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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.
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ACRONYMS AND ABBREVIATIONS

cm	centimeter
d	day
DOE	U. S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ft	foot
GM	geometric mean
GSD	geometric standard deviation
HEU	highly enriched uranium
Hp(d)	personal dose equivalent at tissue depth *d* (*d* = 10 mm or 0.07 mm)
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
kg	kilogram
MDL	minimum detection level
MED	Manhattan Engineering District (a DOE predecessor agency)
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
min	minute
mm	millimeter
mrem	millirem
NIOSH	National Institute for Occupational Safety and Health
NTA	nuclear track emulsion, type A (film)
ORNL	Oak Ridge National Laboratory
PGDP	Paducah Gaseous Diffusion Plant
POC	probability of causation
QF	quality factor
RU	recycled uranium
TBD	technical basis document
TLND	thermoluminescent neutron dosimeter

TEPC	tissue-equivalent proportional counter
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
yr	year
§	section

6.1 INTRODUCTION

6.1.1 Purpose

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 historic background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

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

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

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

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

6.1.2 Scope

Paducah Gaseous Diffusion Plant (PGDP) workers, especially those employed 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 (PGDP 1976). Before about July 1960, personnel dosimeters were not assigned to all workers (PGDP 1957a). 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. *External Dose Reconstruction Implementation Guideline* (NIOSH 2006) indicates that these records represent the highest quality records for assessment and reconstruction of doses.

Initial radiation dosimetry practices were based on experience gained during 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 Engineering District (MED) program to develop nuclear weapons, which began in about 1940.

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 in the 1940s, 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 1993). 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 (Martin Marietta 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 (2006) 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
- **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/University of Indiana 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 Administrative Practices

The PGDP radiation monitoring program used portable instruments, contamination surveys, zone controls, and personnel dosimeters to measure exposure in the workplace (Harris 1957; PGDP 1957a,b, 1964, 1976; UCND 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
- 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/University of Indiana 2000).

In 1953, PGDP began using dosimeter and processing technical support provided by the Oak Ridge National Laboratory (ORNL) (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 used at ORNL and other major sites at that time (PGDP 1957a). 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 developed at the Metallurgical Laboratory at the University of Chicago (Pardue, Goldstein, and Wollan 1944). ORNL has provided PGDP with dosimeters from early in the operations period through the present.

Table 6-1. Dosimeter type, period of use, exchange frequency, MDL, and potential annual missed dose.

Dosimeter	Period of use	Monitored population	Exchange frequency	Laboratory MDL (rem)[a]	Maximum annual missed dose equivalent (rem)[b]
				***Hp(10)* beta/photon dosimeters**	
Four-element film	1953 through 7/1960	Selected workers based on activities performed	Weekly (n = 50)	0.04	1.0
Four-element film	After 7/1960 through 1980	Workers in C-340, C-400, and C-410	Monthly (n =12)	0.04	0.24
	After 7/1960 through 1980	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n =4)	0.04	0.08
	After 7/1960 through 1980	Workers and visitors not likely to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.04	0.02
Harshaw two-chip TLD	Beginning 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	Beginning 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	Beginning 1989 through present	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n =4)	0.02	0.04
				***Hp(0.07)* beta/photon dosimeters**	
Four-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 1980	Workers in C-340, C-400, and C-410	Monthly (n =12)	0.12	0.72
	After 7/1960 through 1980	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n =4)	0.12	0.24
	After 7/1960 through 1980	Workers and visitors not likely to exceed 0.1 of applicable guidelines	Annual (n = 1)	0.12	0.06
Harshaw two-chip TLD	Beginning 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	Beginning 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	Beginning 1989 through present	Workers and visitors with potential to exceed 0.1 of applicable guidelines	Quarterly (n =4)	0.02	0.04
				Neutron dosimeters[c]	
Harshaw TLND	Beginning 1998 to 2003 (ongoing)	Selected workers based on activities performed	Quarterly (n =4)	0.015	0.03

a. Estimated film dosimeter detection levels based on NIOSH (1993), NRC (1989), and Wilson et al. (1990). TLD detection levels from Martin Marietta (1994) and personal communication with site personnel.

b. Maximum annual missed dose (NIOSH 2006).

c. The potential annual missed dose based on laboratory irradiations is not applicable to workplace missed neutron dose.

