This value is likely to be slightly lower than reality, and since multiple filters were used throughout the DOE complex from the early 1960s, this is likely a considerable overestimate for the later years (1970s).

6.4.1.1 Photon Radiation Performance Studies

Within the IAAP dosimetry records, NIOSH has found that the IAAP Rad-Safe group intermittently exposed spare badges to known quantities of radiation and submitted them with other dosimeters as a quality assurance check. NIOSH's review of this information indicates a slight overall bias (-7%) and a standard deviation of ±20%, with an increase to -60% and +40% observed for two of the three low level exposures at 50mR. This greater uncertainty is common with film dosimeter as the exposure approaches the limit of detection. This effect is discussed in NIOSH's External Dose Reconstruction Implementation Guideline (OCAS-IG-001, 2002) which provides guidance on how to incorporate this uncertainty in dose reconstructions. Figure 6.6 illustrates the reasonable agreement between the irradiations and the reported readings.

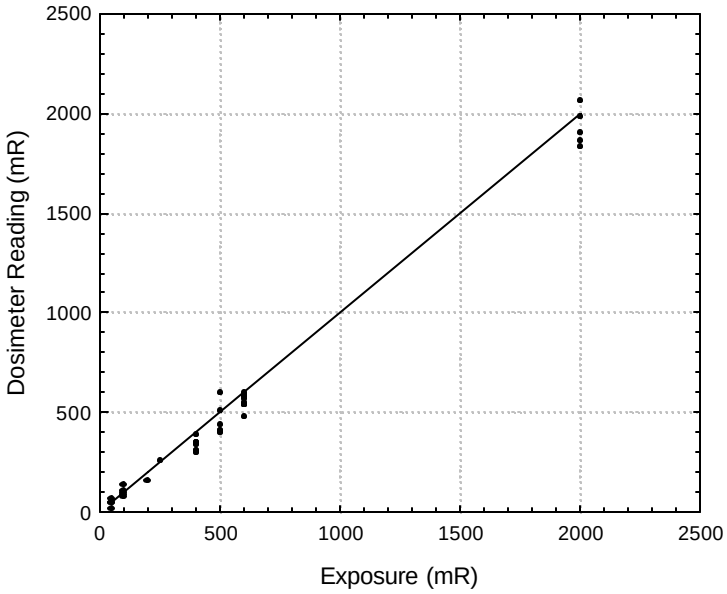


Figure 6.6 Comparison of film dosimeter badges exposed to a known quantity of radiation and the reported readings by the commercial vendor.

6.4.2 Neutron Radiation

Some neutron doses were reported by Tracerlab and Landauer on dose reports beginning about 1962. Eastman Kodak nuclear emulsion type A (NTA) film was used for these measurements. NTA was basically the only common dosimeter method available to measure neutron dose in AEC facilities at that time. Results reported at the first AEC Neutron Dosimetry Workshop in 1969 indicated that SRS calibration laboratory dose measurements made with NTA film were about one-half to one-fourth of those measured with other methods, including the neutron thermoluminescent dosimeter (TLD)(Vallario, Hankins, and Unruh 1969). This underestimation is due to the NTA film's inability to accurately measure lower energy neutrons (<800 keV). As a result, IAAP's neutron dosimeters underestimated the total neutron dose.

One of the most important parameters related to performance of NTA is the difference between calibration and workplace neutron energy spectra. There are no known measurements of neutron spectra at IAAP, and the method(s) used to calibrate the Landauer NTA film known is currently unknown. Neutron dose calculations have been conducted using MCNP for various IAAP pits in a configuration that would incorporate albedo effects of surrounding materials to increase the neutron scatter. As noted in section 6.2, these MCNP calculations indicate that no more than 40% of the neutron dose was less than 800 keV, thus indicating that IAAP's neutron readings would be of similar quality to those reported by other DOE facilities such as the Savannah River Site.

Although neutron spectra were not measured at IAAP, neutron spectra measurements have been conducted at various Pantex Plant facilities (classified), and the performance of the Pantex 809/812 thermoluminescent dosimeter used beginning in 1993 has been validated for weapon and workplace exposures. As a result, this data is considered to be of higher

quality than is calculated from MCNP. Significant neutron exposure from nuclear weapons components at IAAP was typically associated with a photon dose that would be readily and reliably measured with film dosimeters.

6.5 IAAP WORKER EXTERNAL DOSE RECONSTRUCTION

The primary objective of dose reconstruction for IAAP workers is to utilize a claimant-favorable method to retrospectively:

- Calculate the dose to unmonitored or intermittently monitored workers prior to the routine use of personnel dosimeters.
- Interpret the measured dose for monitored workers for input to IREP for probability of causation calculations.
- Calculate the unmeasured or poorly measured neutron dose to both unmonitored and monitored workers.

6.5.1 Unmonitored or Intermittently Monitored Workers

At IAAP, there are basically three categories of unmonitored or intermittently monitored workers; 1) unexposed workers who did not need to be monitored, 2) intermittently or routinely exposed workers who did not directly handle radioactive materials, and 3) workers who directly handled radioactive materials.

Unexposed workers who did not need to be monitored
This category includes office and administrative personnel who may have been occasionally exposed to radioactive materials as they walked throughout the plant. Since their main responsibilities would put them in an office setting, assigning onsite ambient dose levels based on area monitoring data located throughout the plant in production areas would constitute a reasonable upper bound. The dose estimate for these workers should be based on onsite ambient dose levels as described in section 4.2.

Intermittently or routinely exposed workers who did not directly handle radioactive materials
These workers would likely have been monitored by modern standards but may not have been due to higher regulatory dose limits or past radiological control practices. Most IAAP unmonitored workers fall into this category. This category includes workers who did not directly handle radioactive materials but may have conducted some work in proximity to sources of radiation. Their work around radioactive materials may have been intermittent or routine. This category generally includes trade workers, and engineers whose time was spent between plant operations and an office setting. The dose estimate for these workers should be based on the exposure distributions in Table 6.4 and Table 6.6.

Workers who directly handled radioactive materials
In general, since 1963, workers who directly handled radioactive materials at IAAP are expected to have been routinely monitored and dosimetry data should be available. However, some may have only been partially monitored. This category included workers

involved directly in the assembly, disassembly, inspection, or transport of radioactive materials.

6.5.1.1 Dose estimation prior to 1963

Prior to 1963, only a relatively small number of workers were monitored for exposure to external radiation and only intermittent monitoring data is available. As a result, the external dose is estimated based on a ratio methodology using dose rate information and annual doses. Since the magnitude of the radiation dose could have changed over time due to the handling of various pits, a generic or bare unshielded pit[2] is assumed in order to estimate a claimant favorable annual dose prior to 1963. Details and specifications of the generic pit can be found in Appendix D. Based on these specifications, the dose rate for the generic pit was calculated using MCNP (Traub et al, 2005). With this dose rate, the only other factor necessary to estimate annual dose is the relative work factor or number of hours in which a worker would have handled the pit or been in close proximity to the pit. To estimate the relative time spent in proximity to a pit (work factor), dosimetry data post 1962 was compared with an era dose rate. The methodology and assumptions used to develop the era dose rates remain classified as described in Traub et al (2005). Using the era dose rates of 1.08 mrem/hr and 1.48 mrem/hr for era #3 and #4 respectively, the work factor was calculated as follows:

$$W_f = \frac{D_{Annual}\, x\, C_f}{\dot{D}_{ERA}\, x\, 2000\ \text{hours}} \qquad \textit{Equation 6.1}$$

Where: W_f = *Work Factor*
D_{Annual} = *Annual Dose (Lognormal Distribution)*
D_{ERA} = *Era Dose Rate*
C_f = *Conversion factor Roentgen to rem (1.05)*

The correction factor was obtained by integrating the area under the curve from 70 keV to 2 MeV which represents a combination of the measured dose (energy threshold) and that most of the external dose photon spectra is less than 2 MeV.

[2] The terminology for pit is used as a general term to indicate the fissile materials which are the dominant contributor to the external dose.

Table 6.5 Work factor calculation using the two era dose rates.

Year	Annual Dose	Modified for Hp(10)	Era Dose Rate	Estimated Annual Dose	Work Factor	Mean Work Factor
1962	619	650	1.08	2160	0.301	
1963	468	491	1.08	2160	0.228	
1964	176	185	1.08	2160	0.086	ERA #3
1965	226	237	1.08	2160	0.110	
1966	247	259	1.08	2160	0.120	
1967	261	274	1.08	2160	0.127	0.162
1968	354	372	1.48	2960	0.126	
1969	296	311	1.48	2960	0.105	
1970	514	540	1.48	2960	0.182	ERA #4
1971	586	615	1.48	2960	0.208	
1972	400	420	1.48	2960	0.142	
1973	553	581	1.48	2960	0.196	
1974	527	553	1.48	2960	0.187	0.164
					Geometric Mean	0.153
					Geometric Standard Deviation	1.43

The average annual dose changed on an annual basis. This introduces uncertainty into the calculation. As a result, the work factor is treated as a distribution. Figure 6.7 depicts the work factor distribution, which has geometric mean of 0.153 and a geometric standard deviation of 1.43.

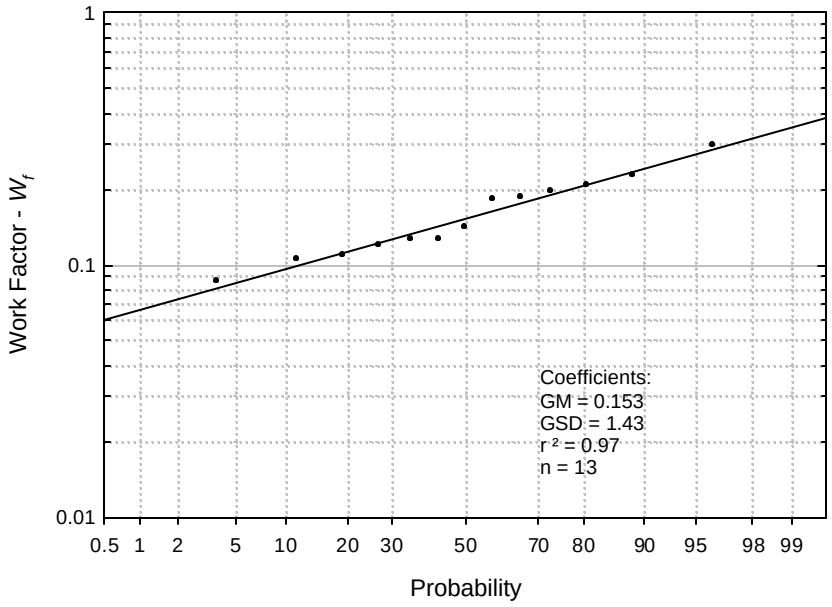


Figure 6.7 Work Factor Probability Distribution

The annual dose is then calculated by multiplying the dose rate from the generic pit by a 2000 hr. work year and the work factor (Equation 6.2). Since in the equation the generic pit is treated as a constant, and the work year is also a constant, the uncertainty is simply the geometric standard deviation of the work factor.

$$D_{Annual} = \dot{D}_{Generic} \times 2000 \text{ hours} \times W_f \times R_{Am\text{-}241} \qquad \text{Equation 6.2}$$

Where D_{Annual} = Estimated annual dose (mrem)
$D_{Generic}$ = Estimated Measured dose equivalent from the generic pit
W_f = Work Factor
$R_{Am\text{-}241}$ = Ratio of Am-241 in growth

Equation 6.2 is used to reconstruct doses prior to 1963 and the annual dose information from Table 6.4 is used to reconstruct doses from 1963 through 1974. Using these two methodologies, Table 6.6 and Figure 6.8 provides the total annual dose distribution parameters for IAAP workers from 1949 through 1974.

