ORAU TEAM
Dose Reconstruction
Project for NIOSH

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
08/24/2004	00	New technical basis document for the Paducah Gaseous Diffusion Plant – Occupational External Dose. First approved issue. Initiated by Jay J. Maisler.
03/29/2005	01	Approved issue of Revision 01. Initiated by Jay J. Maisler.
08/30/2006	02	Revision initiated to address comments from earlier unresolved comments received on TBD. Approved issue of Revision 02. Incorporates additional information obtained through additional data capture for the Paducah Gaseous Diffusion Plant. Incorporates revised standard language into Purpose section and adds a Scope section. This revision addresses comments from the Worker Outreach meeting with the United Steelworkers Local 5-550 and Security, Police, Fire Professionals of America Local 111 held on February 10, 2005. Constitutes a total rewrite of the document. This revision results in a reduction in assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Paul A. Szalinski.
04/04/2007	03	Approved Revision 03 initiated to incorporate Attributions and Annotations section. Constitutes a total rewrite of the document. This revision results in no change to the assigned dose and no PER is required. No further changes occurred as a result of formal internal review. Training required: As determined by the Task Manager. Initiated by Daniel S. Mantooth.
08/24/2012	04	This document was predominantly revised to address SC&A concerns about the document as identified in SC&A-TR-TASK1-0016, *Review of the NIOSH Site Profile for the Paducah Gaseous Diffusion Plant.* Wording was revised throughout the document to add clarity. Acronyms and Abbreviations, References, and Glossary Sections have been updated. Table 6-1 has been revised to change four-element film to two-element film from 1953 through July 1, 1960, and to add an annual badge exchange option beginning in 1989. Building start dates in Table 6-3 have been clarified. Guidance for assigning neutron dose has been revised. Guidance for ^{99}Tc exposure has been added as Attachment A. Guidance for coworker dose assignment based on ORAUT-OTIB-0031 has been incorporated into this TBD as Attachment B. The application of construction trade worker doses has been added. Guidance for skin dose assignment based on ORAUT-OTIB-0017 has been incorporated into this TBD as Attachment C. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jodie L. Phillips.

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
02/13/2017	05	Revision initiated to update direction in Sections 6.3.4.2 and 6.4.1 to apply exposure to organ dose conversion factors to recorded deep dose results for workers for the period through 1988. For 1989 and after, reported deep dose values are considered equivalent to *Hp(10)*. Incorporates formal internal and NIOSH review comments. Added dose conversion factor for the skin in Attachment C. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jodie L. Phillips.

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ACRONYMS AND ABBREVIATIONS

AWE	atomic weapons employer
cm	centimeter
cpm	counts per minute
CTW	construction trade worker
d	day
DCF	dose conversion factor
DOE	U. S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DOL	U.S. Department of Labor
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ft	foot
HEU	highly enriched uranium
$Hp(d)$	personal dose equivalent at depth d in millimeters in tissue
$Hp(0.07)$	personal dose equivalent at 0.07 millimeters depth in tissue
$Hp(10)$	personal dose equivalent at 10 millimeters depth in tissue
hr	hour
IARC	International Agency for Research on Cancer
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1 thousand electron-volts
LOD	limit of detection
MDL	minimum detection level
MED	Manhattan Engineer District
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
min	minute
mm	millimeter
mrem	millirem
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH-Division of Compensation and Analysis Claims Tracking System
NTA	nuclear track emulsion, type A
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
OW	open window
PER	program evaluation report
PGDP	Paducah Gaseous Diffusion Plant

POC	probability of causation
PORTS	Portsmouth Gaseous Diffusion Plant
PPE	personal protective equipment
QF	quality factor
RU	recycled uranium
S	shielded
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
TBD	technical basis document
TEPC	tissue-equivalent proportional counter
TLD	thermoluminescent dosimeter
TLND	thermoluminescent neutron dosimeter
U.S.C.	United States Code
yr	year
§	section

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 Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies). These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of the individual work required for each dose reconstruction.

In this document the word "facility" is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an "AWE facility" or a "DOE facility." The terms AWE and DOE facility are defined in sections 7384l(5) and (12) of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), respectively. An AWE facility means "a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling." 42 U.S.C. § 7384l(5). On the other hand, a DOE facility is defined 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 [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program);" and with regard to which DOE has or had a proprietary interest, or "entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services." 42 U.S.C. § 7384l(12). The Department of Energy (DOE) determines whether a site meets the statutory definition of an AWE facility and the Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Accordingly, a Part B claim for benefits must be based on an energy employee's eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility's designated time period and location (i.e., covered employee). After DOL determines that a claim meets the eligibility requirements under EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. Unlike the abovementioned statutory provisions on DOE facility definitions that contain specific descriptions or exclusions on facility designation, the statutory provision governing types of exposure to be included in dose reconstructions for DOE covered employees only requires that such exposures be incurred in the performance of duty. As such, NIOSH broadly construes radiation exposures incurred in the performance of duty to include all radiation exposures received as a condition of employment at covered DOE facilities in its dose reconstructions for covered employees. For covered employees at DOE facilities, individual dose reconstructions may also include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees (NIOSH 2010):

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

6.1.1 Purpose

The purpose of this technical basis document (TBD) is to provide technical data and other key information that may be used in the evaluation of external occupational dose for EEOICPA claimants who were employed at the Paducah Gaseous Diffusion Plant (PGDP).

6.1.2 Scope

PGDP workers, especially those during the peak production decades (1950s, 1960s, and 1970s), have been exposed to radiation types and energies associated with enrichment of natural and recycled uranium (RU). PGDP used facility and individual worker monitoring methods to measure and control radiation exposure to workers (UCC 1976). Before about July 1960, personnel dosimeters were not assigned to all workers (PACE and University of Utah 2000). Records of radiation dose to individuals who wore dosimeters are available beginning in 1953. Doses from these dosimeters were recorded at the time of measurement, routinely reviewed by PGDP operations and radiation safety personnel for compliance with radiation control limits, and routinely made available to individual workers. OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007), indicates that these represent the highest quality records for assessment and reconstruction of doses.

Initial radiation dosimetry practices were based on experience from several decades of radium and X-ray medical diagnostic and therapy applications. In general, these practices were well advanced at the start of the Manhattan Engineer District (MED) program to develop nuclear weapons, which began on August 13, 1942.

6.1.3 Special Exposure Cohort

PGDP is one of the original sites that was designated by Congress as part of the SEC under EEOICPA [42 U.S.C. § 7384l(14)]. This designation is as follows:

> *(A) The employee was so employed for a number of work days aggregating at least 250 work days before February 1, 1992, at a gaseous diffusion plant located in Paducah, Kentucky, Portsmouth, Ohio, or Oak Ridge, Tennessee, and, during such employment—*
>
> *(i) was monitored through the use of dosimetry badges for exposure at the plant of the external parts of employee's body to radiation; or*
>
> *(ii) worked in a job that had exposures comparable to a job that is or was monitored through the use of dosimetry badges.*

Dose reconstruction guidance in this TBD is presented to provide a technical basis for dose reconstructions for nonpresumptive cancers that are not covered in the SEC class through January 31, 1992. Dose reconstructions for individuals who were employed at PGDP before February 1, 1992, but who do not qualify for inclusion in the SEC, may be performed using this guidance as appropriate.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 6.9.

6.2 BASIS OF COMPARISON

Since the start of the MED on August 13, 1942, various radiation dose concepts and quantities have been used to measure and record occupational dose. The basis of comparison for reconstruction of dose 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 is noted as *Hp(0.07).* For penetrating radiation of significance to whole-body dose, *d* = 10 mm and is noted as *Hp(10).* Both *Hp(0.07)* and *Hp(10)* are the radiation quantities the International Commission on Radiation Units and Measurements (ICRU) has recommended for use as operational quantities for radiological protection (ICRU 1998). In addition, *Hp(0.07)* and *Hp(10)* are the radiation quantities the DOE Laboratory Accreditation Program (DOELAP) has used to accredit the Department's personnel dosimetry systems since the 1980s (DOE 1986). The International Agency for Research on Cancer (IARC) Three-Country Combined Study (Fix et al. 1997) and the IARC Collaborative Study (Thierry-Chef et al. 2002) selected *Hp(10)* as the quantity to assess error in historical recorded whole-body dose for workers in IARC nuclear worker epidemiologic studies. This TBD uses *Hp(10)* and *Hp(0.07)* as deep dose and shallow dose, respectively.

6.3 DOSE RECONSTRUCTION PARAMETERS

Examinations of beta, photon (X- and gamma rays), and neutron energies and geometries of exposure, and the characteristics of PGDP dosimeter responses, are crucial for assessment of the original recorded doses. Bias and uncertainty for current dosimetry systems are typically well documented (MMES 1994). The performance of current dosimeters can often be compared to the performance of dosimetry systems in the same, or highly similar, facilities or workplaces. In addition, current performance testing techniques can be applied to earlier dosimetry systems to achieve a consistent evaluation of all dosimetry systems. Dosimeter response characteristics for radiation types and energies in the workplace are crucial to the overall analysis of error in recorded dose.

Overall, accuracy and precision of the original recorded individual worker doses and their comparability to be considered in using NIOSH (2007) guidelines depend on the following factors (Fix et al. 1997):

- **Administrative practices** adopted by facilities to calculate and record personnel dose based on technical, administrative, and statutory compliance considerations;
- **Dosimetry technology**, including 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 methods of calibration to sources of exposure in the workplace; and
- **Workplace radiation fields** that could include mixed types of radiation, variations in exposure geometries, and environmental conditions.

The accuracy of PGDP worker doses has been the subject of DOE investigations (PACE and University of Utah 2000). An evaluation of the original recorded doses as available, combined with detailed examinations of workplace radiation fields, is the recommended option to provide the best estimate of *Hp(0.07)* for the shallow dose and *Hp(10)* for the deep dose for individual workers.

6.3.1 <u>Administrative Practices</u>

The PGDP radiation monitoring program has used portable instruments, contamination surveys, zone controls, and personnel dosimeters to measure exposure in the workplace (Ely et al. 1957; UCNC 1957a, 1957b; Becher and Baker 1964; UCC 1976; Affel et al. 1980). The program improved as better technology and more information became available. Results from personnel dosimeters were used to measure and record doses from external radiation exposure to PGDP workers. These dosimeters included one or more of the following:

- Personnel whole-body beta/photon dosimeters,
- Pocket ionization chamber dosimeters, and
- Personnel neutron dosimeters.

For low-energy beta radiation, the dosimeters were probably incapable of furnishing accurate doses in terms of *Hp(0.07)*. This TBD analysis does not include extremity doses, which were generally not assessed (PACE and University of Utah 2000).

In 1953, PGDP began using Oak Ridge National Laboratory (ORNL) to provide and process dosimeters (Baker ca. 1995). There is evidence that PGDP might have processed its own dosimeters for a period; a review of the limited documentation available indicated that practices were similar to those at ORNL and other major sites at that time (UCNC 1957a). PGDP used the ORNL services through the implementation of a newer film badge in 1961 for all Union Carbide Corporation – Nuclear Division (UCC-ND) facilities (ORAUT 2009). Beginning in 1980, the TLD system for UCC-ND facilities was issued by the Y-12 Plant laboratory (UCND 1980). Beginning in 1989, the Centralized External Dosimetry System (CEDS) operated by the Y-12 Plant laboratory was implemented. (ORAUT-TKBS-0014-6, ORAUT 2007a). The CEDS dosimetry system is DOELAP-accredited, so reported deep and shallow dose values can be treated as representative of *Hp(10*) and *Hp(0.07)* for dose reconstruction. Table 6-1 summarizes PGDP personnel beta/photon and neutron dosimeter characteristics [dosimeter type, exchange, minimum detection level (MDL), and potential missed annual dose]. ORNL, which was then the Clinton Laboratory, had based its dosimetry methods on the personnel beta/photon dosimeter design from the Metallurgical Laboratory at the University of Chicago (Pardue, Goldstein, and Wollan 1944).