The precise detection levels listed 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 (Martin Marietta 1994) based on a DOELAP-accredited laboratory testing protocol (DOE 1986). During earlier years, MDLs were subject to additional uncertainty because factors involving radiation field and film type, as well as 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), NRC (1989), Wilson et al. (1990), and site personnel. Examination of older records, where 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 (Martin Marietta 1994). The film badge was replaced by the TLD in 1980 (PGDP 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
- 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/University of Utah 2000). This population of monitored individuals represents those with the highest exposure potential. After July 1960, PGDP routine practices required the assignment of dosimeters to all workers who entered a controlled radiation area (BJC 2000). Dosimeters were exchanged on a routine schedule (PGDP 1957a, 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 2000a). All dosimeters were processed, and measured results were recorded and used to estimate dose.

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

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 dosimeter services from ORNL. 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 thermoluminescent dosimetry systems.

Figure 6-1 shows the energy response characteristics of the PGDP beta/gamma dosimeters based on the essentially identical two-element film dosimeter 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).

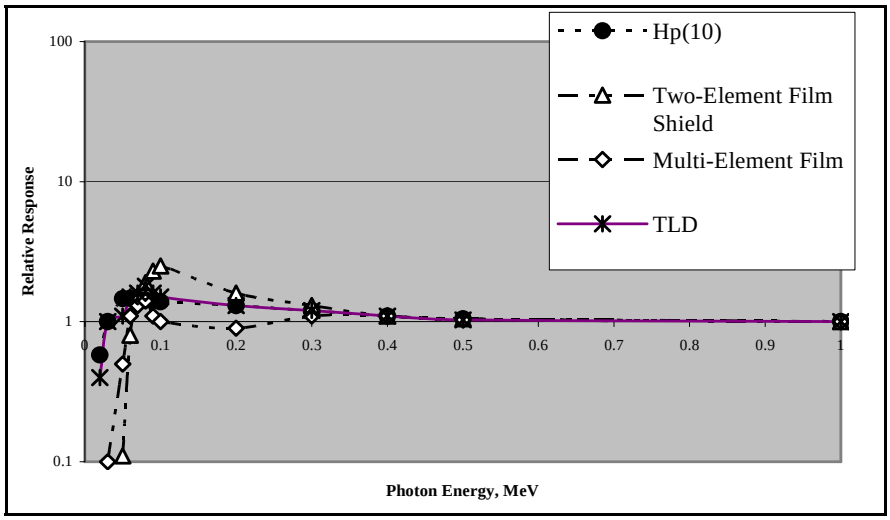

Figure 6-1. Estimated dosimeter photon response characteristics.

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 used in the two-element dosimeter. The figure shows that the two-element dosimeter over-responded in relation to *Hp(10)* from 0.05 to 0.3 MeV, followed *Hp(10)* for higher energies, and under-responded 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 comes from handling uranium of low enrichment. The photon energy spectrum is almost entirely in the range from 30 to 250 keV (Schleien et al. 1998).

The nonpenetrating response of the two-element dosimeter was calculated as the difference between the *unshielded* and *shielded* 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 over-response of the unshielded portion of the film to any 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 used at PGDP historically had a neutron-sensitive element that was processed on request. After 1989, this capability has been provided with 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].

The quality factors (QFs) used historically for neutrons have changed significantly. 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 (1991) are about 2 times higher than the ICRP (1983) values. Because a QF of 10 was used for the referenced radiation measurements, the PGDP personnel dosimetry, an adjustment to ICRP (1991) of at most a factor of 2 times higher would be necessary [3].

Average neutron energy is 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 neutrons from depleted and natural uranium cylinders ranged from 210 to 360 keV (Cardarelli 1997, p. 9). Unmoderated and deuterium (water) ^{252}Cf neutrons created were between 1,306 and 1,403 keV. This means the dose as monitored at PGDP since 1998 was overestimated and, therefore, is favorable to the claimant.

6.3.3 <u>Calibration</u>

Potential error in recorded dose is dependent on dosimetry technology response characteristics to each radiation type, energy, and geometry; the methodology used to calibrate the dosimetry system; and the extent of similarity between the radiation fields used for calibration and that present 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 and TLND 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 2004b). 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 1993).

Introduction of on-phantom calibration of film dosimeters and 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 from ^{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 in NIOSH (2006, Appendix B) 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 for 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 needed 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). Thus, the determination of beta dose was specific to uranium.

6.3.3.2 Neutron Dosimeters

Calibration of neutron dosimeters for use at PGDP was appropriate for the work locations in which those dosimeters were worn (Martin Marietta 1994). Dosimeter response was characterized in a manner that would represent the workplace (Martin Marietta 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 (PNL 1995). The basis for the calibration factor was developed using data obtained at the Y-12 Plant in a room 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 (PNL 1990).

In 1989, field measurements for neutron flux were made by PNL 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 (PNL 1990, Measurement M23 in Table 5.9; ORNL 2005). These calibration factors apply to the PGDP TLNDs (Martin Marietta 1994).

