ORAU Team **Dose Reconstruction Project for NIOSH** Paducah Gaseous Diffusion Plant – Occupational External Dose	Document Number: ORAUT-TKBS-0019-6 Effective Date: 03/29/2005 Revision No.: 01 Controlled Copy No.:________ Page 1 of 25
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ACRONYMS AND ABBREVIATIONS

cm	centimeter
DOE	U. S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
GM	geometric mean
GSD	geometric standard deviation
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
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1 thousand electron volts
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
RU	recycled uranium
TLND	thermoluminescent neutron dosimeter
TEPC	tissue-equivalent proportional counter
TLD	thermoluminescent dosimeter
U.S.C.	United States Code
yr	year

6.1 INTRODUCTION

Technical Basis Documents and Site Profile Documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health (NIOSH) in the completion of the individual work required for each dose reconstruction.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy facility” as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) [42 U.S.C. Sections 7384l (5) and (12)].

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. Before about July 1960, personnel dosimeters were not assigned to all workers. Records of radiation dose to those 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 staff for compliance with radiation control limits, and routinely available to individual workers. The NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2002) indicates that these records represent the highest quality record 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. These practices were generally well advanced at the start of the Manhattan Engineering District (MED) program to develop nuclear weapons, beginning in about 1940.

6.2 BASIS OF COMPARISON

Since the start of the MED in the early 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 recommended for use as the operational quantity for radiological protection by the International Commission on Radiation Units and Measurements (ICRU 1993). In addition, *Hp(0.07)* and *Hp(10)* are the radiation quantities used in the U.S. Department of Energy (DOE) Laboratory Accreditation Program (DOELAP) 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 technical basis document 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 of 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 (2002) 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 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 Administrative Practices

PGDP had a radiation monitoring program using portable instruments, contamination surveys, zone controls, and personnel dosimeters to measure exposure in the workplace. The program improved as better technology and more information became available. Results from the personnel dosimeters were used to measure and record doses from external radiation exposure to PGDP workers. These dosimeters include 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 likely incapable of furnishing accurate doses in terms of *Hp(0.07)*. Extremity doses, which were generally not assessed (PACE and University of Utah 2000), were not treated in this analysis.

In 1953, PGDP began using dosimeter and processing technical support provided by the Oak Ridge National Laboratory (ORNL). There is evidence that Paducah might have processed its own dosimeters for a period; however, a review of the limited documentation available indicated that practices were similar to those used at ORNL and other major sites at that time. 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.

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 the 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).

Parameters of the PGDP administrative practices significant to dose reconstruction involve policies:

- To assign dosimeters to workers
- To exchange dosimeters
- To record notional dose (i.e., some identified value for lower dosed workers, often based on a small fraction of the regulatory limit)
- To estimate dose for missing or damaged dosimeters
- To replace destroyed or missing records
- To evaluate and record dose for incidents
- To 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, 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. For workers in some areas the frequency was monthly, but for the general population it was quarterly. 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 historic practices at PGDP (PACE and University of Utah 2000; BJC 2000).

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 dosimeter	1953-7/1960	Selected workers based on activities performed	Weekly (n = 50)	0.04	1.0
Four-element film dosimeter	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 dosimeter	1953-7/1960	Selected workers based on activities performed	Weekly (n = 50)	0.12	3.0
Four-element film dosimeter	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 2002).
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 as improved technology was developed and complex radiation fields were better understood. 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). 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” is representative of 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 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. Last, the figure 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 keV to 250 keV.

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 claimant-favorable because of the significant over-response of the unshielded portion of the film to any lower energy photons that could have been present. 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.

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. The use of the commercial Harshaw thermoluminescent neutron dosimeters (TLND) to assess neutron dose routinely (along with deep and shallow dose) began in 1998. ORNL has provided the dosimeter and associated services. The 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.