The precise detection levels in Table 6-1 are difficult to estimate, particularly for older systems. Current PGDP commercial thermoluminescent dosimeter (TLD) system MDLs are identified in ORNL documentation (MMES 1994) based on a DOELAP-accredited laboratory testing protocol (DOE 1986). During earlier years, MDLs were subject to additional uncertainty because factors such as radiation field, film type, and processing, developing, and reading systems cannot now be tested (Thornton, Davis, and Gupton 1961). The estimates of film dosimeter MDLs in Table 6-1 were based on information from NIOSH (1993), Lalos (1989), Wilson et al. (1990), and site personnel. Examination of older records, when available, indicated that the *Hp(0.07)* MDL values were about 3 times those for *Hp(10)* for film. The current TLD MDLs were obtained from ORNL (MMES 1994). The film badge was replaced by the TLD in 1980 (UCC 1980). Parameters of the PGDP administrative practices significant to dose reconstruction involve policies to:

- Assign dosimeters to workers,

- Exchange dosimeters,

- Record notional dose (i.e., some identified value for lower dosed workers, often based on a small fraction of the regulatory limit),

- Estimate dose for missing or damaged dosimeters,

- Replace destroyed or missing records,

- Evaluate and record dose for incidents, and

- Obtain and record occupational dose to workers for other employer exposure.

PGDP policies appear to have been in place for all these parameters. From startup until July 1960, PGDP issued dosimeters to a limited number of individuals (PACE and University of Utah 2000). This population of monitored individuals represents those with the highest exposure potential. After July 1960, routine practice required the assignment of dosimeters to all workers who entered a controlled radiation area (BJC 2000). Dosimeters were exchanged on a routine schedule (UCNC 1957a; UCC 1977; DOE 2000a). For workers in some areas the frequency was monthly, but for the general population it was quarterly. Employees on the monthly exchange cycle were primarily involved in chemical processing, maintenance of chemical processing facilities, and uranium metal production (DOE 2000). All dosimeters were processed, and measured results were recorded and used to estimate dose.

Current administrative practices are generally available (MMES 1994), as is detailed information for each worker in the PGDP exposure history documentation. Summary documents provide information on historical practices at PGDP (PACE and University of Utah 2000; BJC 2000; UCNC 1957a; Affel et al. 1980; Baker ca. 1995).

Table 6-1. Dosimeter types, periods of use, exchange frequencies, MDLs, and potential annual missed doses.

***Hp(10)* beta/photon dosimeters**

Dosimeter	Period of use	Monitored population	Exchange frequency	Laboratory MDL(rem)[a]	Maximum annual missed dose equivalent (rem)[b]
Two-element film	1953 through 07/1960	Selected workers based on activities performed	Weekly (n = 50)	0.04	1.0
Four-element film	After 07/1960 through 1979	Workers in Buildings C-340, C-400, and C-410	Monthly (n = 12)	0.04	0.24
Four-element film	After 07/1960 through 1979	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n = 4)	0.04	0.08
Four-element film	After 07/1960 through 1979	Workers and visitors not likely to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.04	0.02
Harshaw two-chip TLD	1980 through 1988	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n = 4)	0.02	0.04
Harshaw two-chip TLD	1980 through 1988	Workers and visitors not likely to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.02	0.01
Harshaw four-chip TLD, 8800 series	1989 through present	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n = 4)	0.02	0.04
Harshaw four-chip TLD, 8800 series	1989 through present	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.02	0.01

_Hp(0.07)_ beta/photon dosimeters

Dosimeter	Period of use	Monitored population	Exchange frequency	Laboratory MDL(rem)[a]	Maximum annual missed dose equivalent (rem)[b]
Two-element film	1953 through 7/1960	Selected workers based on activities performed	Weekly (n = 50)	0.12	3.0
Four-element film	After 7/1960 through 1979	Workers in Buildings C-340, C-400, and C-410	Monthly (n =12)	0.12	0.72
Four-element film	After 7/1960 through 1979	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n = 4)	0.12	0.24
Four-element film	After 7/1960 through 1979	Workers and visitors not likely to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.12	0.06
Harshaw two-chip TLD	1980 through 1988	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n = 4)	0.03	0.06
Harshaw two-chip TLD	1980 through 1988	Workers and visitors not likely to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.03	0.015
Harshaw four-chip TLD, 8800 series	1989 through present	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n = 4)	0.02	0.04
Harshaw four-chip TLD, 8800 series	1989 through present	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.02	0.01

Neutron dosimeters[c]

Dosimeter	Period of use	Monitored population	Exchange frequency	Laboratory MDL(rem)[a]	Maximum annual missed dose equivalent (rem)[b]
Harshaw TLND	1998 through present	Selected workers based on activities performed	Quarterly (n = 4)	0.015	0.03

a. Estimated film dosimeter detection levels based on NIOSH (1993), Lalos (1989), and Wilson et al. (1990). TLD detection levels from MMES (1994) and personal communication with site personnel.
b. Maximum annual missed dose (NIOSH 2007).
c. The potential annual missed dose based on laboratory irradiations is not applicable to workplace missed neutron dose.

6.3.2 Dosimetry Technology

PGDP dosimetry methods evolved with the development of improved technology and better understanding of complex radiation fields. The adequacy of dosimetry methods to measure radiation dose accurately is determined from radiation type, energy, exposure geometry, and other factors described in this section. The dosimeter exchange frequency gradually lengthened, corresponding in general to the period of regulatory dose controls.

6.3.2.1 Beta/Photon Dosimeters

PGDP has historically used personnel dosimetry services from ORNL or the Y-12 Plant laboratory. (UCND 1980) In 1945, ORNL implemented the beta/gamma film dosimeter design, which was developed originally at the Metallurgical Laboratory at the University of Chicago (Pardue, Goldstein, and Wollan 1944). ORNL followed a research and development process that led to gradual upgrades in dosimetry capabilities for complex radiation fields (Thornton, Davis, and Gupton 1961). Other DOE sites followed this evolution in dosimetry capabilities, which led to site-specific multielement film and TLD systems.

Figure 6-1 shows the energy response characteristics of the PGDP beta/gamma dosimeters based on the essentially identical two-element film dosimeter that was designed at the University of Chicago and used at the Hanford Site (as well as ORNL, Los Alamos National Laboratory, and probably other MED sites).

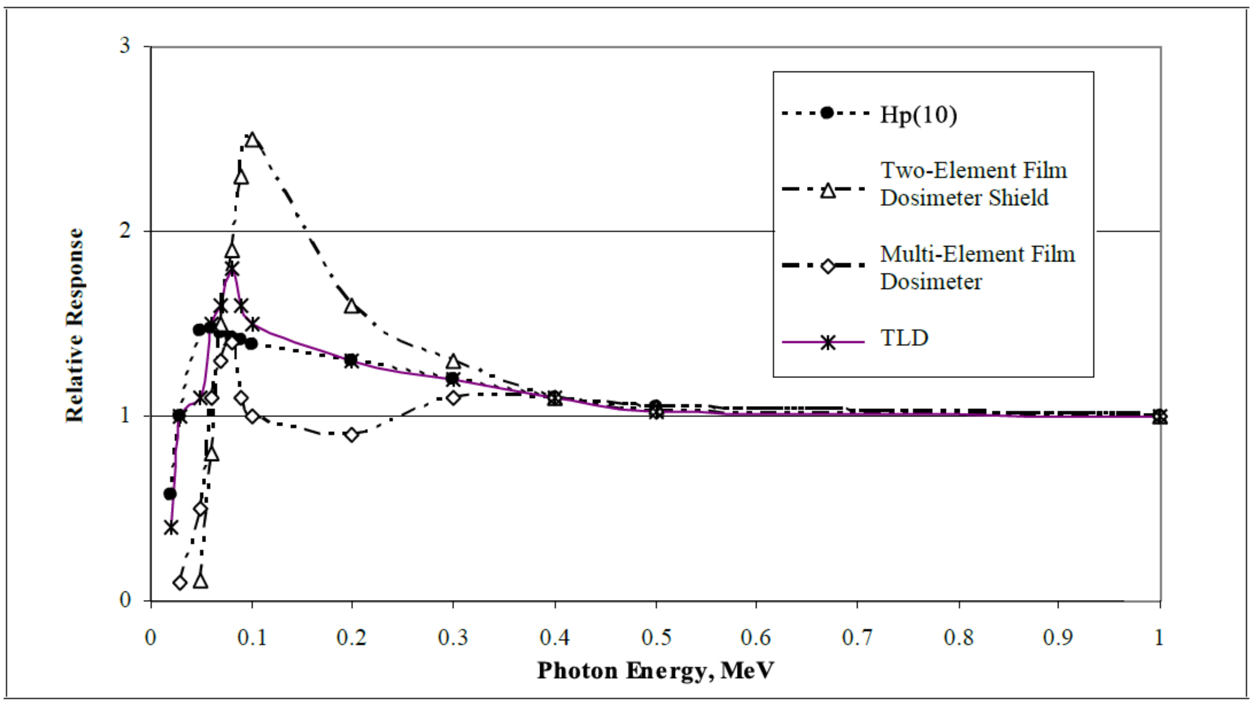

Figure 6-1. Estimated dosimeter photon response characteristics (Wilson, et al. 1990).

In addition, Figure 6-1 shows the *Hp(10)* response. Further, the figure shows the energy response of Hanford multielement film and TLDs (Wilson et al. 1990). The curve labeled “Two-Element Film Shield” represents ORNL dosimeters from 1945 through 1978. ORNL used a multielement film dosimeter after 1953 (Thornton, Davis, and Gupton 1961), but processed photon response as it did for the two-element dosimeter and used the same shielding as that in the two-element dosimeter. The figure shows that the two-element dosimeter overresponded in relation to *Hp(10)* from 0.05 to 0.3 MeV, followed *Hp(10)* for higher energies, and underresponded for lower energies. It also shows that TLDs are capable of following *Hp(10)* over the energy range of interest. The majority of PGDP worker photon dose has come from handling uranium of low enrichment. Therefore, the photon energy spectrum has been almost entirely in the range from 30 to 250 keV (Shleien, Slaback, and Birky 1998).

The nonpenetrating response of the two-element dosimeter was calculated as the difference between the open window (OW) and shielded (S) portions of the film based on a uranium calibration. The two-element dosimeter workplace nonpenetrating (i.e., beta or shallow) dose response based on the uranium calibration should adequately represent *Hp(0.07)* or at least be favorable to claimants because of the significant overresponse of the OW portion of the film to lower energy photons that could have been present (Wilson et al. 1990). The multielement film dosimeters and TLDs, which were also calibrated to uranium slabs, had the ability to correct more accurately for mixed photon and beta radiation (Wilson et al. 1990).

6.3.2.2 Neutron Dosimeters

Dosimeters at PGDP historically had a neutron-sensitive element that was processed on request. From 1989 to 1998, this capability was provided by a TLD that contained a ^{6}LiF chip, which is very responsive to low-energy neutrons. There is no indication of recorded neutron doses for PGDP workers wearing either of these dosimeters [1]. The use of commercial Harshaw thermoluminescent neutron dosimeters (TLNDs) to assess neutron dose routinely (along with deep and shallow dose) began in 1998. ORNL has provided the dosimeters and associated services. The albedo dosimeter has been worn with a belt to minimize distance from the worker's body, which optimizes the albedo effect for which the dosimeter is calibrated [2]. Before 1998, the beta/photon badge assembly contained a neutron-sensitive element (NTA; Eastman Kodak Type 2 film) with an energy threshold of about 0.5 MeV. This element was processed only when requested. A review of data does not indicate the assignment of neutron dose before 1998.