6.3.4 Workplace Radiation Fields

6.3.4.1 Beta/Photon Fields

PGDP operations are 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/ 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.

Table 6-2. Major radiation sources.

Nuclide	Source	Half-life	Energies (MeV) and abundances of major radiations		
			Alpha	Beta (max)	Gamma
U-238	Primary U isotope	4.51E9 yr	4.15 (21%)		
			4.20 (79%)		
U-235	Primary U isotope	7.1E8 yr	4.21 (6%)		0.144 (11%)
			4.37 (17%)		0.163 (5%)
			4.40 (55%)		0.186 (57%)
			4.60 (5%)		0.205 (5%)
U-234	Primary U isotope	2.47E5 yr	4.72 (28%)		0.053(0.12%)
			4.77 (72%)		
Th-234	Decay product	24.1 d			0.013 (9.8%)
				0.103 (21%)	0.063 (3.5%)
				0.193 (79%)	0.092 (3%)
					0.093 (4%)
Pa-234m	Decay product	1.17 min		2.29 (98%)	0.765 (0.3%)
					1.001 (0.60%)
Th-231	Decay product	25.5 hr		0.206 (13%)	
				0.287 (12%)	0.026 (2%)
				0.288 (37%)	0.084 (10%)
				0.305 (35%)	
Tc-99	Impurities from RU	2.12E5 yr		0.294 (100%)	None

The major facilities and associated activities at PGDP are (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
- 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/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 (PGDP 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 6-3. Buildings and periods of operation.

Site facilities	Source for external dose	Operation	
		Begin	End
C-310 Purge and Product Withdrawal	UF_6 process equipment and cylinders	1953	1999
C-315 Surge and Tails Withdrawal	UF_6 process equipment and cylinders	1953	1999
C-331, C-333, C-335, C-337 Gaseous Diffusion Process Buildings	UF_6 process equipment and cylinders	1953	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)	Ongoing

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/University of Utah 2000). However, some routine chemical processes would concentrate them (PACE/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 2000b). 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 nor TLD efficiently detects 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. Table 6-4 lists the principal locations where and periods during which recovery operations at PGDP are believed to have taken place (PACE/University of Utah 2000).

Table 6-4. Technetium-99 recovery operations.

Building	Began	Terminated
C-710	Before 1959	~1959
C-400	~1959	~1975

6.3.4.2 Workplace Beta/Photon Dosimeter Response

Essentially all PGDP radiological work areas involved photon and beta radiation characteristic of operations involving uranium at low enrichments. As discussed in Section 6.3.3.1, the recorded

responses of the PGDP beta/photon film dosimeters are favorable to claimants and need no adjustment.

6.3.4.3 Neutron Fields

While neutrons occur in some areas at PGDP, measured levels are low. There are no identified locations where measurable neutron dose was encountered (Martin Marietta 1994). Studies have evaluated neutron fields at gaseous diffusion plants (PNL 1995; Cardarelli 1997); these studies confirm Martin Marietta (1994). 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 range from 0.007 to 0.34 mrem/hr; associated QFs range 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 2000b). A representative average value is 0.2 mrem/hr based on a QF of about 10 (PNL 1995; Cardarelli 1997). Estimates of average neutron energies ranged from 0.25 to 0.56 MeV (PNL 1995). Neutron monitoring of individuals was performed during a UF_6 cylinder-painting project (Meiners 1999). Results of this project 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 mounted 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 6-5 lists dose fractions for the neutron energy groups (indicated by the dashed lines in Figure 6-4). 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 2004a).

Exposure to low enriched UF_6 (less than 5%) will result in a lower neutron flux than the neutron field expected 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 listed in Table 6-5 are favorable to claimants (Soldat and Tanner 1992).

The neutron study performed in 1990 at X10 and Y12 was the only definitive study of neutron energy spectra documented over the history of PGDP. It is assumed that the energy spectra are valid for the earlier years, given the presence of enriched uranium. 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). This element was processed only when requested. (NTA film had an energy threshold of about 0.5 MeV.) A review of data does not indicate the assignment of neutron dose before 1998.

