ORAU TEAM
Dose Reconstruction Project for NIOSH

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
01/18/2005	00	Technical Basis Document for Portsmouth Gaseous Diffusion Plant – Occupational External Dose. First approved issue. Initiated by Mark D. Notich.
11/07/2006	01	Approved Revision 01 to incorporate worker outreach comments addressed in CT-0189. Amends Section 6.3.2.2 based on worker outreach comments. Adds required language to the Introduction. Adds purpose and scope subsections to the Introduction. Adds four references. Incorporates formal internal, NIOSH and DOL review comments. Table 6-37 has been changed to include footnote citations. Constitutes a total rewrite of the document. Sections 6.5, 6.5.1, and 6.5.2 and the reference section were changed to include ORAUT-OTIB-0040, Rev. 00, “External Coworker Dosimetry Data for the Portsmouth Gaseous Diffusion Plant” coworker dosimetry data information for dose reconstruction. ORAUT-OTIB-0017, Rev. 01, and ORAUT-OTIB-0040, Rev. 00 citations were also added. This revision results in an increase in assigned dose and a PER is required. Initiated by Paul J. Demopoulos. Training required: As determined by the Task Manger. Initiated by Paul J. Demopoulos.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AP	anterior-posterior (X-ray view)
BJC	Bechtel Jacobs Company
cm	centimeter
cpm	counts per minute
DDE	deep dose equivalent
DE	dose equivalent
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
ft	foot
g	gram
GAT	Goodyear Atomic Corporation
GM	Geiger-Mueller detector
HEU	highly enriched uranium
HP	health physics
Hp(d)	personal dose equivalent at depth d in tissue
hr	hour
IARC	International Agency for Research on Cancer
ICN	International Chemical and Nuclear Corporation
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
in.	inch
keV	kiloelectron-volt, 1,000 electron-volts
kV	kilovolt
LAW	low-assay withdrawal
LOD	limit of detection
LODR	limit of dose rate
mCi	millicurie
MDL	minimum detectable level
MED	Manhattan Engineer District
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
mm	millimeter
mo	month
mR	milliroentgen

mrem	millirem
MTU	metric tons of uranium
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Laboratory Accreditation Program
ORNL	Oak Ridge National Laboratory
OW	open window (i.e., no filter) nonpenetrating dose
PGDP	Paducah Gaseous Diffusion Plant
POC	probability of causation
PORTS	Portsmouth Gaseous Diffusion Plant
QF	quality factor
R	roentgen
rem	radiation equivalent man
rep	radiation equivalent physical
RN	radionuclide
RU	recycled uranium
SDE	shallow dose equivalent
SRS	Savannah River Site
TBD	technical basis document
TEPC	Tissue Equivalent Proportional Counter
TLD	thermoluminescent dosimeter
TLND	thermoluminescent neutron dosimeter
U.S.C.	United States Code
USEC	United States Enrichment Corporation
wk	week
yr	year
α	alpha particle
µCi	microcurie
µg	microgram
§	section or sections

6.1 INTRODUCTION

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

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

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

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

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

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

Radiation dosimetry practices at the Portsmouth Gaseous Diffusion Plant (PORTS) were based initially on experience gained during several decades of radium and X-ray medical diagnostic and therapy applications. These methods were well advanced at the start of the Manhattan Engineer District (MED) program to develop nuclear weapons in about 1940. The primary new challenges encountered by MED, and later the U.S. Atomic Energy Commission (AEC), operations were to measure worker dose to external radiation involved:

- Comparatively large quantities of high-level radioactivity
- Mixed radiation fields involving beta, photon (gamma and X-ray), and neutron radiation with low, intermediate, and high energies
- Neutron radiation

From 1954 until 1986, Goodyear Atomic Corporation (GAT) operated the PORTS site. In 1986, Martin Marietta Energy Systems assumed responsibility for PORTS operations. The Energy Policy Act of 1992 transferred responsibility for the site from the U.S. Department of Energy (DOE) to a newly created entity, the United States Enrichment Corporation (USEC), which leased the nonoperational portion of PORTS to Bechtel Jacobs Company (BJC), the DOE primary contractor (USEC 2003). On July 1, 1993, the operational side of PORTS officially transferred to USEC (USEC 2003), which is regulated by the U.S. Nuclear Regulatory Commission (NRC). From 1993 to 1999, Martin Marietta Utility Services, which became Lockheed Martin Utility Services, operated the USEC portion of the site as the primary contractor. In May 1999, USEC assumed control without a primary contractor. The remaining portion of PORTS (including cylinder storage and legacy wastes) is under the operation of DOE; BJC has been the primary contractor since April 1998.

PORTS employees have been exposed to gamma, beta, neutron, and X-ray radiation. Early employees were exposed to higher levels of radiation due to (1) recycled uranium that could have included transuranic elements and ^{99}Tc, (2) highly enriched uranium, or (3) greater amounts of processed materials. From the beginning of operations in 1954 to 1980, PORTS monitored its employees for external exposures beginning with film dosimetry and, from 1981 to the present, with various forms of thermoluminescent dosimetry.

Records of surveys, investigations, procedures, and facility controls are available. The types of instrumentation and procedures used over the history of the PORTS site have varied. The *External Dose Reconstruction Implementation Guideline* (NIOSH 2002) recognizes that personnel external dosimetry issued to workers is the best way to determine external dose. Methods of calibration and limitations of dosimetry systems have been documented with on- and offsite references, as indicated in this TBD.

6.1.1 Purpose

This TBD represents a specific support mechanism concerning documentation of external dosimetry historical practices at the PORTS plant. This external dose TBD can be used to evaluate external dosimetry data for monitored workers and can serve as a supplement to individual monitoring data. For unmonitored workers, this document presents information that can provide for estimations of external doses. This document provides a site profile of PORTS that contains technical basis information to be used to evaluate the total occupational external radiation dose for EEOICPA claimants.

6.1.2 Scope

This document provides supporting technical data to evaluate the total PORTS plant occupational radiation dose that can reasonably be associated with the worker's radiation exposure. This dose results from exposure to external radiation sources in PORTS facilities that would be added to PORTS occupationally required diagnostic X-ray examinations, and to onsite environmental releases, if applicable, in order to determine the total external dose. Also included are techniques to estimate the doses that could have occurred while an employee was not monitored, was inadequately monitored, and whose monitoring records are incomplete or missing (i.e., missed dose), as well as dose that could have been missed due to analytical detection limits. Over the years new and more reliable scientific methods and protection measures have been deployed. The methods needed to account for these changes are also identified in this document.

This Occupational External Dosimetry TBD describes the external dosimetry program at PORTS. It discusses dose reconstruction, practices and policies at PORTS, and dosimeter types and technologies for measuring dose from the different types of radiation present in the work environment. It also discusses the specific details of the evaluation of doses measured from exposure to beta, gamma, and neutron radiation; sources of bias; workplace radiation field characteristics; responses to different beta/gamma and neutron dosimeters in the workplace fields; and adjustments to the recorded dose measured by these dosimeters during specific years.

6.2 BASIS OF COMPARISON

A basis of comparison for dose reconstruction is the personal dose equivalent *Hp(d)* where *d* identifies the depth (in millimeters) and represents the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose d = 0.07 mm, and skin dose is noted as *Hp(0.07).* For penetrating radiation of significance to whole-body dose d = 10 mm, and whole-body dose is noted as *Hp(10).* The International Commission on Radiological Units and Measurements (ICRU) recommends the use of both *Hp(0.07)* and *Hp(10)* as the operational quantities to be recorded for radiological protection (ICRU 1993). In addition, *Hp(0.07)* and *Hp(10)* are the quantities used in the DOE Laboratory Accreditation Program (DOELAP), which the Department has used to accredit personnel dosimetry systems since the 1980s. The National Laboratory Accreditation Program (NVLAP), which is the NRC equivalent to DOELAP, uses the same operational quantities.

PORTS has monitored photon (gamma) and beta radiation since 1954. Therefore, comparisons with similar dosimetry systems monitoring nearly equivalent radiation work environments are possible. PORTS operated its own film dosimetry system from 1954 to 1980 and its own thermoluminescent dosimetry system from 1981 to January 1, 1999. The Plant did not monitor neutron dose with personal dosimetry until 1992, then operated an onsite neutron dosimetry system from 1992 to 1994. By 1995, vendors provided dosimetry for USEC [with the International Chemical and Nuclear Corporation (ICN) as the vendor] and BJC [Oak Ridge National Laboratory (ORNL) with Y-12 laboratories as the vendor]. As early as 1963, PORTS recognized that the health physics (HP) group did not have adequate instrumentation to monitor for neutrons. The policy was that neutron monitoring was not necessary (Cardarelli 1997). Pacific Northwest Laboratories conducted a neutron survey in 1992 (Soldat and Tanner 1992). PORTS did not routinely monitor for neutrons until 1997.

6.3 DOSE RECONSTRUCTION PARAMETERS

Examinations of the beta, photon (X-ray and gamma ray), and neutron radiation types, energies, and geometries of exposure in the workplace, and the characteristics of the PORTS dosimeter responses, are crucial to the assessment of bias and uncertainty of the original recorded dose in relation to the radiation quantity *Hp(10)*. Earlier dosimetry systems can be compared to current systems to evaluate their performance based on the premise that current systems have more stringent criteria as indicated in DOELAP and NVLAP programs.

Accuracy and precision of the recorded individual worker doses depend on (Fix et al. 1997):

- **Administrative practices** that facilities adopt to calculate and record personnel dose based on technical, administrative, and statutory compliance considerations

- **Dosimetry technology**, which includes the physical capabilities of the dosimetry system, such as the response to different types and energies of radiation, in particular in mixed radiation fields

- **Calibration** of the respective monitoring systems and similarity of the methods of calibration to sources of exposure in the workplace

- **Workplace radiation fields**, which can include mixed types of radiation, variations in exposure geometries, and environmental conditions

An evaluation of the available original recorded doses combined with detailed examinations of workplace radiation fields and dosimeter responses to those fields is the recommended option to provide the best estimate of *Hp(d)* for individual workers.

6.3.1 PORTS Historic Administrative Practices

Monitoring at PORTS included radiation level monitoring with portable and area instrumentation, use of pocket ionization chambers when necessary, establishment of radiation areas and high radiation zones, and criticality monitoring with personnel and area instrumentation along with personnel dosimeters.

PORTS based the frequency of personnel dosimetry on safety policy. When operations began in 1954, personnel received dosimetry based on job assignments. In 1960, personnel dosimetry was assigned to all employees, contractors, and visitors as a picture identification and personnel badge. PORTS analyzed all badges assigned to radiation workers, but not all visitor or contractor badges were analyzed. In 1981, personnel dosimetry changed from film dosimeters to thermoluminescent dosimeters (TLDs) with exchange frequencies varying from monthly to quarterly based on work assignments. As indicated below, a number of frequency and monitoring selection changes have occurred over time at PORTS (GAT 1964).

One administrative tool PORTS used to maintain control of film badge and later thermoluminescent dosimeter (TLD) badges was a color code. Badges for selected departments used the color codes listed in Table 6-1. This was first listed in the film badge procedure dated May 3, 1963 (GAT 1963), and continued until PORTS used outside personal dosimetry vendors (January 1, 1999, for USEC employees and January 1, 1997, for BJC employees for neutron monitoring).

Table 6-2 lists letter prefixes used for badge inserts (GAT 1971); these codes might occur in dosimetry records or databases.

In another administrative method, PORTS places badges that cannot be assigned to personnel, visitors, or contractors in the “bucket” file. In addition, since at least December 29, 1969, PORTS has processed badges at random for employees not among the normally selected employees at a rate of 100 per quarter for spot checks; the Plant records positive results only for readings greater than the

Table 6-1. Film badge meter calendar (GAT 1971).

Insert colors	Quarter
Black – Red	1
Blue – Green	2
Black – Red	3
Blue – Green	4

Table 6-2. Security badge letter prefix[a].

Badge	HP insert	Personnel
A	A	Akron GT&R officials
PM	AO	AEC personnel
SO	SO	Federal Bureau of Investigation personnel
B	B	Nationwide and all food vendors
C	C	OVEC
CC	CO	Construction contractors
F	F	All temporary visitors and employee lost badges, etc.
J	J	Miscellaneous (IBM, General Telephone, etc.)
H	H	Hurst dosimeters (Criticality Dosimeters)
E	E	Equipment and emergency badges

a. No prefix was used for PORTS employees.

limit of detection (LOD) of 30 mrem (GAT 1971). *Selected employees* were determined by department, job category, or job assignments. Anyone likely to exceed 10% of the contemporary regulatory limits probably received a permanent badge. This policy continued from the film dosimetry program though the TLD program until it ended on January 1, 1999.

On occasion, PORTS has assigned extremity dosimeters to workers based on their job categories and potential to exceed 10% of the DOE/NRC regulatory limit. [The LOD is about 30 mrem. This is the same as the minimum detectable level (MDL).] Table 6-3 lists the dosimeters used at PORTS.

6.3.2 PORTS Dosimetry Technology

PORTS maintained onsite personnel dosimetry from 1954 to 1998. The dosimetry section followed operational and technical guidelines, as indicated in the available procedural information. As listed in Table 6-4, which summarizes major events in the PORTS personnel dosimetry program, the program was dynamic. Changes occurred due to changes in dosimetry technology, regulatory guidance, and plant operations.

In the mid-1990s, an internal investigation indicated that PORTS did not keep dosimetry records in accordance with procedures, which resulted in improper assignment of doses. From 1993 to 1995, some employee exposures were recorded as zero for exposed damaged TLDs. The HP group reconstructed these records, which resulted in minor adjustments. All doses were much less than DOE or NRC limits (DOE 2002a, p. 37).

Table 6-3. PORTS dosimeter type, period of use, exchange frequency, LOD, and potential annual missed DE (rem).

Dosimeter	Period of use	Exchange frequency	Laboratory LOD	Maximum annual missed dose[a]
Beta/photon dosimeters				
PORTS film 2-element[b]	9/22/54–7/16/57	Weekly (n=52) {selected groups}	0.03[c]	0.78
	7/17/57–9/30/58	Biweekly (n=26) {selected groups}	0.03[c]	0.39
	10/01/58–4/8/59	Weekly (n=52) {chemical operators and material handlers} Monthly (n=12) {remainder of selected groups}	0.03[c] 0.03[c]	0.78 0.18
	4/9/59–7/31/60	Every 4 wk (n=13) {all selected groups}	0.03[c]	0.195
	8/1/60–7/5/64	Monthly (n=12) {all selected groups} Quarterly (n=4) {all other employees}	0.03[c] 0.03[c]	0.18 0.06
	7/6/64–12/28/69	Quarterly (n=4) {all employees}	0.03[c]	0.06
	12/29/69–12/30/73	Quarterly (n=4) {selected employees}	0.03[c]	0.06
	12/31/73–6/29/75	Quarterly (n=4) {selected employees} Semiannual (n=2) {unselected employees}	0.03[c] 0.03[c]	0.06 0.06
	6/30/75–12/31/80	Quarterly (n=4) {selected employees} Monthly (n=12) {selected female employees only}	0.03[c] 0.03[c]	0.06 0.18
PORTS Harshaw 2276 4-element TLD without window	1/1/81–12/31/82	Monthly (n=12) {all monitored} Quarterly (n=4) {all monitored}	0.015[d] 0.015	0.09 0.03
PORTS Harshaw 2276, 8000, 8800 4-element TLD with window	1/1/83–12/31/98 {1/1/93–12/31/96 for BJC employees}	Quarterly (n=4)	0.010[e] (0.04 SDE)	0.04 (0.08 SDE)
ICN TLD 760	1/1/99–present {USEC employees}	Quarterly (n=4)	0.01[f] (0.03 SDE)[f]	0.02 (0.06 SDE)
ORNL Panasonic 8805/8806 4-element TLD with window	1/1/99–present {BJC employees}	Quarterly (n=4)	0.01[h] (.03 SDE) [h]	0.02 (0.06 SDE)
Neutron dosimeters				
PORTS TLD albedo dosimeter {USEC and BJC}	1/1/1992–12/31/94 {unmoderated Cf-252 calibrated}	Quarterly (n=4)	0.02[i]	0.04
	1/1/95–12/31/96 {moderated Cf-252 calibrated}	Quarterly (n=4)	0.02[i]	0.04
ICN TLD 760 {USEC}	1/1/97–present {moderated Cf-252 calibrated}	Quarterly (n=4)	0.01[f]	0.02
Y-12 Panasonic TLND {BJC employees}	1/1/97–12/31/98	Quarterly (n=4)	0.01[g]	0.02
ORNL Panasonic TLND 8806 4-element TLD {BJC employees}	1/1/1999–present	Quarterly (n=4)	0.01[h]	0.02

a. Maximum annual missed dose (NIOSH 2002): For photon/beta missed dose = LOD/2 × n (frequency (p. 18); for neutron missed dose = LOD/2 × n(frequency) (p. 29).
b. Kodak personnel type 2 film with gold sandwiched with cadmium for high-energy gamma, OW with aluminum for low-energy gamma and beta. LOD for SDE and DDE are the same (0.03 rem), as the reporting level.
c. GAT (1963), 30 mrem reporting level for gamma and beta.
d. (Wagner 2003).
e. Bassett (1986a, p. 3).
f. ICN (2003).
g. Souleyrette (2003).
h. McMahon (2003).
i. Cardarelli (1997).

