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RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	11/03/2003	00-A	New Technical Basis Document for the K-25 Site for Occupational External Dose. Initiated by Joseph L. Alvarez.
Draft	01/26/2004	00-B	Incorporates NIOSH and internal comments. Initiated by Joseph L. Alvarez.
Draft	05/06/2004	00-C	Incorporates changes for consistency with other gaseous diffusion plants and completes comment resolution. Initiated by Jay J. Maisler.
Draft	07/26/2004	00-D	Incorporates additional changes for consistency with other gaseous diffusion plants and completes comment resolution. Initiated by Jay J. Maisler.
Draft	09/02/2004	00-E	Provides additional data for recorded deep doses and recorded shallow doses to workers. Initiated by Jay J. Maisler.
Draft	10/11/2004	00-F	Incorporates responses to NIOSH comments. Initiated by Jay J. Maisler.
11/24/2004	11/24/2004	00	First approved issue. Initiated by Jay J. Maisler.

ACRONYMS AND ABBREVIATIONS

CEDR	Comprehensive Epidemiologic Data Resource
DCF	Dose Conversion Factor
DOE	U. S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DOL	U.S. Department of Labor
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)
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
Kg	kilogram
MDL	minimum detection limit
MED	Manhattan Engineering District
MeV	million electron volts
mm	millimeter
mrem	millirem
n	neutron
NIOSH	National Institute for Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NTA	nuclear track, type A emulsion (also, NTB and NTB-2)
NVLAP	National Voluntary Laboratory Accreditation Program
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
PIC	Pocket Ionization Chamber dosimeters
RU	recycled uranium
s	second
Sv	Sievert
TBD	technical basis document
TEPC	tissue-equivalent proportional counter
TLD	thermoluminescent dosimeter
TLND	thermoluminescent neutron dosimeter
U.S.C.	United States Code
WB	whole body

6.0 OCCUPATIONAL EXTERNAL DOSIMETRY

6.1 INTRODUCTION

Site Profile documents, which include technical basis documents (TBDs), 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 the 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 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. § 7384l (5) and (12)).”

Workers at the K-25 site, especially those employed during peak production in the 1950s, 1960s, and 1970s, were exposed to radiation types and energies associated with natural and recycled uranium enrichment processes. In addition, K-25 workers occasionally worked at the X-10 and Y-12 facilities and might have received exposures to other sources of radiation. Personnel dosimeter records are available for K-25 workers beginning in 1945. The operations and radiation safety staff routinely reviewed dosimeter results for compliance with radiation control limits and investigated doses approaching annual or quarterly dose limits. The number of individuals assigned dosimeters varied over the years. Dosimeter monitoring intervals also varied, depending on the potential to receive significant dose. During the early years, only workers entering controlled areas and likely to receive measurable dose received dosimeters. Beginning in 1951, dosimeters were issued to the entire work force as part of the security badge, although only those likely to have received measurable dose were processed. Beginning in 1980, all dosimeters were processed. The *External Dose Reconstruction Implementation Guide* (NIOSH 2002) identified dosimeter records as the highest quality records for a retrospective dose assessment.

Radiation dosimetry practices were initially based on experience gained during several decades of radium and X-ray medical diagnostic and therapy applications. These methods were generally well advanced at the start of the Manhattan Engineering District (MED) program to develop nuclear weapons beginning about 1940.

The EEOICPA gives responsibility to NIOSH for producing radiation dose estimates that the U.S. Department of Labor (DOL) will use in adjudicating certain cancer claims under the Act. The doses are those incurred by employees in the performance of duty at U.S. Department of Energy (DOE) and predecessor agencies and atomic-weapons employer facilities. EEOICPA covers two cancer claimant groups: (1) a Special Exposure Cohort and (2) all others. Employees in group 1 who have a specified cancer and meet certain other criteria qualify for compensation under EEOICPA. Unlike group 2, their claims do not require NIOSH dose reconstruction for DOL adjudication. Oak Ridge Associated Universities (ORAU) has developed this TBD for the second group, which excludes the special cohort established for K-25 and the other gaseous diffusion plants.