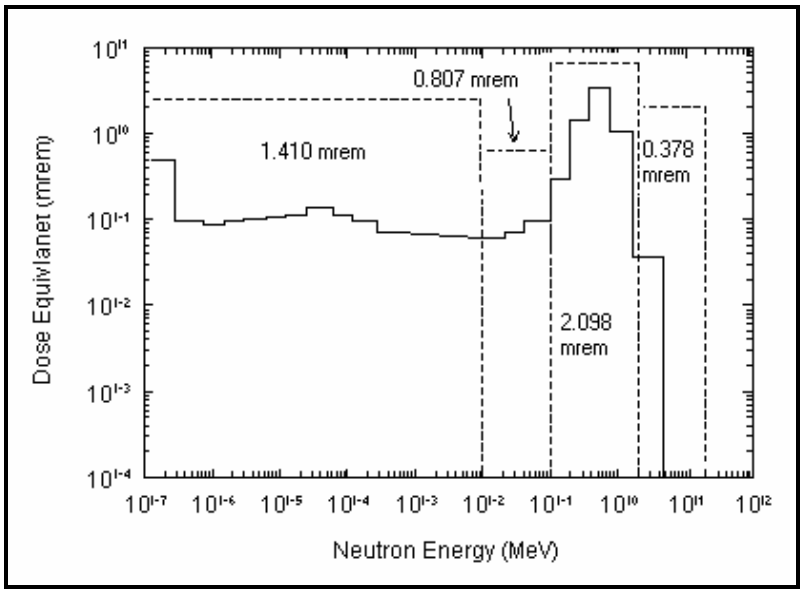

Figure 6-2. Results of neutron spectrum measurements made about 24 in. in front of 93% – 96% HEU cylinders (Soldat and Tanner 1992).

Table 6-5. Dose fractions for PORTS HEU storage vault in Building 345.

Neutron energy group	Near unshielded Cf-252 source
< 10 keV	0.300
10-100 keV	0.172
0.1-2 MeV	0.447
2-14 MeV	0.081
Favorable to claimant dose fractions	
< 10 keV	0.300
0.01- 2 MeV	0.610
0.1-2 MeV	0.081

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 reporting unit used by DOE is the millirem, a unit of dose equivalent. The international unit of dose equivalent is the millisievert, which is equivalent to 100 mrem. Since 1986, deep dose equivalents at PGDP have been based on DOELAP calibration to *Hp(10)* and require no adjustment. Before 1986, TLDs were calibrated in air to ^{137}Cs, which is nearly equivalent to an *Hp(10)* on-phantom ^{137}Cs calibration. No adjustment to the measured TLD penetrating photon dose is necessary. As discussed in Section 6.3.3.1, the earlier film badge deep doses are favorable to claimants and require no numerical adjustment.

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 0.10 and 2.0 MeV, for which the ICRP Publication 60 radiation weighting factor is 20 (ICRP 1990). Therefore, dose reconstructors should multiply the reported neutron dose equivalent by a factor of 2 to be consistent with the ICRP (1990) recommendations to be used for reconstruction (NIOSH 2006). Apply this factor to measured and missed neutron doses.

6.5 MISSED DOSE

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

1. A zero dose was recorded but the worker was not monitored (most workers from 1953 to July 1960).

2. A zero dose was recorded for the dosimeter system for any response less than the MDL.

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 (Martin Marietta 1994). Neutron dosimeters were not routinely assigned and doses were not recorded until about 1998. Neutron doses reported 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, favorable to claimant option 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

Watson et al. (1994) examined methods to be considered when there is no recorded dose for a period during a working career. 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 description and work locations. NIOSH (2006) cites several different models.

For Group 2, the missed dose for dosimeter results that are less than the MDL is particularly important for earlier years, when MDLs were higher and dosimeter exchange was more frequent. NIOSH (2006) describes an acceptable, favorable to claimant estimate of the maximum potential missed dose as one-half the MDL multiplied by the number of zero dose results (the MDL/2 method). 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 persons in Groups 1 and 3. Figure 6-3 shows the distribution of individual annual deep dose equivalent for