Table 6-4. PORTS historical dosimetry events[a].

Date	Description
9/22/1954	PORTS Film 2-element system started.
8/1/1960	Film badge combined with security badge (picture).
7/6/1964–12/28/1969	All employees monitored.
1/1/1981	Harshaw 2276 TLD system/4-element enacted (without window), Sr-90 calibration; depleted uranium beta, radium gamma (1985). In addition, company photo was placed over beta chip during 1981 preventing beta (skin) determinations for 1981.
1/1/1983	Window added to TLD badge.
4/20/1986	DOELAP accreditation process begins.
1/1/1997	Neutron monitoring begins with ICN TLD 760; beta/gamma monitoring continues.
1/1/1997	BJC splits with USEC, utilizes Y-12 Panasonic TLND dosimetry.
1/1/1999	USEC ends PORTS TLD program, utilizes ICN TLD 760 or equivalent.
1/1/1999	BJC utilizes ORNL Panasonic 8805 (beta/gamma)/8806 (neutron).

See Table 6-3 for references. Based on GAT and USEC/BJC procedures.

6.3.2.1 Beta/Photon Dosimeters

6.3.2.1.1 Film Dosimetry, Two-Element, 1954-1980

The film dosimetry program began in 1954. The dosimeter description from the documentation obtained is cryptic. *A Description of Co-Operative Work Assignments in Industrial Hygiene and Health Physics* (Wooldridge 1964) describes the dosimeter as a “film badge with Kodak Type-2 personal monitoring film combined with aluminum, cadmium, and gold filters for beta-gamma, low energy gamma, and high energy gamma radiation. There are also sulfur and gold filters for neutron exposures.” This document indicates a detection range from 30 to 2,000 mrem. None of the documentation indicated any analysis for neutron or extremity exposure during the film badge era (Wooldridge 1964, p. 3).

The film badge had two elements with an open window (OW) and a shield of gold sandwiched between cadmium to admit mostly high-energy gamma. To measure high-energy gamma deep dose equivalent (DDE), the shield should have been at least 1 mm thick (Wooldridge 1964, p. 5).

There is some description of film processing. Kodak Type 2 film had two emulsions. The first was fast and, when developed and monitored for density, would yield results for an exposure range from 30 to 2,000 mrem. The second, which was a fast emulsion that would measure gamma radiation from 5 to 300 R, was used for an extended exposure range (Wooldridge 1964, p. 3).

Series of films, including a control for every 150 film badges, were developed for each batch. The control film was taken from the same emulsion as the series. Controls were used to determine the amount of natural film darkening from the normal wear cycle and to enable differences in emulsions and developing.

A densitometer was used to measure the density of the control badge and then zeroed. The zeroed densitometer was used to read the badges in that series. If the measurement in the OW was less than or equal to zero, the beta, gamma, and combined beta/gamma measurements were recorded as zero. In addition, total exposures less than 30 mR were recorded as zero.

If the shielded portion of the film read zero, the cause of OW darkening was believed to be beta exposure, so gamma exposure was recorded as zero. A calibration graph generated twice a year was used to determine the beta dose.

If the shielded portion of the film read the same as the OW, or within the ratio of about 1.3 (OW to shielded portion) on the graph of the OW reading, all darkening was caused by gamma exposure. Beta exposure was recorded as zero. The densitometer reading from the shielded portion was found on the vertical axis of the graph and the corresponding gamma exposure was derived from the gamma or shielded labeled curve.

If the shielded portion of the film read greater than zero, but the ratio of the OW reading to the shielded portion reading was greater than the ratio on the calibration graph, the following formula was used to determine the darkening due to the gamma exposure:

$$OW - S(R) = beta \tag{6-1}$$

where

OW = OW densitometer reading
S = shielded portion densitometer reading
R = ratio given on the graph

The corresponding beta exposure was derived from the graph and recorded. The gamma exposure was obtained by using the shielded portion densitometer reading and the curve labeled *gamma*. Total beta/gamma exposure was obtained by adding these exposure readings (GAT 1964, 1971, p. 2-6).

PORTS used emergency and equipment badges in potentially high radiation areas to determine the cause of high exposure readings in personnel badges because those badges remained in the same locations and were exposed to the same conditions for known periods.

One way to determine the effectiveness of an external dosimetry system is to compare similar systems used in comparable environments. The International Agency for Research on Cancer (IARC) evaluated a two-element film dosimeter used at the Hanford Site to monitor several photon energies in exposure orientations as a combination of anterior-posterior (AP), rotational, and isotropic geometries (Thierry-Chef et al. 2002). As indicated in Table 6-5, the two-element dosimetry system overestimated the *Hp(10)* for most exposure orientations, especially for lower energies. The listed 118-keV energy is the most applicable energy for PORTS exposure potential. (Table 6-16 in Section 6.3.4 lists radionuclide information.)

Wilson et al. (1990) conducted another study on dosimetry at the Hanford Site. That study used only the AP orientation for low energies for film dosimeters (see Table 6-6). This is the only orientation that PORTS used for film dosimeter calibration. The lower energies of 16 and 59 keV that represent plutonium photons are close to the 13, 30, 53, 63, and 68 keV representative of low-energy uranium, uranium progeny, and contaminant photon energies present at PORTS. There are also many intermediate energy photons, most of which fall in the 30- to 250-keV energy bin (see Table 6-16).

6.3.2.1.2 Thermoluminescent Dosimeters, 1981-present

PORTS Four-Element TLD, January 1, 1981, to December 31, 1998

The TLD program began in 1981 under GAT. Most of the following information is from *Thermoluminescent Dosimeter Response and Calibration* (Bassett 1986a).

Table 6-5. IARC testing results for U.S. beta/photon dosimeters[b,c].

		118 keV		208 keV		662 keV	
Geometry	Phantom	Mean[a]	SD/Mean	Mean[a]	SD/Mean	Mean[a]	SD/Mean
US-2 (Hanford two-element film dosimeter)							
AP	Slab	3.0	2.1	1.3	1	1.0	0.8
AP	Anthropomorphic	3.0	4.2	1.2	1.9	1.0	1.8
Rotational	Anthropomorphic	2.2	2	1.4	3	1.2	3.2
Isotropic	Anthropomorphic	1.5	4.4	1.1	1.6	1.0	2.7
US-8 (Hanford multielement film dosimeter)							
AP	Slab	1.0	1.5	1.0	0.8	0.8	1.7
AP	Anthropomorphic	0.8	9.5	0.9	6	0.8	1.8
Rotational	Anthropomorphic	1.2	1.9	1.2	17	1.1	1.8
Isotropic	Anthropomorphic	1.0	3	1.2	9	1.0	2.3
US-22 (SRS multielement thermoluminescent dosimeter)							
AP	Slab	0.9	4.4	0.9	3.9	0.9	3.5
AP	Anthropomorphic	0.8	3.1	0.9	2.1	0.9	3.9
Rotational	Anthropomorphic	1.1	3.1	1.2	1.5	1.0	4.1
Isotropic	Anthropomorphic	0.9	0.3	1.0	2.5	0.9	1.6

a. Ratio of recorded dose to *Hp(10)*
b. (Thierry-Chef et al. 2002).
c. (ORAUT 2003a or ORAUT 2003b)

Table 6-6. Testing results for Hanford two-element and multielement film dosimeters for energy and angular response.[a,b]

	AP exposure			Rotational exposure		
	Film dosimeters			Film dosimeters		
Beam (energy, keV)	Two-element 1944–1956	Multielement 1957–1971	TLD 1972–present	Two-element 1944–1956	Multielement 1957–1971	TLD 1972–93
16	0.1	0.9	N/A[c]	N/A	N/A	N/A
59	0.5	1.1	N/A	N/A	N/A	N/A
M150 (70)	0.7	0.70	0.95	1.31	1.31	1.77
H150 (120)	1.6	0.64	0.87	3.00	1.20	1.64
^{137}Cs (662)	1.0	1.0	1.0	1.46	1.46	1.46

a. Divide recorded dose by table value to estimate *Hp(10).*
b. Based on Wilson et al (1990).
c. N/A = not applicable

PORTS used the Harshaw Type L card with three TLD-700 chips (^{7}Li) and one TLD-600 chip (^{6}Li). In general, chips 1 and 2 were used for skin dose, chip 3 for deep dose, and chip 4 for lens dose. Although not indicated in the calibration procedure, the TLDs were probably irradiated in the AP geometry. Table 6-7 lists the TLD shielding configurations.

Manufacturer performance specifications for the Harshaw Type L card for response are as follows: 10 mrem ±15% at 90% confidence for ^{60}Co and 40 mrad ±20% at 90% confidence for natural uranium beta particles.

The Panasonic 802D four-element TLD system, similar to that used at the Hanford and Savannah River Site (SRS), was evaluated. The TLD was irradiated by several photon energies in exposure orientations as a combination of AP, rotational, and isotropic geometries. As indicated in Table 6-5, the TLD dosimetry system closely estimated *Hp(10)* for all exposure orientations within ±20% of the

Table 6-7. Harshaw Type L card element and filter description.[a]

Chip position	TLD type	Shield density thickness (mg/cm^2)	Total density thickness (mg/cm^2)
1	TLD-700	Mylar/8, Teflon/8	16
2	TLD-700	Laminated Photo/75, Polyethylene/84, Teflon/8	167
3	TLD-600	Laminated Photo/75, Polyethylene/52, Cadmium/790, Gold/245, Tape and Teflon/15	1,177
4	TLD-700	Laminated Photo/75, Polyethylene/65, Aluminum/281, Tape and Teflon/15	436

a. (Bassett 1986a, p. 3).

expected exposure. The 118-keV listed energy (AP and rotational orientation) is the most applicable energy for PORTS exposure potential. A similar French Harshaw TLD with a plastic filter of 1,000 mg/cm^2 was within ±10% of the U.S. Panasonic TLD. Although the Panasonic system might have different filters, the overall response was similar (Thierry-Chef et al. 2002, Table 2, p. 106).

ORNL (Y-12) Four-Element TLD, January 1999 to present – BJC Employees
PORTS used TLDs in a few configurations, including this configuration with a Panasonic 8805/8806 four-element card. It was designed to monitor beta, photon, and neutron radiation (McMahon 2003).

Commercial ICN Four-Element TLD System, January 1999 to Present – USEC Employees
PORTS implemented a commercial ICN TLD system on January 1, 1999. This system includes a four-chip beta/photon dosimeter and a separate neutron dosimeter. Technical characteristics are described at the ICN Internet site (www.ICN.com). This dosimetry is NVLAP-accredited.

6.3.2.2 Neutron Dosimeters

PORTS Four-Element TLD, January 1, 1992, to December 31, 1998
PORTS used the Harshaw Type L card with three TLD-700 chips (^{7}Li) and one TLD-600 chip (^{6}Li) (thermal neutron sensitive) with a 235-mg/cm^2 density thickness. The Plant performed neutron monitoring with this dosimeter but only of selected groups. From January 1, 1992, to December 31, 1994, the PORTS thermoluminescent neutron dosimeter (TLND) system was calibrated against an unmoderated ^{252}Cf neutron source. From January 1, 1995, to December 31, 1996, a moderated ^{252}Cf source was used. The change was in response to a perceived over-response of area neutron dosimeters.

ORNL (Y-12) Four-Element TLD, January 1999 to present – BJC Employees
PORTS used TLDs in a few configurations. This configuration has a Panasonic 8805/8806 four-element card. It was designed to monitor beta, photon, and neutron radiation (McMahon 2003).

Commercial ICN Four-Element TLD System, January 1999 to Present – USEC Employees
PORTS implemented a commercial ICN TLD system on January 1, 1999. This system includes a four-chip beta/photon dosimeter and a separate neutron dosimeter. Technical characteristics are described at the ICN Internet site (www.ICN.com). This dosimetry is NVLAP-accredited.

Despite changes in neutron dosimetry systems at PORTS, neutron sensitivities have remained essentially the same.

6.3.2.2.1 PORTS UF_6 Neutron Surveys

Neutron dosimetry was not used at PORTS until January 1, 1992. In 1996, at the request of several union representatives, NIOSH prepared a hazard evaluation report (Cardarelli 1997). This report referred to a survey of neutron radiation levels around 5-in. cylinders of highly enriched uranium (HEU) in storage in 1985. This survey resulted in measurements of 3 mrem/hr at the surface and 0.5 mrem/hr at 1 m. Lower enrichment 10-ton storage cylinders produced radiation levels of 0.5 mrem/hr neutron dose equivalent (DE) at the surface. Based on this information, a worker who spent about 3 hr/wk exposed to 0.5 mrem/hr would receive 75 mrem of neutron dose per year. Because this is 1.5% of the regulatory limit of 5,000 mrem/yr, the HP group deemed neutron personnel monitoring to be unnecessary (Cardarelli 1997). [Cylinder lot surveys measured an average gamma DE rate of about 22 mrem/hr at the surface and 3 to 12 mrem/hr at 1 m (Author unknown 1998a, p. 1; 1998b, p. 8)].

Actual personnel exposures monitored from November 1996 to February 1997 indicated no measurable neutron dose above the 20-mrem/quarter LOD or MDL for the dosimeter used (Table 6-8). This indicates possibly 80 mrem of neutron dose missed per year, a value close to that estimated by surveys conducted about 10 yr earlier, as mentioned above.

Table 6-8. Personal neutron dosimetry results November 1996 to February 1997[a].

Job title	Department	Buildings	Comments	Dose[b] (mrem)
Process operator	720, 730, 740	X-333, X-330, X-326	Cold recovery, tails, product, low-assay withdrawal, unit operator, extended range product station	<=20 mrem
Security guard	151, 152	X-326, X-705, X-345	Product withdrawal, rotation P-12	<=20 mrem
Chemical operator	721, 771, 791	X-344, X-345, X-744G, X-326	Recovery, cylinder lots, small parts, tunnel	<=20 mrem
Health physics technician	300	X-342, X-343, X-344, X-705	HP coverage	<=20 mrem
Laborer	147	Cylinder yards	Paint and scrap in yards (20 hr/wk)	<=20 mrem
Uranium material handler	791	X-344, X-745C, X-745E, X-744G, X-326	Cylinder lots, L-cage, warehouse storage, autoclave, shipping and receiving, vault	<=20 mrem

a. Cardarelli (1997).
b. LOD or MDL = 20 mrem.