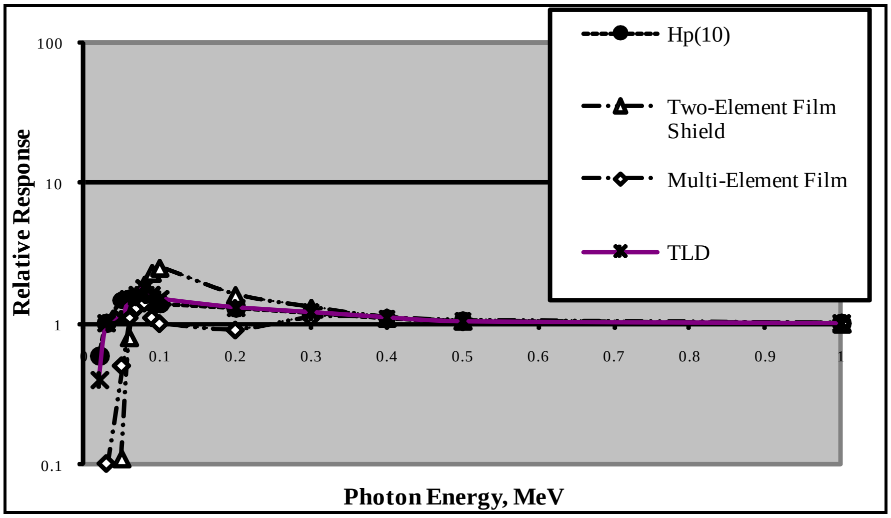

Figure 6-1. Estimated dosimeter photon response characteristics.

6.3.3 Calibration

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.

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 was changed to ^{137}Cs. The calibration to both ^{226}Ra and ^{137}Cs was free in air (no phantom) until the DOELAP procedures adopted in 1986 required phantoms. *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 ^{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 relative to calibration with ^{137}Cs. Because the photons at PGDP are of intermediate energies, some numerical dose adjustment should be considered for the early film dosimetry. The overall dose adjustment depends on these two factors, which act in opposite directions, as well as the spectrum of registered photons and the film dosimeter itself.

In the 1980s, studies were carried out at a number of laboratories to assess changes from the on-phantom calibration mandated by the new 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 Hanford showed 8% and 4% increases, respectively, in dosimeter response for on-phantom exposures using ^{226}Ra and ^{137}Cs. 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 Savannah River, on the other hand, indicated that film badge doses underestimated *Hp(10)* by 11.9% before 1986 and by 3.9% for 1986 (Taylor et al. 1995). Lacking site-specific data for PGDP, it is recommended that exposure-to-organ dose conversion factors in Appendix B of (NIOSH 2002) be employed for dose reconstruction at PGDP with no numerical adjustment to the recorded doses. This procedure is expected to be claimant-favorable. It allows for an overestimate of exposure, such as assessed in the Hanford studies, which should also 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 found to be 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 also 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). The determination of beta dose was thus 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 these dosimeters were worn. 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 quality factor (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 quality factor was 11, and the average energy range was 0.6 to 1.4 MeV (PNL 1990).

In 1993, field measurements were made by ORNL representatives at the end row of the K-25 cylinder yard with a TEPC and a phantom with TLDs about 4 feet from the outside of a cylinder at about the middle of its length. The results were evaluated qualitatively because the dose rate was very low and an appropriate power supply was not available. The correction factors were similar to those in the Y-12 UF_4 storage area and confirmed the appropriateness of these values. These correction factors apply to the PGDP TLNDs.

6.3.4 <u>Workplace Radiation Fields</u>

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 and University of Utah 2000). The photon energy range of principal interest is 30 keV 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 day			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 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 recovery and uranium recovery were in Building C-340. 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. 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	Operation 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	UF6 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. However, some routine chemical processes would concentrate them. From an external dose standpoint, the most significant impurity found 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 2000). 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 detect them, particularly in the presence of uranium. Clothing and gloves provide adequate shielding. Skin contamination is the only credible scenario where significant shallow dose could occur from ^{99}Tc. Table 6-4 shows the principal locations where and periods during which the recovery operations at PGDP are believed to have taken place (PACE AND 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 claimant-favorable and need no adjustment.

6.3.4.3 Neutron Fields

While neutrons occur in some areas at PGDP, the measured levels are low. There are no locations identified where measurable neutron dose was encountered (Martin Marietta 1994). Several studies have evaluated neutron fields at gaseous diffusion plants (PNL 1995; Cardarelli 1996); these studies confirm Martin Marietta (1994). Cylinder yards, feed and withdraw areas, and locations where uranium forms deposits in the cascade have been investigated (Cardarelli 1996). 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 quality factors range from 7 to 10. A representative average value is 0.2 mrem/hr based on a quality factor of about 10 (PNL 1995; Cardarelli 1996). 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 quality factor of 10. The associated neutron-to-photon absorbed dose ratio is 1 to 50.

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, NTB, Eastman Kodak Type 2). 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.

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Photon Dose

The recorded doses varied in reporting units depending on regulatory requirements and dose definitions (both national and international). The current reporting unit used by DOE is the millirem, a unit of dose equivalent. The international unit of dose equivalent is the milliseivert, 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 claimant-favorable 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 can be consulted for investigations of nonroutine beta exposures and dose assessment.