The quality factors (QFs) for neutrons have changed significantly over time. In current regulations, QFs that are used to convert radiation dose (millirad) to dose equivalent (millirem) are based on International Commission on Radiological Protection (ICRP) Publication 38 (ICRP 1983). The most current QFs from ICRP Publication 60 (1991) are about 2 times higher than the ICRP Publication 38 (ICRP 1983) values: therefore, an adjustment of a factor of 2 is necessary.

Average neutron energy has been less than 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 neutron energy from depleted and natural uranium cylinders has ranged from 210 to 360 keV (Cardarelli 1997, p. 9). Process-created unmoderated and deuterium (water) ^{252}Cf neutrons were between 1,306 and 1,403 keV. Therefore, assuming that 100% of the neutron doses at PGDP were delivered by neutrons in the 0.1-to-2-MeV energy range is favorable to claimants.

6.3.3 Calibration

Potential error in recorded dose is dependent on:

- Dosimetry technology response characteristics to each radiation type, energy, and geometry;
- Calibration methodology; and
- Extent of similarity between the radiation fields for calibration and those in the workplace.

The potential error is much greater for dosimeters with significant variations in response, such as film dosimeters for low-energy photon radiation and the nuclear track emulsion (NTA) and TLNDs for neutron radiation [4].

6.3.3.1 Beta/Photon Dosimeters

The beta/photon film dosimeters at PGDP were calibrated to ^{226}Ra until 1980 when the calibration source changed to ^{137}Cs (ORAUT 2007a). The calibration to both ^{226}Ra and ^{137}Cs was free in air (no phantom) until the DOELAP procedures adopted in 1986 required phantoms (DOE 1986). *Hp(10)* is defined with a phantom, in particular the ICRU slab phantom, which is a conservative practical definition of anterior-posterior whole-body dose to the standard ICRU spherical phantom (ICRU 1998).

Introduction of on-phantom calibration of film dosimeters, and the replacement of ^{226}Ra by ^{137}Cs as the calibration source, changed the relationship between recorded dose and *Hp(10)*. In addition to registration of the additional backscattered radiation, the generally lower energy photon spectrum from ^{226}Ra in comparison with that of ^{137}Cs (662 keV) gave a greater optical density for the same dose during calibration (Figure 6-1). In contrast, the effect of backscatter is to overestimate dose, and calibration with ^{226}Ra tends to underestimate the dose in relation to calibration with ^{137}Cs.

In the 1980s, studies at a number of laboratories assessed changes from the on-phantom calibration mandated by the DOELAP testing criteria (Fix et al. 1982; Wilson 1987; Wilson et al. 1990; Taylor et al. 1995). While not exactly the same at all sites, most film dosimeters, like those at PGDP, had common features due to their evolution from the original work of Pardue, Goldstein, and Wollan (1944). The early badges were calibrated to exposure in free air. Laboratory tests at the Hanford Site showed 8% and 4% increases in dosimeter response for on-phantom exposures using ^{226}Ra and ^{137}Cs, respectively (Fix et al. 1982). With free-air calibration, the exposure to the wearer tends to be overestimated by this amount, which is assumed to be similar for Paducah. Tests at the Savannah River Site, on the other hand, indicated that film badge doses underestimated *Hp(10)* by 11.9% before 1986 and by 3.9% in 1986 (Taylor et al. 1995). Lacking site-specific data for PGDP, this TBD recommends the use of exposure-to-organ dose conversion factors (DCFs) in Appendix B of OCAS-IG-001 (NIOSH 2007) for dose reconstruction at PGDP with no numerical adjustment to the recorded doses; this procedure should be favorable to claimants (Fix et al. 1982). It allows for an overestimate of exposure, as assessed in the Hanford studies, that should be sufficient to offset effects due to the calibration source if they are in the opposite direction.

For a number of years, ORNL used uranium beta as well as ^{226}Ra gamma calibration curves to interpret film densities (Thornton, Davis, and Gupton 1961). The ratio of beta-to-gamma responses was tested in several ways. Films wrapped in a 7-mg/cm^2 absorber were placed in contact with a slab of natural uranium. The densities per rad were nearly the same as those produced from ^{226}Ra gamma rays measured behind a cadmium filter. In addition, stacks of film were exposed on a uranium surface, and the densities at various depths were used to extrapolate to the value for a depth of 7 mg/cm^2. This value was nearly equal to that produced by the same dose from ^{226}Ra photons behind the cadmium filter. Therefore, for beta radiation from natural uranium, the density produced per rad in film was equal to the density produced per rad behind the cadmium filter by ^{226}Ra gamma rays. Analysts concluded that, for routine personnel dosimetry, film was equally sensitive to beta and gamma radiations. Because the film badge had a minimum absorber thickness of 80 mg/cm^2 between the film and the source, the effective beta energy is necessary to interpret the film density in terms of *Hp(0.07)*. The radiation was routinely treated as 1.7-MeV beta particles from uranium, which are about 40% absorbed in 80 mg/cm^2 (Thornton, Davis, and Gupton 1961). Therefore, this determination of beta dose was specific to uranium.

6.3.3.2 Neutron Dosimeters

Calibration of neutron dosimeters for PGDP was appropriate for the work locations (MMES 1994). Dosimeter response was characterized in a manner that would represent the workplace (MMES 1994). Reference dosimetry for these measurements was evaluated with tissue-equivalent proportional counters (TEPCs). TEPCs provide an absolute measure of absorbed dose in a tissue-like material and, with an appropriate algorithm, an estimate of the neutron QF (Scherpelz and Murphy 1995). The basis for the calibration factor was developed using data from a room at the Y-12 Plant that was used to store an array of small canisters of UF_4. Measurements were made with Bonner spheres at the same location. The average QF was 11, and the average energy range was 0.6 to 1.4 MeV (Soldat et al. 1990).

In 1989, field measurements for neutron flux were made by Pacific Northwest National Laboratory representatives at the end row of the cylinder yard at the K-25 Plant. The measurements were completed with a TEPC and a phantom with TLDs approximately 4 ft from the outside of a cylinder; the phantom was near the center of the cylinder’s length. The results were evaluated qualitatively because the dose rate was low and an appropriate power supply was not available. The calibration factors were similar to those in Y-12 Building 9212 in the UF_4 storage area container array and confirmed the appropriateness of these values (Soldat et al. 1990). These calibration factors apply to the PGDP TLNDs (MMES 1994).

6.3.4 Workplace Radiation Fields

6.3.4.1 Beta/Photon Fields

PGDP operations have been characterized by the relatively low-level external beta and photon radiation fields associated with uranium in feed materials, products, wastes, and contaminated equipment and systems. Processed RU was present with natural, depleted, and enriched (up to 2% ^{235}U by weight) abundances. (Section 6.3.4.3 describes potential sources for neutron exposure.)

Table 6-2 summarizes the major sources of external radiation throughout PGDP operations (PACE and University of Utah 2000). The photon energy range of principal interest is 30 to 250 keV. Handling uranium material of these types did not, in general, produce areas with significantly elevated photon radiation.

The major facilities and associated activities at PGDP have been (BJC 2000):

- C-331, C-333, C-335, and C-337 – gaseous diffusion process buildings,
- C-410/420 – UF_6 Feed Plant,
- C-310 – Purge and Product Withdrawal Building,
- C-315 – Surge and Tails Withdrawal Building,
- C-340 – Metals Plant,
- C-400 – Decontamination and Cleaning Building, and
- C-720 – Maintenance Building.

The buildings with the greatest potential for elevated direct radiation levels were C-340, C-410, C-420, and the cascade buildings (PACE and University of Utah 2000). From 1952 to approximately 1980, the major sites of potential exposure to radioactive material were buildings involved in the conversion of UO_3 powder to enriched UF_6 in solid or gaseous form, UF_4 and uranium metals recovery operations, and the decontamination building. Feed and enrichment operations were in Buildings C-410, C-420, C-331, C-333, C-335, C-337, C-310, and C-315, while UF_4 and uranium recovery were in Building C-340 (UCNC 1957b). The decontamination operation was in Building C-400. The oxide conversion building, C-420, was where UO_3 powder (clean or recycled) was received and converted to UF_4.

From Building C-420, material went to Building C-410, the feed plant, for conversion to UF_6. Last, UF_6 was processed through the cascade buildings (C-331, C-333, C-335, and C-337). Enriched UF_6 was withdrawn in Building C-310, the product withdrawal building, while depleted UF_6 was removed in Building C-315, the tails withdrawal building. Radiation surveys were performed near the UF_6 cylinders to evaluate the potential for exposure to personnel working adjacent to the shipping containers and area exposure rates in the cylinder yards (McDougal 1980; Frazee 1982; Mason 1986). Table 6-3 lists the principal buildings, sources for external dose, and periods of operation.

Table 6-2. Major radiation sources, half-lives, energies (MeV), and abundances (%).

Nuclide	Source	Half-life	Alpha	Beta (maximum)	Gamma
U-238	Primary U isotope	4.51E+09 yr	4.15 (21%), 4.20 (79%)	Not applicable	Not applicable
U-235	Primary U isotope	7.1E+08 yr	4.21 (6%), 4.37 (17%), 4.40 (55%), 4.60 (5%)	Not applicable	0.144 (11%), 0.163 (5%), 0.186 (57%), 0.205 (5%)
U-234	Primary U isotope	2.47E+05 yr	4.72 (28%), 4.77 (72%)	Not applicable	0.053 (0.12%)
Th-234	Decay product	24.1 d	Not applicable	0.103 (21%), 0.193 (79%)	0.013 (9.8%), 0.063 (3.5%), 0.092 (3%), 0.093 (4%)
Pa-234m	Decay product	1.17 min	Not applicable	2.29 (98%)	0.765 (0.3%), 1.001 (0.60%)
Th-231	Decay product	25.5 hr	Not applicable	0.206 (13%), 0.287 (12%), 0.288 (37%), 0.305 (35%)	0.026 (2%), 0.084 (10%)
Tc-99	Impurities from RU	2.12E+05 yr	Not applicable	0.294 (100%)	0

Table 6-3. Buildings and periods of operation.

Site facilities	Source for external dose	Period
C-310 Purge and Product Withdrawal	UF_6 process equipment and cylinders	1953–present
C-315 Surge and Tails Withdrawal	UF_6 process equipment and cylinders	1953–present
C-331, C-333 Gaseous Diffusion Process Buildings	UF_6 process equipment and cylinders	1952–1964
		1969–1970
		1972–1976
C-335, C-337 Gaseous Diffusion Process Buildings	UF_6 process equipment and cylinders	1954–1964
		1969–1970
		1972–1976
C-340 Reduction and Metals Facility	Process equipment, contaminated floors	1957–1962
		1967–1977
C-400 Decontamination and Cleaning Buildings	UF_6 process equipment and cylinders	1952–1990
C-410 UF_6 Feed Plant and C-420 Oxide Conversion Plant	Process equipment, contaminated floors	1953–1964
		1968–1977
C-415 Feed Plant Storage Building	Radioactive source storage area	1953–1977
C-745 A-V Cylinder Yards	UF_6 cylinders	1953 (estimated)–present

PGDP also processed RU. The feed material 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 (PACE and University of Utah 2000).

However, some routine chemical processes would concentrate them (PACE and University of Utah 2000). From an external dose standpoint, the most significant impurity in RU is the pure beta emitter ^{99}Tc, which tends to deposit in enrichment equipment and "pocket" in the higher sections of the diffusion cascade (DOE 2001). Technetium-99 was also 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 badges nor TLDs efficiently detect them, particularly in the presence of uranium. Clothing and gloves provide adequate shielding. Skin contamination is the only credible scenario in which significant shallow dose could occur from ^{99}Tc.

6.3.4.2 Workplace Beta/Photon Dosimeter Response

PGDP has historically used personnel dosimeter services from ORNL or the Y-12 Plant laboratory. (UCND 1980) Essentially all radiological work areas involved photon and beta radiation characteristic of operations with low-enrichment uranium.