6.2 BASIS OF COMPARISON

Various radiation dose concepts and quantities have been in use to measure and record occupational dose since the initiation of the MED in the early 1940s. 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. Hp(d) is an operational quantity used for the purpose of determining the worker's actual dose equivalent from a dosimeter reading. It is defined as a practical method for calibration of instruments and dosimeters to dose equivalent in International Commission on Radiological Protection (ICRP) Report 74 (ICRP 1996) and in International Commission on Radiation Units and Measurements (ICRU) Report 57 (ICRU 1998). Weakly penetrating radiation is significant to shallow dose equivalent, defined at d = 0.07 mm or Hp(0.07). Penetrating radiation is significant to deep dose equivalent and is defined at d = 10 mm or Hp(10). Hp(0.07) and Hp(10) are the radiation quantities used in the DOE Laboratory Accreditation Program (DOELAP) used to accredit DOE personnel dosimetry systems since the 1980s (DOE 1986). The National Voluntary Laboratory Accreditation Program (NVLAP), which is the U.S. Nuclear Regulatory Commission (NRC) equivalent to DOELAP, uses the same operational quantities. This TBD uses Hp(10) and Hp(0.07) as deep and shallow dose equivalents, respectively.

6.3 DOSE RECONSTRUCTION PARAMETERS

Examinations of the beta, photon (X-ray and gamma ray), and neutron radiation type, energy, and geometry of exposure in the workplace, and the characteristics of the K-25 dosimeter responses are crucial to the assessment of bias and uncertainty of the original recorded dose in relation to the radiation quantity Hp(d). Dose reconstructors can compare earlier dosimetry systems to current systems to evaluate their performance, based on the premise that current systems have more stringent criteria, as indicated in the DOELAP and NVLAP programs.

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

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

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

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

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

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

6.3.1 K-25 Historical Administrative Practices

The K-25 Site used personnel dosimeters to measure and record doses from external radiation to designated workers throughout the history of its operations. These dosimeters include one or more of the following:

- Personnel whole-body (WB) beta/photon and neutron dosimeters
- Pocket Ionization Chamber (PIC) dosimeters
- Personnel extremity dosimeters

- Personnel neutron dosimeters.

K-25 began operations in 1945 using dosimeter and processing technical support provided by the Oak Ridge National Laboratory (ORNL). ORNL, then the Clinton Laboratory, had implemented its dosimetry methods based on the personnel beta/photon dosimeter design developed at the Metallurgical Laboratory at the University of Chicago (Pardue, Goldstein, and Wollan 1944). ORNL provided K-25 with beta/photon film dosimeters and neutron nuclear track, type A (NTA) emulsion.

Table 6-1 summarizes the personnel dosimeters used at K-25 over the years, along with their periods of use, exchange frequencies, Minimum Detection Levels (MDLs), and estimated annual missed doses. (Note: The dosimeter exchange frequency discussed in this TBD is undergoing additional research to verify its accuracy. In Table 6-1, when the actual frequency is questionable, weekly is assumed (n=50), which provides a claimant-favorable estimate for the potential annual missed dose.) The listed detection levels were measured against calibrated sources and documented in laboratory reports (Gupton 1978; Thornton, Davis, and Gupton 1961). The ORNL external dosimetry technical basis document (Martin Marietta 1994) reported the current Site thermoluminescent dosimeter (TLD) system MDLs, based on DOELAP protocol (DOE 1986).

Table 6-1. K-25 dosimeter type, period of use, exchange frequency, MDL, and potential annual missed dose.