PORTS determined from a study of dosimeters in the eight work areas that neutron dose would be 12.5% of the photon DE (Cardarelli 1997, p. 8). A Hanford study yielded a 26% average neutron-to-photon dose ratio (reactor average; Fix, Wilson, and Baumgartner 1996). SRS experience indicates a somewhat lower ratio of 10% neutron-to-photon dose (reactor average; ORAUT 2003a, Section 5.3.4.2.3.2). From the *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities* (DOE 2000, p. 2-19), a 0.01- to 0.2-mrem/hr neutron DE rate for natural to 5% enriched cylinders in cold storage is likely. A neutron DE rate of 2 to 5 mrem/hr at contact and 1 to 2 mrem/hr at 3 ft is likely for 97% enriched cylinders. Table 6-9 lists neutron dosimetry results for the work areas.

Additional neutron radiation level surveys by PORTS health physics are presented in Tables 6-10 and 6-11. One survey was conducted in 1986 and was used to estimate the potential exposure to workers who transported highly enriched uranium cylinders (Bassett 1986b). It was assumed that each worker would take 3 hr/wk, for 50 wk in a year to transport cylinders (Bassett 1986c). The 0.5 mrem/hr DE

Table 6-9. Area neutron dosimetry results November 1996 to February 1997[a].

Buildings	Locations/comments	Dose (mrem)
X-326	Extended range product {Dynamics St. 2} Product withdrawal {T-57-8-3 Bed 3}	80 mrem (3 mo) 60 mrem (3 mo)
X-330	Tails Deposit;dd-4;29AB-1 G-33;29-3-7-7 {randomly selected} Low-assay withdrawal (LAW)	<= 20 mrem[b] < =20 mrem[b] <=20 mrem[b] <=20 mrem[b]
X-343	Typical movement (300 cylinders/mo)	<= 20 mrem[b]
X-345	Behind Phantom 5 on wall	<= 20 mrem[b]
X-705	Portal DOE Lot 11,200;2-3%, row 22-23 Sec 44	<= 20 mrem[b] 420 mrem[b] 210 mrem[b] 710 mrem[b]
X-745G	DOE Lot; 11,200;2-3%, row 20-21 Sec 3, heel	210 mrem[b] 510 mrem[b] 320 mrem[b]

a. Cardarelli (1997, pp. 21–22).
b. Measurement period of 1 mo.

Table 6-10. Neutron and gamma results for six full and four empty 5-in. product cylinders stored in the X-326 Product Withdrawal Vault, February 14, 1986.[a]

Survey Point and Radiation Type	Empty	Full
Surface/Neutron	<= 0.5 mrem/hr	<= 3.0 mrem/hr
Surface/Gamma	<= 1.0 mR/hr	<= 2.4 mR/hr
Surface/Neutron and Gamma	<= 1.5 mrem/hr	<= 5.4 mrem/hr
1 Meter/Neutron and Gamma	<= 0.2 mrem/hr	<= 0.5 mrem/hr

a. Bassett (1986b).

Table 6-11. Neutron survey results for full 5-in 2.5- and 10-ton product cylinders stored in various locations, March 26, 1987.[a]

Location	Description	Fast Neutrons (mrem/hr)	Slow Neutrons (mrem/hr)	Comments
X-342 North Cylinder Lot	Full 10-ton cylinder	0.47	0.34	Readings taken between 2 full cylinders, 1 ft from each
X-342 North Cylinder Lot	Full 2.5-ton cylinder	11.3	7.8	Readings taken approximately 1 ft from each cylinder
X-326	Full 5-in. cylinder HEU	-	2.5–3.0	Readings taken at contact with cylinder
X-345 Center Vault	Full 5-in. cylinder HEU in overpack	-	1.0–1.5	Readings taken at contact with overpack

a. AHJ (1987).

rate measurement from the one meter reading of six full 5-in. cylinders was used in the calculation. This resulted in a 75-mrem/yr DE:

DE in mrem = 0.5 mrem/hr x 50 wk/yr x 3 hr/wk = 75 mrem/yr.

From sharing of the work, it was expected that the average dose was more likely 25 mrem/yr.

A neutron survey result was also conducted March 26, 1987, of 5-in., 2.5- and 10-ton full cylinders. The measurements distinguished fast from thermal neutrons as listed in Table 6-11.

Guards could have been more routinely exposed workers than previously assessed by PORTS health physics. Three estimates of DE (without the applied correction as discussed in the next section) are presented below.

Using the 11.3mrem/hr fast and 7.8-mrem/hr thermal neutrons at 1 ft from full 2.5-ton cylinders from Table 6-11 as a maximizing approach, one would obtain the following DE estimate:

19.1 mrem/hr x 50 wk/yr x 3 hr/wk = 2,865 mrem/yr

Using 3.0-mrem/hr full surface neutron values from Table 6-10 as a best estimate, one would obtain the following DE estimate:

3.0 mrem/hr x 50 wk/yr x 3 hr/wk = 450 mrem/yr

Using 0.47 mrem/hr fast and 0.34 mrem/hr thermal neutrons at 1 ft from full 10-ton cylinders from Table 6-11 full as an underestimate, one would obtain the following DE estimate:

0.81 mrem/hr x 50 wk/yr x 3 hr/wk = 121.5 mrem/yr

These can be considered potential neutron doses to guards because it has been suggested that guards were often told to guard the cylinders for entire shifts.

Note the above calculations are still likely to be conservative in all three cases based on:

1. Transport of cylinders per year estimate is probably high.
2. Cylinders are assumed full.
3. Surface readings or 1-ft readings were used.

As a comparison, if photon doses from ORAUT-OTIB-0040 (ORAUT 2005a) are used:

- 1954, 95th percentile: 1,736 mrem x 0.2 (neutron/photon ratio on measured personnel dose)
  = 347.2 mrem routine or high-level exposure.

- 1954, 50th percentile: 780 mrem x 0.2 (neutron/photon ratio on measured personnel dose)
  = 156 mrem average or intermittent low-level exposure.

If photon dose is used from this TBD:

- 1954: 750 mrem (adjusted by 0.5 or LOD/2 method) x 0.2 (neutron/photon ratio on measured personnel dose)
  = 150 mrem

As a comparison using the PORTS Occupational Environmental Dose TBD (ORAUT 2004b):

- For cylinder yard workers, a DDE of 1,056 mrem for both neutron and (beta/gamma) for 2001
- 267 mrem DDE neutron and beta/gamma or less before 2001, a DDE of 178 mrem/yr neutron and beta/gamma ambient radiation level to be assigned to unmonitored personnel

6.3.2.2.2 Quality Factors for Neutrons

The quality factors (QFs) used historically for neutrons have changed significantly. In current regulations, QFs that are used to convert radiation dose (mrad) to DE (mrem) are based on International Commission on Radiological Protection (ICRP) Publication 38 (ICRP 1983). The most current QFs from Publication 60 (ICRP 1991) are about 2 times higher than the ICRP (1983) values. Because a QF of 10 was used for the referenced radiation measurements, the personnel dose results (Table 6-8), and PORTS personnel dosimetry, an adjustment to ICRP (1991) of at most a factor of 2 times higher would be necessary.

Average neutron energy is about <1 MeV (510 keV for 2% ^{235}U, 770 keV for 5% ^{235}U, and 860 keV for 97% ^{235}U) (Cardarelli 1997, p. 9). QF equals 10 for ICRP (1983), or about 20 for the ICRP (1991) revision. The average neutrons from depleted and natural uranium cylinders ranged from 210 to 360 keV (Cardarelli 1997, p. 9). Unmoderated and deuterium (water) ^{252}Cf neutrons created were between 1,306 and 1,403 keV. This means the dose as monitored at PORTS since 1992 (for calibration facility personnel) and 1994 (others included) was overestimated and therefore favorable to claimants.

From November 1996 to February 1997, specific process areas and personnel were monitored using the Landauer/Neutrak ER badge (Landauer 2003). This badge combines an albedo dosimeter with a Columbia Resin No. 39 chip, which monitors neutrons from >30 to 50 keV up to 35 MeV. The TLD albedo chip monitors neutrons in the energy range of 0.5 to 100 keV. Thermal neutrons were not monitored due to the assumption that most neutron exposures at PORTS are from intermediate to fast neutrons (Cardarelli 1997).

6.3.3 <u>Calibration</u>

Potential error in recorded dose is dependent on dosimetry technology response characteristics to each radiation type, energy, and geometry; the methodology used to calibrate the dosimetry system; and the similarity between the radiation fields used for calibration and those in the workplace.

6.3.3.1 Film Badges Beta/Photon

For PORTS film dosimeter calibration, eight badges were exposed to a 22.5-mCi ^{226}Ra source in an AP geometry at various distances free in air to yield zero (control), 30, 60, 100, 200, 500, 750, and 1,000 mR of gamma exposure (GAT 1964). To correct for this free-in-air exposure [a conversion from roentgen (87.6 ergs/g) to rad (95 ergs/g)], ratios of the differences of rad to roentgen were taken, as follows:

$$(87.6 - 95)/87.6 = 8.45\%$$

In other words, if the badges had been calibrated with an anthropomorphic phantom, they would have received 95 ergs/g (plus scatter) rather than the 87 ergs/g they actually received. Assuming a QF of 1 for rad-to-rem conversion, the correction factor due to not using a phantom would be –8.45% relative to using a phantom.

Because ^{226}Ra or ^{137}Cs was used for calibration, an under-response of 8% or 3.9%, respectively, would result due to higher energy calibration energies, in comparison to the actual lower exposure energies encountered in the workplace (ORAUT 2003a). A correction of 16.5% for film dosimetry (1954 to 1980), 12.5% for TLD dosimetry (1981 to 1986), and 4% for TLD dosimetry (1987 to present) is recommended.

Figure 6-1 shows a comparison of a two-element film dosimetry system to the other film dosimetry systems. Wilson et al. (1990) measured the AP photon energy response of the Hanford systems. As indicated in Figure 6-1, the dosimeter OW response shows a significant over-response to lower energy photon radiation. Most of the photon energy spectrum and the DE at PORTS are from the 30- to 250-keV range where there was an over-response from the two-element film-shielded portion of the dosimeter (silver shield). This indicates that recorded results for the PORTS two-element film would overall be favorable to claimants and no corrections are needed for the response of the dosimetry to the radiation work environment. The under-response of the two-element film shielded portion to low energies (less than 50 keV) should be of little consequence because the vast majority of the photon energy spectrum and the DE are from photons greater than 50 keV. This is because most radiation work environments involve shielded or self-shielded uranium sources that allow little exposure from less-than 30-keV photons. In the case of open systems such as processing, recovery, and maintenance, the exposure to low-energy photons is more probable. In these situations, because the low-energy photons are monitored conservatively by the OW, it might be favorable to the claimant to equate the shallow dose to the deep dose if the deep dose was less than the shallow dose in the claimant's records.

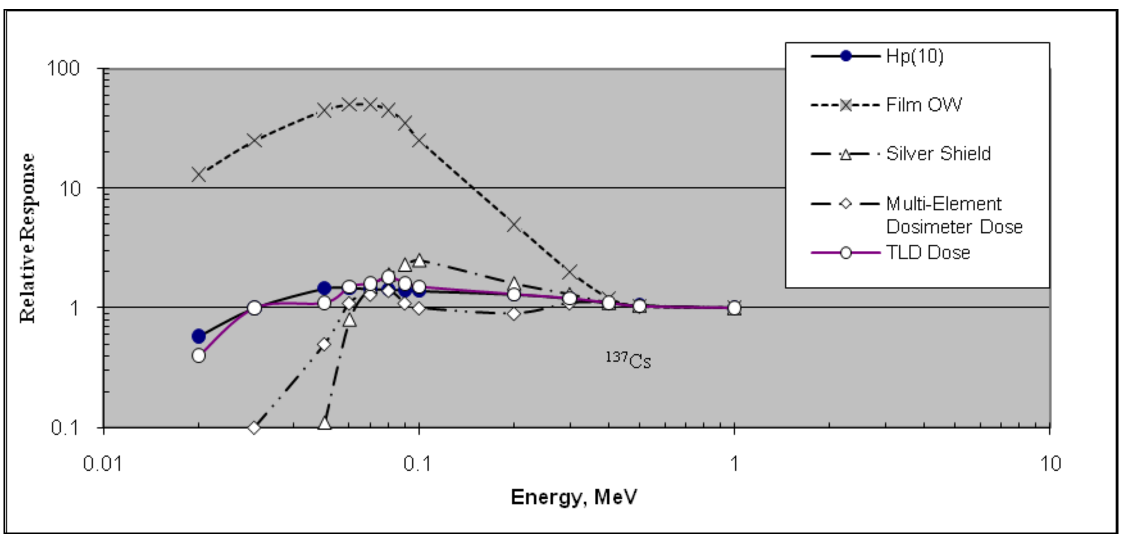

Figure 6-1. Measured Hanford two-element dosimeter photon response characteristics (Wilson et al. 1990).

The nonpenetrating (OW) and penetrating (gold-cadmium filter) film responses were used to estimate skin dose from beta and photon radiation.

For beta calibration, six badges were exposed to depleted uranium slugs (240 mR/hr) in an AP geometry at various times to yield zero, 60, 120, 240, 480, and 960 mR of beta radiation exposure.

The slugs were placed directly on the badges (GAT 1964). This meant that the calibration closely matched the work environment and no corrections for beta response are necessary.

6.3.3.2 TLD Badges Beta/Gamma

TLD badges were calibrated in free air with a ^{137}Cs source, National Bureau of Standards (NBS)-filtered X-rays, and depleted uranium using slab geometry. Because different energies were used for calibration, exposure-to-DE conversion factors were used in the calibration algorithms, as listed in Table 6-12.

Table 6-12. Roentgen-to-rem conversion factors used for TLD dosimeter calibration[a].

NBS-filtered X-rays	Conversion factors (rem/R) shallow (0.07 mm)	Conversion factors (rem/R) deep(10 mm)
M30 (20 keV)	1.08	0.45
S60 (36 keV)	1.15	1.07
M150 (70 keV)	1.41	1.47
H150 (120 keV)	1.41	1.41
Cs-137 662	(b)	1.03

a. Bassett (1986a, p. 5).
b. This value, which depends on source geometry, should be measured.

Controls were placed at the plant entrance portals: Drive gate, X-100 lobby, X-108B gate, H-Lot gate, and C-Lot gate. About 16 combinations of sources were used, such as ^{137}Cs and 20-, 36-, 70-, and 120-keV X-rays and U-slab; U-slab and 20-, 36-, 70-, 120-keV X-rays are representative of possible mixed fields encountered at PORTS.

The general form of the DDE equations used in the TLD calibration system was:

$$DDE = D_o + D_1(CR_3) + D_2(CR_4) \tag{6-2}$$

$$SDE = S_0 + S_1(CR_1) + S_2(CR_2) + S_3(CR_3) + S_4(CR_4) \tag{6-3}$$

where D_i and S_i are the deep and shallow multiple linear regression coefficients for their respective chip positions and CR_i is the nanocoulomb thermoluminescent response for chips 1 to 4.

TLD-700 chips 1 and 2 (16 and 167 mg/cm^2, respectively) are not used for DDE determination. The basis for dose algorithm calculations is chip proportionalities, which enables discrimination of radiation energy types to unknown fields (Basset 1986a, pp. 22–23).

Table 6-13 summarizes laboratory sources of uncertainty parameters in beta and photon calibrations. If uncertainty or bias is positive, dose reconstructors should make no change. In the case of negative bias, dose reconstructors should make the appropriate corrections using the values in Table 6-13. Uncertainty was estimated using information from PORTS procedures or performance information from similar dosimetry systems.

6.3.3.3 TLD Badges Albedo Neutron

In 1992, some workers in the PORTS Radiation Calibration Facility and the Applied Nuclear Technology Department were placed on a routine neutron dose monitoring program. In October

1994, only workers entering a Controlled or Restricted Area were monitored. From 1992 to 1994, the albedo dosimeter was calibrated with an unmoderated ^{252}Cf source, which resulted in higher doses than expected. In 1995, the calibration procedure was modified to use a moderated ^{252}Cf source that would result in lower doses than expected due to the fast-neutron environment at PORTS (210 to 860 keV) (Cardarelli 1997).

Table 6-13. Laboratory sources of uncertainty for beta/photon dosimeter calibration parameters.