All personnel dosimetry PGDP (from ORNL) used over its history included suitably filtered elements for the determination of deep dose. The ORNL film dosimeters, which were calibrated in terms of exposure from ^{226}Ra photons, show either good agreement with or an overresponse to *Hp(10)* (see Figure 6-1). However, because dosimeters were calibrated in terms of exposure over most of ORNL history, dose reconstructors should apply exposure to organ DCFs to the recorded deep dose results for the period through 1988 (i.e., for the period before DOELAP accreditation). From 1989 to the present, reported deep dose values are considered equivalent to *Hp(10)* (ORAUT 2007a, 2007b)

6.3.4.3 Neutron Fields

While neutron radiation has occurred in some areas at PGDP, measured levels have been low. There are no identified locations where measurable neutron dose was encountered (MMES 1994). This is confirmed by studies of neutron fields at gaseous diffusion plants (Scherpelz and Murphy 1995; Cardarelli 1997). Cylinder yards, feed and withdrawal areas, and locations where uranium forms deposits in the cascade have been investigated (Cardarelli 1997). These studies identified the storage cylinders, which contained either depleted UF_6 (tails) or enriched UF_6 (product), as areas where neutron fields could represent an exposure hazard. Estimates of dose equivalent rates ranged from 0.007 to 0.34 mrem/hr with QFs from 7 to 10. Radiation measurements indicated that the neutron flux increased as a function of uranium enrichment; neutron flux increased from 0.2 mrem/hr for cylinders with as much as 5% enrichment to 4 mrem/hr on contact with 97% enrichment (DOE 2001). A representative average value is 0.2 mrem/hr based on a QF of about 10 (Scherpelz and Murphy 1995; Cardarelli 1997). Estimates of average neutron energies ranged from 0.25 to 0.56 MeV (Scherpelz and Murphy 1995). Neutron monitoring of individuals during a UF_6 cylinder-painting project (BJC 1999) indicated a neutron-to-photon dose equivalent ratio of approximately 1 to 5 based on a QF of 10 [5].

Cylinders of highly enriched (93% to 96%) uranium (HEU) were measured with a TEPC on a phantom about 24 in. from the cylinders (Soldat and Tanner 1992). The dose equivalent from the cylinders was about 0.8 mrem/hr with a total dose equivalent 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 dose equivalent rate of 0.5 mrem/hr.

The solid lines in Figure 6-2 show the calculated energy spectrum from the multisphere detectors (Bonner spheres). Table 6-4 lists dose fractions for the neutron energy groups (indicated by the dashed lines in Figure 6-2). The dose fractions for the lower (less-than-10-keV) and intermediate (10- to 100-keV) energy neutron groups were about 47% of the total dose from the measurements (ORAUT 2007a).

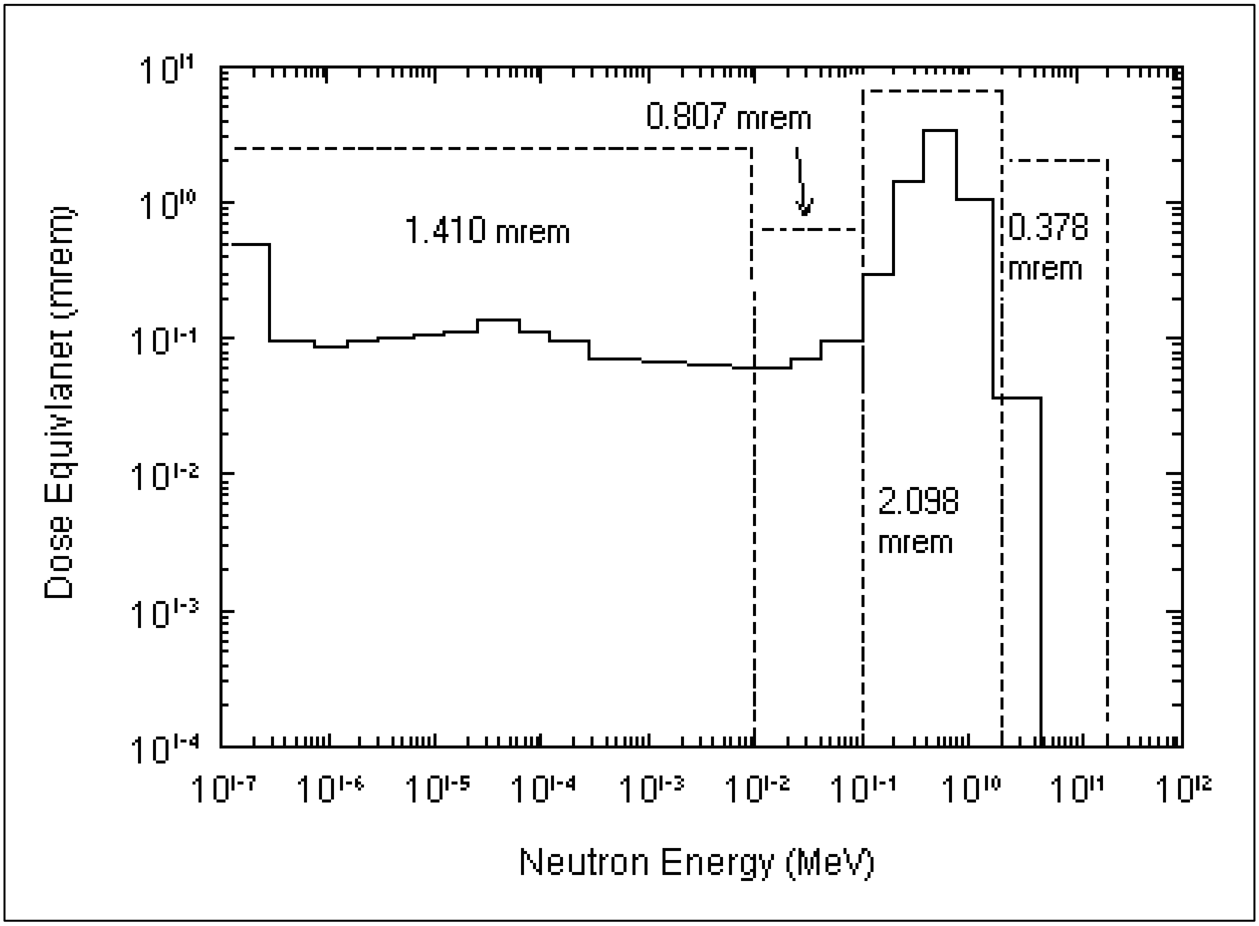

Figure 6-2. Results of neutron spectrum measurements about 24 in. in front of 93%–96% HEU cylinders (Soldat and Tanner 1992).

Table 6-4. Dose fractions for PORTS HEU storage vault in Building 345 near an unshielded ^{252}Cf source.

Neutron energy group	Actual fractions	Fractions favorable to claimants
<10 keV	0.300	0.300
10–100 keV	0.172	0.610
0.1–2 MeV	0.447	0.610
2–20 MeV	0.081	0.081

Exposure to low-enriched UF_6 (less than 5%) results in a lower neutron flux than that from highly enriched UF_6 (greater than 97%) as surveyed at the Portsmouth Gaseous Diffusion Plant (PORTS) by Soldat and Tanner (1992). The dose fractions in Table 6-4 are favorable to claimants (Soldat and Tanner 1992).

A neutron study in 1990 at ORNL and the Y-12 Plant was the only definitive study of neutron energy spectra documented over the history of PGDP (Soldat et al. 1990). The energy spectra are assumed valid for the earlier years due to the presence of enriched uranium.

6.3.4.4 Workplace Neutron Dosimeter Response

Quantitative monitoring for neutron dose began at PGDP in 1998. TLNDs were used in conjunction with appropriate work field calibration factors. Before 1998, the beta/photon badge assembly contained a neutron-sensitive element (NTA; Eastman Kodak Type 2 film) with an energy threshold of about 0.5 MeV. This element was processed only when requested. A review of data does not indicate the assignment of neutron dose before 1998.

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Photon Dose

Recorded doses varied in reporting units depending on regulatory requirements and dose definitions (national and international). The DOE reporting unit is the millirem, a unit of dose equivalent. The international unit of dose equivalent is the millisievert, which is equivalent to 100 mrem. Dose reconstructors should apply exposure to organ DCFs to deep dose results for PGDP workers for the period through 1988. From 1989 to the present, reported deep dose values are considered equivalent to *Hp(10)* (ORAUT 2007b).

6.4.2 Nonpenetrating Dose

The early film dosimeters were calibrated to uranium for nonpenetrating radiation. No numerical adjustment of recorded shallow doses is recommended. Incident reports are a possible source that dose reconstructors can consult for investigations of nonroutine beta exposures and dose assessment.

6.4.3 Neutron Dose

Measured neutron energies at PGDP are between 0.1 and 2.0 MeV, for which the ICRP Publication 60 radiation weighting factor is 20 (ICRP 1991). Therefore, dose reconstructors should multiply the reported neutron dose equivalent by the appropriate ICRP (1991) correction factor to be used for reconstruction (NIOSH 2007). Apply this factor to measured, missed, and unmonitored neutron doses.

6.5 MISSED DOSE

Missed deep and shallow doses have been examined for three groups of PGDP workers as follows:

- Group 1. A zero dose was recorded but the worker was not monitored (most workers from 1953 to July 1960).
- Group 2. A zero dose was recorded for the dosimeter system for any response less than the MDL.
- Group 3. There was no recorded dose because workers were not monitored or the dosimetry record is not available.

Neutron dose rates at PGDP were low (MMES 1994). Neutron dosimeters were not routinely assigned and doses were not recorded until about 1998. Neutron doses from before 1998 were based on a conservative calibration associated with a neutron-sensitive element in the beta/gamma dosimeter. Application of a neutron-to-gamma dose equivalent ratio of 1 to 5 appears to be a satisfactory option that is favorable to claimants because the photon dose is reliably measured. This ratio can be applied to selected work activities [6].

6.5.1 Estimating Missed and Unmonitored Photon Deep Dose

6.5.1.1 Non-Construction Trade Workers

Watson et al. (1994) examined methods to be considered when there is no recorded dose for a period during a working career (Group 1). In general, estimates of unmonitored dose can be made by using dose results for coworkers or the recorded dose before and after the period when they were not monitored. However, these situations require careful examination. The dose reconstructor should consider all reasonable methods and assign the most appropriate dose based on employee job descriptions and work locations. NIOSH (2007) cites several different models.

For Group 2, the missed dose for dosimeter results less than the MDL is particularly important for earlier years when MDLs were higher and dosimeter exchange was more frequent. NIOSH (2007) describes an acceptable estimate that is favorable to claimants of the maximum potential missed dose as one-half the MDL (or limit of detection, LOD) multiplied by the number of zero dose results. The right-hand column in Table 6-1 lists estimates of the annual missed dose for Group 2 at PGDP.

If it is definite that the employee was not a radiation worker, the unmonitored deep dose for that period can be assigned as the onsite ambient dose.

Otherwise, dose reconstructors should treat an individual in Group 1 or 3 as a radiation worker, then approach the unmonitored deep dose in two ways. First, consider the same assignment of missed dose as that for Group 2 (from the right-hand column of Table 6-1). However, for 1953 through July 1960, with the frequent (weekly) dosimeter exchange and relatively large MDL, the resulting implied annual missed dose of 1 rem is probably unrealistically large for many unmonitored workers in Groups 1 and 3. Figure 6-3 shows the distribution of individual annual deep dose equivalent for monitored workers from 1953 to 1974 (Baker ca. 1995). Few of these individuals received as much as 1 rem in a year.