Dosimeter	Period of use	Exchange frequency	Reported laboratory MDL (rem)	Potential annual missed dose (rem)
Beta/photon dosimeters				
ORNL two-element film dosimeter	1945-1951	Weekly (n = 50) Controlled areas	0.03	0.75
ORNL two-element film dosimeter	Nov. 1951-1953	Weekly (n = 50) All employees for ID	0.03	0.75
ORNL multi-element film dosimeter	1953-1960	Weekly (n=50)	0.03	0.75
ORNL multi-element film dosimeter II	1960-1974	Weekly (n=50)	0.03	0.75
	1975-1979	Quarterly (n=4)	0.03	0.06
ORNL TLD Class 1	1980-1987	Quarterly (n=4)	0.03	0.06
Harshaw TLD	1988	Monthly (n=12)	0.005	0.06
Neutron dosimeters				
NTA	1951-1955	Weekly (n = 50)	~0.05	1.25
NTB	1956-1958	Weekly (n = 50)	~0.05	1.25
NTB-2	1958	Weekly (n = 50)	~0.05	1.25
NTA	1958-1976	Weekly (n = 50)	~0.05	1.25
NTA	1976-1978	Quarterly (n=4)	~0.05	0.1
Eastman Type 2	1978-1988	Monthly (n=12)	~0.05	0.3
Harshaw TLD	1988	Monthly (n=12)	0.015	0.09

Dose reconstruction parameters concerning K-25 administrative practices significant to dose reconstruction involve policies to:

- Assign dosimeters to workers
- Exchange dosimeters
- Estimate dose for missing or damaged dosimeters
- Replace any destroyed or missing records
- Evaluate and record dose for incidents
- Obtain and record occupational dose to workers for other employer exposure

K-25 policies were in place for all of these parameters. Routine practices appear to have required assigning dosimeters to all workers who entered a controlled radiation area until November 1951, when all workers wore dosimeters in their photo-identification badges (Thornton, Davis, and Gupton 1961). Dosimeters were exchanged on a routine schedule: weekly until 1976, then quarterly, and finally monthly after 1983. Some individual workers or designated groups of workers were on other exchange frequencies, up to annually, depending on exposure potential. From 1945 to 1979, ORNL processed dosimeters only on request. Universal processing began in 1980. ORNL recorded all measured results and used them to estimate dose. Doses below the detection limit were recorded as zero.

6.3.2 K-25 Dosimetry Technology

The dosimeters provided to the K-25 site by ORNL evolved with dosimetry technology. The purpose of dosimeter designs was to accommodate the various radiation fields that workers might encounter throughout the ORNL complex. The dosimeters provided to the K-25 site were generally simple, designed primarily to measure uranium beta/gamma fields with relatively low potential for significant neutron doses. The history and development was similar for the entire ORNL area because the same contractor was responsible for all sites. Dosimetry practices varied somewhat between sites due to differing radionuclide mixes.

6.3.2.1 Beta/Gamma Dosimeters

In 1945, ORNL implemented the beta/gamma film dosimeter design originally developed at the Metallurgical Laboratory at the University of Chicago (Pardue, Goldstein, and Wollan 1944). ORNL followed a process of dosimetry research and development leading to gradual upgrades in dosimetry capabilities for the complex radiation fields encountered at the Site. Other DOE sites followed this evolution in dosimetry capabilities, leading to site-specific multielement film and thermoluminescent dosimetry systems. Figure 6-1 shows the energy response characteristics of the K-25 beta/gamma dosimeters, based on the essentially identical two-element film dosimeter designed at the University of Chicago (Pardue, Goldstein, and Wollan 1944). Figure 6-1 also shows the Hp(10) response, and the energy response of the thermoluminescent dosimeters (Wilson et al. 1990). The curve labeled "Two element film" is representative of the ORNL dosimeters used from 1945 through 1978. ORNL used a multielement film dosimeter after 1953 (Thornton, Davis, and Gupton 1961), but processed the photon response as it did the two-element dosimeter, using identical shielding. Figure 6-1 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 most of the energy range.