6.4.3 Neutron Dose

The measured neutron energies at PGDP are between 0.10 MeV and 2.0 MeV, for which the International Commission on Radiological Protection Publication 60 (ICRP 1990) radiation weighting factor is 20. Therefore, the reported neutron dose equivalent should be multiplied by a factor of 2 to be consistent with the ICRP (1990) recommendations to be used for reconstruction (NIOSH 2002). This factor should be applied to both 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 (majority of 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 recorded until about 1998. Neutron doses reported before 1998 were based on a conservative calibration associated with a neutron-sensitive element incorporated in the beta/gamma dosimeter. Application of a neutron-to-gamma dose equivalent ratio of 1 to 5 appears to be a satisfactory, claimant-favorable option, because the photon dose is reliably measured. This ratio can be applied to selected work activities.

6.5.1 Estimating Missed and Unmonitored Photon Deep Dose

Methods to be considered when there is no recorded dose for a period during a working career have been examined by Watson et al. (1994). 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 the employees' job description and work locations. NIOSH (2002) 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 (2002) describes an acceptable, claimant-favorable 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 last column in Table 6-1 lists the resultant estimates of the annual missed dose for Group 2 for different years at PGDP.

If it is definitely established that the employee was not a radiation worker, then the unmonitored deep dose for that period can be assigned as the on-site ambient dose.

Otherwise, an individual in Group 1 or 3 should be treated as a radiation worker. The unmonitored deep dose can then be approached in two ways. First, the same assignment of missed dose as for Group 2, from the last column of Table 6-1, can be considered. However, for the period 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 both Groups 1 and 3. Figure 6-2 shows the distribution of individual annual deep dose equivalent for monitored workers for the years 1953 to 1974 (Baker c. 1995). Few of these individuals received as much as 1 rem in any given year.

A second, alternative, approach for Group 1 or 3 is to base the unmonitored dose estimate on exposure data compiled in (PACE and University of Utah 2000) for monitored PGDP workers. The first four columns in Table 6-5 (taken from Table 7.4 of the PACE Report) show 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 use in the present dose reconstruction, it was assumed that the exposure data for each year could be represented by a lognormal distribution with a geometric mean (GM) equal to the average shown in column 3 of Table 6-5 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. The two parameters, GM and GSD, thus determine the dose distribution assumed for the monitored workers for each year. The values from columns 3 and 5 in Table 6-5 can be entered directly into the

Interactive RadioEpidemiological Program (IREP). Unmonitored dose values obtained both from the use of Table 6-5 and from Table 6-1 should be considered. Knowledge of specific conditions known for some workers at some periods could have a bearing on which value should be considered more appropriate. In the absence of such information, the larger of the two values should be used as claimant-favorable.

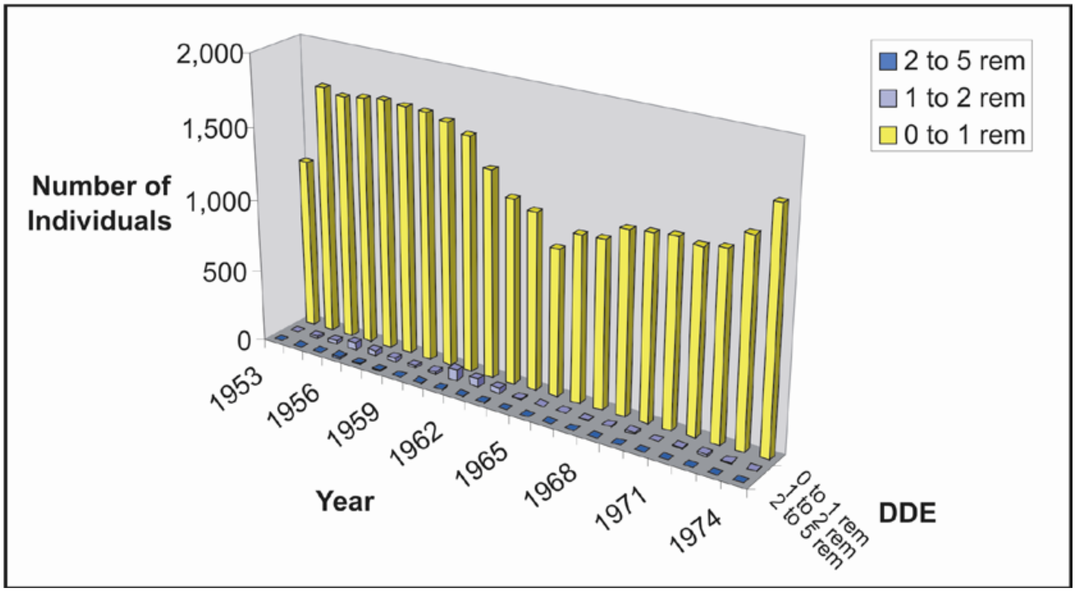

Figure 6-2. Historical distribution of deep dose equivalent (Baker c. 1995).