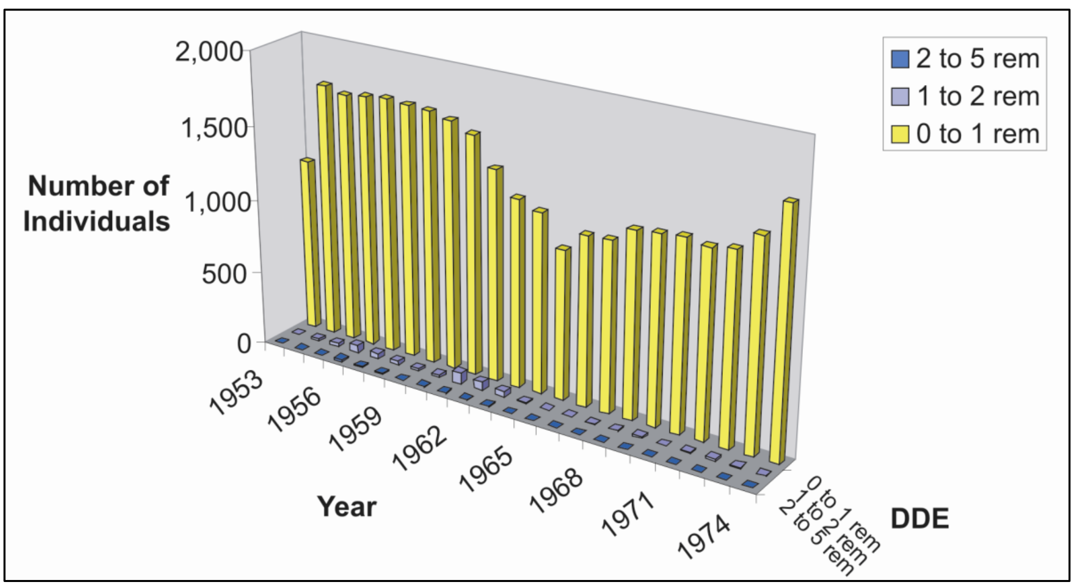

Figure 6-3. Historical distribution of deep dose equivalent (Baker ca. 1995).

An alternative approach for Group 1 or 3 is to base the unmonitored dose estimate on exposure data compiled for monitored PGDP workers. ORAUT-OTIB-0020, *Use of Coworker Dosimetry Data for External Dose Assignment* (ORAUT 2011), provides general instructions to evaluate the measured and missed doses for monitored PGDP workers to arrive at a dose that is favorable to claimants to be assigned to unmonitored workers. Attachment B contains the details of the evaluation of PGDP coworker dose to be assigned to unmonitored workers. These measured doses include an analysis of the missed dose, which is particularly significant for the earlier years with higher LODs and frequent dosimeter exchanges. Table B-2 provides the 50th- and 95th-percentile coworker doses.

6.5.1.2 Construction Trade Workers

Measured doses to construction trade workers (CTWs) have been increased to account for uncertainty for reasons described in ORAUT-OTIB-0052, *Technical Information Bulletin: Parameters to Consider When Processing Claims for Construction Trade Workers* (ORAUT 2014). For extended employment periods without a measured dose, consideration about whether to assign an unmonitored dose using the coworker doses in Attachment B is necessary. In this case, the measured coworker penetrating annual dose has been multiplied by a factor of 1.4 (ORAUT 2014), and the missed dose was determined using NIOSH (2007) guidance. Table B-3 lists the 50th- and 95th-percentile doses for CTWs.

6.5.2 <u>Estimating Missed and Unmonitored Shallow Dose</u>

The procedure for assessing missed and unmonitored shallow dose is similar to that for missed deep dose.

For Group 2, the last column of Table 6-1 lists the missed annual shallow dose equivalent in keeping with the MDL/2 method of evaluation. Attachment B contains guidance on determining the reconstructed skin dose. Figure 6-4 shows the historical data for the distribution of shallow dose equivalent for monitored workers (Baker ca. 1995). In comparison with Figure 6-4, the Table 6-1 values for annual missed shallow dose for Group 2 is favorable to claimants.

For nonradiological workers in Groups 1 and 3, the unmonitored shallow dose can be assigned as the environmental dose. Dose reconstructors should regard other individuals in these groups as radiation workers and consider the same estimate as that for Group 2. As an alternative, use Attachment B, Table B-2. Significant nonroutine beta doses, such as from skin contamination events (particularly during ^{99}Tc recovery and removal), could be addressed in specific incidence reports. Attachment A provides guidance.

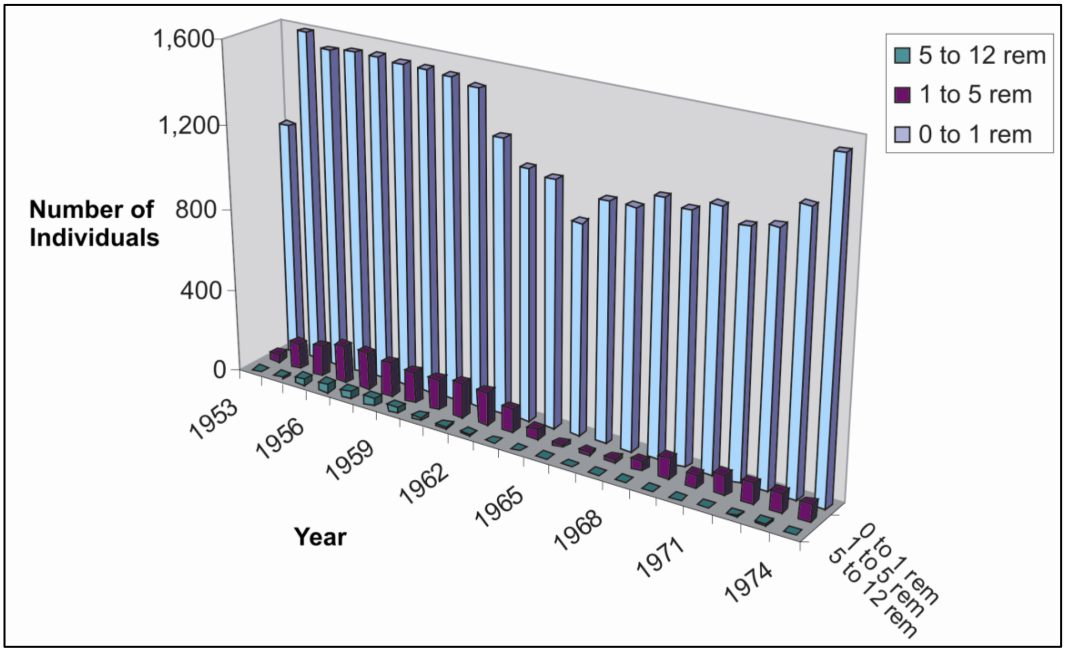

Figure 6-4. Historical distribution of shallow dose equivalent (Baker ca. 1995).

6.5.3 Estimating Missed and Unmonitored Neutron Dose

Dose reconstructors should add a neutron component to annual doses prior to 1998. Table 6-5 lists the criteria for assigning missed and unmonitored neutron dose. The affected radiological areas include, but are not limited to, cascade facilities, feed and production withdrawal areas, oxide conversion facilities, feed manufacturing facilities, decontamination or cleaning facilities, cylinder yards, and neutron source storage areas.

Table 6-5. Missed and unmonitored neutron dose assignment (BJC 1999).

Description	Neutron dose assignment
Unmonitored employee – all radiological areas	Assign neutron dose based on coworker dose. Apply a neutron-to-photon ratio of 0.2 for dose equivalent.
Monitored (photon and electron) employee	Assign neutron dose if positive photon results are recorded. Apply a neutron-to-photon ratio of 0.2 for dose equivalent.
Monitored (photon and electron) employee having no dosimetry results greater than half the LOD	Assign neutron dose based on the missed dose. Apply a neutron-to-photon ratio of 0.2 for dose equivalent.

Beginning in 1998, employees with neutron exposure were monitored, and these dosimetry records should be used for dose assignment. The neutron dose equivalent should be multiplied by a factor of 2 as previously discussed in Section 6.4.3.

6.6 UNCERTAINTY

PGDP has historically used ORNL personnel dosimeter services. ORNL has assessed the standard error in the recorded film-badge dose as ±30% for photons of all energies (ORAUT 2007b). The standard error for beta dose is the same (or somewhat larger for unknown mixtures of beta/gamma dose). Therefore, the film badge dose uncertainty is 1.3. The uncertainty in the TLD dose is 1.15 (ORAUT 2007b), which is consistent with NIOSH (2007).

6.7 DOSE RECONSTRUCTION

As much as possible, dose reconstructors should base dose to individuals 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 actions for dose reconstructors:

- Consider dosimetry records that provide nonzero beta/photon values for *Hp(10)* and *Hp(0.07)* to be adequate. No numerical adjustment of the doses is required. Beta energies are greater than 15 keV and photon energies range from 30 to 250 keV.

- Assign missed dose to workers for whom dosimetry records provide zero beta/photon values for *Hp(10)* and *Hp(0.07)* based on MDL/2 times the number of zero results, as described in Sections 6.5.1 and 6.5.2 (NIOSH 2007).

- Assign only the missed environmental dose from the environmental TBD (ORAUT 2012b) if it is certain that the individual was not a radiation worker. However, individuals with no recorded dose might or might not have been radiological workers, so if there is uncertainty, estimate the missed dose as described in Section 6.5. No numerical adjustments to the missed dose are necessary.

- Multiply reported, missed, and unmonitored neutron dose equivalents by the appropriate ICRP (1991) correction factors.

- Base the assignment of missed and unmonitored neutron dose equivalent on a neutron-to-photon ratio of 0.2 for dose equivalent (BJC 1999) for years before 1998. Beginning in 1998, base the neutron assignment on the dosimetry records. Assign missed neutron dose using Table 6-1. Multiply the estimated neutron dose equivalent by 2 to adjust for ICRP (1991).

- Pay special attention to the possibility of skin contamination incidents for workers involved with ^{99}Tc recovery operations (Attachment A).

- See Section 6.6 for a discussion of uncertainty.

6.8 ORGAN DOSE

NIOSH (2007) discusses the conversion of measured doses to organ dose equivalent, and Appendix B of that document contains the appropriate DCFs for each organ, radiation type, and energy range based on the type of monitoring performed. In some cases, simplifying assumptions are appropriate [7].

6.9 ATTRIBUTIONS AND ANNOTATIONS

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

[1] Turner, James E. Oak Ridge Associated Universities (ORAU) Team. Consultant. 2003.
The reviewed records did not reveal recorded neutron doses for either dosimeter.

[2] Turner, James E. ORAU Team. Consultant. 2003.
Proximity of the albedo dosimeter is important for its response; standard practice ensured this.

[3] Turner, James E. ORAU Team. Consultant. 2003.
ICRP (1991) recommends a weighting factor of 20 for neutron energies between 0.1 and 2 MeV. Doses of record used a QF of 10; therefore, a factor of 2 correction is indicated.

[4] Turner, James E. ORAU Team. Consultant. 2003.
The importance of energy response to accurate measurement of dose equivalent is well known, and the response of the historical dosimeters is shown in Figure 6-1.

[5] Turner, James E. ORAU Team. Consultant. 2003.
The determination of a neutron-to-photon ratio for absorbed dose was based on a dose equivalent ratio that can be used to estimate neutron dose from photon measurements.

[6] Turner, James E. ORAU Team. Consultant. 2003.
Empirical neutron and photon worker dose equivalent data provide a basis from which neutron dose equivalent can be inferred from a better known photon dose (based on interpretation of BJC 1999).

[7] Turner, James E. ORAU Team. Consultant. 2003.
Appendix B of NIOSH (2007) contains tables for numerous organs. Some professional judgment is needed to fit particular conditions.