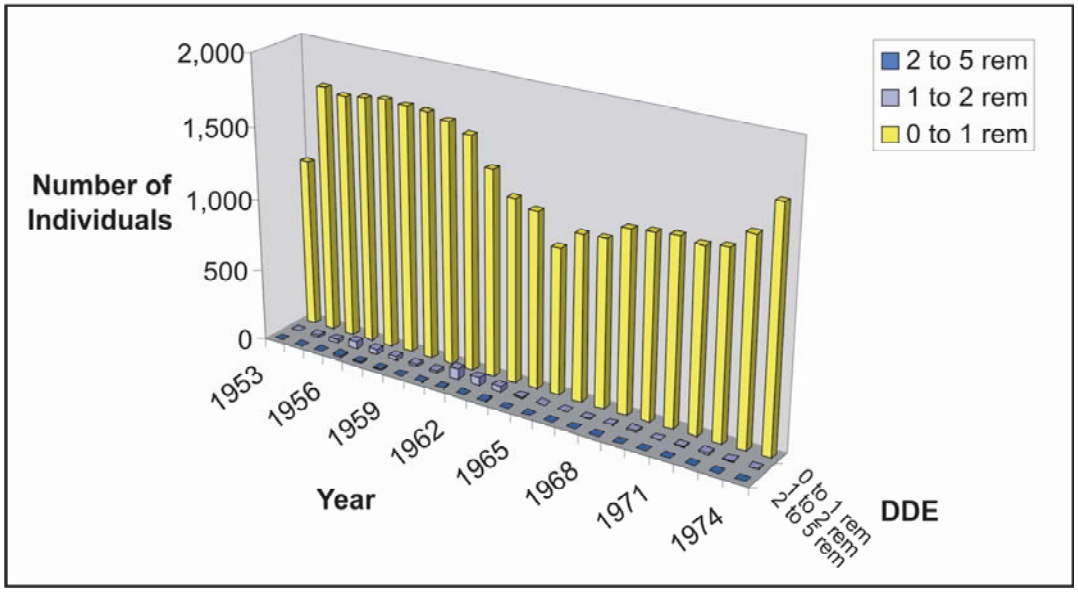

Figure 6-3. Historical distribution of deep dose equivalent (Baker ca. 1995).

monitored workers from 1953 to 1974 (Baker ca. 1995). Few of these individuals received as much as 1 rem in any year.

An alternative approach for Group 1 or 3 is to base the unmonitored dose estimate on exposure data compiled in PACE/University of Utah (2000) for monitored PGDP workers. The first four columns in Table 6-6 (from Table 7.4 of the PACE report) list the number of monitored workers, their average recorded deep dose, and the maximum individual deep dose for each year from 1953 through 1988 (zero doses were not included). For dose reconstruction, this TBD analysis assumed that exposure data for each year could be represented by a lognormal distribution with a geometric mean (GM) equal to the average listed in column 3 of Table 6-6 and a 99th percentile equal to the maximum in column 4. With these assumptions, the geometric standard deviation (GSD) of the lognormal distribution, shown in the last column, was computed [7]. The two parameters, GM and GSD, determine the dose distribution assumed for the monitored workers for each year. The values from columns 3 and 5 in Table 6-6 can be entered directly into the Interactive RadioEpidemiological Program (IREP). Dose reconstructors should consider unmonitored dose values obtained from Tables 6-1 and 6-6. Knowledge of specific conditions for some workers at some periods could have a bearing on which value would be more appropriate. In the absence of such information, use the larger of the two values as favorable to claimants [8].

6.5.2 Estimating Missed and Unmonitored Shallow Dose

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. Figure 6-4 shows the historical data for the distribution of shallow dose equivalent for monitored workers at PGDP (Baker ca. 1995). When compared with Figure 6-4, this assessment of 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 used for Group 2. As an alternative, use Table 6-7,

Table 6-6. Average recorded deep dose and maximum for any single worker by year (PACE/University of Utah 2000).

Year	Number of workers	Average dose, GM (rem)	Maximum dose (rem)	GSD (rem)
1953	223	0.1398	0.820	2.14
1954	284	0.2835	1.580	2.09
1955	417	0.2419	2.500	2.72
1956	471	0.3586	4.700	3.02
1957	669	0.2517	3.190	2.97
1958	661	0.1853	3.630	3.59
1959	570	0.2015	2.360	2.88
1960	526	0.2011	2.510	2.95
1961	1,690	0.1770	2.530	3.13
1962	1,479	0.1495	2.980	3.61
1963	1,311	0.1441	3.040	3.70
1964	1,289	0.0734	1.860	4.00
1965	1,128	0.0341	1.610	5.23
1966	1,138	0.0371	1.470	5.19
1967	1,143	0.0498	1.120	3.80
1968	1,241	0.0618	1.400	3.82
1969	1,270	0.0733	1.970	4.11
1970	1,273	0.0417	0.840	3.63
1971	1,254	0.0624	1.380	3.78
1972	1,288	0.0589	1.760	4.30
1973	1,404	0.0530	1.830	4.57
1974	1,624	0.0265	1.030	4.81
1975	2,013	0.0501	1.049	3.69
1976	2,426	0.0351	1.224	4.59
1977	2,643	0.0232	0.742	4.42
1978	2,613	0.0399	0.359	2.57
1979	2,487	0.0082	0.364	5.09
1980	2,308	0.0182	0.344	3.53
1981	1,840	0.0076	0.420	5.60
1982	1,617	0.0065	0.350	5.53
1983	1,452	0.0067	0.340	5.39
1984	1,434	0.0092	0.420	5.15
1985	1,365	0.0061	0.350	5.69
1986	1,244	0.0096	0.490	5.41
1987	1,275	0.0080	0.470	5.74
1988	1,359	0.0065	0.720	7.54

which is based on the shallow-dose data for monitored workers in Baker (ca. 1995) and shown in Figure 6-4. This TBD analysis assumed that the dose distribution for each year could be represented by a lognormal function with GM equal to the average listed in column 2 of Table 6-7 and a 95th percentile equal to the maximum in column 3. The resultant GSD is listed in the last column. Enter the GM and GSD into IREP. Compare values for unmonitored shallow dose obtained for Groups 1 and 3 from Table 6-7 with those determined by the MDL/2 method from Table 6-1. Use knowledge of specific job conditions and location in judging which of the two estimates is more appropriate. In the absence of such information, assign the larger estimate as favorable to claimants [9].