6.5.2 Estimating Missed and Unmonitored Shallow Dose

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

For Group 2, the missed annual shallow dose equivalent is given in the last column of Table 6-1, in keeping with the MDL/2 method of evaluation. Figure 6-3 shows the historical data for the distribution of shallow dose equivalent for monitored workers at PGDP (Baker c.1995). When compared with Figure 6-3, this assessment of annual missed shallow dose for Group 2 is seen to be claimant-favorable.

For non-radiological workers identified in Groups 1 and 3, the unmonitored shallow dose can be assigned as the environmental dose. Other individuals in these groups should be regarded as radiation workers, for whom the same estimate as that used for Group 2 should be considered. Alternatively, Table 6-6 can be used, which is based on the shallow-dose data for monitored workers taken from Baker (c. 1995) and shown in Figure 6-3. It was assumed that the dose distribution for each year could be represented by a lognormal function with GM equal to the average shown in column 2 of Table 6-6 and a 95th percentile equal to the maximum in column 3. The resultant calculated GSD is shown in the last column. The GM and GSD can be entered into IREP. Values for unmonitored shallow dose obtained for Groups 1 and 3 from Table 6-6 should be compared with those determined by the MDL/2 method from Table 6-1. Knowledge of specific job conditions and location should be considered in judging which of the two estimates is more appropriate. In the absence of such information, the larger estimate should be assigned as claimant-favorable.

Table 6-5. Average recorded deep dose and maximum for any single worker by year (PACE and University of Utah 2000).[a]

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

a. As explained in text, columns 3 and 5, respectively, show the GM and GSD of lognormal distribution used to describe the data.

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

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 has proven to be 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.

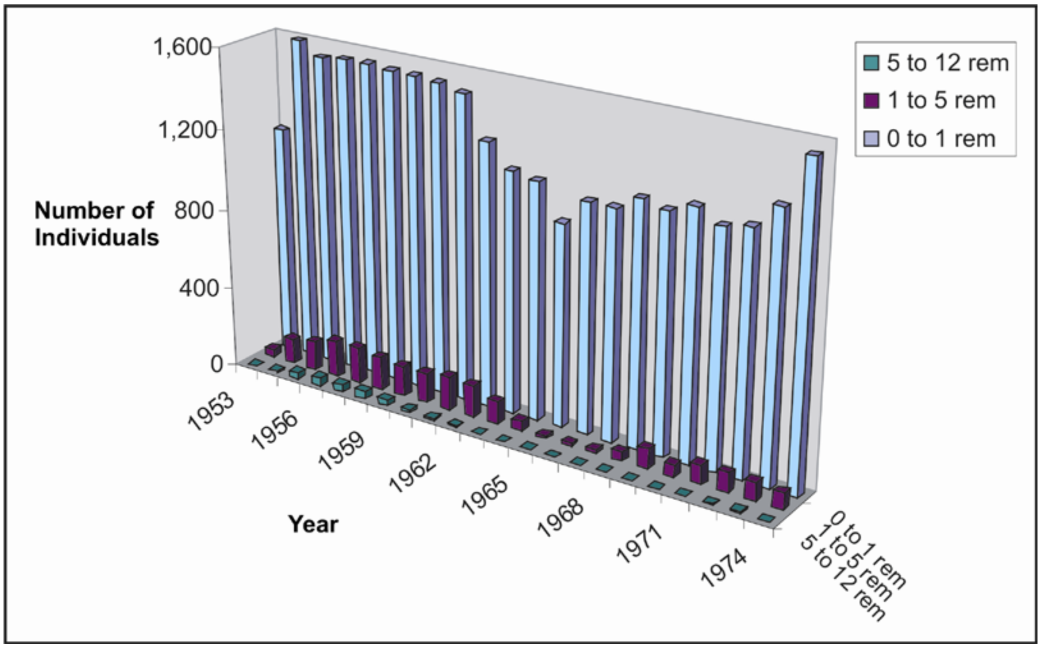

Figure 6-3. Historical distribution of shallow dose equivalent (Baker c. 1995).