The dosimeter response for photon energies greater than 0.06 MeV is of primary interest for K-25 workers. Because lower energy photons (<0.06 MeV) would be unlikely to contribute significantly to dose at K-25, the under-response in this region is of no consequence. In most areas at K-25, the majority of the photon dose is attributable to photons in the 0.06- to 0.25-MeV range, energies at which dosimeters will overestimate exposure, reported in units of roentgen. Under some circumstances (see Section 6.3.4), particularly when recycled uranium is involved or where ^{238}U progeny tend to concentrate, the photon dose from higher energy photons can be significant. For these higher energy photons, the dosimeter response would be likely to represent exposure and Hp(10) accurately. Overall, dosimeters used at K-25 tended to overestimate photon exposure.

The beta response of the dosimeters was calculated as the difference between the "unshielded" and "shielded" portions of the film with adjustment for the contribution of low-energy photon radiation to the unshielded region of the dosimeter. The dosimeter shallow dose was calibrated to a uranium metal slab, making the adjustment appropriate for K-25. The multielement film dosimeters and TLDs were

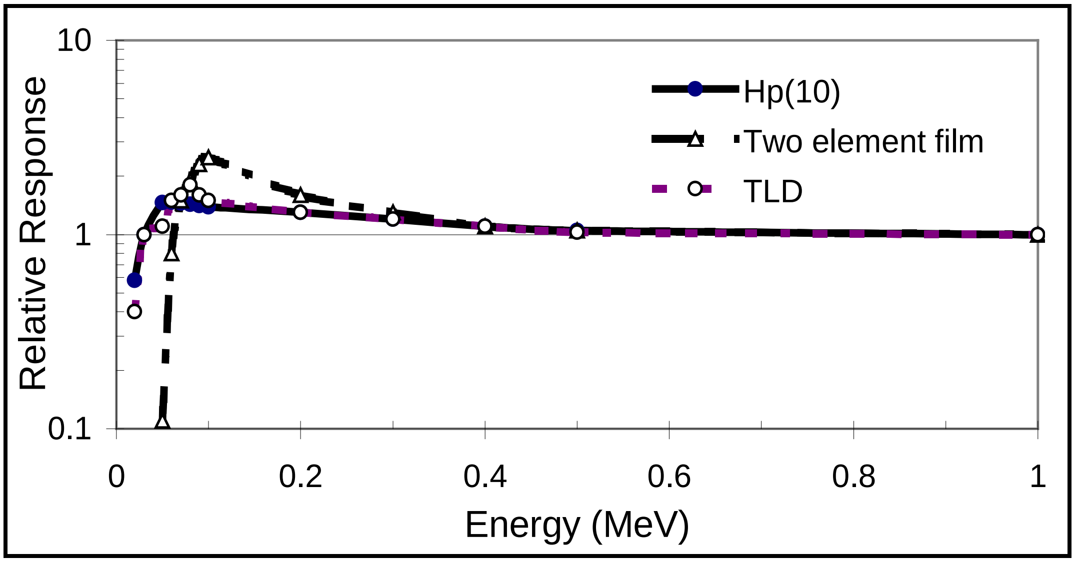

Figure 6-1. K-25 film and TLD dosimeters compared to Hp(10).

also calibrated to uranium slabs, and had the ability to correct more accurately for the beta energy if known.

6.3.2.2 K-25 Neutron Dosimeters

There was minimal potential for significant neutron dose at K-25. Neutron dose rates ranged from zero to less than 0.05 mSv/hr at K-25 locations (Martin Marietta 1994; DOE 2000). K-25 used two types of neutron dosimeters over the years, NTA film and TLD.

NTA Film - The nuclear track emulsion, NTA types A, B, and B-2, neutron dosimeter was an NTA film packet enclosed in the beta/gamma dosimeters, both film and TLD, until the adoption of the Harshaw TLD dosimeter in 1989.

TLD - The Harshaw TLD neutron dosimeter is an integral part of the dosimeter card placed in the identification badge. Processing for neutron dose is on an as-needed basis. TLDs with higher neutron sensitivity might be used in some cases.