Table 6.6 Modeled Annual Dose Using Work Factor Methodology

ERA	Year	Generic Pit Dose Rate ($D_{Generic}$)	Work Factor (W_f)	Am^{241} Ingrowth ($R_{Am\text{-}241}$)	Modeled Annual Dose	GSD	Annual Dosimeter Data[1]	
							Median	GSD
#1	1949	14.8	0.153	0.67	3034	1.43		
	1950	14.8	0.153	0.71	3215	1.43		
	1951	14.8	0.153	0.75	3397	1.43		
	1952	14.8	0.153	0.78	3532	1.43		
	1953	14.8	0.153	0.81	3668	1.43		
	1954	14.8	0.153	0.84	3804	1.43		
#2	1955	14.8	0.153	0.87	3940	1.43	1664	1.26
	1956	14.8	0.153	0.90	4076	1.43	1347	1.30
	1957	14.8	0.153	0.93	4212	1.43	910	1.36
	1958	14.8	0.153	0.95	4302	1.43	988	1.85
	1959	14.8	0.153	0.98	4438	1.43	520	1.54
	1960	14.8	0.153	1.00	4529	1.43	582	1.52
	1961	14.8	0.153	1.00	4529	1.43		
	1962	14.8	0.153	1.00	4529	1.43	619	1.32
#3	1963						468	1.67
	1964						176	1.44
	1965						226	1.70
	1966						247	1.62
	1967						261	1.78
#4	1968						354	2.03
	1969						296	1.88
	1970						514	2.93
	1971						586	2.75
	1972						400	2.47
	1973						553	2.36
	1974						527	2.40

a. The data has been converted to mrem by multiplying the dosimeter dose by a correction factor of 1.05

Table 6.6 and Figure 6.8 indicate a continuous increase of the annual dose between 1949 and then constant for 1960 through 1962. This ramp effect is due to the estimate in growth of Am-241 from the decay of Pu-241 in weapons grade plutonium. The generic pit assumes 15 year aged plutonium, whereas in 1949, the maximum age plutonium in the United States could have been 4 years old. By 1960, however, there could have been 15 year age plutonium in the stockpile.

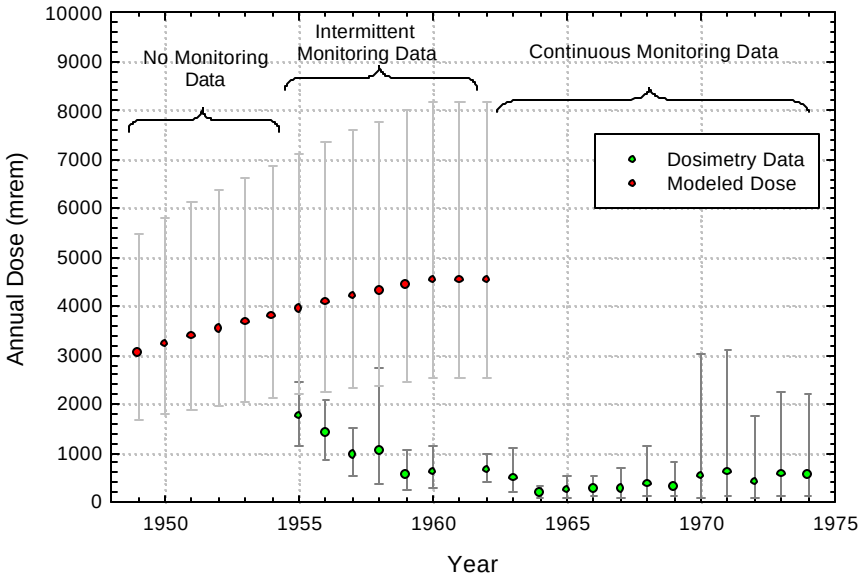


Figure 6.8 Summary of annual dose distributions using the model between 1949 -1962 and actual dosimetry data from 1963 to 1974

Table 6.7 provides the change in dose rate of the generic pit from Am-241 in-growth. It is also important to note that the main mission of IAAP and Pantex was to assemble nuclear weapons. The Clarksville and Medina facilities were primarily responsible for retrofit and disassembly during this early time period (Mitchell, 2003). As a result, the ramp effect and assumption of 15 years is considered claimant favorable.

Table 6.7 Am-241 Build-up from decay of Pu-241

Decay Year	Year	Photon Dose Rate (mrem/hr)					Simulated Measured[a] >70 keV + 0.3*Am	Ratio of Am^{241} in-growth
		< 30 keV	30-250 keV Pu Component	30-250 keV Am Component	30-250 keV Total	> 250 keV		
1		1.18	3.14	2.28	5.42	4.49	8.1	0.55
2		1.18	3.14	4.46	7.60	4.49	8.8	0.59
3		1.18	3.14	6.52	9.66	4.49	9.4	0.63
4	1949	1.18	3.14	8.49	11.63	4.49	10.0	0.67
5	1950	1.18	3.14	10.36	13.50	4.49	10.5	0.71
6	1951	1.18	3.14	12.14	15.28	4.49	11.1	0.75
7	1952	1.18	3.14	13.84	16.98	4.49	11.6	0.78
8	1953	1.18	3.14	15.44	18.58	4.49	12.1	0.81
9	1954	1.18	3.14	16.97	20.11	4.49	12.5	0.84
10	1955	1.18	3.14	18.43	21.57	4.49	13.0	0.87
11	1956	1.18	3.14	19.81	22.95	4.49	13.4	0.90
12	1957	1.18	3.14	21.12	24.26	4.49	13.8	0.93
13	1958	1.18	3.14	22.37	25.51	4.49	14.2	0.95
14	1959	1.18	3.14	23.56	26.70	4.49	14.5	0.98
15	1960	1.18	3.14	24.68	27.82	4.49	14.8	1.00

a. Simulated Measured Dose Rate is the C7 value as indicated in Appendix D. Also note values may not sum due to rounding.

In reality, between the years from 1949 to 1961, there would also be some additional variance if the actual pit data could be presented. Since the generic pit was used, the data in Table 6.6 should be considered a reasonable upper bound of what the measured dose would have been if pits were onsite, and they were assembled into the weapons.

6.5.1.2 Dose estimation between 1963-1974

Starting in late 1962, a moderate number of workers were monitored for radiation exposure at IAAP. As a result, the annual dosimeter data in Table 6.6 can be used as a reasonable surrogate of co-worker data for unmonitored or intermittently monitored workers. For partial year exposures, the dose should be prorated by dividing the annual dose by 52 weeks to obtain a weekly rate and then multiplied by the number of weeks in the partial year.

6.5.2 Monitored Workers

The IAAP reported dose for monitored workers should be adjusted for any missed photon dose in accordance with the NIOSH External Dose Reconstruction Implementation Guidelines (NIOSH 2002). This adjusted dose should then be compared against the annual dosimeter doses in Table 6.6. Since the worker may not have always worn their dosimeter, the higher of the two values should be assigned unless there is an indication in the dosimetry records which would warrant a lower dose assignment.

6.5.3 Adjustments to Photon Dose

The first correction is to adjust the measured or estimated doses by the Roentgen to rem conversion factor of 1.05. This factor was derived by integrating the area under the curve from 70 keV to 2 MeV which represents a combination of the measured dose (energy threshold) and that most of the external dose photon spectra is less than 2 MeV.

As noted earlier in this report, the film badge dosimeter used at IAAP may or may not have been capable of measuring all of the 60 keV photons from Am-241. The claimant favorable assumption is made that the dosimeter could only measure about 30% of this photon. As a result, some adjustment to the measured photon dose is necessary for IREP inputs. Using the generic pit (i.e. no cladding and minimal attenuation), dosimeter adjustment factors were developed using MCNP to translate the measured annual doses (> 70 keV) into IREP energy intervals. Table 6.8 provides the claimant favorable adjustment factors. These factors incorporate both the ratio to account for the lower (< 70 keV) photons that were not measured as well as the energy fractions from Table 6.1. The ratios were developed using the dose rate data found in Table D.1 of Appendix D and by adding 30% of the Am-241 60 keV photopeak into the measured dose (P4). To obtain the IREP input, multiply the values in Table 6.8 by the measured or estimated annual dose after conversion to Hp(10).

Table 6.8 IAAP Photon Adjustment Factors.

	Energy Interval	Adjustment Factor
Low Energy Photon Dose	< 30 keV	0.080
Intermediate Energy Photon Dose	30-250 keV	1.878
High Energy Photon Dose	> 250 keV	0.303

6.5.3.1 Bladder Cancer Example

In order to summarize the ratios and adjustments, a bladder cancer example is provided in Appendix G.

6.5.4 Estimation of Neutron Dose

The recommended approach to estimate potential neutron dose for IAAP monitored workers is to utilize the distribution of neutron-to-photon dose ratio calculated from Pantex dosimeters during the period of 1993 through 2003. These measurements were made with the performance validated Pantex 809-812 dosimetry system. These dosimeters effectively evaluate the neutron to photon doses from a given source term. Dosimetry records were analyzed for each dosimeter with a positive neutron and photon dose greater than 50 mrem for the period of 809/812 use. Analysis of this information is shown in Figure 6.9 as a lognormal probability plot of the ratio of neutron-to-photon doses. A regression analysis yields a geometric mean of 0.79 and a geometric standard deviation (GSD) of 1.57. The measured Pantex neutron to photon ratio from 1993 through 2003 is expected to be greater than the actual IAAP neutron to photon ratio due to the use of lead aprons at Pantex which reduce the total photon dose and therefore increase the overall neutron to photon ratio. As a result, assigning the distribution of modeled and measured dose from Table 6.6 to IAAP workers prior to the routine use of personnel dosimeters and using the distribution of the neutron-to-photon dose ratio based on the Pantex dosimeter measurements is claimant-favorable.