Parameter	Historical description	Uncertainty[a]	Comment
In-air calibration	Prior to 1/1/1987, calibrations were performed free in air. In 1997 for BJC and 1999 for USEC, use of commercial or outside vendors for dosimetry began. Phantom for calibration introducing increase of response from backscatter occurred relative to dosimetry used before 1997 and 1999.	±10%	Before 1987, recorded dose of record **too low** (see Section 6.3.3.1). After 1997, recorded dose of record **too high.** Backscatter radiation from worker's body is highly dependent on dosimeter design (ORAUT 2003b, Table 6-2).
Radiation quantity	Before 1981, PORTS used Cs-137 and Ra-226 for beta and photon beam calibration.	±5%	For higher energy Cs-137 and Ra-226 for beta and photon beam calibration, this caused about 4% or 8% **under-response** in recorded dose (Basset 1986a).
Tissue depth of dose	Historically, PORTS used specified depth of 10 mm to estimate deep dose.	±5%	Numerical effect of this for photon radiation is comparatively low. PORTS dosimeter designs had filtration density thickness of about 1,000 mg/cm^2 that would relate closely to 1-cm depth in tissue.
Angular response	PORTS dosimeter system is calibrated using AP laboratory irradiations.	~ –25% 100 keV	Recorded dose of record likely **too low** because dosimeter response is usually lower at non-AP angles. Effect is highly dependent on radiation type and energy (Tables 6-15 and 6-18)
Environmental effects	Workplace heat, humidity such as dosimeter fading impact dosimeter results.	±5%	Fading should have been less than 2% for TLDs. Heat effect should have been much less than 1%. Recorded dose due to these effects likely **too low**.

a. Uncertainty estimate in recorded dose compared to *Hp(10)* based on judgment from Hanford dosimeter laboratory studies.

The use of the ICN TLND, the Y-12 TLND, and the ORNL TLND from 1997 to the present should not present much variation because all are either NVLAP- or DOELAP-certified and all use phantoms for calibration and similar geometries. ICN is a large commercial vendor with references to its dosimetry program available at its Internet site (www.ICN.com). Table 6-14 lists possible dosimeter lab parameter uncertainty.

6.3.4 Workplace Radiation Fields

The PORTS radiation work environment consists of a variety of combinations of complex beta, gamma (photon), X-ray, neutron, and *bremsstrahlung* (X-ray) radiations. The worker's job, location, and occurrences of incidents affect the external exposure. Worker location in relation to radiation sources during the course of the work history is one of the most challenging parameters to establish, as is estimation of actual radiation source levels. Assumptions related to these parameter definitions need to be based on available worker information and the work environment. This section describes the basis of the assumptions.

For the PORTS areas listed in Table 6-15, assumptions favorable to the claimants were used. As suggested in NIOSH (2002), the 30- to 250-keV energy range for photons is used if a specific gamma

Table 6-14. Laboratory sources of uncertainty for neutron dosimeter calibration parameters.

Parameter	Historical description	Uncertainty[a]	Comment
Source energy spectra	PORTS used unmoderated and moderated Cf-252 source for calibration. Unmoderated source led to increased bias.	±100%	Delivered dose used in calibrating neutron dosimeters is **uncertain.** Uncertainty listed is favorable to claimants.
Radiation quantity	QF or spectrum used for PORTS by outside vendors.	±50%	Recorded dose of record likely **too low** because dosimeter values based upon pre ICRP 60 quality factors.
Angular response	PORTS TLND dosimeters calibrated using AP laboratory irradiations.	±50%	Recorded dose of record likely **too high** because dosimeter response is often higher at angles other than AP. Effect is highly dependent on energy.
Environmental stability	TLD systems are subject to signal fade with time, heat, humidity, light, etc.	±50%	Recorded dose of record likely **too low** because of fading; however, this effect depends strongly on such routine dosimetry practices as when calibration dosimeters were irradiated.

a. Uncertainty in recorded dose compared to *Hp(10)* based on judgment from laboratory studies based on ORAUT (2003b, Table 6-3).

Table 6-15. PORTS workplace beta/photon dosimeter response.

Process/ buildings	Description	Operations Begin	Operations End	Radiation type	Energy selection	%
X-326	High assay withdrawal station, High Assay Storage Area{HEU}	1954	2001	Beta Photon	> 15 keV 30 – 250 keV	100 100
	Purge system (top and side) {EU – HEU}	1954	2003	Beta Photon	> 15 keV 30 – 250 keV	100 100
X – 330	Tails withdrawal station (depleted uranium)	1954	2001	Beta[a] Photon	> 15 keV 30 – 250 keV >250 keV[b]	100 85 15
X-333	Reactor grade LAW (EU) [when used as backup for tails withdrawal; see X-330]	1954	2001	Beta Photon	> 15 keV 30 – 250 keV	100 100
X-342, 342A, & 343	Fluorine generation (342) and fixed feed (EU)	1954	2001	Beta Photon	> 15 keV 30 – 250 keV	100 100
X-344	Feed manufacturing plant	1958 1962	1962 2003	Beta Photon	> 15 keV 30 – 250 keV	100 100
X-345	Special Nuclear Material storage and HASA	1978	2003	Beta Photon	> 15 keV 30 – 250 keV	100 100
X-700	Maintenance	1954	2003	Beta Photon	> 15 keV 30 – 250 keV	100 100
X-705[c,d] & 705E[c]	Decontamination, cleaning and recovery, oxide conversion plant	1954	2003	Beta Photon	> 15 keV 30 – 250 keV	100 100
X-710[e]	Analytical labs	1954	2003	Beta Photon	> 15 keV 30 – 250 keV >250 keV	100 75 25
X-720[d]	Compressor shop	1954	2003	Beta Photon	> 15 keV 30 – 250 keV > 250 keV	100 75 25

X-744	Smelter for aluminum recovery	1961	1983	Beta Photon	> 15 keV 30 – 250 keV	100 100

a. Technetium-99 is expected, especially from 1977 – 1980s.
b. Uranium-238 in equilibrium with daughters ^{234m}Pa and ^{234}Th.
c. Expect all RNs including transuranic materials and ^{99}Tc.
d. Beta exposures are more probable with the treatment of wastes and opening of equipment.
e. Calibration sources such as ^{137}Cs and ^{226}Ra and X-ray equipment from 40 to 200 kV have been used in parts of this facility.

^{238}U (^{234m}Pa, and ^{234}Th) have photon energies greater than 250 keV when in equilibrium with ^{238}U, the energy bin of 30 to 250 keV was allocated to 85% of the photon field, and the over 250-keV range was allocated to 15% of the field. Buildings X-710 and X-720 use ^{137}Cs and ^{226}Ra calibration sources that yield photons of energy greater than 250 keV. Therefore, 25% of the photon field was allocated to these areas. This is favorable to claimants because exposure from or usage in areas other than calibration facilities or structural analysis areas is limited. *Bremsstrahlung* could be present in the ambient work area but would be of a lower level than the ambient gamma radiation.

The energy bin selection for beta radiation for every area at PORTS is greater than 15 keV. Of special note for beta areas is Building X-330 or any area of tails with potential for exposure during processing. Technetium-99 exposure could have occurred during cascade maintenance, removal of the magnesium traps, and waste processing.

PORTS processed recycled uranium (RU) that contained trace amounts of radioactive impurities not present in natural uranium feed material. Because these impurities were present in such minute concentrations, their radiological impact was usually negligible. However, some routine chemical processes can concentrate the impurities. The most significant impurity found in RU is the pure beta emitter ^{99}Tc, which tends to deposit in enrichment equipment and to “pocket” in the higher sections of the diffusion cascade (DOE 2002a). In addition, ^{99}Tc was concentrated for recovery and removal. The relatively low-energy beta particles (maximum 294 keV) from ^{99}Tc pose minimal external exposure potential because of their limited range. Neither film nor TLD efficiently detect them, particularly in the presence of uranium. Clothing and gloves provide adequate shielding. Skin contamination is the most credible scenario in which a significant shallow dose could occur from ^{99}Tc. Table 6-16 lists the principal locations and periods for which recovery operations at PORTS are believed to have occurred (DOE 2002b).

Table 6-16. Major facilities at PORTS where ^{99}Tc could have accumulated.

Building No.	Name	Dates of operation	Activities
X-326	Gaseous Diffusion Process Bldg.	1954–1991	High Assay Product
X-330	Gaseous Diffusion Process Bldg.	1954–2001	Intermediate process & tails withdrawal
X-333	Gaseous Diffusion Process Bldg.	1954–2001	Initial enrichment & reactor product
X-344	UF_6 Feed Manufacturing Plant	1958–1962	Conversion of UF_4 to UF_6
X-345	Special Nuclear Material Storage	1978–2003	HEU storage
X-700	Maintenance Building	1954–2003	Large component repairs
X-705	Decontamination & Cleaning Bldg.	1954–2003	Equipment wash & uranium recovery
X-705E	Oxide Conversion Plant	1957–1978	Conversion of U_3O_8 to UF_6
X-720	Compressor Shop	1954–2003	Disassembly & repair of compressors
X-744G	Smelter & Aluminum Recovery	1954–1978	Recover aluminum from scrap

Source: DOE (2002b), p. 16.

Table 6-17 summarizes the RU reactor returns [also see ORAUT (2004a, Table 2.4.2-1). RU comprised about 1,094 MTU of the 330,000 MTU fed to the PORTS cascade (BJC 2000, p. 22).

Table 6-17. Reactor returns fed to cascade.

Fiscal year	Amount fed (MTU)	Enrichment (% U-235)	Source	Remarks
1955	105.8	0.64–0.68	Paducah	May–Sept. 1955
1956	54.5	0.64–0.68	Paducah	
1956	293.4	0.64–0.68	Oak Ridge	
1957	6.2	0.64–0.68	Paducah	
1958	64.2	0.64–0.68	Paducah	
1970	168.1	0.64–0.68	Paducah	Oct. & Nov. 1969
1974	398.8	0.64–0.68	Paducah	Jan. 1974
1974–1978	1.86	2–50	PORTS Oxide Conversion	
1968–1977	0.15	78–80	Division of International Affairs	
1977–1998	0.15	78–97	Babcock & Wilcox	
1969–1993	0.07	78	AEC Office of Safeguards & Materials Management	
1997–1998	1.10	56–82	France	
1997–1998	0.33	80	NUMEC	
TOTAL	1,094.66			

Source: BJC (2000, Table 2.2.2.5-1).

6.3.4.1 Workplace Beta/Photon Dosimeter Response

All PORTS radiological work areas involve beta/photon radiation covering a wide range of energies, which are characterized by radionuclides in the work environment. Table 6-18 lists the properties of the radionuclides and the machine sources at PORTS. Information on the actual radiation environment can be reviewed in the Site Description TBD (ORAUT 2004a).

Table 6-19 summarizes the common beta/photon personnel dosimeter parameters important to *Hp(10)* performance in the workplace. PORTS dosimetry has made reasonable measurements of workplace radiation fields. The two-element dosimeter would have over-responded to the average photon field of about 100 keV by about 50% or more. No corrections to over-response are recommended. As of 1987, PORTS estimated shallow dose by adding the gamma (deep) and beta (shallow) doses together. This policy would be an overestimate of shallow dose. Extremity exposures were monitored infrequently. BJC has not monitored for extremity exposure.

6.3.4.2 PORTS Workplace Neutron Response

In general, PORTS radiation workers were exposed to ambient neutron radiation produced primarily from three reactions: Spontaneous fission of ^{235}U, subcritical fission of ^{235}U, an alpha reaction on fluorine from the decay of uranium [$^{19}F(\alpha,n)$ ^{22}Na] and an alpha reaction on oxygen from the decay of uranium [$^{18}O(\alpha,n)^{21}Ne$]. The most likely places for neutron exposures are in storage areas or cylinder yards (X-345, Cylinder Lot 745), feed and withdrawal process areas (X-326, X-330, and X-333), calibration and laboratory assay areas where ^{252}Cf sources were used, and areas where uranium deposits formed in the cascades.

The assay laboratory in Building 710 and the storage vault for HEU in Building 345 have been characterized for neutron energy (Soldat and Tanner 1992). The ^{252}Cf calibration facility in Building 710 energy characterizations was based on measurements of a similar facility.

Table 6-18. Properties of radionuclides that might be found at uranium facilities.[a]

Nuclide	Half-life	Energies (MeV) and abundances of major radiations		
		Alpha	Beta	Gamma
Primary uranium isotopes				
U-238	4.51×10^9 yr	4.15 (25%)	--	0.013 (8.8%)
		4.20 (75%)		
U-235	7.1×10^8 yr	4.37 (18%)	--	0.144 (11%)
		4.40 (57%)		0.185 (54%)
				0.013 (31%)
U-234	2.47×10^5 yr	4.58 (8%)	--	0.204 (5%)
		4.72 (28%)		0.053 (0.20%)
		4.77 (72%)		0.013 (10%)
U-236	2.34×10^7 yr	4.49 (76%)		0.013 (10%)
		4.44 (24%)		
U-232	72 yr	5.26 (31%)		0.013 (12%)
		5.32 (69%)		
Decay Products				
Th-234 (U-238 parent)	24.1 d	--	0.103 (21%)	0.063 (3.5%)
			0.193 (79%)	0.093 (4%)
Pa-234m (U-238 parent)	1.17 m		2.29 (98%)	0.765 (0.30%)
				1.001 (0.60%)
Th-231 (U-235 parent)	25.5 hr		0.140 (45%)	0.026 (2%)
			0.220 (15%)	0.084 (10%)
			0.305 (40%)	
Th-230 (U-234 parent)	77,000 yr	4.62 (23.4%)		0.068 (0.4%)
		4.69 (76.3%)		0.012 (8.4%)
Th-228 (U-232 parent)	1.913 yr	5.34 (26.7%)		0.012 (9.6%)
		5.42 (72.7%)		0.084 (1.2%)
Impurities				
Tc-99	2.12×10^5 yr	--	0.294	--
Np-237	2.14×10^6 yr	4.78 (75%)		0.030 (14%)
		4.65 (12%)		0.086 (14%)
				0.145 (1%)
Pu-238	86.4 yr	5.50 (72%)		
		5.46 (28%)		
Pu-239	24.4×10^4 yr	5.16 (73%)		
		5.14 (15%)		
		5.10 (12%)		
Pu-240	6.6×10^3 yr	5.17 (76%)		
		5.12 (24%)		
Pu-241	13.2 yr		0.021	
Cs-137/calibration sources	30.6 yr		0.514 (95%)	0.662 (85%)
Ra-226	1,600 yr	4.60 (6%)		0.186 (3.6%)
		4.78 (94%)		
Machine generated X-rays				0.07–0.2

a. In most part from Bassett (1986a).

Table 6-19. Common workplace beta/photon dosimeter *Hp(10)* performance.[a]

Parameter	Description	Uncertainty[c]	Workplace bias[b]
Exposure geometry	PORTS dosimeter system calibrated AP lab irradiations. Workplace exposure geometries are highly variable.	>100 keV: Two–element film dosimeter, ~ +200% Others, ±25% <100 keV: Probably too low	Recorded dose probably **too low** because dosimeter response can be much lower at rotational and lateral angles. This effect is highly energy dependent. Highest doses are probably associated with AP geometry where work is performed close to radiation source. Effect is highly dependent on radiation energy.
Energy response	PORTS film dosimeter response to photon radiation <100 keV too low and >100 keV was too high.	Depends on actual field 100 keV ~ –25%	Bias in recorded dose depends on photon spectrum in workplace, especially for film dosimeter. Reasonable estimate of *Hp(10)* dose is likely (Figure 6-1) (Wilson et al. 1990). Estimate based on 100 keV and Table 6-6.
Mixed fields	PORTS dosimeters responded to both beta and photon radiation. TLD PORTS system was calibrated against number of different possible mixed fields.	Depends on actual field	About 16 types of mixed fields were calibrated under PORTS TLD system. PORTS film system was not calibrated against mixed fields. Reasonable estimate of *Hp(10)* dose is likely.
Missed dose	Doses less than MDL or LOD are recorded as zero dose.	Recorded dose of record probably too low	PORTS recorded doses <MDL for all years. Issue is significant, primarily in earlier years with frequent dosimeter exchange and film dosimeters with higher MDLs.
Environmental effects	Workplace heat, humidity such as dosimeter fading impact dosimeter results.	±5%	Fading should have been less than 2% for TLDs. Heat effect should have been much less than 1%. Recorded dose due to these effects probably **too low**.

a. Judgments based on PORTS dosimeter response characteristics and workplace radiation fields.
b. Recorded dose compared to *Hp(10).*
c. Uncertainty based on recorded dose in comparison to *Hp(10)*, judgment on similar dosimeter laboratory studies, and onsite procedures.