6.3.3 Calibration

Potential error in recorded dose is dependent on the dosimeter 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 the field 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 both the NTA film and TLDs for neutron radiation.

6.3.3.1 Beta/Photon Dosimeters

Dosimeters provided to K-25 were calibrated using ^{226}Ra or ^{137}Cs in-air (i.e., no phantom) until the 1986 adoption of DOELAP procedures requiring calibration with phantoms. Photon calibrations with the use of a phantom tended to cause overestimated doses, because backscatter from the body (the intent of a phantom) results in additional dose to the dosimeter for any given exposure in air.

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). Beta calibrations were routinely to a slab of uranium. Thus, the determination of beta dose was specific for uranium.

6.3.3.2 Neutron Dosimeters

Calibration of neutron dosimeters for use at K-25 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 K-25 thermoluminescent neutron dosimeters (TLNDs).

6.3.4 <u>Workplace Radiation Fields</u>

Beta Fields

During normal operations at K-25, beta radiation fields were the dominant external radiation hazard. The source of these beta fields were the ^{238}U daughters ^{234m}Pa and, to a lesser extent, ^{234}Th. The contact beta dose rate from depleted uranium metal is 240 mrem/hr. This dose rate decreases with enrichment. During melting and casting operations, the ^{238}U progeny can concentrate and produce beta radiation fields as high as 20 rads per hour (DOE 2000).

In addition to what would be considered “normal” operations, K-25 processed recycled uranium (RU), which contained trace amounts of radioactive impurities that are not present in natural uranium feed material. Because these impurities were present at minute concentrations, their radiological impact was usually negligible. However, because many routine chemical processes would concentrate them, their presence is worthy of discussion. The most significant beta-emitting impurity found in RU is ^{99}Tc, which tends to deposit in enrichment equipment and will “pocket” in the higher enrichment sections of the gaseous diffusion cascade (DOE 2000). In addition, ^{99}Tc was concentrated at K-25 for purposes of recovery and removal. The relatively low-energy beta (maximum 294 keV) from ^{99}Tc poses minimal external exposure potential due to its limited range. Neither film dosimeters nor TLDs will detect ^{99}Tc betas efficiently, 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 6-2 lists locations and periods of recovery operations at K-25 (Bechtel Jacobs 2000) at which the highest concentrations of ^{99}Tc probably existed.

Table 6-2. ^{99}Tc recovery operations.

Trap locations	Period
K-1031	1952 - 1962
K-1410	1952 - 1962
K-1420	1960 - 1985

Photon Fields
In many areas at K-25, the primary cause of photon doses during normal operations was gamma rays from ^{235}U, with primary photon energies ranging from 144 to 205 keV. This is particularly true in areas where enriched uranium was present (with a higher than natural abundance of ^{235}U). The photon dose rate from uranium is proportional to the level of enrichment. For 93% enriched uranium, the most highly enriched uranium at K-25, the contact photon dose rate is about 30 mrem/hr. The contact photon dose rate from depleted uranium metal is less than 10 mrem/hr.

In some cases, especially when activities involve maintenance and decontamination of equipment contaminated with ^{238}U progeny or melting and casting operations, gamma radiation and *bremsstrahlung* from ^{234m}Pa can be the most significant source of photon dose. In addition, ^{238}U progeny will condense out in process equipment, causing elevated photon dose rates. Gamma dose rates from empty UF_6 cylinders can be significantly greater than those from full cylinders because the UF_6 in the full cylinders effectively shields high-energy photons from uranium progeny on internal surfaces. *Bremsstrahlung*, primarily from the energetic ^{234m}Pa beta, can contribute as much as 40% of the photon dose from uranium metal (DOE 2000).