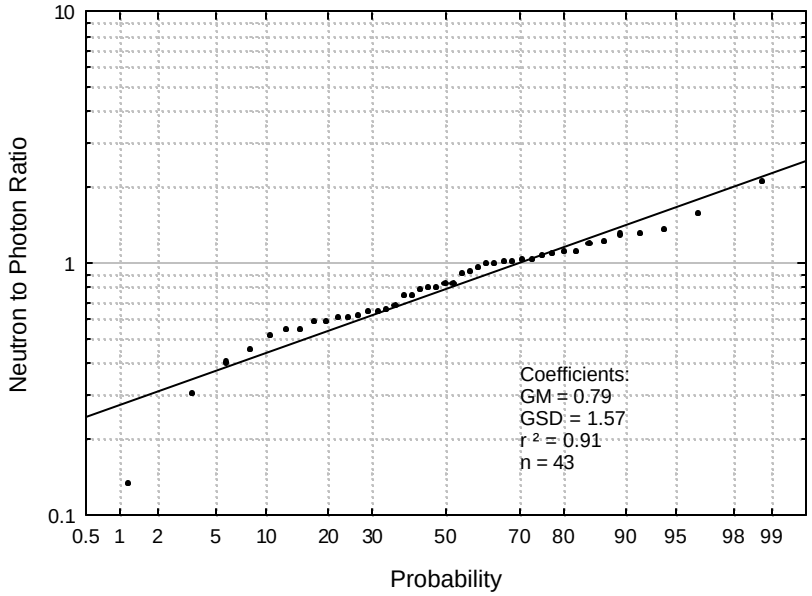


Figure 6.9 Probability plot of neutron to photon ratios calculated from Pantex 809/812 neutron and photon dosimeter data.

The respective statistical parameters for the lognormal distribution of neutron-to-photon dose ratios for application to dose reconstruction for IAAP monitored workers are presented in Table 6.9.

Table 6.9 Statistical Parameters for IAAP neutron-to-photon dose ratios

Parameter	Neutron to Photon Dose Ratio
Geometric Mean	0.79
Geometric Standard Deviation	1.57
Upper 95% percentile	1.66

The estimate of the neutron dose must be adjusted to include the conversion to the ICRP Publication 60 (1990) neutron weighting factor required for input of the dose into the Interactive RadioEpidemiological Program (IREP) by using the assumed neutron energy and dose fraction listed in Table 6.10 (ICRP 1990). Fission spectrum neutrons were selected for several reasons. As noted in Table D.1 of Appendix D, the neutron energy distribution for the generic pit was relatively evenly split between the fission and fast neutrons. The generic pit, however, does not have any cladding material surrounding it to slow the neutron spectra like a real pit would have. In order to ensure claimant favorability, 100% of fission spectra neutrons is assumed.

Table 6.10. IAAP neutron dose fractions and associated ICRP 60 correction factors.

Process	Description/ buildings	Neutron Energy (MeV)	Default dose Fraction (%)	ICRP 60 correction Factor (CF)
Nuclear weapon component assembly	Neutron exposure associated with weapon assembly and disassembly activities.	0.1 – 2 MeV	100	1.91

6.5.5 Skin and Extremity Dose (Reserved)

This section is currently reserved and will be updated in a revision to the TBD. The major difficulties with estimating the skin dose at this point evolve around beta exposures to the extremities (hands and forearms). The ORAU team is currently working on this issue as part of the skin dose Technical Information Bulletin (TIB).

Although this section is reserved, this reservation does not pre-empt dose reconstruction for likely compensable cases. The dose reconstructor should not that the deep photon and neutron dose may be sufficient for the dose reconstruction to be completed.

6.6 UNCERTAINTY IN PHOTON AND NEUTRON DOSE

For the usual analysis of measured film badge doses, the minimum detection levels (MDLs) quoted in the literature range from about 10 to 50 mrem for beta/photon irradiation; it is possible to read a photon dose of 100 mrem to within ±15 mrem if the exposure involved photons with energies between several hundred keV and several MeV (Morgan 1961). For dose reconstruction, the uncertainty in photon doses is estimated to be ±20%. This value is based on the IAAP Quality Assurance checks that were conducted periodically by IAAP's Rad-Safe Department. The uncertainty in the neutron dose is incorporated in a claimant favorable manner in the neutron to photon ratio.

6.7 ORGAN DOSE

Once the photon and neutron doses and their associated standard errors have been calculated for each year, the values are used to calculate organ doses of interest using NIOSH (2002). There are many complexities and uncertainties when applying organ dose conversion factors to adjusted doses of record. Many of the factors that affect the dose of record have been discussed in tables in this TBD. ICRU (1988) indicated that film badge dosimeters, while not tissue-equivalent, can be used for personnel dosimetry. It also indicated that it is more difficult to ensure that the variation in response with energy and angle of incidence with low energy. Considering the IAAP film badge response and the typical photon energy spectra, the measured dose can be reasonably converted to Hp(10) using a correction factor of 1.05. Since the generic pit calculations were conducted using ICRP 74 (1996) conversion factors from kerma to Hp(10), Hp(10) is the recommended organ dose conversion factors for IAAP dose reconstructions.

REFERENCES

AEC (US Atomic Energy Commission), 1955, "Intercomparison of Film Badge Interpretations," *Isotopics*, April.

AEC (US Atomic Energy Commission), 1962-1973a, *Summary of Radiation Exposures Resulting in the Internal Body Deposition of Radioactive Material For The Year,* AEC Form 191 submitted by Mason & Hanger-Silas Mason Co. Inc., Burlington, Iowa.

Ahlstrand, Franz A., (1955) "Project History of Line 1 Operations at Iowa Ordnance Plant, January 1955 through June 1955," Contract No. W-49-010-ORD-68, Silas Mason Company, Inc., Burlington, Iowa.

Ahlstrand, Franz A., (1956a) "Project History of Line 1 Operations at Iowa Ordnance Plant, January 1956 through June 1956," Contract No. W-49-010-ORD-68, Silas Mason Company, Inc., Burlington, Iowa.

Ahlstrand, Franz A., (1956b) "Project History of Line 1 Operations at Iowa Ordnance Plant, July 1956 through December 1956," Contract No. W-49-010-ORD-68, Silas Mason Company, Inc., Burlington, Iowa.

Ahlstrand, Franz A., (1957b) "Project History of Line 1 Operations at Iowa Ordnance Plant, July 1957 through December 1957," Contract No. W-49-010-ORD-68, Silas Mason Company, Inc., Burlington, Iowa.

Ahlstrand, Franz A., (1958b) "Project History of Line 1 Operations at Iowa Ordnance Plant, July 1958 through December 1958," Contract No. W-49-010-ORD-68, Silas Mason Company, Inc., Burlington, Iowa.

Ahlstrand, Franz A., (1959b) "Project History of Line 1 Operations at Iowa Ordnance Plant, July 1959 through December 1959," Contract No. W-49-010-ORD-68, Silas Mason Company, Inc., Burlington, Iowa.

ATSDR (Agency for Toxic Substances and Disease Registry), 2003, *Health Consultation – Environmental Pathway Evaluation for Beryllium and Depleted Uranium,* Iowa Army Ammunition Plant, August 15, 2003.

Brinck, W. L., and G. A. Jacobson, 1977, *Trip Report – Iowa Army Ammunition Plant – September 7, 1977,* U. S. Environmental Protection Agency.

Carbaugh, E. H., 2003 (or current version), *Methods and Models of the Hanford Internal Dosimetry Program,* PNL-MA-860, available from Radiological Records files or on the Internet at http://www.pnl.gov/eshs/cap/dss/epub.html.

Dahl, D. A., Johnson, L. J., LA-UR-77-681 "Aerosolized U and Be from LASL Dynamic Experiments," Los Alamos Scientific Laboratory, 1977.

Davis, C. E., 1969, *Health Protection Survey Report, Burlington AEC Plant,* report on annual health protection survey by the AEC Albuquerque Office of Burlington operations held September 17-19, 1969 (ORAU file locator 000923).

Decisioneering Inc., 2000, *Crystal Ball 2000 User Manual*, Denver, Colorado.

DOE (U. S. Department of Energy), 1994, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, Volume 1 – Analysis of Experimental Data,* DOE Handbook DOE-HDBK-3010-94, Washington, D.C.

DOE (U. S. Department of Energy), 2003, *Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components,* EIS-0225, Washington, D.C.

DOE (U. S. Department of Energy), 1997, *Linking Legacies,* DOE/EM-0319, Washington, D.C.

DOE (U. S. Department of Energy), 2001, *Gravel Gertie Experiment,* DOE/NV – 710.

Eckerman, K. F., R. W. Leggett, C. B. Nelson, J. S. Puskin, and A. C. B. Richardson, 1999, Cancer Risk Coefficients for Environmental Exposure to Radionuclides, Federal Guidance Report No. 13, CD Supplement, U. S. Environmental Protection Agency Report EPA 402-R-99-001 (National Technical Information Service, Springfield, Virginia).

Fix, J. J., and D. E. Bihl, 2003, "Transcription of telephone discussion on December 19, 2003 with Don Bihl and Jack Fix," letter to J. Shannan with attached notes from telephone interview.

Fuortes, L., 2001, Burlington Atomic Energy Commission Plant – Former Worker Program – Needs *Assessment – December 2001,* College of Public Health, University of Iowa, Iowa City, Iowa.

Holmberg, R. L. Letter dated February 6, 1974, to Contracting Officer, Burlington AEC Plant, "Areas Contaminated with Radioactive Material at the Burlington AEC Plant."

Holmberg, R. L. Letter dated September 20, 1974, to Contracting Officer, Burlington AEC Plant, "Decontamination of FS-12."

Groves, Leslie (1945) Memorandum concerning the Trinity test dated July 16, 1945.

ICRP (International Commission on Radiation Protection), 1968, Report of Committee IV on *Evaluation of Radiation Doses to Body Tissues from Internal Contamination due to Occupational Exposure,* Publication 10, Pergamon Press, Oxford, England

ICRP (International Commission on Radiological Protection), 1981, *Limits for Inhalation of Radon Daughters by Workers,* Publication 32, Pergamon Press, Oxford, England

ICRP (International Commission on Radiation Protection), 1982, *Protection of the Patient in Diagnostic Radiology,* Publication 34, Pergamon Press, Oxford, England

ICRP (International Commission on Radiological Protection), 1991, "1990 Recommendations of the International Commission on Radiological Protection," *Annals of the ICRP,* Publication 60, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiation Protection), 1996, Conversion Coefficients for use in Radiological Protection against External Radiation, Publication 74, Pergamon Press, Oxford, England

ICRU (International Commission on Radiation Units and Measurements), 1988, *Determination of Dose Equivalents from External Radiation Sources,* Part 2, Report 43, Bethesda, Maryland.

Larson, H.V., Roesch, W. C., (1954) Gamma Dose Measurement with Hanford Film Badges, HW-32516, Hanford Works, Richland, Washington.

Landauer (1955-1974) (R. S. Landauer, Jr. and Company), *Reports of film badge exposure for selected periods during 1955 – 1975,* Matteson, Illinois.