Table 6-6. Average recorded shallow dose and maximum for any worker by year (Baker c.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

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 6-7 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 6-7. 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}

It is possible that some skin contamination events involving ^{99}Tc 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 that an affected skin region could have been contaminated and the extent of each contamination. For example, the 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 × 0.081 mrem per dpm/cm^2). The direct external dose rate at a distance of 10 cm from a surface contaminated at this same level would be 0.025 mrem/hr (250 dpm/cm^2 × 10^{-4} mrem/h per dpm/cm^2). At 30 cm, the rate would be 0.00025 mrem/hr.

6.5.3 Estimating Missed Neutron Dose

A neutron component should be added 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 the survey conducted at PGDP (Meiners 1999). The neutron dose equivalent should then 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 (ORAU 2004). 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 (ORAU 2004), which is consistent with NIOSH (2002).

6.7 DOSE RECONSTRUCTION

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

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

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

- Individuals with no dose recorded might or might not have been radiological workers. If it is definitely established that the individual was not a radiation worker, then the assigned missed dose is the environmental dose discussed in the Occupational Environmental Dose portion of this PGDP Site Profile. Otherwise, the missed dose is to be estimated as described in Section 6.5. No numerical adjustments to the missed dose are necessary.

- Reported and missed neutron dose equivalents should be multiplied by 2 to adjust for ICRP (1990).

- Cylinder yard workers for whom no neutron dose is recorded should have missed neutron dose equivalent estimate assigned based 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).

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

- Uncertainty is discussed in Section 6.6.

6.8 ORGAN DOSE

NIOSH (2002) 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.

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GLOSSARY

absorbed dose
Radiation energy deposited in tissue or other material divided by the mass of the tissue or material.

albedo dosimeter
A dosimeter that measures slow neutrons generated by higher energy neutrons incident on the body and that reflect back into the dosimeter.

beta radiation
Radiation consisting of electrons emitted spontaneously from the nuclei of certain radioactive elements.

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

deep dose equivalent (Hd)
The dose equivalent at the depth of 10 mm in tissue.

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

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

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

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

film
In general, a "film packet" that contains one or more pieces of film in a light-tight wrapping. When developed, the film has an image caused by radiation that can be measured using an optical densitometer.

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

gamma rays
Electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Physically, gamma

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

gray (Gy)
The special name for the SI unit of absorbed dose (1 gray = 1 joule/kilogram).

kerma
Sum of initial kinetic energies of all charged particles (including Auger electrons) liberated by uncharged radiation per unit mass. Units are rad and gray. The word derives from *kinetic energy released per unit mass*.

minimum detection level
The minimum quantifiable exposure or neutron flux that can be detected by a dosimeter.

neutron
A basic nuclear particle that is electrically neutral, having nearly the same mass as the hydrogen atom.

neutron film dosimeter
A film dosimeter that contains a nuclear track emulsion film packet, such as NTA, NTB, or Eastman Type 2.

nuclear emulsion
Often referred to as NTA film and used to measure personnel dose from neutron radiation.

nuclear track emulsion, type A (NTA)
A film that is sensitive to fast neutrons. The developed image has tracks caused primarily by recoil protons. Tracks can be seen by using an appropriate imaging capability such as oil immersion and a 1,000-power microscope or a projection capability.

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

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

rad
The traditional unit of absorbed dose (1 rad = 100 ergs per gram of material absorbing the radiation energy; 1 rad = 0.01 gray).

radiation
Alpha, beta, neutron, and photon radiation.

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

rem

The traditional unit of dose equivalent, which is equal to the product of the absorbed dose in rad and the quality factor of the radiation (1 rem = 0.01 sievert).

rep (roentgen-equivalent-physical)

A historical unit used to report beta exposures, usually recorded in millirep; equivalent to 93 ergs of energy per gram and roughly the same as a rem.

roentgen (R)

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

recycled uranium (RU)

Uranium recovered from used reactor fuel; isotopic activity ratios listed 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)

shallow absorbed dose (D_s)

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

shallow dose equivalent (H_s)

Dose equivalent at a depth of 0.07 mm in tissue.

sievert (Sv)

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

skin dose

Absorbed dose at a tissue depth of 7 mg/cm^2 at about 0.07 mm depth in tissue.

thermoluminescence

Phosphorescence developed by heating a previously excited material.

thermoluminescent dosimeter (TLD)

A device used to measure radiation dose. It consists of a holder containing solid chips of material that when heated will release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

whole-body dose

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

X-ray

Ionizing electromagnetic radiation of external nuclear origin or a radiograph.