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GLOSSARY

absorbed dose

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

albedo dosimeter

Thermoluminescent dosimeter that measures the thermal, intermediate, and fast neutrons scattered and moderated by the body or a phantom from an incident fast neutron flux.

albedo effect

In relation to health physics, dosimeter response caused by the moderating and backscattering of neutron radiation by a human chest or a phantom.

alpha radiation

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

attenuation

Process by which absorption and scattering reduces the number of particles or photons passing through a body of matter.

background radiation

Radiation from cosmic sources, naturally occurring radioactive materials including naturally occurring radon, and global fallout from the testing of nuclear explosives. Background radiation does not include radiation from source, byproduct, or Special Nuclear Materials regulated by the U.S. Nuclear Regulatory Commission. The average individual exposure from background radiation is about 360 millirem per year.

beta radiation

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

Bonner sphere

See *multi-sphere neutron spectrometer*.

cascade

At PGDP, series of compressor, heat exchanger, control valve and motor, converter stages, and supporting piping arranged in stages, cells, and units that progressively increase the concentration of ^{235}U in a uranium hexafluoride (UF_6) feed. Enrichment occurs as UF_6 passes through semiporous barriers in the converter stage. These barriers allow the lighter ^{235}U molecules to pass through more easily, which results in a gas with a slightly higher percentage of ^{235}U (enriched) on one side of the barrier and a slightly lower percentage (depleted) on the other side. The enriched UF_6 gas flows toward the top of the cascade while the depleted UF_6 gas travels toward the bottom of the cascade.

curie (Ci)

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

deep dose equivalent [DDE, *Hd*, *Hp(10)*)]
Dose equivalent in units of rem or sievert for a 1-centimeter depth in tissue (1,000 milligrams per square centimeter). See *dose*.

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

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

dose conversion factor (DCF)
Multiplier for conversion of potential dose to the personal dose equivalent to the organ of interest (e.g., liver or colon). In relation to radiography, ratio of dose equivalent in tissue or organ to entrance kerma in air at the surface of the person being radiographed.

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

dosimeter
Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual. See *albedo dosimeter*, *film dosimeter*, *neutron film dosimeter*, *pocket ionization chamber*, *thermoluminescent dosimeter*, and *track-etch dosimeter*.

dosimetry
Measurement and calculation of internal and external radiation doses.

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

enrichment
Isotopic separation process that increases the percentage of a radionuclide in a given amount of material above natural levels. For uranium, enrichment increases the amount of ^{235}U in relation to ^{238}U. Along with the enriched uranium, this process results in uranium depleted in ^{235}U. At PGDP this involves a process that occurs as UF_6 passes through barriers in converters that allow isotopes of lower molecular weight to pass through.

external dose
Dose received from radiation emitted by sources outside the body.

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

film dosimeter

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

gamma radiation

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

gaseous diffusion plant

Facility where uranium hexafluoride (UF_6) gas is filtered to enrich the ^{235}U and separate it from ^{238}U. The process requires enormous amounts of electric power and results in an increase in ^{235}U enrichment from 1% to about 3%.

gray (Gy)

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

highly enriched uranium (HEU)

Uranium enriched to at least 20% ^{235}U for use as fissile material in nuclear weapons components and some reactor fuels. Also called high-enriched uranium.

ionizing radiation

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

limit of detection (LOD)

Minimum level at which a particular device can detect and quantify exposure or radiation. Also called lower limit of detection and detection limit or level. See *minimum detectable level*.

minimum detectable activity or amount (MDA)

Smallest amount (activity or mass) of an analyte in a sample that can be detected with a probability β of nondetection (Type II error) while accepting a probability α of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error). See *action level*, *decision level*, and *minimum reporting level*.

minimum detectable level (MDL)

See *minimum detectable activity*.

multi-sphere neutron spectrometer

Spectrometer that consists of a series of neutron-moderating spheres of tissue-equivalent material with a neutron detector in the middle of the respective spheres. Algorithms are used to calculate the neutron spectra.

neutron (n)

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

neutron film dosimeter

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

neutron radiation

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

nonpenetrating dose (NP, NPEN)

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

nuclear track emulsion, type A (NTA)

Film made by the Eastman Kodak Company that is sensitive to fast neutrons. The developed image has tracks caused by neutrons that become visible under oil immersion with about 1,000-power magnification. The number of tracks in a given area is a measure of the dose from that radiation.

occupational dose

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

on-phantom

Exposure of a dosimeter on a phantom to simulate the dosimeter’s response when worn on a person.

open window (OW)

Area of a film dosimeter that has little to no radiation shielding (e.g., only a holder and visible light protection). The open window measures nonpenetrating as well as penetrating dose, which minimizes the potential for beta radiation to contribute to the interpreted penetrating dose. See *film dosimeter*.

penetrating dose (PEN)

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

personal dose equivalent [*Hp(d)*]

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

phantom

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

photon

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

photon radiation

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

pocket ionization chamber (PIC)

Cylindrical monitoring device commonly clipped to the outer clothing of an individual to measure ionizing radiation. A PIC may be self-reading or require the use of an outside device to be able to read the dosimeter. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

probability of causation (POC)

For purposes of dose reconstruction for the Energy Employees Occupational Illness Compensation Program Act of 2000, the percent likelihood, at the 99th percentile, that a worker incurred a particular cancer from occupational exposure to radiation.

proton

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

quality factor (Q, QF)

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

rad

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

radiation

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

radioactivity

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

recycled uranium (RU)

Uranium first irradiated in a reactor, then recovered through chemical separation and purification. RU contains minor amounts of transuranic material (e.g., plutonium and neptunium) and fission products (e.g., technetium) or uranium products (e.g., ^{236}U) after purification. PGDP lists the isotopic activity ratios as:

Isotope	Activity fraction
^{234}U	0.8489
^{235}U	0.0120
^{236}U	0.1388
^{238}U	0.0003 or 0.0004 (both listed)

rem

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

rep

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

roentgen (R, sometimes r)

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

shallow absorbed dose (D_s)

Absorbed dose at a depth of 0.07 millimeters (7 milligrams per square centimeter) in a material of specified geometry and composition.

shallow dose equivalent [SDE, *Hs, Hp(0.07)*]

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

sievert (Sv)

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

skin dose

See *shallow dose equivalent*.

thermoluminescence

Property that causes a material to emit light as a result of heat.

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

thermoluminescent neutron dosimeter (TLND)
Thermoluminescent dosimeter for measurement of neutron dose.

tissue-equivalent proportional counter (TEPC)
Device that measures absorbed dose from neutron radiation in materials nearly equivalent to tissue. Analysis of the counter data determines the effective weighting factor and the dose equivalent for that radiation.

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

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

ATTACHMENT A
EXTERNAL EXPOSURE TO TECHNETIUM-99

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ATTACHMENT A
EXTERNAL EXPOSURE TO TECHNETIUM-99 (continued)

A.1 INTRODUCTION

This attachment provides guidance for the assignment of external dose from ^{99}Tc for employees of PGDP. Due to its nonpenetrating characteristics, combined with the routine use of personal protective equipment (PPE) by affected employees, the dose potential from ^{99}Tc is low. Information on the assignment of ^{99}Tc dose in this attachment is based on information about work location, job title, and job description together with shallow dose data that can be used by dose reconstructors to identify employees who could have been exposed to ^{99}Tc.

A.2 BACKGROUND

Technetium-99 is present at PGDP as a contaminant from the introduction of RU into the cascade at various times throughout site operations. It is a long-lived fission product with a radiological half-life of 213,000 years and is a pure beta emitter with average and maximum energies of 84.6 keV and 293.6 keV, respectively. Although it is difficult to detect due to its low-energy beta emission, this characteristic results in minimum potential for external dose. Studies have shown that the outer layer of skin affords significant protection to the germinal skin layers from ^{99}Tc, and the wearing of PPE such as coveralls and gloves provides further skin protection. In this way, the vast majority of radiation from ^{99}Tc is attenuated before it can interact with the body (ORAUT 2012b).

A.3 POTENTIAL FOR EXPOSURE

Operations procedures at PGDP required special precautions for working around significant quantities of ^{99}Tc, especially during plant maintenance and repairs on the upper cascade equipment. These included engineering controls, administrative controls, and the use of PPE. The control measures routinely used to protect workers from exposure to uranium and its progeny provided an even greater protection factor for exposure to ^{99}Tc (Saraceno 1981).

Specific work activities that could have resulted in exposure to ^{99}Tc included:

- Technetium recovery operations,
- Removal of equipment from the cascade for routine maintenance, and
- Removal and replacement of cascade equipment during the Cascade Improvement Program and Cascade Upgrade Program.

Table A-1 lists the facilities at PGDP with ^{99}Tc exposure potential.

Table A-1. Facilities with ^{99}Tc exposure potential.

Facility	Description
C-409	Stabilization Building
C-410	Feed Plant
C-420	Oxide Conversion Plant
C-331	Gaseous Diffusion Process Building
C-333	Gaseous Diffusion Process Building
C-335	Gaseous Diffusion Process Building
C-337	Gaseous Diffusion Process Building
C-310	Purge and Product Withdrawal Building
C-710	Analytical Laboratory
C-400	Decontamination and Cleaning Building
C-720	Maintenance Building

ATTACHMENT A
EXTERNAL EXPOSURE TO TECHNETIUM-99 (continued)

Employees with any of the job titles listed in Table A-2 could have had exposure to ^{99}Tc while working in the facilities listed in Table A-1. The highest exposure potential would have been to maintenance workers in the top purge cells and to those doing change-outs of trapping media near the top purge cells.

Table A-2. Job titles for workers with possible ^{99}Tc exposure.

Job title
Cascade worker/operator
Chemical operator
CTW
Decontamination and decommissioning worker
Feed plant operator
Maintenance mechanic
Radiological worker

A.4 MAGNITUDE OF EXPOSURE

Based on a review of the properties of ^{99}Tc and the routine controls that were in place, automatic assignment of ^{99}Tc dose due to skin contamination is not warranted. Guidance for assignment of ^{99}Tc skin contamination dose on a case-by-case basis is provided below. It is apparent, however, that external exposure to ^{99}Tc was unlikely to be measured by dosimetry due to its low-energy electron characteristics. In certain cases, as described below, an annual external dose assignment from^{99}Tc should be included in the dose estimate under EEOICPA.

Site evaluations at PGDP assessed the potential for an external exposure problem from ^{99}Tc recovery operations and found that the likelihood of high exposure was low due to the following reasons (Baker et al. 1978):

- Gloves were worn routinely for all operations involving the handling of containers.
- All material was transferred remotely from point to point, with one exception. For movement from one container to another, the transfer was done by pumping; the container was never dumped by hand.
- The solutions were dilute.
- Less than 20% of employee work time was spent at jobs with the potential to generate ^{99}Tc contamination.

Information about the magnitude of ^{99}Tc exposure is available in *Tc-99 Contamination* (Swinth 2004). Measured contamination exposure levels at PGDP ranging from 10,000 to 335,849 cpm/100 cm^2 were considered, which resulted in average dose rates to the skin (calculated using the VARSKIN program) that ranged from 0.212 mrem/hr on contact to 0.013 mrem/hr at a distance of 10 cm in air. These dose rates account for the use of coveralls with a density thickness of 28 mg/cm^2. To estimate the skin dose from a contamination event, a contamination level of 25,000 dpm/100 cm^2 (250 dpm/cm^2) was assumed based on the action limit for ^{99}Tc contamination on work surfaces and hand tools (GAT 1963).

ATTACHMENT A
EXTERNAL EXPOSURE TO TECHNETIUM-99 (continued)

The dose from a contamination event is calculated as follows (Swinth 2004):

$$25{,}000 \text{ dpm/100 cm}^2 \times 0.081 \text{ mrem per dpm/cm}^2 = 20 \text{ mrem} \quad \text{(A-1)}$$

The assumed contamination value is greater than the average contamination level of 13,540 cpm/100 cm^2 identified by Swinth (2004). The value of 0.081 mrem/dpm/cm^2 is derived from a value of 1.6 × 10^{-3} mrem/dpm/cm^2 multiplied by a residence half-time of 1.5 days. This half-time is assumed because ^{99}Tc can be difficult to remove from the skin.