One phenomenon that occurs at gaseous diffusion plants is the formation of *slow cookers*, which are uranium deposits that can accumulate in the cascade. At a fission rate just below critical, a slight increase of neutron production occurs. All of these processes produce fast neutrons with energies usually less than 2 MeV.

6.3.4.2.1 Calibration Laboratory in Building 710

Measurements were made of the bare ^{252}Cf calibration source in the low-scatter room, where it was used with a tissue-equivalent proportional counter (TEPC) at 1 m. On the day of the measurement (March 12, 1992), the source was rated at 27.54 µg, corresponding to a DE rate of 64.9 mrem/hr at 1 m. The TEPC measurements ranged from 48 to 57 mrem/hr (Soldat and Tanner 1992, p. 3.3).

Pacific Northwest Laboratory measured the ORNL bare ^{252}Cf calibration source for a neutron energy spectrum. The results are indicated by the solid lines in Figure 6-2. Table 6-20 lists the dose fractions for the neutron energy groups indicated by the dashed lines in Figure 6-2. The dose fractions for the lower (<10 keV) and intermediate (10 to 100 keV) energy neutron groups were less than 1% of the total dose from the measurements. Therefore, combining the lower and intermediate energy groups into the fast group of 0.1 to 2 MeV is a reasonable simplification of the neutron dose calculation.

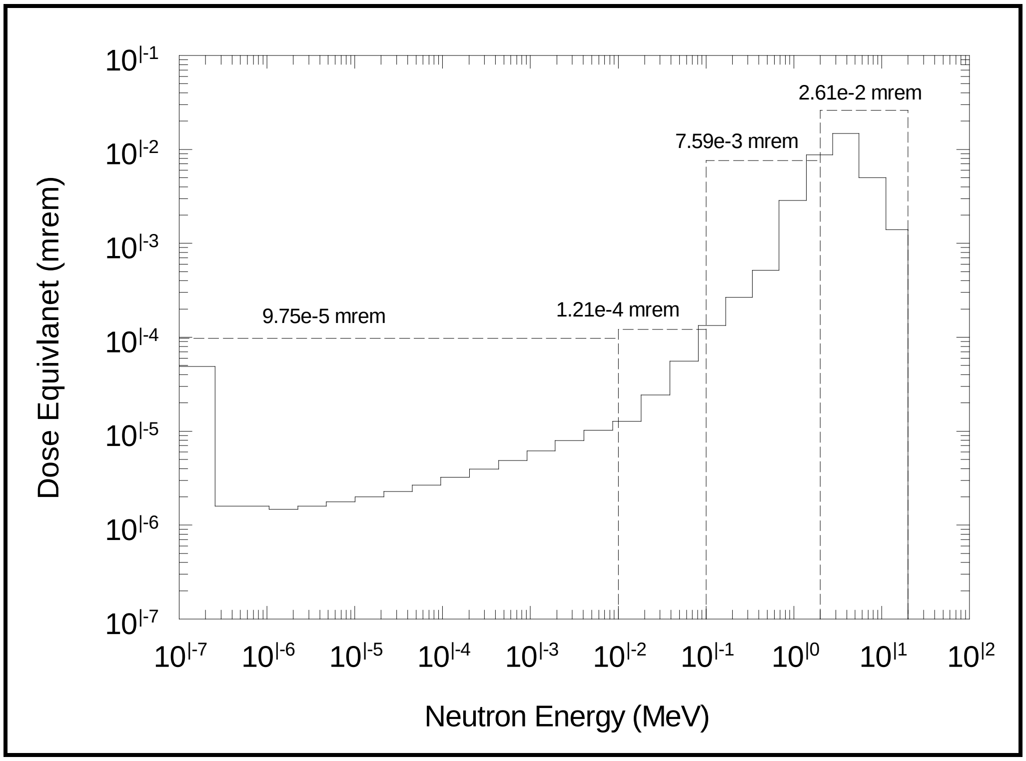

Figure 6-2. Results of neutron spectrum measurements made at 1 m from bare ^{252}Cf fission neutron source (Soldat and Tanner 1992) (See Y-12 TBD, Fig. 6.3.4.2-3).

Table 6-20. Dose fractions for PORTS calibration facility.

Neutron energy group	Near unshielded Cf-252 source
<10 keV	0.003
10–100 keV	0.004
0.1–2 MeV	0.224
2–20 MeV	0.769
Dose fractions favorable to claimants	
0.1–2 MeV	0.23
2–20 MeV	0.77

The use of the ORNL ^{252}Cf calibration source is reasonable because the energy spectra would be the same. However, neutron distribution would differ from calibration facility to facility based on scatter of the neutrons or room rate return. This difference should be minimal because calibration facilities are designed to minimize scatter and maintain consistency. This difference would probably not be more than 20% (NCRP 1991).

6.3.4.2.2 Nondestructive Assay Laboratory in Building 710

The assay laboratory contained a californium shuffler unit and a segmented gamma scanner separated by a concrete wall that extended halfway down the room. Measurements of the ^{252}Cf shuffler unit in its fixed-open position yielded a DE rate of 0.2 mrem/hr and an integrated dose on environmental dosimeters on a phantom of about 1 mrem. The multisphere measurement at the same location as the phantom resulted in an average neutron energy of 0.52 MeV and a DE rate of 0.15 mrem/hr.

The solid lines in Figure 6-3 show the calculated energy spectrum from the multisphere detectors (Bonner spheres). Table 6-21 lists the dose fractions for the neutron energy groups (indicated by the dashed lines in Figure 6-3). The dose fractions for the lower (<10 keV) and intermediate (10 to 100 keV) energy neutron groups were about 20% of the total dose from the measurements. Combining the lower and intermediate energy groups into the 0.1- to 2-MeV group is a reasonable simplification of the neutron dose calculation.

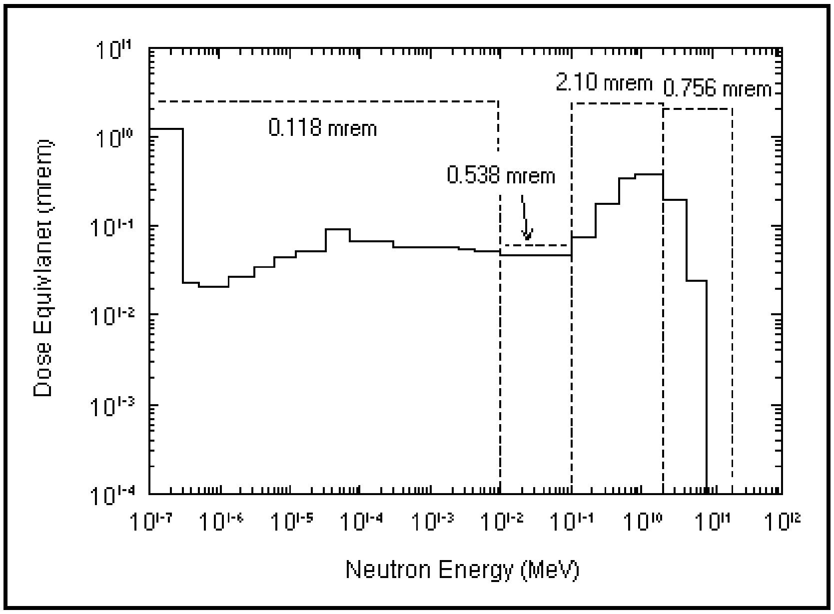

Figure 6-3. Results of neutron spectrum measurements made in the front of shuffler unit (Soldat and Tanner 1992).

Table 6-21. Dose fractions for PORTS nondestructive assay laboratory facility in Building 710.

Neutron energy group	Near unshielded Cf-252 source
<10 keV	0.034
10–100 keV	0.153
0.1–2 MeV	0.598
2–20 MeV	0.215
Dose fractions favorable to claimants	
0.01–2 MeV	0.785
2-20 MeV	0.215

6.3.4.2.3 HEU Storage Vault in Building 345

Cylinders of highly enriched (93% to 96%) uranium were measured with a TEPC mounted on a phantom about 24 in. from the cylinders. The DE from the cylinders was about 0.8 mrem/hr with a total DE of 14 mrem. The multisphere measurement at the same location as the phantom resulted in an average neutron energy of 0.53 MeV and a DE rate of 0.5 mrem/hr.

The solid lines in Figure 6-4 show the calculated energy spectrum from the multisphere detectors (Bonner spheres). Table 6-22 lists the dose fractions for the neutron energy groups (indicated by the dashed lines in Figure 6-4). The dose fractions for the lower (<10 keV) and intermediate (10 to 100 keV) energy neutron groups were about 47% of the total dose from the measurements.

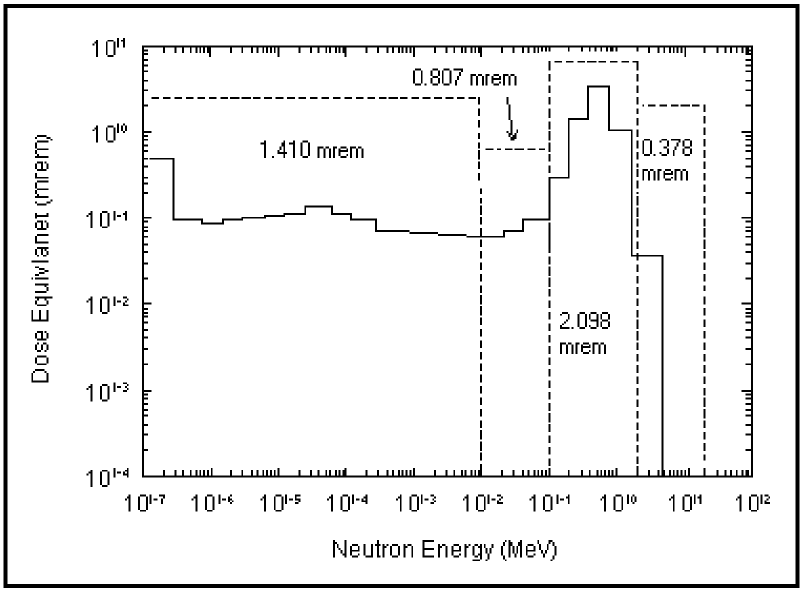

Figure 6-4. Results of neutron spectrum measurements made about 24 in. in front of 93% to 96% HEU cylinders (Soldat and Tanner 1992).

Table 6-22. Dose fractions for PORTS HEU storage vault in Building 345.

Neutron energy group	Near unshielded Cf-252 source
<10 keV	0.300
10–100 keV	0.172
0.1–2 MeV	0.447
2–14 MeV	0.081
Dose fractions favorable to claimants	
< 10 keV	0.300
0.01–2 MeV	0.610
0.1–2 MeV	0.081

6.3.4.2.4 Neutron-to-Photon Dose Ratio

A 0.125 neutron-to-photon dose ratio was calculated in a Centers for Disease Control and Prevention study of PORTS from data obtained from an onsite study conducted in 1995. Eight locations at the site were monitored. The observed median quarterly neutron doses were at the Feed Plant (5 mrem), X-344 autoclave area (7 mrem), Shipping and Receiving (7 mrem), cylinder lots (7 mrem), X-345 (3 mrem), X-744G (2 mrem), X-326 L Cage (<LOD), and the burn area (4 mrem). The mean total beta and gamma dose was 40 mrem. The mean neutron dose was 5 mrem; therefore, the calculated neutron-to-photon ratio was 0.125 (Cardarelli 1997).

A neutron dose can be calculated for all PORTS facilities with a potential for such a dose using the neutron-to-photon dose ratio of 0.125 applied to a fraction of the recorded photon dose. For example, a worker in a process facility with a recorded photon dose of 500 mrem in 1992 would be assigned a 62.5-mrem neutron dose (500 × 0.125 = 62.5). The photon dose should be adjusted for missed dose before estimation of the neutron dose. Because the routine monitoring for neutron exposure began in

1997, this ratio method should be used before 1997. The LOD method can apply after 1997 for neutron exposures.

PORTS measurement of the process and storage areas was actually about 0.125:1. For the PORTS neutron-to-photon dose ratio, a default ratio of 0.2 obtained from a Paducah Gaseous Diffusion Plant (PGDP) cylinder survey (Meiners 1999) would be reasonable. Table 6-23 lists these values for PORTS facilities.

Table 6-23. PORTS neutron-to-photon DE ratios.

Facility	Neutron-to-photon DE ratio[a]
General areas	0.20
Calibration facility	0.20
Nondestructive lab	0.20
HEU/EU storage	0.20
Process facilities	0.20

a. Multiply adjusted (i.e., for any missed dose) annual photon dose by the tabulated value and use this fraction times the neutron-to-photon dose ratio to estimate neutron dose.

6.3.4.3 PORTS Workplace Neutron Dosimeter Response

Table 6-24 summarizes typical neutron personnel dosimeter parameters important to *Hp(10)* performance in the workplace. The most important parameter in relation to *Hp(10)* performance of neutron dosimeters is the difference between calibration and workplace neutron energy spectra.

Table 6-24. Typical workplace neutron dosimeter *Hp(10)* performance.[a]

Parameter	Description	Potential workplace bias[b]
Workplace neutron energy spectra	TLD response increases with decreasing neutron energy.	Depends on workplace neutron spectra.
Exposure geometry	TLD response decreases with increasing exposure angle.	TLD recorded dose is lower at angles other than AP. Effect is highly dependent on neutron energy.
Missed dose	Doses less than MDL recorded as zero dose.	Recorded dose of record is probably **too low**. Impact of missed dose is greatest in earlier years because of higher MDLs of neutron dosimeters.
Environmental effects	Workplace environment (heat, humidity, etc.) fades dosimeter signal.	Recorded dose of record is probably **too low**.

a. Judgment based on Y-12 dosimeter response characteristics.
b. Recorded dose compared to *Hp(10)*.

6.3.5 PORTS Workplace Dose Uncertainty

Uncertainty in the recorded dose is due primarily to two sources, laboratory bias and workplace radiation field composition and geometry. Sections 6.3.3 and 6.3.4 discuss potential effects of these parameters on the recorded dose. Table 6-25 lists the judged estimates of uncertainty in the recorded dose at PORTS based on the combination of parameters.

Table 6-25. Estimates of uncertainty.

Dosimeter	PORTS	Laboratory uncertainty[a]	Workplace uncertainty[b]
Beta/gamma dosimeters			
Two-element film	Used 1954–1980	±20%	±50%
TLD	Used 1981–2003	±10%	±50%
Neutron dosimeters			
TLND	Used 1992–2003	±25%	±50%

a. In relation to *Hp(10)* response of the dosimeter.
b. 95% confidence level.

6.4 ADJUSTMENTS TO RECORDED DOSE

With the exception of beta dose, adjustments to the PORTS reported doses are necessary in consideration of the uncertainty associated primarily with the complex workplace fields and exposure orientations.

6.4.1 Beta Dose Adjustments

No adjustment in recorded nonpenetrating or skin dose is recommended. PORTS incident reports would typically address nonroutine worker exposure to significant beta or photon radiation. The assessed doses in the incident reports, based on investigations conducted at the time of the incident, probably provide the best estimate of dose received.

6.4.2 Photon Dose Adjustments

Using only AP geometry, the only major dose adjustments necessary to use are corrections of 16.5% for film dosimetry (1954 to 1980), 12.5% for TLD dosimetry (1981 to 1986) and 4% for TLD dosimetry (1987 to present) as listed in Table 6-26.

Table 6-26. Adjustments to reported PORTS deep photon dose.