Landauer (1965) (R. S. Landauer, Jr. and Company), *Reports of film badge exposure for 1965,* Matteson, Illinois.

Lemert, Ann Arnold (1979), First You Take A Pick & Shovel: The Story of the Mason Companies, John Bradford Press, Lexington, Kentucky

Lincoln, T. A. and E. D. Gupton, 1958, "Radiation Doses in Diagnostic X-ray Procedures," Radiology. 71:208-215.

Loeber, Charles R. (2002), Building the Bombs: A History of the Nuclear Weapons Complex, Sandia National Laboratories, SAND2002-0307P, Albuquerque, New Mexico.

Mitchell, Kris C. (2003) Rhetoric to Reality: A Cold War Context Statement for the Pantex Plant (1951-1991) BWXT Pantex, LLC.

Meek, L., and L. S. Shahan, 1972, *Natural and Depleted Uranium (D-38) Concentration in the IAAP Area,* BAECP Technical Report 181 Part 5, Burlington Atomic Energy Commission Plant, May 11.

Morgan, K. Z., 1961, "Dosimetry Requirements for Protection from Ionizing Radiation," in *Selected Topics in Radiation Dosimetry,* International Atomic Energy Agency, Vienna, Austria, pp. 3-23.

NBS (National Bureau of Standards), 1959, *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water,* Handbook 52, U. S. Department of Commerce, Government Printing Office, Washington, D.C.

NCRP (National Council on Radiation Protection and Measurements), 1989, Screening Techniques for Determining Compliance with Environmental Standards – Releases of Radionuclides to the Atmosphere, NCRP Commentary No. 3, Bethesda, Maryland.

NCRP (National Council on Radiation Protection and Measurements), 1996, Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground, Report No. 123, Volume 1, Bethesda, Maryland.

NIOSH (National Institute for Occupational Safety and Health), 2002, *External Dose Reconstruction Implementation Guidelines,* OCAS-IG-001, Revision 1, Office of Compensation Analysis and Support, Cincinnati, Ohio.

NRC (U. S. Nuclear Regulatory Commission), 1983, *Information for Establishing Bioassay Measurements and Evaluations of Tritium Exposure,* NUREG-0938, National Technical Information Service, Springfield, Virginia.

ORAU (Oak Ridge Associated Universities), 2003a, *Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures,* ORAUT-OTIB-0006, Rev. 0, Oak Ridge, Tennessee.

ORAUT-TKBS-0001, 2003b Technical Basis Document: *Basis for Development of an Exposure Matrix for Bethlehem Steel Corporation.* Rev 00, March 31, 2003, Oak Ridge Associated Universities (ORAU), Oak Ridge, TN.

ORAU (Oak Ridge Associated Universities), 2004a, *Technical Basis Document for Atomic Energy Operations at the Iowa Army Ammunition Plant (IAAP),* ORAUT-TKBS-0018, Rev. 0, Oak Ridge, Tennessee.

ORAU (Oak Ridge Associated Universities), 2004b, *Technical Basis Document for the Rocky Flats Plant – Occupational Medical Dose,* ORAUT-TKBS-0011-3, Rev. 0, Oak Ridge, Tennessee.

ORAU (Oak Ridge Associated Universities), 2004c, *Estimation of Ingestion Intakes,* ORAUT-OTIB-009, Rev. 0, Oak Ridge, Tennessee.

ORAU (Oak Ridge Associated Universities), 2004d, *Technical Basis Document for the Pantex Plant – Occupational Internal Dosimetry,* ORAUT-TKBS-0013-5, Rev. 0, Oak Ridge, Tennessee.

Pantex, 2001, *Internal Dosimetry Technical Basis and Quality Assurance Document,* RSD-TBD-0001, Rev. 03, BWXT Pantex, Radiation Safety Department, Amarillo, Texas.

Poole, C. R. and K. C. Harrison, 1954. "Project History of Line 1 Operations at Iowa Ordnance Plant, January 1, 1947 through July 1, 1954," Silas Mason Company, Inc., Burlington, Iowa, ORAU file locator 001487.

Shahan, L. S. 1970. "The Fluorometic Determination of Uranium in Soil and Water," Technical Report No. 181, Burlington AEC Plant, Burlington, Iowa.

Shahan, L. S. 1971. "Natural and Depleted Uranium (D-38) Concentration in the IAAP Area," Technical Report No. 181, Part 3, Burlington AEC Plant, Burlington, Iowa.

Shleien, B., L. A. Slayback, and B. K. Birky, 1998, *Handbook of Health Physics and Radiological Health,* Third Edition, Williams & Wilkins, Baltimore, Maryland.

Shaykin, J. 1969, *Health Protection Appraisal Report – Burlington AEC Plant,* ORAU file locator 000922, January.

Tech/Ops Landauer, Inc. 1991. "Radon Monitoring Report for Mason & Hanger-Silas Mason, Co. Inc. – c/o Iowa Army Amm. Plant." Process Nos. A11370 and A11371, March 22, 1991, (file locator 000xxxx).

Traub, RJ, Scherpelz, RI, and Taulbee, TD, 2005 "Health Physics Analysis of Doses Received at the Iowa Ordnance Plant" PNNL-ETD-0385 (Classified Report)

TN & Associates, Inc., 2001, *Work Plan for Supplemental Remedial Investigation for Line 1 (Including Historical Site Assessment) for the US Army Corps of Engineers,* final draft, September (file locator 000910)

UNC Geotech, 1990, *Results of the U.S. Department of Energy Indoor Radon Study Volume 2,* DOE/ID/12584-75 v.1, U.S. Department of Energy, Grand Junction, Colorado.

UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation), 1993, *Sources and Effects of Ionizing Radiation.*

Vallario, E. J., D. E. Hankins, and C. M. Unruh, 1969, *AEC Workshop on Personnel Dosimetry,* BNWL-1340, Battelle Northwest Laboratory, Richland, Washington.

Wilson, R. H., 1958, "The Hanford Uranium Bioassay Program," unpublished report to file available in Radiation Records Historical Files, Hanford Laboratories, Richland, Washington.

GLOSSARY

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

baratol
A castable mixture of explosives used in nuclear weapons.

beta (ß) dose
A designation (i.e.beta) on some Pantex external dose records referring to the dose from less-energetic beta, X-ray, or gamma radiation.

beta radiation
Radiation consisting of charged particles of very small mass (i.e. the electron) emitted spontaneously from the nuclei of certain radioactive elements. Physically, the beta particle is identical to an electron moving at high velocity.

cladding
Material surrounding or encasing another material. In this TBD, cladding generally refers to the material or materials surrounding the fissile material of the pit.

curie
A special unit of activity. One curie (1 Ci) exactly equals 3.7×10^{10} nuclear transitions per second.

deep absorbed dose (D_d)
The absorbed dose at the depth of 1.0 cm in a material of specified geometry and composition.

deep dose equivalent (H_d)
The dose equivalent at the respective depth of 1.0 cm in tissue.

detection limit (lower)
The minimum quantifiable exposure or neutron flux that can be detected.

dose equivalent (H)
The product of the absorbed dose (D), the quality factor (Q), and any other modifying factors. The special unit is the rem. When D is expressed in Gy, H is in sieverts (Sv). (1 Sv = 100 rem).

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

dosimeter
A device used to measure the quantity of radiation received. A holder with radiation-absorbing element (filters) and an insert with radiation-sensitive elements packaged to provide a record of absorbed dose or dose equivalent received by an individual. (See *film dosimeter, neutron film dosimeter, thermoluminescent dosimeter).*

dosimetry

The science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external or internal sources of radiation.

dosimetry system

A system used to assess dose equivalent from external radiation to the whole body, skin, and extremities. This includes the fabrication, assignment, and processing of dosimeters as well as interpretation and documentation of the results.

Depleted Uranium (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.

exchange period (frequency)

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

exposure

As used in the technical sense, a measure expressed in roentgens (R) of the ionization produced by photons (i.e., gamma and X-rays) in air.

extremity

That portion of the arm extending from and including the elbow through the fingertips, and that portion of the leg extending from and including the knee and patella through the tips of the toes.

field calibration

Dosimeter calibration based on radiation types, intensity and energies present in the work environment.

film

Generally means a “film packet” that contains one or more pieces of film in a light-tight wrapping. The film when developed has an image caused by radiation that can be measured using an optical densitometer. (See *Dupont 552, Dupont 558, Eastman Kodak, Nuclear Emulsions).*

film density

See optical density.

film dosimeter

A small packet of film in a holder that attaches to a wearer.

gamma rays (?)

Electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g. fission, radioactive decay, and neutron capture). Physically, gamma rays are identical to X-rays but with higher energy; the only essential differences is that X-rays do not originate in the nucleus.

Gertie

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

Gray

SI unit of absorbed dose. Unit symbol, Gy. 1 Gy = 100 rad.

hydroshot

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

ionizing radiation

Electromagnetic or particulate radiation capable of producing charged particles through interactions with matter.

Line 1

Facilities and operations taken over by the AEC in 1947 for casting of baratols and processes related to the assembly of nuclear weapons.

Minimum Detectable Level (MDL)

A term used in this document and other NIOSH documents to refer to a statistically determined minimum detection level, Lower Limit of Detectability (L_D), and related quantities.

Minimum Reportable Dose (MRD)

A general term used to identify the minimum dose recorded and reported, normally based on site-specific policy.

neutron

A basic particle that is electrically neutral weighing nearly the same as the hydrogen atom.

neutron, fast

Neutrons with energy equal or greater than 10 keV.

neutron, intermediate

Neutrons with energy between 0.5 eV and 10 keV.

neutron, thermal

Strictly, neutrons in thermal equilibrium with surroundings. Generally, neutrons with energy less than about 0.5 eV.

neutron film dosimeter

A film dosimeter that contains a Neutron Track Emulsion, type A, film packet.

nuclear emulsion

Often referred to as “NTA” film and used to measure personnel dose from neutron radiation.

nuclear track emulsion, type A (NTA)

A film that is sensitive to fast neutrons. The developed image has tracks caused by neutrons that can be seen by using an appropriate imaging capability such as oil immersion and a 1000X power microscope or a projection capability.

open window

Designation on film dosimeter reports that implies the use of little shielding. It commonly is used to label the film response corresponding to the open window area.

optical density

The quantitative measurement of photographic blackening with the density defined as $D = Log_{10} (I_o/I)$.