Because the low-energy ^{99}Tc electrons would not have been detected by dosimetry, the potential unmeasured external electron dose can be estimated by assuming an ambient dose rate level of 0.2 mrem/hr, a technetium-to-uranium progeny ratio of 0.4, and a 2,000-hour work year (Bassett 1986):

$$0.2 \text{ mrem/hr (maximum ambient level)} \times 0.4 \text{ (Tc:U progeny ratio)} \times 2{,}000 \text{ hr/yr} = 160 \text{ mrem/yr} \quad \text{(A-2)}$$

Because the facilities, processes, and contaminants were similar at all three gaseous diffusion plants, the magnitude of exposure discussed here should be valid for PGDP.

A.5 ASSIGNMENT OF EXTERNAL DOSE FROM TECHNETIUM-99

The assignment of external dose due to the presence of ^{99}Tc is warranted under certain circumstances for cancer sites on the hand. Dose assignment is limited to the hand because the ^{99}Tc dose rate at distances beyond 30 cm is less than 0.08 mrem/hr and drops off rapidly at greater distances. The following conditions must be met to assign an external dose from ^{99}Tc:

1. Claimant has skin cancer on the hand(s); and
2. Claimant worked in a facility where ^{99}Tc was present (Table A-1); and
3. Claimant performed a job function that could have involved ^{99}Tc exposure (Table A-2); and
4. Claimant dosimetry indicates a relatively high ratio (more than 2) of shallow to deep dose (NIOSH 2007).

If, and only if, all four of the above conditions are met, the dose reconstructor should:

- Assign an external electron dose of 8 mrem/yr.

This value derives from an annual external dose of 160 mrem reduced by a protection factor of 95% to account for the use of PPE. The external dose should be assigned as electrons >15 keV and a constant distribution.

A.6 ASSIGNMENT OF SKIN CONTAMINATION DOSE FROM TECHNETIUM-99

Skin contamination dose due to ^{99}Tc should be applied under certain circumstances for cancer sites where a documented skin contamination event occurred. The following conditions must be met to assign a skin contamination dose from ^{99}Tc:

1. Claimant has skin cancer on a potentially uncovered area of the skin; and

ATTACHMENT A
EXTERNAL EXPOSURE TO TECHNETIUM-99 (continued)

2. Claimant worked in a facility where ^{99}Tc was present (Table A-1); and

3. Claimant performed a job function that could have involved ^{99}Tc exposure (Table A-2); and

4. Claimant records indicate a contamination incident involving the area of the skin cancer site.

If, and only if, all four of the above conditions are met, the dose reconstructor should:

- Assign a skin dose of 20 mrem per documented incident.

The skin contamination should be assigned as electrons >15 keV and a constant distribution.

ATTACHMENT B
COWORKER DOSE ASSIGNMENT

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ATTACHMENT B
COWORKER DOSE ASSIGNMENT (continued)

B.1 PURPOSE

The purpose of this attachment is to provide information, based on site coworker data, about assignment of dose to PGDP workers who have no or limited monitoring data. The data in this attachment are to be used in conjunction with ORAUT-OTIB-0020, *Use of Coworker Dosimetry Data for External Dose Assignment* (ORAUT 2011).

B.2 BACKGROUND

An analysis of external coworker dose was performed to permit dose reconstructors to complete certain cases for which external monitoring data are unavailable or incomplete. Cases not having complete monitoring data can fall into one of several categories, including:

- The worker was unmonitored and, even by today's standards, did not need to be monitored (e.g., a nonradiological worker).
- The worker was unmonitored, but by today's standards would have been monitored.
- The worker might have been monitored, but the data are not available to the dose reconstructor.
- Partial information is available, but it is insufficient to facilitate a dose reconstruction.

As described in ORAUT-OTIB-0020 (ORAUT 2011), some cases without complete monitoring data can be processed based on assumptions and methodologies that do not involve coworker data. For example, many cases in the first category above can be processed by assigning ambient external and internal doses based on information in the relevant site profiles.

As described in Section 6.3.1, radiological operations at PGDP began in September 1952, and in 1953 the site began using dosimeter and processing technical support from ORNL. Until July 1960, dosimeters were issued to a limited number of workers (i.e., those with the highest potential for exposure), and the badges were exchanged weekly. After that time, dosimeters were assigned to all workers who entered a controlled area, and the badges were exchanged and processed on an annual, monthly or quarterly schedule. There does not appear to be any significant administrative practice that would jeopardize the integrity of the recorded dose of record.

B.3 APPLICATIONS AND LIMITATIONS

1. Some workers might have worked at one or more other major sites within the DOE complex during their employment history. Therefore, this guidance must be used with caution to ensure that, for clearly noncompensable cases, unmonitored external doses from multiple site employments have been overestimated. This typically requires the availability of external coworker dosimetry data for all relevant sites.

2. Summary statistics based on the dosimetry data in this attachment do not extend beyond 1995 because data beyond 1997 were not available, and the data for 1996 and 1997 included too few data points to be considered reliable. However, the absence of these data (and the subsequent development of dose distributions) should not interfere with the processing of most cases with a lack of external dosimetry data because well before 1995 the monitoring and reporting practices at the site ensured that essentially all workers with a potential for

ATTACHMENT B
COWORKER DOSE ASSIGNMENT (continued)

external radiation exposure were monitored and the results are readily accessible. Coworker doses can be extended to later years if needed. However, the vast majority of PGDP employees with a potential for radiological exposure were likely to have been monitored in recent years.

3. The data in this attachment address penetrating radiation from gamma radiation and nonpenetrating radiation from beta radiation. Neutron data are not presented. However, Section 6.5.3 should be used as the basis for assigning neutron doses, when relevant, in addition to the photon and beta doses assigned in accordance with this attachment.

4. External onsite ambient doses should not be included in addition to the coworker doses assigned in accordance with this attachment because such doses would have been included in the dosimetry results reported by the site, which were used as the basis for the coworker dose distributions presented below (ORAUT 2012a, 2006).

B.4 COWORKER DATA DEVELOPMENT

Dosimetry data for monitored workers from various sources were evaluated (see Section 7.0). The data selected for development of coworker doses were (1) a “history tape” containing annual data between 1953 and 1975 and quarterly data from 1976 to 1988, and (2) a database titled “OHIS_External” containing mostly quarterly data from 1989 to 1997 (although the data for 1996 and 1997 were excluded from consideration, as discussed above). In all cases, the reported data corresponded to deep doses (i.e., penetrating gamma radiation) and shallow doses (i.e., penetrating plus nonpenetrating radiation).

The annual data from 1953 to 1975 were prorated to account for partial years of employment based on an analysis of the length of monitored employment associated with the data (see Section B.5 for further discussion). The reported quarterly data from 1976 to 1988 were also prorated, but with a different approach (also described in Section B.5). The data from 1989 to 1995 included specific monitoring start and end dates, so they were prorated based on 365 d/yr. The data were prorated so coworker doses representing a full year of monitored employment could be derived; this permits the dose reconstructor to assign appropriate doses based on specific employment dates and job descriptions.

The validity of the data selected for coworker dose development was confirmed by selecting a sampling of claimant dosimetry data submitted by the site as part of the EEOICPA Subtitle B program and comparing it with the data selected as described above. A review of annual data for 10 claimants with more than 150 worker-years of monitored employment at PGDP indicated excellent agreement between the two datasets. Specifically, a perfect match was found for more than 95% of the reported values. It was concluded that the data cited above are acceptable for the development of coworker doses for PGDP.

Adjustment for Missed Dose

According to OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007), missed doses should be assigned for dosimeter readings less than the LOD to account for the possibility that doses were received but not recorded by the dosimeter or reported by the site. Annual missed doses are calculated by multiplying the number of <LOD dosimeter readings by the dosimeter LOD and summing the results. These values are used as the 95th percentile of a lognormal distribution for calculating the probability of causation (POC). Therefore, in the Interactive

ATTACHMENT B
COWORKER DOSE ASSIGNMENT (continued)

RadioEpidemiological Program (IREP), the calculated annual missed doses should be multiplied by 0.5 and entered in Parameter 1 and a value of 1.52 should be entered in Parameter 2 to represent the geometric mean and geometric standard deviation, respectively.

The assignment of missed doses for monitored workers is particularly significant for PGDP workers before August 1960 when they were monitored weekly. Table B-1 lists the maximum annual missed dose by era and type of radiation (penetrating gamma and nonpenetrating) based on information in Section 6.3.1 and Attachment C.

Table B-1. Missed external doses (rem) (based on Section 6.3.1 and Attachment C).

Period	Penetrating LOD	Nonpenetrating LOD[a]	Exchange frequency	Maximum annual penetrating	Maximum annual nonpenetrating
1953–1959	0.04	0.05	Weekly	2.080	2.600
1960	0.04	0.05	Varied[b]	1.280	1.600
1961–1980	0.04	0.05	Varied[c]	0.160	0.200
1981–1988	0.02	0.03	Varied[d]	0.080	0.120
1989–present	0.02	0.02	Quarterly	0.080	0.080

a. Attachment C provides an explanation for nonpenetrating LODs.
b. The exchange frequency was weekly through July 1960, then became less frequent (see footnote c).
c. Section 6.3.1 indicates that monthly, quarterly, or annual exchange frequencies were used during this period depending on work locations and the potential for exposure. A review of the data indicates that quarterly exchanges were predominant. Therefore, quarterly exchanges have been assumed here to calculate the maximum annual missed dose.
d. Section 6.3.1 indicates that either quarterly or annual exchange frequencies were used during this period depending on the potential for exposure. A review of the data indicates that quarterly exchanges were predominant. Therefore, quarterly exchanges have been assumed here to calculate the maximum annual missed dose.

Special Considerations

Certain aspects of the external dosimetry practices at PGDP (Section 6.3.1) were considered in the analysis of the site data. These include:

- In some cases, values less than the dosimeter LODs (Table B-1) were reported by the site. For example, values as low as a few millirem were reported even though the penetrating LOD was considered to be 20 or 40 mrem (depending on the era).

- As discussed above, before 1976 the data available to analyze coworker doses represent annual dose summaries for individual workers. Because these data include partial work years, the reported average annual doses tend to underestimate the received average annual doses by employees who worked an entire year.

As described in Section B.5, an approach that is favorable to claimants was adopted in the development of coworker dose summaries; this approach is intended to account for any underestimate of doses to radiological workers at PGDP based on the above considerations.

B.5 COWORKER ANNUAL DOSE SUMMARIES

Based on the above-described information and approaches described above, PGDP coworker annual external dosimetry summaries were developed for use in the evaluation of external dose for certain claimants potentially exposed to workplace radiation, but with no or limited monitoring data from DOE. These summaries were developed using the following steps:

ATTACHMENT B
COWORKER DOSE ASSIGNMENT (continued)

1. As described in Section B.4, for data from 1953 to 1975 the reported deep and shallow doses, which represent annual summary data, were modified to account for partial years of employment. This adjustment was made by analyzing NIOSH-Division of Compensation and Analysis Claims Tracking System (NOCTS) employment data for PGDP workers and adjusting the reported doses upward by an appropriate multiplier corresponding to the average fraction of a year an employee worked at the site. For example, if in a particular calendar year the average employment period for all PGDP employees in NOCTS was 11 months, the reported annual doses were multiplied by 12/11, or 1.09. This permits the dose reconstructor to assign an appropriate prorated dose to account for partial years of employment or potential exposure.

2. For data from 1976 to 1988, the reported deep and shallow doses, which represent quarterly summary data, were modified to account for partial years of employment. Consistent with the guidelines in ORAUT-OTIB-0020 (ORAUT 2011), doses for individuals with less than 4 quarters of data for a particular year were converted to annual doses by extrapolation (i.e., 1 quarterly result was multiplied by 4; 2 quarterly results were multiplied by 2; and 3 quarterly results were multiplied by 1.333).