Period	Dosimeter	Facility	Adjustment to reported dose
Prior to 1/1/1981	All beta/photon dosimeters	All facilities	Multiply reported film/TLD deep dose by factor of 1.165.
1/1/1981–1986	All beta/photon dosimeters	All facilities	Multiply reported film/TLD deep dose by factor of 1.125.
1987–present	All beta/photon dosimeters	All facilities	Multiply reported film/TLD deep dose by factor of 1.04.

6.4.3 Neutron Dose Adjustments

Adjustment to the neutron dose is necessary to account for the change in neutron QFs between historic and current scientific guidance, as described in NIOSH (2002). The QF is incorporated in the calibration methodology, which used flux-to-dose-rate conversion factors for varying neutron energies for each calibration source. Flux-to-dose-rate conversion factors were based on National Council on Radiation Protection (NCRP) Report 38 (NCRP 1971). This report lists both flux-to-dose-rate conversion factors and associated QFs that vary from 2 at energies less than 1 keV to 11 at 1 MeV. To convert from NCRP (1971) QFs to ICRP (1991) radiation weighting factors, a curve was fit to describe the neutron QFs as a function of neutron energy. The average QF for each neutron energy group was developed by integrating the area under the curve and dividing by the neutron energy range as shown in equation 6-4:

$$\bar{Q}(E_{n,0.1-2.0MeV}) = \frac{\int_{0.1}^{2.0} Q_f(E)dE}{Range(2.0-0.1)} \tag{6-4}$$

Table 6-27 summarizes historical changes in the QFs and the average NCRP (1971) QF for the neutron energy groups used in dose reconstruction. In addition, Table 6-27 lists the average QFs for the four neutron energy groups that encompass PORTS neutron exposures. The neutron DE correction factor can be calculated by dividing the dose fractions from Section 6.3.4.2 for each neutron energy group [$D_f(E_n)$] by the corresponding energy specific average (NCRP 1971) QF [$\bar{Q}(E_n)$] and then multiplying by the ICRP (1991) radiation weighting factor w_R, as shown in equation 6-5.

$$C_f(E_n) = \frac{D_f(E_n)}{\bar{Q}(E_n)} \times w_R \tag{6-5}$$

Table 6-27. Historical neutron quality or weighting factors.

Neutron energy (MeV)	Historical dosimetry guideline[a]	NCRP 38 QFs[b]	Average QF used at PORTS $\bar{Q}(E_n)$	ICRP 60 neutron weighting factor, w_R [c]
2.5×10^{-8}	3	2	2.35	5
1×10^{-7}	10	2		
1×10^{-6}		2		
1×10^{-5}		2		
1×10^{-4}		2		
1×10^{-3}		2		
1×10^{-2}		2.5	5.38	10
1×10^{-1}		7.5	10.49	20
5×10^{-1}		11		
1		11		
2		10	7.56	10
2.5		9		
5		8		
7		7		
10		6.5		
14		7.5		
20		8	Not applicable	5
40		7		
60		5.5		

a. Trilateral meeting in 1949 radiation protection guidelines (Fix, Wilson, and Baumgartner 1997).
b. Recommendations of NCRP Report 38 (NCRP 1971).
c. ICRP Publication 60 (ICRP 1991).

Table 6-28 summarizes the default neutron dose fractions by energy for PORTS work areas where field measurements of neutron spectra occurred, using the associated ICRP (1991) correction factors. The neutron DE is calculated by multiplying the recorded neutron dose by the area-specific correction factors. For example, consider a 50-mrem recorded neutron dose to a worker at the calibration facility. The corrected neutron dose is 22 mrem from neutrons between 0.1 MeV and 2.0 MeV (50 × 0.44) and 51 mrem from neutrons with energy between 2 MeV and 20 MeV (50 × 1.02). These adjustments should be applied to measured dose, missed dose, and dose determined based on a neutron-to-photon dose ratio.

Table 6-28. PORTS facility neutron dose fractions and associated ICRP (1991) correction factors.

Process	Description/buildings	Operations		Neutron energy (MeV)	Default dose fraction	ICRP (1991) correction factor
		Begin	End			
Calibration Area	X-710 (40 mCi Cf-252 source)	1954	Present	<0.01–2 2–20	0.23 0.77	0.44 1.02
Nondestructive Laboratory Area	X-710 (Cf-252 source)	1954	Present	<0.01–2 2–20	0.78 0.22	1.50 0.28
HEU/EU Storage Areas	X-345 and Cylinder Yards	1954	Present	<0.01–2 2–20	0.92 0.08	1.81 0.11
General Facilities including process areas[a]	Uranium enrichment, recovery, testing, maintenance, transport and storage areas	1954	Present	<0.01–2 2–20	0.92 0.08	1.81 0.11

a. Neutron dose fraction is assumed to be that of the HEU/EU storage areas.

6.5 MISSED AND UNMONITORED DOSE

There are undoubtedly missed doses for PORTS monitored workers. Missed dose applies to workers who were monitored but had results below the LOD of their personal radiation monitors. For the early years of radiation monitoring, when relatively high detection limits were combined with short monitoring durations, missed doses can be significant. Methodologies for estimating missed doses are discussed in this section and in ORAUT-OTIB-0040, *External Coworker Dosimetry Data for the Portsmouth Gaseous Diffusion Plant*, (ORAUT 2005a).

Unmonitored dose could have occurred because workers could have had the potential of receiving less than 10% of the radiation protection guidelines or because they worked in uncontrolled areas, where they were not considered radiation workers and therefore were not assigned dosimetry. The following sections discuss beta, photon, and neutron missed or unmonitored dose. Dose reconstructors should apply the adjustments to beta, photon, and neutron dosimetry discussed in Section 6.4 to the missed dose calculations discussed below. For these cases, dose reconstructors must rely on coworker data and/or population data to estimate a worker's potential unmonitored dose. The methods to be used are discussed in ORAUT (2005a).

For the special case when a worker's exposure potential has been determined to be low, the environmental dose should be assigned.

6.5.1 Missed and Unmonitored Shallow Dose

Missed shallow dose can occur if a zero dose was recorded for a dosimeter reading less than the LOD. Estimates of missed dose can be computed from personnel dose data from either before or after the missed dose period or from coworker data. NIOSH (2002) recommends calculating missed dose by multiplying LOD/2 by the number of zero dose results. The missed shallow dose for dosimeter results less than the LOD is particularly important for earlier years because LODs were higher and dosimeter exchange frequency was higher. The last column in Table 6-29 lists the resultant estimates of this annual missed dose for different years at PORTS. The LOD/2 method seems to be more conservative for most periods. The methods to be used are also discussed in ORAUT (2005a).

Table 6-29. Missed beta (or nonpenetrating) dose according to dosimeter type.

Dosimeter	LOD (rem)	Period of use	Exchange frequency	Max. annual missed dose (rem)
Two-element film	0.03	9/22/54–7/16/57	Weekly (n=50)	0.75
		7/17/57–9/30/58	Biweekly (n=25)	0.38
		10/01/58–4/8/59	Weekly (n=50)	0.75
		10/01/58–4/8/59	Monthly (n=12)	0.18
		4/9/59–7/31/60	Every 4 wk (n=13)	0.20
		8/1/60–7/5/64	Monthly (n=12)	0.18
		8/1/60–7/5/64	Quarterly (n=4)	0.06
		7/6/64–12/28/69	Quarterly (n=4)	0.06
		12/29/6912/30/73	Quarterly (n=4)	0.06
		12/31/73–6/29/75	Quarterly (n=4)	0.06
		12/31/73–6/29/75	Semiannual (n=2)	0.03
		6/30/75–12/31/80	Quarterly (n=4)	0.06
		6/30/75–12/31/80	Monthly (n=12)	0.18
TLD	0.04	1/1/81–12/31/82	Monthly (n=12)	0.24
		1/1/81–12/31/82	Quarterly (n=4)	0.08
	0.03	1/1/83–12/31/98	Quarterly (n=4)	0.06
		1/1/93–12/31/96 (BJC)	Quarterly (n=4)	0.06
	0.03	1/1/99–present (USEC)	Quarterly (n=4)	0.06
	0.03	1/1/99–present	Quarterly (n=4)	0.06

To calculate unmonitored dose, the dose reconstructor should consult ORAUT-OTIB-0040 (ORAUT 2005a) for instructions. Table 6-30, “Reported SDE dose by year in rem,” Table 6-31, “Departmental SDE dose ratios,” and Figure 6-5, “PORTS reported SDE average and maximum reported dose by year,” are for informational purposes only.

Table 6-31 is based on the average accumulated skin dose of PGDP employees from 1953 to 1988. The ratios were calculated by dividing the average cumulative dose column by the Mechanical Inspection group cumulative total. Because PORTS and PGDP had similar operations, the relative dose in each departmental area should be similar.

The beta dose reported from 1981 through the end of 1982 might not be as reliable because there was no OW for the PORTS TLD badge for this period. However, the average SDE for 1981 to 1982 is close to those recorded from subsequent years (as indicated in Table 6-30). ORAUT-OTIB-0017 (ORAUT 2005b) and ORAUT-OTIB-0040 (ORAUT 2005a) contain instructions on reconstructing the beta or nonpenetrating dose during 1981 and 1982. Table 6-29 includes the potential missed beta dose for the respective periods of use, dosimeter types, LODs, and exchange frequencies.

If an individual has no recorded dose, and if it is definitely established that the individual was not a radiation worker, then the assigned dose is the environmental dose discussed in the occupational environmental dose portion of this PORTS Site Profile (ORAUT 2004b) and ORAUT-OTIB-0040 (ORAUT 2005a).

Table 6-30. Reported SDE dose by year (rem).[a]

Year	Number monitored	Total SDE	Maximum SDE	Average SDE (Geometric mean)	Geometric standard deviation
1954	172	57.374	1.640	0.334	1.98
1955	903	128.187	0.867	0.142	2.17
1956	610	53.579	0.655	0.088	2.37
1957	482	77.845	2.095	0.162	3.00
1958	873	229.719	5.770	0.263	3.76
1959	866	442.237	10.075	0.511	3.60
1960	488	106.240	3.420	0.218	3.26
1961	549	113.291	4.890	0.206	3.89
1962	693	94.380	1.340	0.136	2.67
1963	640	78.012	1.510	0.122	2.94
1964	621	49.253	1.388	0.117	2.89
1965	356	37.018	1.150	0.104	2.80
1966	236	24.590	0.790	0.104	2.39
1967	141	18.606	1.135	0.132	2.52
1968	611	90.098	1.040	0.147	2.32
1969	901	101.743	1.890	0.113	3.35
1970	414	70.396	2.028	0.170	2.90
1971	167	30.015	2.385	0.180	3.03
1972	234	35.116	1.440	0.150	2.64
1973	207	36.700	2.140	0.177	2.91
1974	189	27.925	2.185	0.148	3.18
1975	285	59.540	2.060	0.209	2.67
1976	363	63.585	1.350	0.175	2.40
1977	523	71.895	1.410	0.137	2.72
1978	618	127.100	1.840	0.206	2.56
1979	393	58.970	1.120	0.150	2.37
1980	1,004	122.872	1.500	0.122	2.94
1981	587	26.203	0.535	0.045	2.89
1982	784	37.638	1.709	0.048	4.63
1983	1,092	61.048	1.749	0.056	4.38
1984	1,002	46.394	0.958	0.046	3.68
1985	1,428	45.647	0.452	0.032	3.12
1986	1,308	49.436	2.198	0.038	5.71
1987	1,147	51.602	2.160	0.045	5.27
1988	1,189	42.922	0.608	0.036	3.36
1989	1,491	53.851	0.593	0.036	3.33
1990	1,862	81.447	0.694	0.044	3.27
1991	1,509	61.066	0.517	0.040	3.00
1992	889	50.365	0.910	0.057	3.28
1993	667	41.737	1.551	0.063	3.95
1994	583	24.610	0.507	0.042	2.91
1995	2,112	67.011	1.996	0.032	5.89
1996	3,461	100.426	1.951	0.029	6.09
1997	3,394	116.483	1.281	0.034	4.75
1998	355	14.238	0.300	0.040	2.37
1999	189	9.423	0.247	0.050	1.98
2000	703	19.103	0.338	0.027	2.96
2001	182	5.479	0.199	0.030	2.25
2002	224	10.710	0.353	0.048	2.35
2003	385	23.498	0.569	0.061	2.61

a. Data provided by PORTS HP department. This dataset contains only positive dose workers (Litton 2004; Demopoulos 2004)

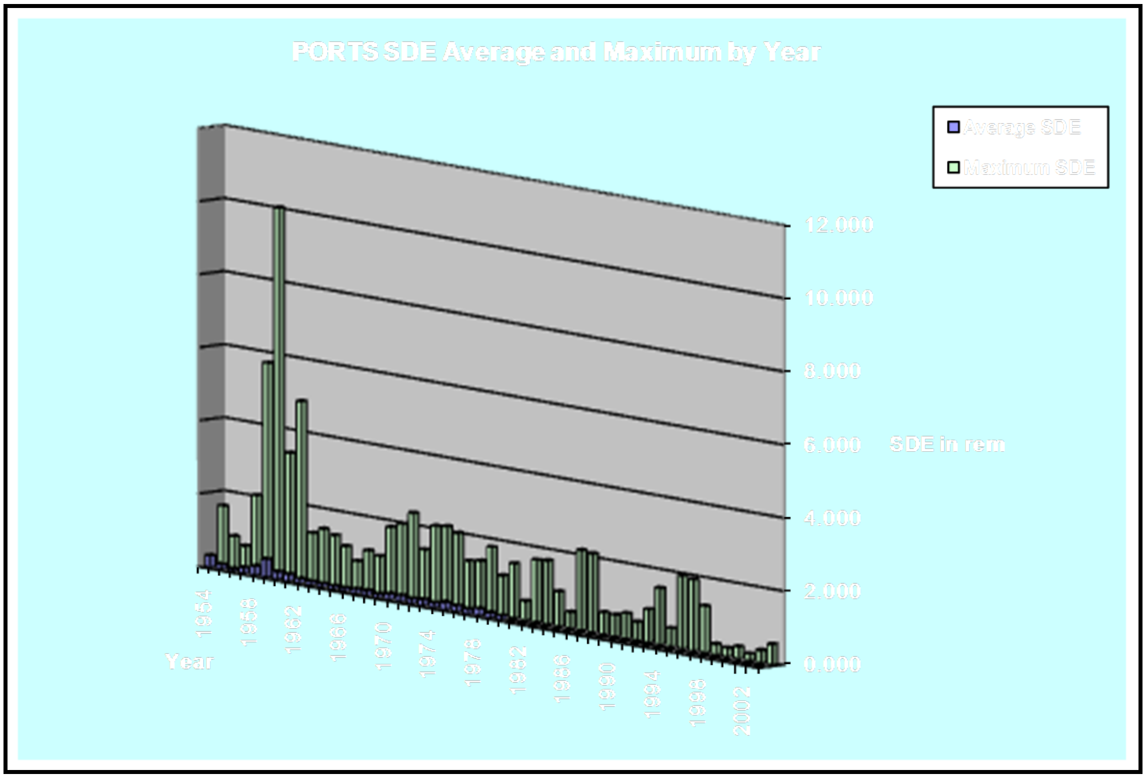

Figure 6-5. PORTS reported SDE average and maximum dose by year. Data provided by PORTS HP department (Demopoulos 2004).

Table 6-31. Departmental SDE dose ratios.

Description	Average cumulative dose (mrem)[a]	Ratio (using mechanical inspection employees as the basis)
Feed Plant Operators	15,834	30.99
Decontamination	12,369	24.21
Feed Plant Mechanics	9,767	19.11
Chemical Processors	3,794	7.42
Feed Plant Mechanics	1,968	3.85
Process Maintenance	1,954	3.82
Cascade Operators	1,824	3.57
Instrument	1,407	2.75
PEMU Decontamination	1,223	2.39
Electrical	987	1.93
Converter Shop	933	1.83
Material Term Serv.	931	1.82
Analytical Chemistry	877	1.72
Laundry	851	1.67
Converter Test	836	1.64
Equipment Maintenance	586	1.15
Equipment Control	572	1.12
Transportation Pool	557	1.09
Fabrication Shop	517	1.01
Mechanical Inspection	511	1.00

a. Adapted from PACE and University of Utah (2000, Table 7-3).