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.

personal dose equivalent $H_p(d)$

Represents the dose equivalent in soft tissue below a specified point on the body at an appropriate depth d. The depths selected for personnel dosimetry are 0.07 mm and 10 mm, respectively, for the skin and body. These are noted as $H_p(0.07)$ and Hp(10), respectively.

photon

A unit or "particle" of electromagnetic radiation consisting of X- or gamma rays.

photon – X-ray

Electromagnetic radiation of energies between 10 keV and 100 keV whose source can be an X-ray machine or radioisotope.

pit

Term used to describe the nuclear physics package of a nuclear weapon. This component contains the fissile material that begins the fission chain reaction. Sometimes also referred to as the primary.

quality factor, Q

A modifying factor used to derive dose equivalent from absorbed dose.

radiation

Alpha, beta, neutron, and photon radiation with sufficient energy to ionize atoms. See also ionizing radiation.

radioactivity

The spontaneous emission of radiation, generally alpha or beta particles, gamma rays, and neutrons from unstable nuclei.

rem

A special unit of dose equivalent, which is equal to the product of the number of rad absorbed and the "quality factor."

roentgen (R)

A unit of exposure to gamma (or X-ray) radiation. It is defined precisely as the quantity of gamma (or x) rays that will produce a total charge of 2.58×10^{-4} coulomb in 1 kg of dry air. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher (>100 keV) energy photons.

shallow absorbed dose (D_s)

The absorbed dose at a depth of 0.007 cm in a material of specified geometry and composition.

shallow dose equivalent (H_s)

Dose equivalent at a depth of 0.007 cm in tissue.

shielding

Any material or obstruction that absorbs (or attenuates) radiation and thus tends to protect personnel or materials from radiation.

skin dose

Absorbed dose at a tissue depth of 7 mg/cm^2.

thermoluminescent

Property of a material that causes it to emit light as a result of being excited by heat.

thermoluminescent dosimeter (TLD)

A holder containing solid chips of material that when heated will release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

whole-body dose

Commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1000 mg/cm^2); however, this term is also used to refer to the recorded dose.

Work factor (W_f)

Used in this TBD to describe the relative portion of time that a worker would be in close proximity to a pit.

X-ray

Ionizing electromagnetic radiation that originates external to the nucleus of an atom.

Appendix A – Facility Information

The IAAP archival information associated with the DOE program to screen former workers for potential occupational health conditions provides a summary of IAAP facilities with some potential for radiation exposure to workers. Table A1 lists such facilities.

Table A1. IAAP Line 1 facilities handling radiation.

Building Number	Description	Use	Area Dosimeter Badges	Comments
1-11	Vault storage for components (pits)	Received and unloaded pits, some assembly.	1962 - 1974	Stationary air monitors in building. Some contamination measured at squash press area and squash removal area.
1-12	High explosives, fabrication, pressing and machining	Explosives pressed beginning mid-to-late 1960s.	None	Radiation swipe sampling, 1972-1975.
1-13	Assembly area, U-235 pits			Assembly of larger nonplutonium weapons (U-235) and covers. Radiation swipe data – 1974 for bays A-G.
1-18	Research and development			References to disposing of Radioactive waste (1971)
1-19	Assembly bays	"Experimental Building"		Tritium monitors
1-61	Assembly area – Operating Bays	Nonplutonium-bearing weapons (i.e., uranium)		Tritium monitors. Tritium shipping canisters were sampled and purged in this building.
1-63 Series	Operating bays and assembly cells, plutonium pits post 1956	Cells with Gravel Gerties (i.e. used to contain distribution of radioactive material in event of accident).		Built in 1957 as area to bring together explosives and plutonium physics package
1-64 Series	Storage	Cells with Gravel Gerties (i.e. used to contain distribution of radioactive material in event of accident).		
1-65 Series				
1-66 Series				Radiation swipe data, 1972-73
1-67 Series				Radiation swipe data, 1972-73
1-73	Storage and Receiving	Storage and receiving in early years until about 1957, then used as pit storage 1957 to late 1960s.	1962-	Pits were received in 1958, nuclear weapons built until 1-63 building was built, then was used as storage area for pits.
1-77	Pit storage and Inspection	Pit receiving and inert assembly, Built in mid-1960s		Radiation swipes and air sampling data, tritium bottles were charged here.
1-80				Radiation swipes data 1974
1-85-2				For subcomponents of depleted center items. Building built but not used by AEC.
1-100	X-Ray	Linitron X-ray		Radiation – explosives X-rayed For air cavities

Table A2 Yard C Facilities – Storage of Fissile Materials

Building number	Description	Use	Area Dosimeter Badges	Comments
23-15-7	Storage Igloos	Used for storage of pits	1965-1974	
23-39-15	Storage Igloos	Used for storage of pits	1965-1974	
23-39-20	Storage Igloos	Used for storage of pits	1965-1974	
23-39-21	Storage Igloos	Used for storage of pits	1965-1974	

Appendix B – List of Weapons Programs at IAAP

Designator	Delivery System	Assembly	Disassembly
Mk IV	Bomb	1949	1953
Mk 5	Bomb	1952	1963
Mk 6	Bomb	1951	1962
Mk 7	Bomb	1952	1967
B12	Bomb	1954	1963
B15	Bomb	1955	1965
B17	Bomb	1954	1957
W21	Bomb	1955	1957
W25	Genie	1957	1984
B27	Bomb	1958	1965
W27	Regulus II	1958	1965
B28	Bomb	1958	1991
W28	Hound Dog, Mace	1958	1991
W30	Talos, TADM	1959	1978
W31	Honest John, Nike Hercules, ADM	1958	1989
W34	Lulu, Hotpoint, Astor	1958	1964
B36	Bomb	1956	1962
B39	Bomb	1958	1966
W39	Snark, Redstome	1958	1965
W40	Bomarc, La Crosse	1959	1972
B41	Bomb	1960	1976
B43	Bomb	1961	1972
W44	ASROC	1961	1989
W47	Polaris A1/A2	1960	1975
W49	Thor, Jupitor, Atlas Titan I	1958	1975
W50	Pershing I	1963	1973
W52	Sergeant	1962	1977
B53	Bomb	1962	Retired ?
W53	Titan II	1962	1987
W54	Falcon, Davy Crokett, Special ADM	1964	1978
W56	Minuteman II	1964	1993
B57	Depth/Tactical Bomb	1963	1975
W58	Polaris A3	1964	1968
W59	Minuteman I	1962	1970
B61	Bomb	1967	Enduring
B66	Sprint	1974	1979
W68	Poseidon	1970	1977
W69	SRAM	1972	1992
W71	Spartan	1974	1993
W72	Walleye	1970	1979

*Note that the Designators Mk (Mark), B, W, XW are interchangeable and varied over time.

Appendix C – Summary of Intakes and Details of Ingestion Calculations

C-1.0 INTAKES

Table C-1 lists intakes by work tasks for time periods applicable for input into IMBA. The table includes intakes from environmental releases and exposure from radon. The worker category, "all personnel," applies to all AEC workers even if they are also included in the other categories.

Table C-1. Summary of intakes by AEC workers at IAAP.

Work or worker category[a]	Dose calculation parameters						IREP input Parameters		
	Period	Material	Mode	Absorption type	pCi/d	mg/d	Distribution Type	1	2
All personnel	1/1/1948-7/1/1975	DU	Chronic inhalation	M, S	0.017	4.5E-05	Constant	Dose	
All personnel	12/2/1965-3/3/1969	DU	Chronic inhalation	M, S	0.024	6.4E-05	Lognormal	Dose	4
All personnel	3/4/1969-7/14/1969	DU	Chronic inhalation	M, S	0.0012	3.3E-06	Lognormal	Dose	4
All personnel	7/15/1969-12/31/1973	DU	Chronic Inhalation	M, S	0.0055	1.5E-05	Lognormal	Dose	4
All personnel	1/1/1954-7/1/1975	HTO	Chronic inhalation/ absorption		6.8E+03		Constant	Dose	
All personnel	1/1/1948-7/1/1975	Radon[b]	Chronic inhalation				Lognormal	0.072 WLM/ 12 mo.	3
Machinist	1/1/1948-12/31/1962	DU	Chronic inhalation	M, S	6.6	1.8E-02	Constant	Dose	
Machinist	1/1/1948-12/31/1962	DU	Chronic ingestion	Insoluble	0.2	5.4E-4 (mean)	Triangular	0.5 mean	mean[c]
Burning yard operations	1/1/1948-7/1/1975	DU	Chronic inhalation	M, S	38	0.10	Lognormal	Dose	3
Weapons assembly, disassembly, surveillance, inspection	1/1/1954-7/1/1975	HTO	Chronic inhalation/ absorption		1.34E+7		Constant	Dose	
Weapons assembly, disassembly, surveillance, inspection	1/1/1949-12/31/1974	DU	Chronic inhalation	M, S	223	0.555	Constant	Dose	
Hydroshot operations	12/2/1965-3/3/1969	DU	Chronic inhalation	M, S	0.28	7.5E-04	Lognormal	Dose	4.6
Hydroshot operations	3/4/1969-7/14/1969	DU	Chronic inhalation	M, S	0.014	3.8E-05	Lognormal	Dose	4.6
Hydroshot operations	7/15/1969-12/31/1973	DU	Chronic inhalation	M, S	0.064	1.7E-04	Lognormal	Dose	4.6
Hydroshot cleanup	12/2/1965-3/3/1969	DU	Chronic ingestion	Insoluble	1.2E+03	3.3	Constant	Dose	
Hydroshot cleanup	3/4/1969-7/14/1969	DU	Chronic ingestion	Insoluble	6.4E+01	0.17	Constant	Dose	
Hydroshot cleanup	7/15/1969-12/31/1973	DU	Chronic ingestion	Insoluble	2.9E+02	0.78	Constant	Dose	
FS-12 Operations	12/2/1965-12/31/1974	DU	Chronic inhalation	M, S	30.7	0.076	Constant	Dose	
All personnel	12/2/1965-12/31/1974	DU	Chronic ingestion	Soluble	0.51	0.001	Constant	Dose	

a. Workers performing specific listed tasks, e.g. burning yard operations, are also assigned the intakes for "all personnel."
b. Applies to workers on Line 1. However, unless it is clear that the worker did not work inside the Line 1 buildings, apply the radon intake to everyone.
c. Parameter 3 is 2 x mean.