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

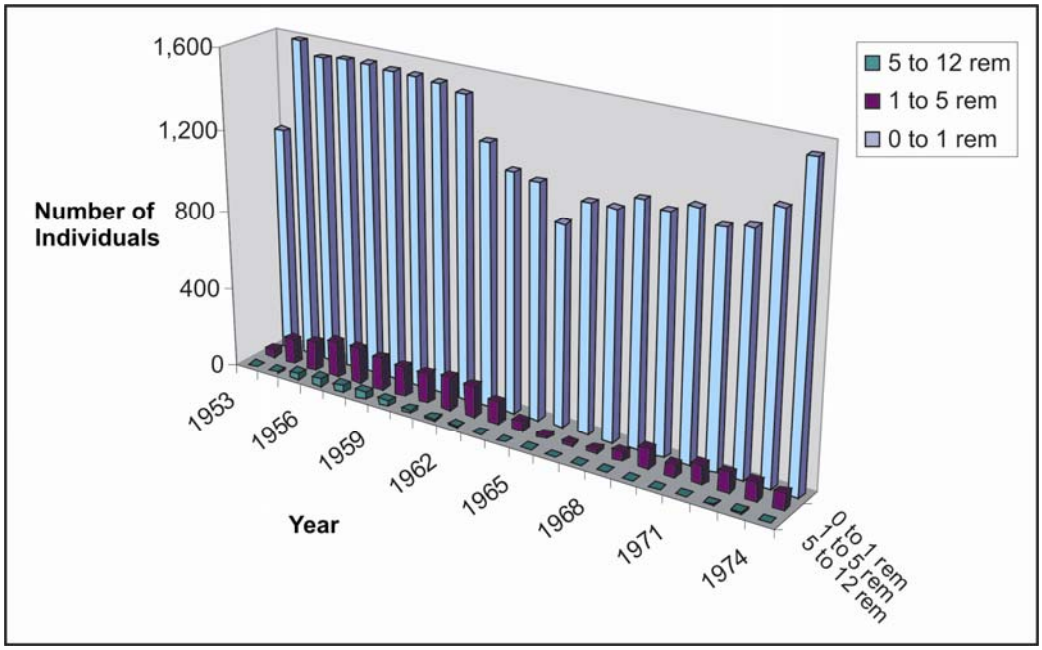

Figure 6-4. Historical distribution of shallow dose equivalent (Baker ca. 1995).

Table 6-7. Average recorded shallow dose and maximum for any worker by year (Baker ca. 1995).

Year	Average dose, GM (rem)	Maximum dose (rem)	GSD (rem)
1953	0.539	4	2.36
1954	0.677	7	2.73
1955	0.776	9	2.86
1956	0.853	12	3.11
1957	0.834	11	3.03
1958	0.809	11	3.07
1959	0.783	10	2.98
1960	0.699	10	3.13
1961	0.734	8	2.79
1962	0.719	10	3.09
1963	0.645	8	2.95
1964	0.547	4	2.35
1965	0.511	2	1.80
1966	0.511	3	2.14
1967	0.528	6	2.84
1968	0.563	8	3.12
1969	0.616	5	2.46
1970	0.552	3	2.07
1971	0.631	7	2.81
1972	0.640	10	3.25
1973	0.679	10	3.17
1974	0.578	7	2.92

Potential doses from ^{99}Tc skin contamination have been evaluated by using the VARSKIN computer code. The calculated shallow dose rate from uniform ^{99}Tc skin contamination is 0.0016 mrem/hr per dpm/cm^2 (Swinth 2004). Technetium-99 is difficult to remove from skin. Therefore, the integrated

shallow dose resulting from ^{99}Tc skin contamination could be relatively large. For example, with a residence half-time of 1.5 days, the dose is 0.081 mrem per dpm/cm^2 of initial contamination.

In general, direct external beta dose from ^{99}Tc is minimal. The unshielded shallow dose rate to bare skin (no clothing) at a distance of 10 cm in air from a uniformly contaminated surface is about 1×10^{-4} mrem/hr per dpm/cm^2, as estimated with VARSKIN. The dose rate at 30 cm is only about 1×10^{-6} mrem/hr per dpm/cm^2. Table 6-8 summarizes these three benchmark values for shallow dose equivalent rate as determined from VARSKIN for skin contamination and for external exposure with intervening air.