The impurity from RU that presents the greatest potential radiological hazard from external sources is ^{232}U (DOE 2000). The health hazards of ^{232}U are due primarily to the rapid buildup of gamma activity of its decay products, particularly ^{228}Th (the gamma dose from ^{228}Th is primarily from the decay product ^{208}Tl). The ^{232}U decay products form nonvolatile fluorides that concentrate in cylinders when UF_6 is vapor-fed. Estimates indicate that the level of gamma activity in the enrichment cascade equipment would increase by about a factor of 3 due to the presence of ^{232}U (DOE 2000).

Neutron Fields
While neutrons occur in some areas at K-25, the measured levels are low. Several studies have evaluated neutron fields at gaseous diffusion plants (PNL 1995; Cardarelli 1996); these studies have shown neutron dose to be minimal in all areas. 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 might 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 at the Paducah Gaseous Diffusion plant during a UF_6 cylinder-painting project (Meiners 1999). Results of this project indicated a neutron-to-photon dose equivalent ratio of approximately 20%, based on a quality factor of 10. The associated neutron-to-photon absorbed dose ratio is 2%. Dose reconstructors should assume that these ratios are the same at K-25.

Table 6-3 summarizes the major K-25 radionuclides associated with external dose and their emissions.

6.4 ADJUSTMENTS TO RECORDED DOSE

Photon Dose
Recorded doses varied in reporting units depending on regulatory requirements and dose definitions, national and international. The current reporting unit used by DOE is the mrem, the unit of dose equivalent. The international unit of dose equivalent is the mSv, which is equal to 100 mrem. Since 1986, deep dose equivalents at K-25 were based on DOELAP calibration to Hp(10) and require no

Table 6-3. Major radiation emissions at K-25.

Nuclide	Half-life	Energies (MeV) and abundances of major radiations		
		Alpha	**Beta (max)**	**Gamma**
Primary uranium isotopes				
U-238	4.51×10^9 yr	4.15 (25%)	--	--
		4.20 (75%)		
U-235	7.1×10^8 yr	4.37 (18%)	--	0.144 (11%)
		4.40 (57%)		0.185 (54%)
U-234	2.47×10^5 yr	4.58 (8%)	--	0.204 (5%)
		4.72 (28%)		0.053 (0.20%)
		4.77 (72%)		
Decay products				
Th-234 (U-238)	24.1 d	--	0.103 (21%)	0.063 (3.5%)
			0.193 (79%)	0.093 (4%)
Pa-234m (U-238)	1.17 m		2.29 (98%)	0.765 (0.30%)
				1.001 (0.60%)
Th-231 (U-235)	25.5 hr		0.140 (45%)	0.026 (2%)
			0.220 (15%)	0.084 (10%)
			0.305 (40%)	
Impurities (RU)				
Tc-99	2.12×10^5 yr	--	0.294 (100%)	--
U-236	2.34×10^7 yr	4.49 (76%)		
		4.44 (24%)		
U-232	72 yr	5.26 (31%)		
		5.32 (69%)		
Th-228 (U-232)	1.9 yr	5.34 (28%)		
		5.43 (71%)		
Pb-212 (U-232)	10.64 hr		0.346 (81%)	0.239 (47%)
			0.586 (14%)	0.300 (3.2%)
Bi-212 (U-232)	60.6 min	6.05 (25%)	1.55 (5%)	0.727 (7%)
		6.09 (10%)	2.26 (55%)	
Tl-208 (U-232)	3.10 min		1.28 (25%)*	0.511 (23%)*
			1.52 (21%)*	0.583 (86%)*
			1.80 (50%)*	0.86 (12%)*
				2.614 (100%)*

* Note: branching ratio of 36% from ^{212}Bi.

adjustment. Prior to 1986, dosimeters might have been calibrated in air (i.e., no phantom). In-air calibration will tend to cause overestimates of photon exposures, which is claimant-favorable when roentgen-to-organ dose conversion factors (DCF) are applied (see Section 6.8). In general, dosimeters overestimated worker exposures to photons. This is particularly the case with film dosimeters for the most common photon energies encountered at K-25. No adjustments to recorded film or TLD photon exposures are necessary or recommended. Refer to Section 6.8 for converting reported exposures to organ dose equivalents.