C2.0 INGESTION INTAKE CALCULATION

Because the ingestion intake calculation for the hydroshot cleanup crew involved picking up source material, not touching surfaces contaminated by settling of particles from the air in a room, the intake calculation was not based on the draft OTIB on ingestion (ORAU 2004c). Instead, an experiment was conducted for this TBD, wherein a small amount of ordinary soil, judged by the author to be similar to what a worker would get on his/her hands by handling a dirty object, was acquired on a small piece of paper with double-sided sticky tape. The net weight of the dirt was measured to be 83 mg with an uncertainty of about 2 mg. The largest

uncertainty was in the judgment of the volume of dirt that would be on someone's hand. Soil has a density of about 2 g/cm^3, so the volume of dirt was about 0.042 cm^3,

For the hydroshot cleanup crew, it was assumed that surfaces of the DU chards were partially covered with unexploded HE and dirt from the impact with the ground. In addition, the action of picking up chards from the ground would have introduced more dirt to the hands. So it was assumed that 10% of the volume of material on the hands would have been DU. Assuming the form of the DU was mostly metal with some oxide, the density of the DU would have been about 18 g/cm^3. So the DU contamination on the hands would have weighed about 76 mg [(18,000 mg/cm^3) (0.0042 cm^3)]. It was then assumed that 10% of the DU on the hands was ingested (as suggested in the draft OTIB on ingestion). This is an upper bound assumption which includes the possibility that the worker ate or drank without washing his/her hands. It was further assumed that this activity occurred after each hydroshot, so the ingestion rate in mg/calendar day is simply

(7.6 mg)(no. shots in period)/(calendar days in period)

Appendix D – Generic Pit Specifications

Specifications for a "bare" (unshielded) unclassified pit used to develop dose ratio for EEOICPA Dose Reconstructions

TD Taulbee

The following discussion is intended to 1) describe the assumptions and 2) provide specifications for a "bare" (unshielded) model pit for dose reconstructions conducted under the Energy Employees Occupational Illness Compensation Program Act (EEOICPA) of 2000. The "bare" (unshielded) pit specifications in conjunction with MCNP calculations will be used to estimate the photon and neutron dose rates and ratios. The dose ratios will subsequently be used in conjunction with co-worker data to estimate external radiation doses during time periods when individual radiation dosimetry (film badges) were not worn and/or were not capable of measuring the low energy photon dose.

Currently, actual pit specifications assembled and disassembled at IAAP remain classified. In order to develop unclassified "bare" pit specifications for the dose reconstruction calculations, classified information was reviewed and discussed with the Department of Energy (DOE). Under NIOSH procedures, dose reconstructions are to give the benefit of the doubt or uncertainty to the claimant. NIOSH considers the classified information as uncertainty and, therefore, has given the benefit of the doubt towards overestimating the dose.

In the development of these specifications, NIOSH has estimated the mass and the surface area of a typical pit handled at IAAP. The effect that each assumption has on the dose rate is discussed in more detail below. The tendency of these two estimations increases the calculated photon dose rate resulting in an overall overestimate of the photon dose rate. Thus when used in conjunction with coworker data from an earlier time period as a ratio, the result will overestimate the true dose experienced by IAAP workers. This overestimate is considered reasonable and necessary to meet the intent of the EEOICPA and to preserve and protect national security interests. If NIOSH used actual pit specifications to calculate the photon dose, the resulting dose will be lower than the estimates derived here.

Assumed Mass for Bare Pit

For the "bare" unshielded pit, NIOSH assumes a mass of 6 kg of 15 year aged weapons-grade plutonium in delta phase (density: ? = 16.00 g/cm^3). This mass was selected since it is the approximate mass of plutonium used in the Trinity and Nagasaki nuclear devices.[1] This assumed mass is greater than the assumed plutonium mass released in accident scenario #1 of the Environmental Impact Statement for the Pantex Plant.[2] In scenario #1, an activity of 300 Ci (5 kg) of plutonium is assumed released following an accidental detonation in a cell. Based on these considerations, a mass of 6 kg is considered claimant favorable. For the bare unshielded pit, weapons-grade plutonium was assumed as opposed to enriched uranium since plutonium has a higher photon dose rate.

Assumed Diameter for Nominal Bare Pit

For this nominal "bare" unshielded pit, NIOSH assumes an outer diameter of one foot (12 inches). The purpose of this assumption is to result in a relatively thin shell such that photons (especially low energy photons) are less self-shielded (attenuated) by the assumed mass. Using the two criteria (mass and assumed outer diameter), the metal shell thickness was calculated to be approximately 1.3 mm (i.e. thin shell). This thin shell will have minimal

photon attenuation and will, therefore, overestimate the true photon dose rate from pits handled at IAAP.

Nominal Bare (Unshielded) Pit Specifications
Assumed Mass of Weapons-grade Plutonium: 6.0 kg
Assumed Outer Diameter: 12 inches
Calculated Shell Thickness: 1.3 mm
Assumed Cladding: None

Using these specifications, dose rates were calculated at 100 cm using MCNP. Table D.1 provides the neutron and photon energy distribution results and dosimetry correction factors.

Table D.1 Generic Pit Dose Rate Information

Description	Designator	Energy Interval	Dose Rate (mrem/h)	Correction Factors (P*/C7)[a]
Total Photon	PT		33.3	
Low Energy	P1	< 30 keV	1.18	0.080
Intermediate Energy	P2	30 – 250 keV	27.8	1.878
High Energy	P3	> 250 keV	4.49	0.303
Measured Energy	P4	> 70 keV	7.4	
Film Badge Energy	C7	> 70 keV + 0.3*(50-70 keV)	14.8	
Total Neutron	NT		0.582	
Thermal	N1	< 10 keV	0.004	
Intermediate Energy	N2	10 – 100 keV	0.008	
Fission	N3	0.1 – 2 MeV	0.334	
Fast	N4	2 - 20 MeV	0.242	
Ultra Fast	N5	> 20 MeV	0.000	

a. Correction factor for dosimeter response according to IREP input energy ranges.

In summary, while the assumptions listed above will result in a dose rate that exceeds the true dose rate, the assumptions are considered reasonable and necessary in order to meet the intent of EEOICPA and preserve and protect national security interest.

References

1. Groves, Leslie (1945) Memorandum concerning the Trinity test dated July 16, 1945.
2. DOE (2003) Supplemental Information to the Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components. DOE/EIS-0225

Appendix D cont. – Generic Pit Calculated Spectral Data (Figure 6.1)

Energy (keV)	Dose (mrem/h)	Energy (keV)	Dose (mrem/h)	Energy (keV)	Dose (mrem/h)	Energy (keV)	Dose (mrem/h)
10	3.948E-04	510	2.227E-03	1010	9.090E-06	1510	0.000E+00
20	1.127E+00	520	9.017E-03	1020	1.265E-05	1520	1.490E-05
30	5.683E-02	530	2.658E-05	1030	8.016E-03	1530	0.000E+00
40	3.118E-02	540	9.524E-03	1040	3.068E-05	1540	4.505E-03
50	1.141E+00	550	2.414E-03	1050	8.160E-06	1550	2.158E-05
60	2.354E+01	560	2.427E-03	1060	8.222E-06	1560	2.198E-05
70	4.134E-02	570	1.476E-02	1070	8.306E-06	1570	0.000E+00
80	1.158E-01	580	1.001E-02	1080	0.000E+00	1580	0.000E+00
90	8.219E-02	590	5.096E-03	1090	2.318E-05	1590	0.000E+00
100	4.204E-01	600	5.152E-03	1100	1.079E-05	1600	1.015E-05
110	5.301E-01	610	1.843E-02	1110	1.379E-05	1610	9.436E-06
120	4.922E-01	620	3.186E-02	1120	1.886E-05	1620	1.906E-05
130	2.082E-01	630	1.892E-02	1130	1.687E-05	1630	7.594E-06
140	1.422E-01	640	1.094E-02	1140	0.000E+00	1640	0.000E+00
150	1.637E-01	650	2.216E-02	1150	4.179E-03	1650	2.692E-05
160	1.092E-02	660	2.535E-02	1160	0.000E+00	1660	0.000E+00
170	1.300E-01	670	1.419E-02	1170	0.000E+00	1670	4.093E-06
180	4.614E-02	680	1.152E-02	1180	4.181E-03	1680	2.597E-05
190	4.075E-02	690	3.507E-02	1190	8.598E-06	1690	2.930E-05
200	3.985E-02	700	3.246E-02	1200	6.808E-06	1700	0.000E+00
210	7.157E-02	710	2.999E-03	1210	2.883E-05	1710	9.785E-06
220	1.024E-01	720	1.516E-02	1220	1.354E-05	1720	4.583E-03
230	8.090E-02	730	2.145E-02	1230	4.223E-03	1730	2.377E-05
240	9.768E-02	740	1.243E-02	1240	1.782E-05	1740	0.000E+00
250	1.369E-01	750	9.370E-03	1250	2.065E-05	1750	1.549E-05
260	2.929E-02	760	1.588E-02	1260	1.494E-05	1760	2.206E-05
270	2.033E-02	770	1.606E-02	1270	0.000E+00	1770	3.645E-05
280	2.476E-02	780	9.725E-03	1280	0.000E+00	1780	5.140E-05
290	2.611E-02	790	1.964E-02	1290	1.835E-05	1790	2.049E-05
300	3.307E-02	800	6.663E-02	1300	3.486E-06	1800	1.363E-05
310	1.239E-01	810	3.386E-03	1310	4.980E-05	1810	4.578E-06
320	1.331E-01	820	3.432E-03	1320	0.000E+00	1820	0.000E+00
330	1.427E-01	830	3.455E-03	1330	1.484E-05	1830	0.000E+00
340	1.737E-01	840	1.114E-05	1340	0.000E+00	1840	9.838E-06
350	1.584E-01	850	2.665E-05	1350	1.300E-05	1850	0.000E+00
360	3.086E-01	860	1.161E-05	1360	1.594E-05	1860	9.859E-06
370	3.272E-01	870	3.557E-03	1370	6.449E-06	1870	0.000E+00
380	3.153E-01	880	4.957E-06	1380	0.000E+00	1880	9.638E-06
390	3.451E-01	890	5.044E-06	1390	2.707E-05	1890	4.609E-06
400	3.527E-01	900	3.617E-03	1400	0.000E+00	1900	0.000E+00
410	2.086E-01	910	7.790E-06	1410	1.044E-05	1910	0.000E+00
420	2.304E-01	920	4.994E-06	1420	1.059E-05	1920	0.000E+00
430	2.450E-01	930	2.835E-05	1430	2.975E-05	1930	6.915E-06
440	2.758E-01	940	1.189E-05	1440	8.050E-06	1940	0.000E+00
450	2.612E-01	950	3.073E-05	1450	6.417E-06	1950	2.934E-05
460	1.350E-02	960	0.000E+00	1460	0.000E+00	1960	2.450E-05
470	1.390E-02	970	5.169E-05	1470	0.000E+00	1970	0.000E+00
480	1.636E-02	980	0.000E+00	1480	0.000E+00	1980	1.892E-05
490	1.471E-02	990	8.862E-06	1490	0.000E+00	1990	7.605E-06
500	2.584E-02	1000	2.087E-05	5000	2.190E-05	2000	0.000E+00

Appendix E – Special Nuclear Material (SNM) Encapsulation and Pit Cladding

During the public meeting in June 2004, there was mention of "bare" pits, and the pits were thermally warm to the touch. In a follow-up worker meeting in July 2004, there was some discussion about this thermal conductivity and that some types of gloves were better than others to protect energy employees' hands during direct handling of pits. The radiological concerns with this issue have both internal and external dose implications. To address this issue, NIOSH has reviewed classified documentation and interviewed Department of Energy (DOE) personnel knowledgeable of pit designs and assembly and disassembly operations.