Energy Range

Although the different elements in a multielement dosimetry system can have specific energy exposure characteristics, this analysis cannot consider shielding and radiation scattering. Attempts were made with the PORTS TLD system to calibrate dosimeters based on a number of mixed field combinations. If the badge read close to the regression results of a particular mixed-field calibration and the person probably worked in an area with that expected mixed field, the dose calculations would use the mixed-field calibration factors. More detail of the procedure was not available.

Skin Contamination

Skin ^{99}Tc contaminations could have gone undetected because of the low-energy beta radiation. Although routine surveys involved monitoring for beta contamination, the uranium daughters ^{234}Th and ^{234m}Pa could have masked the presence of ^{99}Tc.

Potential doses from ^{99}Tc skin contamination have been evaluated by using the VARSKIN computer code (Durham 1992). The calculated shallow dose rate from uniform ^{99}Tc skin contamination is 1.6×10^{-3} mrem/hr per dpm/cm^2 (Swinth 2004). Because the nuclide is difficult to remove from skin, the integrated shallow dose from ^{99}Tc skin contamination can be relatively large. For example, with a residence half-time of 1.5 days, the dose is 8.1×10^{-2} mrem per dpm/cm^2.

Because skin contamination events involving ^{99}Tc could have gone undetected, it might be appropriate to consider an additional skin dose component for a reported shallow dose of a worker who might have had direct contact with ^{99}Tc. To estimate an annual missed shallow dose in the absence of specific data, one must make assumptions about the number of times per year an affected skin region might have been contaminated and the extent of each contamination. For example, one might assume a monthly contamination event at a specific location on the skin with an average level of 25,000 dpm/100 cm^2 (the action limit for ^{99}Tc contamination on work surfaces and hand tools at PORTS, see Table 6-32). With the residence half-time of 1.5 days assumed above, it follows that the annual shallow dose equivalent (SDE) would be 240 rem (12×250 dpm/cm^2 $\times$ 0.081 mrem per dpm/cm^2). The direct external dose rate at a distance of 10 cm from a surface contaminated at this same level would be 0.025 mrem/hr (250 dpm/cm^2 $\times 10^{-4}$ mrem/hr per dpm/cm^2). At 30 cm, the rate would be 0.00025 mrem/hr.

Significant nonroutine worker doses, as can occur from skin contamination events, might be addressed in specific incidence reports. In such cases, dose reconstructors should consider assessments based on investigations conducted at the time of the incident as the best resource.

The limit of dose rate (LODR) appears to be between 0.07 and 0.04 mrem/hr (see Table 6-32) based on the PORTS monitoring limits. One method of skin contamination monitoring was for workers to wrap their hands around a Geiger-Mueller (GM) tube. A typical GM tube is about 2% efficient for ^{99}Tc beta particles. (A contemporary example is a Ludlum Model 44-7 end window GM.)

For the routine meter used from at least the 1960s through the 1980s, the Samson meter appears to have had an LOD of 60 cpm because this was the lowest count rate action limit. This would translate to a 3,000-dpm activity level, which corresponds to a 0.07-mR/hr ^{99}Tc beta radiation level. Table 6-32 points out that the lowest radiation field level used for an action level is 0.04 mR/hr. This is probably the LODR obtainable from a Cutie Pie ion chamber (which was used for beta and photon radiation field monitoring from 1965 to 1985).

External Beta Dose Rates

As a first method, dose reconstructors can estimate dose from probable external beta dose source terms. PORTS personnel have been exposed to some beta radiation levels, typically during

Table 6-32. Beta contamination plant limits for PORTS.[a]

Area	Type of monitoring	Action limit 1963	Action limit 1975	Action limit 1980 and 1990 (Tc-beta)
Floor, work surfaces, etc.	Contamination Radiation level	500 cpm (Samson meter) 0.15 mR/hr @ 1 in.	0.15 mR/hr @ 1 in.	Surface-25,000 dpm 500 cpm Samson Removable-5000 dpm 100 cpm
Hands	Contamination Radiation level	Hand and foot monitor as posted		
Hands	Contamination Radiation level	(Hands wrapped around OW GM tube) 100 cpm 0.04 mR/hr	0.04 mR/hr @ 1 in.	
Body	Radiation level		0.08 mR/hr	3,000 dpm/100 cm^2 (Samson – 60 cpm)
Shoes	Contamination or radiation level	Personal 500 cpm (hand and foot monitor) Issued 5,000 cpm	0.15 mR/hr 1.5 mR/hr	Issued 400,000 dpm/100 cm^2
Clothing, tables, equipment, tools, etc.	Radiation level		0.15 mR/hr	Coveralls, gloves-200,000 dpm/100 cm^2 Personal clothing – 9,000 dpm/100 cm^2
Airborne activity	Contamination Radiation level	20 cpm/ft^3 (filter paper wrapped around GM tube)	2,000 dpm/ft^3	
Hand tools and other equipment	Fixed Removable			25,000 dpm/100 cm^2 3,000 dpm/100 cm^2
Process shop and purge equipment	Fixed Removable			200,000 dpm/100 cm^2 3,000 dpm/100 cm^2

a. PORTS (1963, 1975); GAT (1990).

maintenance. For example, with ^{99}Tc trap maintenance, a typical magnesium trap for ^{99}Tc used for liquid UF_6 is a 10-in.-diameter by 13.25-in.-tall cylinder with about 5,650 g of magnesium trap material. About 1.7 µCi of ^{99}Tc/g of absorbent material is trapped, which leads to about 10 mCi of ^{99}Tc in the trap (gaseous UF_6 ^{99}Tc traps accumulate much less activity). Using the VARSKIN code, the beta dose rate result for 10 mCi in a trap at 30 cm (Swinth 2004) is 0.16 mrem/hr at the face. Because there are three traps for every change-out (PORTS) and 50 hr/yr, the resultant dose is 24 mrem/yr (0.16 mrem/hr × 3 traps/change-out × 50 hr/yr). This method assumes that the personnel were potentially exposed to these beta radiation fields even though most personnel did not perform ^{99}Tc trap maintenance. In addition, this is a conservative estimate because it assumes there was no shielding of the magnesium trap material, the accumulation of activity at the top of the cylinder occurred, and the personnel contact was very close.

In addition, there are alumina traps throughout the cascade buildings. They usually accumulate uranium progeny and some transuranic elements (^{237}Np), but ^{99}Tc is possible. PORTS indicated that external beta fields as high as 2 R/hr have been measured around these alumina traps and that uranium progeny buildup is probably responsible for these fields. However, personal dosimetry would have monitored uranium progeny adequately.

A second method can be based on the LODR for an end geometric mean of 0.08 mrem/hr (Table 6-32) for the whole body. With the assumption of a maximum exposure of 2,000 hr/yr, the exposure calculation would be 0.08 mrem/hr × 2,000 hr/yr, which is 160 mrem/yr. Another approximation can be made using the conservative assumption that the technetium:uranium progeny ratio of about 0.4 (Basset 1986a), along with the maximum exposure from ambient levels of about 0.2 mrem/hr (Basset 1986a), would yield a result of 160 mrem (0.2 mrem/hr × 0.4 × 2,000 hr/yr = 160 mrem/yr.)

This method might be more plausible because it represents a chronic exposure situation that was more common among all PORTS workers.

In general, direct external beta dose from ^{99}Tc is minimal. The unshielded shallow dose rate to bare skin (no clothing) at a distance of 10 cm in air from a uniformly contaminated surface is about 1×10^{-4} mrem/hr per dpm/cm^2, as estimated with VARSKIN. The dose rate at 30 cm is only about 1×10^{-6} mrem/hr per dpm/cm^2. Table 6-33 summarizes these benchmark values for SDE rates as determined from VARSKIN for skin contamination and for external exposure with intervening air.

Table 6-33. SDE rates for ^{99}Tc.

Condition	**Dose rate (mrem/hr per dpm/cm^2)**	**Dose rate (mrem/hr)**
Skin contamination	1.6×10^{-3}	N/A
External, 10 cm air	1×10^{-4}	0.08 mrem/hr lower limit of dose limit whole body
External, 30 cm air	1×10^{-6}	0.20 mrem/hr ambient (× 0.4 ^{99}Tc/U daughter ratio)

For method 1, the total SDE would be 264 mrem/yr (240 mrem skin dose + 24 mrem external beta dose). For method 2, the total SDE would be 400 mrem/yr (240 mrem skin dose + 160 mrem external beta dose).

6.5.2 Missed and Unmonitored Photon Dose

Photon dose could have been missed if a zero dose was recorded for a dosimeter reading less than the LOD. Estimates of missed dose can be computed from dose before or after the missed dose period or from coworker data. NIOSH (2002) recommends calculating missed dose by multiplying LOD/2 by the number of zero dose results. The missed photon dose for dosimeter results less than the LOD is particularly important for earlier years because LODs and dosimeter exchange frequencies were higher. The last column in Table 6-34 lists the resultant estimates of this annual missed dose for different years at PORTS. The LOD/2 method seems to be more conservative for most periods. The methods to be used are also discussed in ORAUT-OTIB-0040, *External Coworker Dosimetry Data for the Portsmouth Gaseous Diffusion Plant*, (ORAUT 2005a).

Individuals with no dose recorded and if it is definitely established that the individual was not a radiation worker, then the assigned dose is the environmental dose discussed in the occupational environmental dose portion of this PORTS site profile and ORAUT-OTIB-0040 (ORAUT 2005a).

For calculating unmonitored dose, the dose reconstructor should consult ORAUT-OTIB-0040 (ORAUT 2005a) for instructions. Table 6-35, “Reported Gamma, Photon or DDE dose by year in rem,” Table 6-36, “Departmental DDE dose ratios” and Figure 6-6, PORTS reported DDE average and maximum reported dose by year” are for informational purposes only.

Table 6-34. Missed photon dose adjustments to recorded deep dose according to PORTS facility.

Facility type	Dosimeter	LOD (rem)	Period of use	Exchange frequency	Max. annual missed dose (rem)
All facilities	PORTS two-element film	0.03	9/22/54–7/16/57	Weekly (n=50)	0.75
			7/17/57–9/30/58	Biweekly (n=25)	0.38
			10/01/58–4/8/59	Weekly (n=50)	0.75
			10/01/58–4/8/59	Monthly (n=12)	0.18
			4/9/59–7/31/60	Every 4 wk (n=13)	0.20
			8/1/60–7/5/64	Monthly (n=12)	0.18
			8/1/60–7/5/64	Quarterly (n=4)	0.06
			7/6/64–12/28/69	Quarterly (n=4)	0.06
			12/29/6912/30/73	Quarterly (n=4)	0.06
			12/31/73–6/29/75	Quarterly (n=4)	0.06
			12/31/73–6/29/75	Semiannual (n=2)	0.03
			6/30/75–12/31/80	Quarterly (n=4)	0.06
			6/30/75–12/31/80	Monthly (n=12)	0.18
	PORTS Harshaw w/o window 4-element TLD	0.015	1/1/81–12/31/82	Monthly (n=12)	0.09
			1/1/81–12/31/82	Quarterly (n=4)	0.03
All facilities	PORTS Harshaw with window 4-element TLD	0.010	1/1/83–12/31/98	Quarterly (n=4)	0.02
			1/1/93–12/31/96 (BJC)	Quarterly (n=4)	0.02
	ICN TLD 760	0.010	1/1/99–present (USEC)	Quarterly (n=4)	0.02
	ORNL TLD Panasonic	0.010	1/1/99–present (BJC)	Quarterly (n=4)	0.02

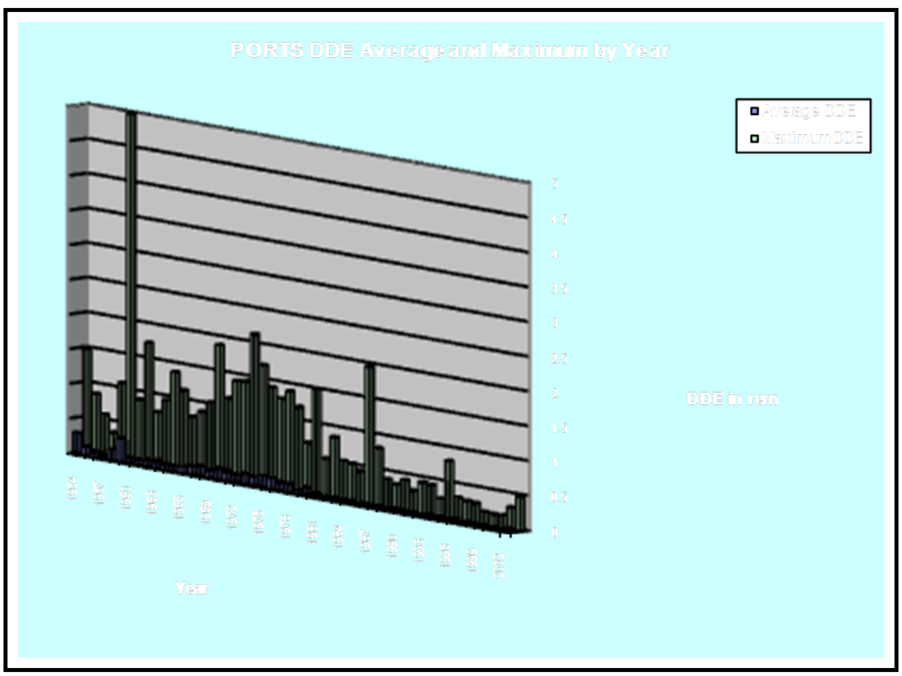

Figure 6-6. PORTS reported DDE average and maximum dose by year. Data provided by PORTS HP department. (Demopoulos 2004)

Table 6-35 is based on the average accumulated deep dose of PGDP employees from 1953 to 1988. The ratios were calculated from dividing the average cumulative dose column by the Steam Plant group cumulative total. Because PORTS and PGDP had similar operations, the relative dose in each departmental area should be similar.

Table 6-35. Reported gamma, photon, or DDE dose by year (rem).[a]

Year	Number monitored	Total DDE	Maximum DDE	Average DDE (Geo Mean)	Geometric Standard Deviation
1954	172	51.945	1.500	0.302	1.99
1955	903	112.711	0.867	0.125	2.30
1956	610	48.002	0.590	0.079	2.37
1957	482	25.805	0.355	0.054	2.24
1958	873	130.699	1.090	0.150	2.34
1959	866	296.067	5.000	0.342	3.16
1960	488	35.840	0.900	0.073	2.94
1961	549	38.691	1.740	0.070	3.97
1962	693	65.015	0.765	0.094	2.46
1963	640	69.772	0.960	0.109	2.54
1964	621	45.244	1.388	0.073	3.00
1965	356	27.710	1.150	0.078	3.17
1966	236	21.884	0.790	0.093	2.50
1967	141	16.460	0.885	0.117	2.38
1968	611	86.630	1.040	0.142	2.35
1969	901	94.836	1.890	0.105	3.46
1970	414	64.633	1.145	0.156	2.35
1971	167	22.595	1.420	0.135	2.75
1972	234	26.931	1.440	0.115	2.96
1973	207	32.415	2.140	0.157	3.07
1974	189	24.395	1.720	0.129	3.04
1975	285	46.225	1.420	0.162	2.54
1976	363	55.424	1.280	0.153	2.49
1977	523	62.290	1.410	0.119	2.89
1978	618	75.170	1.210	0.122	2.68
1979	393	22.835	0.715	0.058	2.94
1980	1,004	117.394	1.500	0.117	2.99
1981	587	26.203	0.535	0.045	2.89
1982	784	18.467	0.864	0.024	4.66
1983	1,092	20.014	0.543	0.018	4.32
1984	1,002	17.855	0.529	0.018	4.27
1985	1,428	16.765	0.452	0.012	4.75
1986	1,308	20.277	1.973	0.016	7.90
1987	1,147	27.207	0.826	0.024	4.57
1988	1,189	21.957	0.395	0.018	3.76
1989	1,491	29.255	0.348	0.020	3.41
1990	1,862	39.745	0.430	0.021	3.65
1991	1,509	24.650	0.306	0.016	3.55
1992	889	19.173	0.446	0.022	3.64

Table 6-35. Continued. Reported gamma, photon, or DDE dose by year (rem).[a]

Year	Number monitored	Total DDE	Maximum DDE	Average DDE (Geometric Mean)	Geometric Standard Deviation
1993	667	18.992	0.428	0.028	3.22
1994	583	8.852	0.259	0.015	3.40
1995	2,112	24.181	0.821	0.011	6.37
1996	3,461	35.346	0.317	0.010	4.41
1997	3,394	30.374	0.292	0.009	4.45
1998	355	10.182	0.285	0.029	2.67
1999	189	6.406	0.183	0.034	2.06
2000	703	15.404	0.173	0.022	2.42
2001	182	4.259	0.176	0.023	2.40
2002	224	7.168	0.316	0.032	2.67
2003	385	17.590	0.496	0.046	2.77

a. Data provided by PORTS HP department. This dataset contains only positive dose workers. (Litton 2004; Demopoulos 2004)

Table 6-36. Departmental DDE dose ratios.