3. For data from 1989 to 1995, the reported deep and shallow doses, which represent primarily quarterly data, were modified to account for partial years of employment by multiplying the data by 365/*x*, where *x* is the number of days the employee was issued a dosimeter. This information is available for this period because the data includes monitoring start and end dates.

4. Half of the maximum annual missed doses in Table B-1 were added to the annual doses from steps 1 through 3 (with the exception of reported positive doses, in which case the maximum missed dose was reduced by the dose corresponding to one badge exchange, because it is not possible that all individual badge results were zero if a positive annual dose was reported).

5. The 50th- and 95th-percentile annual penetrating and shallow doses were derived from the doses calculated in step 4 by ranking the data into cumulative probability curves and extracting the 50th- and 95th-percentile doses for each year.

6. Because the reported shallow doses include penetrating and nonpenetrating radiation, the percentile doses pertaining to penetrating radiation in step 5 were subtracted from the percentile doses pertaining to the reported shallow doses to derive percentile doses pertaining to nonpenetrating radiation.

7. The results are listed in Table B-2. These percentile doses should be used for PGDP workers with no or limited monitoring data in accordance with ORAUT-OTIB-0020 (ORAUT 2011). In general, the 50th-percentile dose can be used as a best estimate of a worker's dose if professional judgment indicates the worker was likely to be exposed to intermittent low levels of external radiation. The 50th-percentile dose should not be used for workers who were routinely exposed. For routinely exposed workers (i.e., those who were expected to have been monitored), the 95th-percentile dose should be applied. For workers who are unlikely to have been exposed, external onsite ambient dose should be used rather than coworker dose.

Doses to organs that are affected only by penetrating radiation (e.g., organs other than the skin, breast, and testes) are calculated based only on the Gamma columns in Table B-2 combined with the appropriate organ DCFs (NIOSH 2007). Doses to the skin, breast, and testes (and any other cancer location potentially affected by nonpenetrating radiation) are determined based on both the Gamma

ATTACHMENT B
COWORKER DOSE ASSIGNMENT (continued)

and Nonpenetrating columns; gamma doses are assigned as photons with an energy range consistent with the information in this document, and nonpenetrating doses are assigned as electrons

Table B-2. Annual external coworker doses modified to account for missed dose (rem).

Year	Gamma 95th percentile	Gamma 50th percentile	Nonpenetrating 95th percentile	Nonpenetrating 50th percentile
1953	1.656	1.128	1.729	0.701
1954	2.218	1.183	4.386	0.970
1955	2.344	1.067	5.574	1.048
1956	2.712	1.073	4.829	1.048
1957	2.224	1.072	4.511	0.580
1958	2.019	1.040	4.021	0.466
1959	1.900	1.083	5.148	0.694
1960	1.544	0.672	3.140	0.452
1961	1.048	0.134	1.647	0.036
1962	1.024	0.080	1.422	0.059
1963	0.868	0.080	0.818	0.037
1964	0.519	0.080	0.514	0.020
1965	0.243	0.080	0.194	0.020
1966	0.225	0.080	0.242	0.020
1967	0.236	0.091	0.343	0.025
1968	0.411	0.080	0.532	0.020
1969	0.541	0.080	0.989	0.020
1970	0.349	0.080	0.763	0.020
1971	0.558	0.080	1.039	0.020
1972	0.451	0.080	1.133	0.020
1973	0.407	0.080	1.254	0.020
1974	0.217	0.080	0.854	0.020
1975	0.247	0.090	0.604	0.055
1976	0.233	0.062	0.553	0.050
1977	0.189	0.062	0.398	0.055
1978	0.193	0.089	0.150	0.031
1979	0.109	0.080	0.265	0.054
1980	0.200	0.080	0.135	0.020
1981	0.090	0.040	0.324	0.020
1982	0.053	0.040	0.712	0.020
1983	0.070	0.040	0.535	0.020
1984	0.156	0.040	0.489	0.020
1985	0.070	0.040	0.615	0.020
1986	0.130	0.040	0.755	0.020
1987	0.070	0.040	0.415	0.020
1988	0.055	0.040	0.590	0.020
1989	0.053	0.040	0.067	0.000
1990	0.040	0.040	0.052	0.000
1991	0.040	0.040	0.033	0.000
1992	0.040	0.040	0.046	0.000
1993	0.043	0.040	0.042	0.000
1994	0.040	0.040	0.037	0.000
1995	0.040	0.040	0.055	0.000

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COWORKER DOSE ASSIGNMENT (continued)

>15 keV with corrections to account for clothing attenuation or other considerations. Further guidance is provided in ORAUT-OTIB-0017, *Technical Information Bulletin: Interpretation of Dosimetry Data for Assignment of Shallow Dose* (ORAUT 2005).

With the methodology described above, null values for nonpenetrating dose can occur because of the subtraction of the reported penetrating doses from the reported shallow doses and the method described above, which is favorable to claimants, to establish coworker doses based on the addition of potential missed doses. However, a zero value in Table B-2 for nonpenetrating dose does not result in a dose of zero to an organ such as the skin. For example, the 50th-percentile dose to the skin in 1989 would be assigned entirely as 0.040 rem of photons. This approach does not result in an underestimation of POC (which is determined by DOL) because assigning beta dose as gamma dose in IREP has no negative effect (because the radiation effectiveness factors are the same for >15-keV electrons and >250-keV photons, and are higher for 30- to 250-keV photons).

B.6 PENETRATING DOSE VALUE FOR SELECTED CONSTRUCTION TRADE WORKERS

Table B-3 lists penetrating dose values that have been adjusted using the guidance in Section 8.0 of ORAUT-OTIB-0052, *Parameters to Consider When Processing Claims for Construction Trade Workers* (ORAUT 2014). This guidance is applicable for CTWs who meet the criteria in ORAUT-OTIB-0052.

B.7 EXTERNAL DOSIMETRY DATA REVIEW

PGDP Dosimetry Data database
There are many tables in this database; they contain internal and external dosimetry data. The external data listings are listed below, with their descriptions:

- DRS_89_THRU_96 – External dosimetry records from 1989 to 1996,
- DRS_97_THRU_98 – External dosimetry records from 1997 to 1998,
- OHIS_EXTERNAL_DOSE – External dosimetry records from 1981 to 1997,
- OHIS_EXTREMITY_DOSE – Extremity dosimetry records from 1990 to 1995,
- OHIS_HP_SCHEDULE – Dosimetry scheduling information from 1987 to 1998,
- OHIS_JOB_HISTORY – Personnel job history information from 1986 to 1998,
- HISTORY_TAPE – External dosimetry records from 1953 to 1988,
- HIS20_EDD_CALCULATED_EXPOSURE – Calculated external dosimetry exposure records from 1980 to 1998, and
- HIS20_EDD_INACTIVE_IRD_EXPOSURE – External dosimetry records from 1953 to 1998.

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COWORKER DOSE ASSIGNMENT (continued)

Table B-3. Annual external penetrating coworker doses modified in accordance with ORAUT-OTIB-0052 (rem) (ORAUT 2014).

Year	Gamma 95th percentile	Gamma 50th percentile
1953	1.910	1.171
1954	2.697	1.248
1955	2.874	1.086
1956	3.389	1.094
1957	2.705	1.093
1958	2.419	1.040
1959	2.252	1.109
1960	1.913	0.693
1961	1.443	0.163
1962	1.409	0.080
1963	1.191	0.080
1964	0.702	0.080
1965	0.316	0.080
1966	0.291	0.080
1967	0.306	0.103
1968	0.552	0.080
1969	0.734	0.080
1970	0.464	0.080
1971	0.757	0.080
1972	0.607	0.080
1973	0.545	0.080
1974	0.280	0.080
1975	0.322	0.102
1976	0.302	0.063
1977	0.240	0.063
1978	0.246	0.101
1979	0.128	0.080
1980	0.256	0.080
1981	0.114	0.040
1982	0.063	0.040
1983	0.086	0.040
1984	0.206	0.040
1985	0.086	0.040
1986	0.170	0.040
1987	0.086	0.040
1988	0.065	0.040
1989	0.062	0.040
1990	0.044	0.040
1991	0.040	0.040
1992	0.040	0.040
1993	0.048	0.040
1994	0.040	0.040
1995	0.040	0.040

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SKIN DOSE ASSIGNMENT

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SKIN DOSE ASSIGNMENT (continued)

C.1 GENERAL INFORMATION

In general, the contribution to skin dose at PGDP from low-energy photons is extremely small in comparison with the contribution from beta particles.

All DCFs for the skin should be assumed to be 1 (ORAUT 2005).

Missed doses should be calculated based on the following LODs:

- 1953 to 1980: 50 mrem for OW, 40 mrem for S;
- 1981 to 1988: 30 mrem for OW, 20 mrem for S; and
- 1989 to present: 20 mrem for OW, 20 mrem for S.

Section 6.3.1 states an OW LOD of 120 mrem for 1953 to 1980. However, this value appears to be speculative when compared with the LOD values for similar dosimetry systems at other sites at that time. As outlined in Section 6.3.1, in 1953, PGDP began using ORNL to provide and process dosimeters. Practices were similar to those at ORNL and other major sites. ORNL has provided PGDP with dosimeters from early in the operations period through the present. Based on the information, it appears that the reported LOD value is based on considerations involving low-energy beta emitters; however, this would significantly overestimate the LOD (and missed dose) when the principal source of exposure is uranium because the dosimeters were calibrated using uranium slabs. Therefore, the value has been reduced in this attachment to 10 mrem above the reported photon LOD. The dose reconstructor should consult Attachment A to address potential exposures to ^{99}Tc. Table C-1 provides examples of skin dose assignments.

Table C-1. Examples of skin dose assignments (mrem) for badge readings in 1970 (assuming Paducah LODs, no clothing correction, and no ^{99}Tc exposure).

OW reading	S reading	Measured dose assigned	Missed dose assigned
50	0	50 (electrons)	40/2 = 20 (30- to 250-keV photons)
0	0	None	50/2 =25 (30- to 250-keV photons)
100	60	40 (electrons) AND 60 (photon energy per main text)	None
100	100	100 (photon energy per main text)	None
0	40	40 (photon energy per main text)	50/2 = 25 (electrons)

C.2 PROCEDURE

Measured Dose

1. Subtract the reported S reading from the reported OW reading. This is the calculated nonpenetrating dose.

2. Assign the calculated nonpenetrating dose as electrons >15 keV. A correction factor should be provided for clothing, if applicable, depending on likely clothing thickness and beta energy.

3. Assign the reported S dose as photons, partitioned by energy according to Section 6.3.4.

4. Assign the reported neutron dose (if applicable) partitioned by energy and corrected for neutron quality according to Section 6.4.3 (using an organ DCF of 1).

ATTACHMENT C
SKIN DOSE ASSIGNMENT (continued)

Missed Dose

1. For a badge cycle with a zero result in the OW or S reading, or both, assign a single missed dose.

2. If only the OW reading was reported as zero, the missed dose should be the appropriate OW LOD for that era (divided by 2, treated as lognormal) and considered to be electrons (corrected for attenuation if applicable).

3. If only the S reading was reported as zero, the missed dose should be the appropriate S LOD for that era (divided by 2, treated as lognormal) and considered to be 30- to 250-keV photons.

4. If both the OW and S readings were reported as zero, the missed dose should be the appropriate OW LOD for that era (divided by 2, treated as lognormal) and considered to be 30- to 250-keV photons.

5. Assign missed or unmonitored neutron dose per the direction in Section 6.5.3.

6. If applicable, assign unmonitored ^{99}Tc dose (as >15-keV electrons) per the direction in Attachment A.