Internal Dose

Based on NIOSH's review, all nuclear capsules and pits were clad with some type of material. From the energy employee viewpoint, the pits could have appeared and felt through thermal radiance (heat) as if they were bare plutonium or uranium metal. According to DOE, all the pits manufactured since the late 1940s are known to be clad. In some instances, this cladding could be a very thin layer of material, thus the pit could be thermally warm and could also look like bare metal. As a result of our analysis, NIOSH has determined that there is no significant potential for internal exposure from pits.

It should be noted that radiological incident records at Pantex have identified a few accidents where the cladding was compromised resulting in contamination and airborne radioactivity. These incidents were not common, and the specific instances have been well documented. NIOSH has reviewed over 200 incident reports from IAAP between 1959 and 1974. From these reports, NIOSH has identified 15 incidents that involved radioactive materials in which special radiological surveys or precautionary building evacuations were conducted. To date, there is no evidence that a radiological incident resulted in the release of special nuclear materials at IAAP.

External Dose

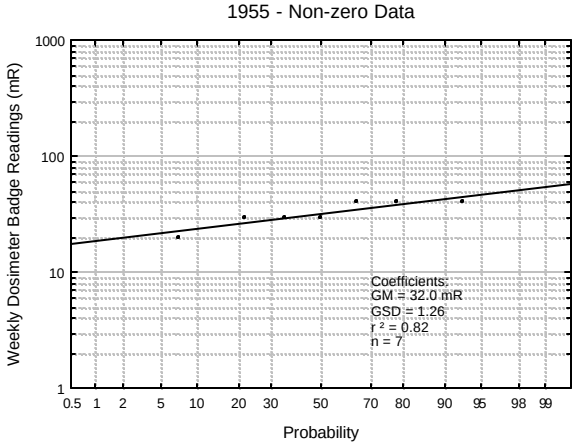
While there is no significant potential for internal exposure from nuclear capsules (pits), the cladding material and thickness can affect the external dose rate. This is particularly problematic for lower energy photons (< 70 keV). High-Z materials such as uranium greatly reduce the photon dose rate and almost completely shield the worker from low energy photon emissions (< 30 keV). Conversely, low-Z materials such as beryllium, do not provide much shielding and allow a relatively large quantity of low energy photons (< 30 keV) to pass. Through discussions with DOE, NIOSH has learned that exact information on the cladding material and thickness for each weapon design assembled and disassembled at IAAP remains classified.

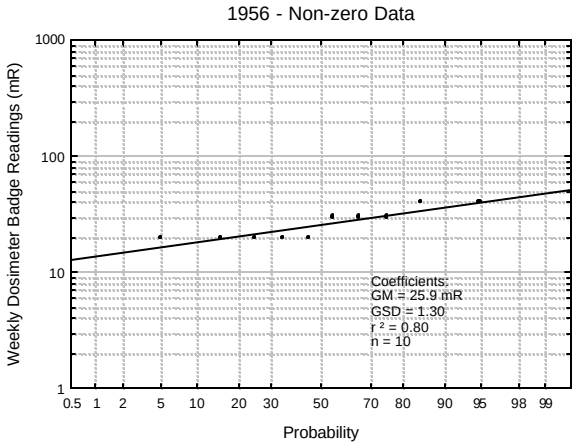
It is important to note that not all components had a significant low energy photon dose. There are three basic types of pits used in assembly at IAAP; 1) enriched uranium pits, 2) plutonium pits, and 3) composite pits (combination plutonium and enriched uranium). In the composite pits, the plutonium always had an outer shell of enriched uranium which would effectively shield worker exposure to lower energy photons. Since the low energy photon dose from enriched uranium is negligible, only the plutonium pits had the potential for significant low energy photon dose.

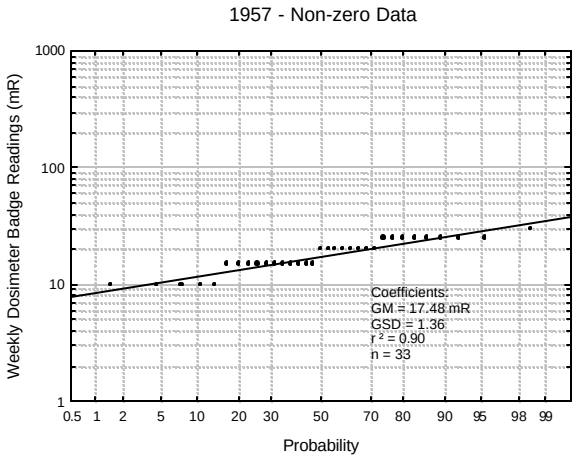
Although the cladding information necessary to precisely estimate the external dose remains classified, an effective maximum photon emission rate (radiation dose) can be determined by assuming that there was no cladding or shielding between the fissile materials (pits) and the energy employee. For external dose reconstruction only, NIOSH assumes that: 1) all pits were plutonium, and 2) there is no cladding of the fissile materials. This assumption will result in an overestimate of the true external dose for both low energy and intermediate energy photons. Furthermore, since low energy photons could not be accurately measured with the film badge dosimeter used at IAAP (i.e. the shielded window blocked low energy photons and attenuated photons less than about 70 keV), no cladding is assumed for low energy photon dose over the entire operations history at IAAP (1949-1974).

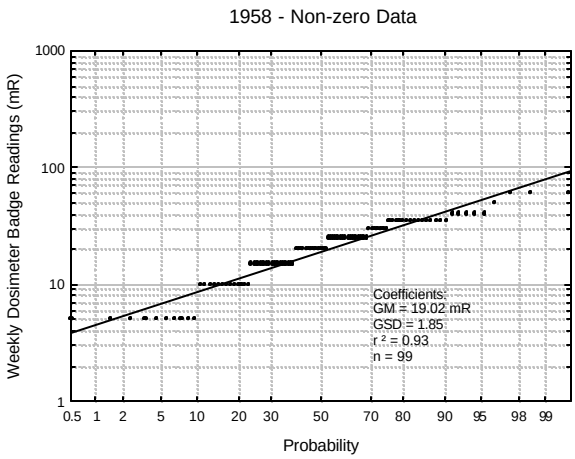
Since most of the fissile materials in weapons assembled and disassembled at IAAP were of a design that greatly diminished the low energy photon dose, the assumptions stated above will overestimate the total external photon dose. The assumption is, however, considered reasonable and necessary in order to meet the intent of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) and still preserve and protect national security interests.

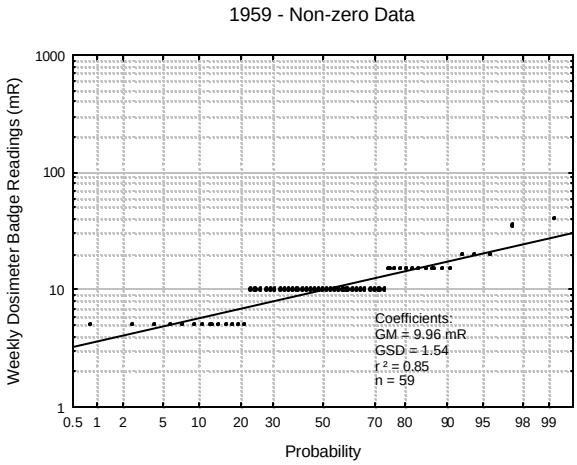
Appendix F – IAAP Annual Dosimeter Badge Distributions