Description	Average cumulative dose in mrem[a]	Ratio (using steam plant employees as the basis)
Plant feed operators	3,814	34.67
Decontamination	2,788	25.35
Feed Plant Mechanics	2,587	23.52
Cascade Operators	627	5.70
Chemical Operators	595	5.41
Instrument	538	4.89
Transportation Pool	371	3.37
Process Maintenance	364	3.31
Environ Control	338	3.07
Electricians	298	2.71
Mat. Term Mgr	295	2.68
PEMU Decontamination	253	2.30
Mechanical Inspection	170	1.55
Plant services	147	1.34
Converter Test	145	1.32
Feed Plant Mechanics	143	1.30
Nitrogen Plant	142	1.29
Metals Building	132	1.20
Fabrication Shops	127	1.15
Steam Plant	110	1.00

a. Adapted from PACE and University of Utah (2000, Table 7-2).

Facility/Location

Table 6-34 lists the missed photon dose at PORTS facilities. The same information is listed in Table 6-3, which contains the table references. The maximum annual missed dose is the LOD/2 multiplied by the exchange frequency. The LOD is based on laboratory irradiations or from citations from dosimetry vendors.

6.5.3 Missed and Unmonitored Neutron Dose

Table 6-37 summarizes missed neutron dose for PORTS (see Table 6-3 for references). For PORTS, neutron-to-photon dose ratio is 0.2. Table 6-23 lists these values for different PORTS facilities. The photon dose should be adjusted for missed dose before estimation of the neutron dose. Because routine monitoring for neutron exposure began in 1997, the neutron-to-photon dose ratio method should be used before 1997. The LOD method can apply after 1997 for neutron exposures. Dose reconstructors should multiply this DE by the neutron dose correction factors listed in Table 6-28 for different processes or buildings.

Table 6-37. PORTS dosimeter type, period of use, exchange frequency, LOD, and potential annual dose missed.

Dosimeter	Period of use	Exchange frequency	Laboratory LOD (rem)	Maximum annual missed dose (rem)[a]
Neutron dosimeters				
PORTS TLD albedo dosimeter {USEC and BJC }	1/1/1992–12/31/94 {Unmoderated Cf-252 calibrated}	Quarterly (n=4)	0.02[a]	0.04
PORTS TLD albedo dosimeter {USEC and BJC }	1/1/95–12/31/96 {Moderated Cf-252 calibrated}	Quarterly (n=4)	0.02[a]	0.04
ICN TLD 760 {USEC}	1/1/97–present {Moderated Cf-252 calibrated}	Quarterly (n=4)	0.01[b]	0.02
Y-12 Panasonic TLND {BJC employees}	1/1/97–12/31/98	Quarterly (n=4)	0.01[c]	0.02
ORNL Panasonic TLND 8806 four-element TLD {BJC employees}	1/1/1999–present	Quarterly (n=4)	0.01[d]	0.02

a. (Wagner 2003).
b. ICN (2003).
c. Souleyrette (2003).
d. McMahon (2003).

Unmonitored neutron dose may be assigned, or it may be calculated with photon coworker data as presented in ORAUT-OTIB-0040 (ORAUT 2005a). The same method as described in the above paragraph can than be applied to the unmonitored neutron dose calculation.

6.6 ORGAN DOSE

6.6.1 Organ Dose Conversion Factors

NIOSH (2002) describes the methodology to calculate the organ dose distribution for the respective radiation types using identified exposure geometries. The missed or unmonitored dose is to be estimated as described in Section 6.5. A correction of 16.5% for film dosimetry (1954 to 1980), 12.5% for TLD dosimetry (1981 to 1986) and 4% for TLD dosimetry (1987 to present) is recommended. After the corrections are applied, the R-to-organ DCFs should be applied for 1954 to 1986 and the *Hp(10)*-to-organ DCFs should be applied for 1987 to the present. The DCFs can be found in Appendix B of NIOSH (2002).

6.7 DOSE RECONSTRUCTION

As much as possible, dose to individuals should be based on dosimetry records. It is important to distinguish between the recorded nonpenetrating and penetrating doses and the actual *Hp(0.07)* and *Hp(10).* The following list summarizes appropriate information:

- Dosimetry records that provide nonzero beta/photon values for *Hp(10)* and *Hp(0.07)* are considered adequate. A correction of 16.5% for film dosimetry (1954 to 1980), 12.5% for TLD dosimetry (1981 to 1986) and 4% for TLD dosimetry (1987 to present) is recommended. Beta energies are greater than 15 keV and photon energies should be considered to be in the range of 30 keV to 250 keV.

- Workers for whom dosimetry records provide zero beta/photon values for *Hp(10)* and *Hp(0.07)* should have missed dose assigned on the basis of LOD/2 times the number of zeros, as described in Sections 6.5.1 and 6.5.2 (NIOSH 2002).

- If an individual has no recorded dose, and if it is definitely established that the individual was not a radiation worker, then the assigned dose is the environmental dose discussed in the Occupational Environmental Dose portion of this PORTS Site Profile (ORAUT 2004b).

- The missed dose is to be estimated as described in Section 6.5. A correction of 16.5% for film dosimetry (1954 to 1980), 12.5% for TLD dosimetry (1981 to 1986), and 4% for TLD dosimetry (1987 to present) is recommended. After the corrections are applied, the R-to-organ DCFs should be applied for 1954 to 1986 and the *HP(10)*-to-organ DCFs should be applied for 1987 to the present. The DCFs can be found in Appendix B of NIOSH (2002).

- Reported and missed neutron DEs should be adjusted according to Section 6.4.3 to account for ICRP (1991).

- Cylinder yard workers, security guards, and general workers for whom no neutron dose is recorded should have missed neutron DE assigned based on a neutron-to-photon dose ratio of 0.2 (Meiners 1999).

- Special attention should be paid to the possibility of skin contamination incidents for workers involved in ^{99}Tc recovery operations (Section 6.5.1).

- Uncertainty is discussed in Tables 6-13, 6-14, 6-19 and 6-25.

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GLOSSARY

absorbed dose, D

Amount of energy imparted by radiation to unit mass of absorbing material (100 ergs per gram), including tissue. The unit used prior to the use of the International System of metric units (SI) is the rad; the SI unit is the gray.

accreditation

In relation to this document, recognition that a dosimeter system has passed the performance criteria of the DOE Laboratory Accreditation Program (DOELAP) standard (DOE 1986) in specified irradiation categories.

accuracy

If a series of measurements has small systematic errors, they are said to have high accuracy. The accuracy is represented by the bias.

albedo dosimeter

A TLD device that measures the thermal, intermediate, and fast neutrons that are scattered and moderated by the body from an incident fast neutron flux.

algorithm

A computational procedure.

Atomic Energy Commission

Original agency established for nuclear weapons and power production; a successor to the Manhattan Engineering District (MED) and a predecessor to the U.S. Department of Energy (DOE).

backscatter

Deflection of radiation by scattering processes through angles greater than 90 degrees, with respect to the original direction of motion.

beta particle

A charged particle of very small mass emitted spontaneously from the nuclei of certain radioactive elements. Most (if not all) of the direct fission products emit (negative) beta particles. Physically, the beta particle is identical with an electron moving at high velocity.

Bonner Sphere

See *Multisphere Neutron Spectrometer*

bremsstrahlung

Secondary photon or X-ray radiation produced by deceleration of charged particles passing through matter.

buildup

In relation to this document, increase in flux or dose due to scattering in the medium.

collective dose equivalent

The sum of the dose equivalents of all individuals in an exposed population. Collective dose is expressed in units of person-rem (person-sievert).

control dosimeter

A dosimeter used to establish the dosimetry system response to radiation dose. The dosimeter is exposed to a known amount of radiation dose.

curie

A special unit of activity. One curie 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 (DDE)

The dose equivalent at the respective depth of 1.0 cm in tissue.

densitometer

Instrument that has a photocell to determine the degree of darkening of developed photographic film.

density reading

See *optical density*.

dose equivalent (DE)

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 gray, *H* is in seivert.
(1 sievert = 100 rem.)

DOELAP

The DOE Laboratory Accreditation Program (DOELAP) accredits DOE site dosimetry programs based on performance testing and onsite reviews performed on a 2- year cycle.

dose equivalent index

For many years the dose equivalent index was used to calibrate neutron sources that were used to calibrate neutron dosimeters. A concept of summing the maximum dose equivalent delivered in the ICRU sphere at any depth for the respective neutron energies was used even though the maximum dose occurred at different depths.

dosimeter

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

dosimetry system

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

error

Term used to express the difference between the estimated and "true" value. *Error* can also be used to refer to the estimated uncertainty.

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 of the ionization produced by gamma (or 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.

fast neutron
Neutron of energy between 10 keV and 10 MeV.

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.

film density
See *optical density*.

film dosimeter
A small packet of film within a holder that attaches to a worker.

filter
Material used to adjust radiation response of a dosimeter to provide an improved tissue equivalent or dose response.

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 of high energy, the only essential difference being that X-rays do not originate in the nucleus.

gamma ray interactions
Interaction of gamma rays with matter occurs through three primary processes as follows:

Photoelectric absorption - Process whereby a gamma ray (or X-ray) photon, with energy somewhat greater than that of the binding energy of an electron in an atom, transfers all its energy to the electron, which is consequently removed from the atom.

Compton scattering - Attenuation process observed for X-ray or gamma radiation in which an incident photon interacts with an orbital electron of an atom to produce a recoil electron and a scattered photon of energy less that the incident photon.

Pair production
Absorption process for X-ray and gamma radiation in which the incident photon is annihilated in the vicinity of the nucleus of the absorbing atom, with subsequent production of an electron and positron pair. This reaction only occurs for incident photon energies that exceed 1.02 MeV.

hurst dosimeter
Film-based criticality dosimeter.

intermediate energy neutron
Neutron of energy between 0.5 electron-volt (assumed to be 0.4 electron-volt because of cadmium cutoff in neutron response) and 10 kiloelectron-volts.

ionizing radiation
Electromagnetic radiation (consisting of photons) or particulate radiation (consisting of electrons, neutrons, protons, etc.) capable of producing charged particles through interactions with matter.

isotopes
Forms of the same element having identical chemical properties but differing in their atomic masses. Isotopes of a given element all have the same number of protons in the nucleus but different numbers of neutrons. Some isotopes of an element may be radioactive.

kiloelectron-volt (keV)
An amount of energy equal to 1,000 electron-volts.

luminescence
The emission of light from a material as a result of some excitation.

minimum detectable level (MDL)
Often confused because the statistical parameters necessary to its calculation are not explicitly defined. Nonetheless, often assumed to be the level at which a dose is detected at the two-sigma level (i.e., 95% of the time). The MDL should not be confused with the minimum recorded dose.

minimum recorded dose
Based on a policy decision, the minimum dose level that is routinely recorded. A closely related concept is the dose recording interval. PORTS has generally recorded minimum doses of 10 mrem and at intervals of 10 mrem (10, 20, 30, etc.).

million-electron volt (MeV)
An amount of energy equal to 1,000,000 electron-volts.

multisphere neutron spectrometer
A series of neutron moderating spheres of tissue-equivalent material with a neutron detector positioned at the middle of the respective spheres. Algorithms are used to unfold the data to calculate the neutron spectra.

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 0.4 electron-volt and 10 kiloelectron-volts.

neutron, thermal
Strictly, neutrons in thermal equilibrium with surroundings. In general, neutrons with energy less than the cadmium cutoff at about 0.4 electron-volts.

open window (OW)
Designation on TLD dosimeter reports is of little use because there was no OW throughout 1981. Otherwise, an OW existed for the film and TLD badges used at PORTS.

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

penetrating dose
Designation (i.e., P or Pen) on film dosimeter reports that implies a radiation dose, typically to the whole body, from higher energy photon radiation.

personal dose equivalent, *Hp(d)*
Radiation quantity recommended for use as the operational quantity to be recorded for radiological protection purposes by the International Commission on Radiological Units and Measurements. Represented by *Hp(d)*, where *d* identifies the depth (in mm) and represents the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose, d = 0.07 mm and is noted as *Hp(0.07)*. For penetrating radiation of significance to whole-body dose, d = 10 mm and is noted as *Hp(10)*.

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

precision
If a series of measurements has small random errors, the measurements are said to have high precision. The precision is represented by the standard deviation.

quality factor, Q
A modifying factor used to derive dose equivalent from absorbed dose.

rad
A unit of absorbed dose equal to the absorption of 100 ergs per gram of absorbing material, such as body tissue.

radiation
One or more of beta, neutron, and photon radiation.

radiation monitoring
Routine measurements and the estimation of the dose equivalent for determining and controlling the dose received by workers.

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

radionuclide
Unstable nuclides that emit radiation, generally alpha or beta particles, gamma rays, and neutrons to eventually form stable nuclei.

random errors
When a given measurement is repeated, the resulting values, in general, do not agree exactly. The causes of the disagreement between individual values must also be causes of their differing from the "true" value. Errors resulting from these causes are *random* errors.

rem
A unit of dose equivalent, which is equal to the product of the number of rads absorbed and the quality factor.

roentgen
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.

scattering
The diversion of radiation from its original path as a result of interactions with atoms between the source of the radiation and a point at some distance away. Scattered radiations are typically changed in direction and of lower energy than the original radiation.

shallow absorbed dose (Ds)
The absorbed dose at a depth of 0.07 mm in a material of specified geometry and composition.

shallow dose equivalent (SDE)
Dose equivalent at a depth of 0.07 mm in tissue.

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

sievert (Sv)
The SI unit for dose equivalent. (1 Sv = 100 rem.)

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

systematic errors
When a given measurement is repeated and the resulting values all differ from the "true" value by the same amount, the errors are systematic.

thermal neutron
Strictly, neutrons in thermal equilibrium with surroundings. In general, neutrons of energy less than the cadmium cutoff of about 0.4 electron-volt.

tissue equivalent
Used to imply that the radiation response characteristics of the material being irradiated are equivalent to tissue. Achieving a tissue equivalent response is typically an important consideration in the design and fabrication of radiation measuring instruments and dosimeters.

tissue-equivalent proportional counter (TEPC)
Device used to measure the absorbed dose from neutron radiation in near tissue equivalent materials and, through analysis of the counter data, determination of the effective quality factor and the dose equivalent.

TLD chip
A small block or crystal made of LiF used in the TLD.

TLD-600 - A TLD chip made from ^{6}Li (>95%) used to detect neutrons.

TLD-700 - A TLD chip made from ^{7}Li (>99.9%) used to detect photon and beta radiation.

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. The solid chips are sometimes called crystals.

whole-body dose
Commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1,000 mg/cm^{2}); also used to refer to the dose recorded.

X-ray
Ionizing electromagnetic radiation of extranuclear origin.