Table 6-8. Shallow dose equivalent rates for ^{99}Tc.

Condition	Dose-equivalent rate (mrem/hr per dpm/cm^2)
Skin contamination	1.6×10^{-3}
External, 10 cm air	1.0×10^{-4}
External, 30 cm air	1.0×10^{-6}

Some skin contamination events involving ^{99}Tc could have occurred without being detected at the time. In some cases, therefore, it could be appropriate to consider an additional skin dose component for a reported shallow dose of a worker who could have had direct contact with ^{99}Tc. In the absence of specific data, the dose reconstructor must make assumptions about the number of times per year an affected skin region could have been contaminated and the extent of each contamination. For example, the dose reconstructor could assume a monthly contamination event at a specific location on the skin with an average level of 25,000 dpm/100 cm^2 (the action limit for ^{99}Tc contamination on work surfaces and hand tools at PGDP). With the assumed residence half-time of 1.5 days, the annual shallow dose equivalent would be 240 mrem (12×250 dpm/cm^2 $\times$ 0.081 mrem per dpm/cm^2). The direct external dose rate at a distance of 10 cm from a surface contaminated at this level would be 0.025 mrem/hr (250 dpm/cm^2 $\times 10^{-4}$ mrem/hr per dpm/cm^2). At 30 cm, the rate would be 0.00025 mrem/hr.

6.5.3 Estimating Missed Neutron Dose

Dose reconstructors should add a neutron component to the annual dose of individuals who worked in the cylinder yard before 1998. However, careful consideration should be given to work history. In general, only workers who were near cylinders for extended periods have the potential for neutron exposure. Estimates should be based on the neutron-to-photon ratio of 1 to 5 for dose equivalent, as determined from Meiners (1999). The neutron dose equivalent should be multiplied by the ICRP (1990) factor of 2.

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 2004b). The standard error for beta dose is the same (or somewhat larger for unknown mixtures of beta/gamma dose). Thus, the film-badge dose uncertainty is 1.3. The uncertainty in the TLD dose is 1.15 (ORAUT 2004b), which is consistent with NIOSH (2006).

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 information 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 are in the 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)* on the basis of MDL/2 times the number of zero results, as described in Sections 6.5.1 and 6.5.2 (NIOSH 2006).

- Individuals with no dose recorded might or might not have been radiological workers. If it is definite that the individual was not a radiation worker, the assigned missed dose is the environmental dose discussed in the Occupational Environmental Dose section of this PGDP Site Profile (ORAUT 2004c). Otherwise, estimate the missed dose as described in Section 6.5. No numerical adjustments to the missed dose are necessary.

- Multiply reported and missed neutron dose equivalents by 2 to adjust for ICRP (1990).

- Base the assignment of missed neutron dose equivalent estimate for cylinder yard workers for whom no neutron dose is recorded on a neutron-to-photon ratio of 1 to 5 for dose equivalent (Meiners 1999). Multiply the estimated neutron dose equivalent by 2 to adjust for ICRP (1990).

- Pay special attention to the possibility of skin contamination incidents for workers involved with ^{99}Tc recovery operations (Section 6.5.2).

- See Section 6.6 for a discussion of uncertainty.

6.8 ORGAN DOSE

NIOSH (2006) discusses the conversion of measured doses to organ dose equivalent, and Appendix B of that document contains the appropriate dose conversion factors for each organ, radiation type, and energy range based on the type of monitoring performed. In some cases, simplifying assumptions are appropriate [10].

6.9 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in the preceding text, bracketed callouts have been inserted to indicate information, conclusions, and recommendations to assist in the process of worker dose reconstruction. These callouts are listed in this section with information that identifies the source and justification for each item. Conventional references are provided in the next section that link data, quotations, and other information to documents available for review on the Oak Ridge Associated Universities Team servers.

[1] Turner, James E. Integrated Environmental Management. Consultant. 2003.
The reviewed records did not reveal recorded neutron doses for either dosimeter.

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

[3] Turner, James E. Integrated Environmental Management. Consultant. 2003.
ICRP (1991) recommended 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. Integrated Environmental Management. Consultant. 2003.
The importance of energy response to accurate measurement of dose equivalent is well known, and the response of the historical dosimeters is illustrated in Figure 6-1.

[5] Turner, James E. Integrated Environmental Management. 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. Integrated Environmental Management. Consultant. 2003.
Empirical neutron and photon worker dose equivalent data provide a basis from which neutron dose equivalent can be inferred from better known photon dose. Interpretation of Meiners (1999).