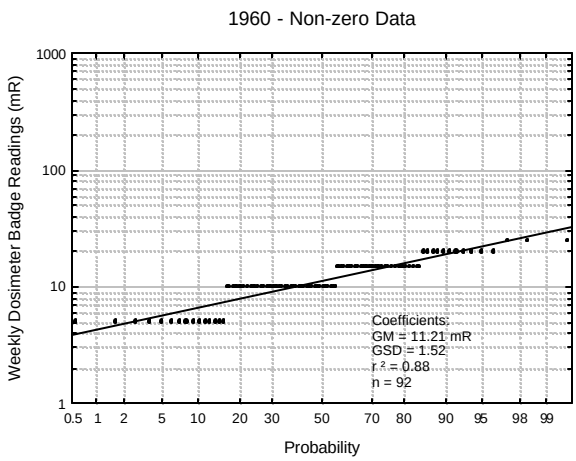






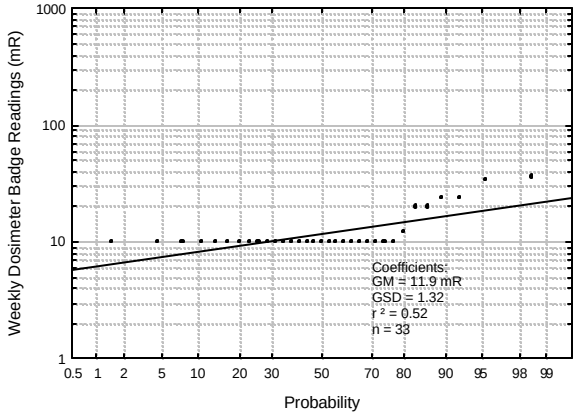




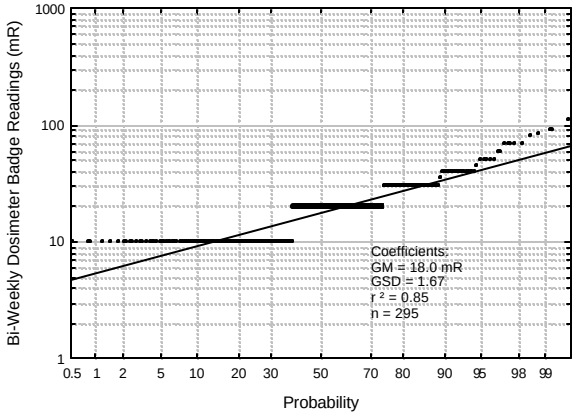


1961 Data Not Available

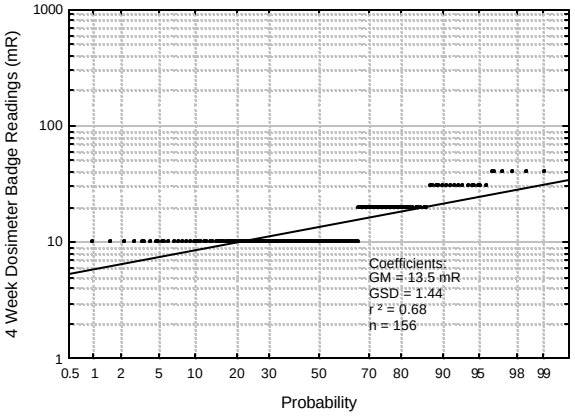
1962 - Non-zero Data



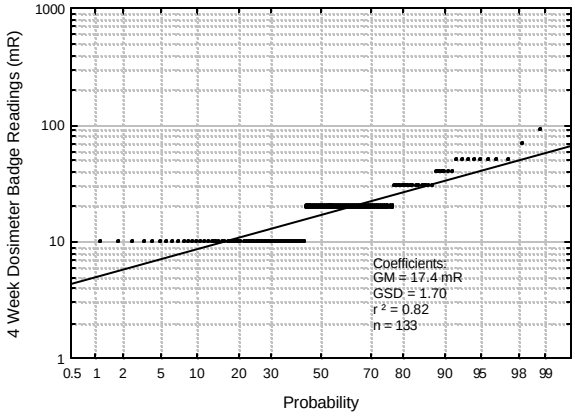
1963 - Non-zero Data



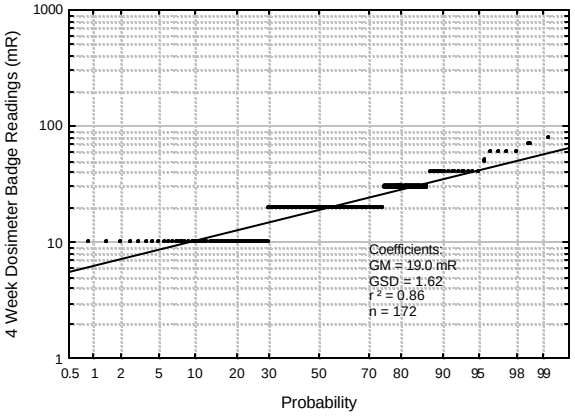
1964 - Non-zero Data



1965 - Non-zero Data



1966 - Non-zero Data



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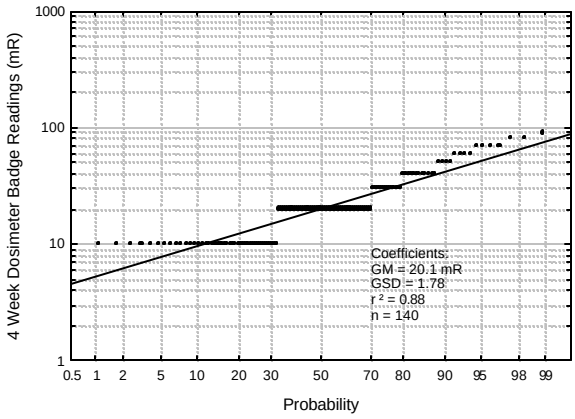

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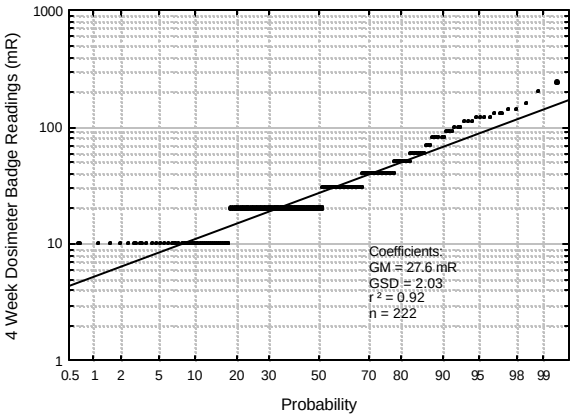
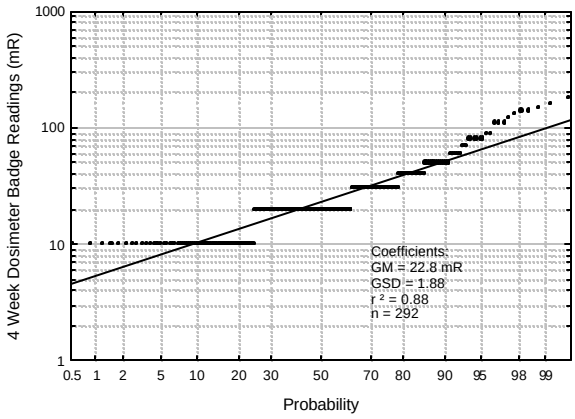
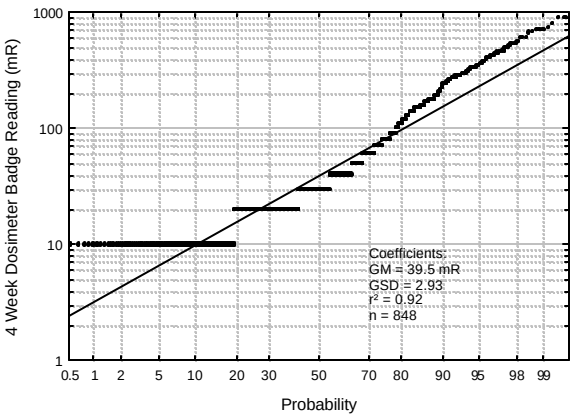


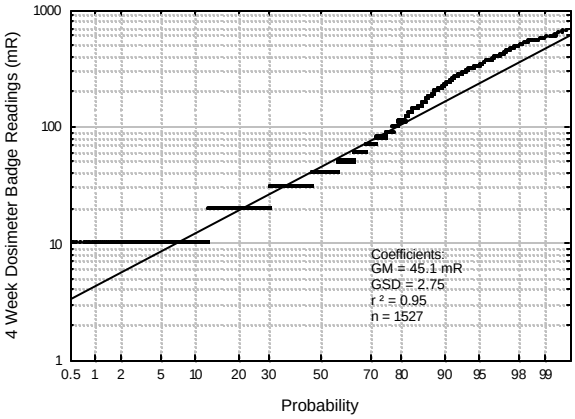
1969 - Non-zero Data



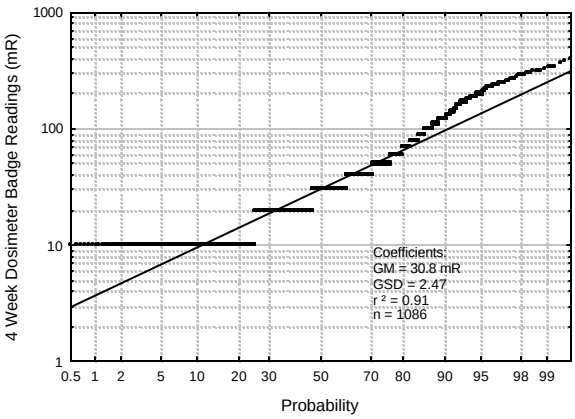
1970 - Non-zero Data

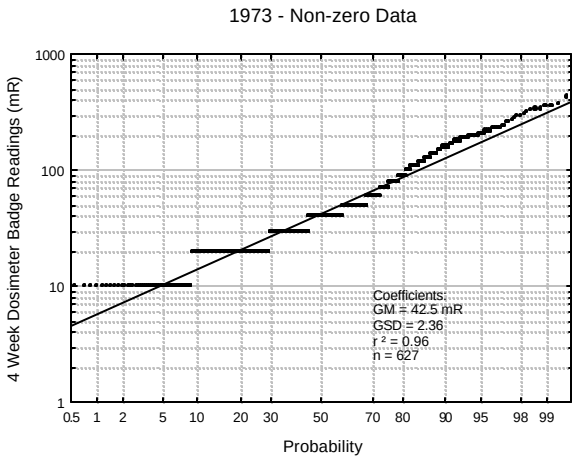


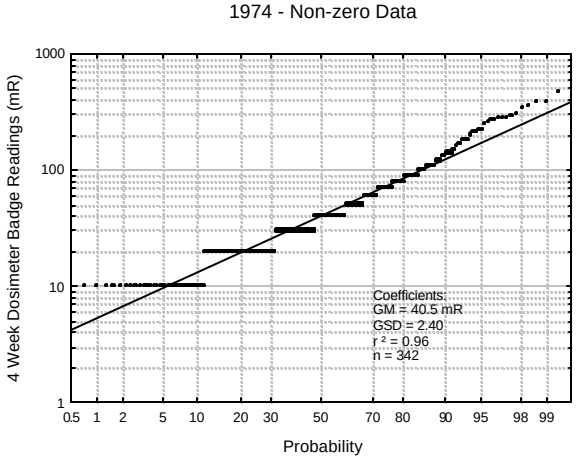
1971 - Non Zero Data



1972 - Non-zero Data




1973 - Non-zero Data
4 Week Dosimeter Badge Readings (mR)
1000
100
10
1
0.5 1 2 5 10 20 30 50 70 80 90 95 98 99
Probability
Coefficients:
GM = 42.5 mR
GSD = 2.36
r ² = 0.96
n = 627


1974 - Non-zero Data
4 Week Dosimeter Badge Readings (mR)
1000
100
10
1
0.5 1 2 5 10 20 30 50 70 80 90 95 98 99
Probability
Coefficients:
GM = 40.5 mR
GSD = 2.40
r ² = 0.96
n = 342

Appendix G –Bladder Cancer Example Calculation for an Unmonitored Worker

Year	Initial Dose (mrem)	Am^{241} in-growth	R to HP(10) Correction	Estimated Annual Dose	Annual Dose (mrem)			Bladder DCFs HP(10)			Bladder Total (mrem)
					< 30 keV	30-250 keV	>250 keV	< 30 keV	30-250 keV	>250 keV	
					0.08	1.88	0.303	0.17	0.873	0.913	
1949	4529	0.67	1	3034	243	5699	919	41	4975	839	5856
1950	4529	0.71	1	3216	257	6039	974	44	5272	890	6205
1951	4529	0.75	1	3397	272	6379	1029	46	5569	940	6555
1952	4529	0.78	1	3533	283	6634	1070	48	5792	977	6817
1953	4529	0.81	1	3668	293	6889	1112	50	6014	1015	7079
1954	4529	0.84	1	3804	304	7145	1153	52	6237	1052	7341
1955	4529	0.87	1	3940	315	7400	1194	54	6460	1090	7604
1956	4529	0.9	1	4076	326	7655	1235	55	6683	1128	7866
1957	4529	0.93	1	4212	337	7910	1276	57	6905	1165	8128
1958	4529	0.95	1	4303	344	8080	1304	59	7054	1190	8303
1959	4529	0.98	1	4438	355	8335	1345	60	7277	1228	8565
1960	4529	1	1	4529	362	8505	1372	62	7425	1253	8740
1961	4529	1	1	4529	362	8505	1372	62	7425	1253	8740
1962	4529	1	1	4529	362	8505	1372	62	7425	1253	8740
1963	468	1	1.05	491	39	923	149	7	806	136	948
1964	176	1	1.05	185	15	347	56	3	303	51	357
1965	226	1	1.05	237	19	446	72	3	389	66	458
1966	247	1	1.05	259	21	487	79	4	425	72	500
1967	261	1	1.05	274	22	515	83	4	449	76	529
1968	354	1	1.05	372	30	698	113	5	609	103	717
1969	296	1	1.05	311	25	584	94	4	510	86	600
1970	514	1	1.05	540	43	1014	164	7	885	149	1041
1971	586	1	1.05	615	49	1156	186	8	1009	170	1187
1972	400	1	1.05	420	34	789	127	6	689	116	810
1973	553	1	1.05	581	46	1090	176	8	952	161	1120
1974	527	1	1.05	553	44	1039	168	8	907	153	1068