[7] Smith, Matthew H. ORAU Team. Statistician. 2004.
The computation of GSD is a standard mathematical formula and can be found in most statistical methods texts.

[8] Turner, James E. Integrated Environmental Management. Consultant. 2003.
The choice that is favorable to the claimant is appropriate.

[9] Turner, James E. Integrated Environmental Management. Consultant. 2003.
The choice that is favorable to the claimant is appropriate.

[10] Turner, James E. Integrated Environmental Management. Consultant. 2003.
Appendix B of NIOSH (2006) contains tables for numerous organs. Some professional judgment is needed to fit particular conditions.

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GLOSSARY

absorbed dose

Amount of energy in rads or grays deposited in a substance by ionizing radiation per unit mass of the substance. See *dose*.

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.

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.

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. Most of the direct fission products are (negative) beta emitters. Exposure to large amounts of beta radiation from external sources can cause skin burns (erythema), and beta emitters can be harmful inside the body. Thin sheets of metal or plastic can stop beta particles.

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 (Hd)

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

dose

In general, the effects of ionizing radiation in terms of the specific amount of energy absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays. Various terms narrow the type of dose, and some are additive:

- Absorbed dose is the amount of energy deposited in a substance by ionizing radiation.
- Collective dose is the sum of the doses to a specific population.
- Committed dose is the dose over time (usually 50 years for workers) to a specific organ or tissue from an intake of radioactive material.
- Cumulative dose is the sum of all doses to the same portion of the body or to the whole body over time.
- Deep dose is the dose at a 1-centimeter depth in tissue (1,000 milligrams per square centimeter).
- Effective dose is the sum of the equivalent doses in the principal tissues and organs of the body, each weighted by a tissue weighting factor that accounts for the probabilities

of fatal and nonfatal cancers according to severity and the average length of life lost due to an induced cancer. It indicates the biological effect of the radiation exposure in that tissue.

- Equivalent dose or dose equivalent is the absorbed dose in a tissue or organ multiplied by a weighting factor for the particular type of radiation.
- Organ dose is the dose to a specific organ.
- Penetrating dose is that from higher energy photon (gamma and X-ray) radiation and neutron radiation that penetrates the outer layers of the skin. Nonpenetrating dose is that from beta and lower energy photon radiation.
- Personal dose equivalent is the dose equivalent in soft tissue below a specified point on the body at a specified depth.
- Shallow dose is the dose at a 0.07-centimeter depth in tissue (7 milligrams per square centimeter).
- Skin dose is the dose to the skin.
- Whole-body dose is the dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder.

dose equivalent (H)
Product of absorbed dose in units of rem or sievert 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*, and *thermoluminescent 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.

film
Radiation-sensitive photographic film in a light-tight wrapping. See *film dosimeter*.

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 rays are very penetrating, but dense materials such as lead or uranium or thick structures can stop them. Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

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.

minimum detection level (MDL)

Lowest amount (mass or activity) of a substance detectable by a specific instrument or process. Often assumed to be the level at which a dose is detected at the two-sigma level (i.e., 95% of the time). Also called minimum detectable limit and minimum detection limit or level.

neutron

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

neutron film dosimeter

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

nuclear track emulsion, type A (NTA)

Film sensitive to fast neutrons made by the Eastman Kodak Company. The developed image has tracks caused by neutrons that visible under oil immersion with about 1,000-power magnification.

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 $H_p(0.07)$ and $H_p(10)$, respectively. The International Commission on Radiological Measurement and Units recommended $H_p(d)$ in 1993 as dose quantity for radiological protection.

photon

Basic unit of electromagnetic radiation. Photons are massless “packages” of light energy that range from low-energy microwave photons to high-energy gamma rays. Photons have energies between 10 and 100 kiloelectron-volts.

pocket ionization chamber

Cylindrical monitoring device commonly clipped to the shirt or laboratory coat pocket to measure ionizing radiation. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

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) that travel from one point to another, some of which can pass through or partly through solid materials including the human body.

radioactivity

Disintegration of certain elements (e.g., radium, actinium, uranium, and thorium) accompanied by the emission of alpha, beta, gamma, and/or neutron radiation from unstable nuclei.

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.

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 average American receives 360 millirem a year from background radiation. 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 83 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 8.38 milligray. The word derives from roentgen equivalent physical.

roentgen (R)

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

shallow absorbed dose (D_s)

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

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

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

skin dose
See *shallow dose equivalent*.

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 by radiation, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

whole-body 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
Penetrating electromagnetic radiation (photons) of short wavelength (0.001 to 10 nanometers) and energy less than 250 kiloelectron-volts. X-rays usually come from excitation of the electron field around certain nuclei. Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.