Shallow Dose

Beta calibrations were routinely to a slab of uranium, the primary source of shallow dose at K-25. No adjustment to recorded dose from dosimeter measurements is recommended.

Neutron Dose
The neutron energies at K-25 are between 0.1 and 2.0 MeV, for which the ICRP Publication 60 radiation weighting factor is 20 (ICRP 1990). The associated dose-correction factor is 1.91, which is rounded to 2. Dose reconstructors should apply this factor to both measured neutron dose equivalent and missed neutron dose equivalent.

6.5 MISSED AND UNMONITORED DOSE

The potential for missed dose exists when workers are exposed to radiation at levels below the detection limit of their personnel dosimeters. In the early years of radiation monitoring, when relatively high detection limits are combined with short monitoring durations, missed doses can be significant. Methodologies for estimating missed doses are discussed in this section.

Unmonitored dose pertains to the potential dose received by workers who did not wear personnel dosimetry. For these cases, dose reconstructors must rely on coworker data and/or population data to estimate a worker's potential unmonitored dose. These methods are also discussed in this section.

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

6.5.1 Estimating Missed and Unmonitored Deep Photon Dose

Watson et al. (1994) examined methods analysts can consider when there is no recorded dose for a period during a working career. The missed dose for dosimeter results less than the MDL is particularly important for earlier years when MDLs were higher and dosimeter exchange was more frequent. NIOSH (2002) describes options to calculate missed dose for this situation. The preferred option estimates a claimant-favorable maximum potential missed dose as (MDL)/2 multiplied by the number of zero-dose results. Table 6-1 lists the results of these calculations.

Methods for estimating unmonitored doses are discussed in some detail in the External Dose Reconstruction Implementation Guide (NIOSH 2002). One approach involves the solicitation and use of co-worker data, if available. In general, the maximum reasonable co-worker dose should be assigned as a claimant-favorable estimate. For cases where dosimetry data is intermittent, i.e., workers appear to have been monitored during some time periods and unmonitored for others, the monitoring data prior to and after the missing data can be used to interpolate the missing data. Another approach to estimating unmonitored dose would be to use population data from K-25.

It is reasonable to assume that monitored workers were more likely to receive significant dose than unmonitored workers. That being the case, the average reported dose for monitored workers should represent a claimant-favorable estimate of the dose that may have been received by an unmonitored worker. The first five columns in Table 6-4, show for each year from 1945 through 1988, the number of monitored workers receiving zero dose, the number of monitored workers receiving positive dose, their average recorded deep dose, and the maximum reported individual deep dose (Maisler 2004). (Zero doses were not included in compiling the average.) For use in dose reconstruction, when the number of workers receiving positive dose exceed 100, it was assumed that the exposure data for the year could be represented by a lognormal distribution with a geometric mean (GM) equal to the average shown in column 4 of Table 6-4 and a 99^{th} percentile equal to the maximum in column 5. With these assumptions, the geometric standard deviation (GSD) of the lognormal distribution, shown in column 6, and the 95^{th} percentile dose, last column, were computed. The two parameters, GM and GSD, thus determine the dose distribution assumed for the monitored workers for each year. Their values from columns 4 and 6 in Table 6-4 can be entered directly into IREP.

Table 6-4. Recorded deep doses for workers by year at K-25

Year	Number of monitored workers receiving zero dose	Number of monitored workers receiving positive dose	Average Deep Dose, GM (rem)*	Maximum Deep Dose (rem)	GSD (rem)	95th Percentile Dose (rem)
1945	0	2	0.475	0.530	**	**
1946	22	30	0.161	1.590	3.39	1.198
1947	22	22	0.086	0.490	2.59	0.412
1948	165	92	0.096	3.000	3.38	0.712
1949	377	258	0.145	1.160	2.44	0.629
1950	573	117	0.186	6.195	4.50	2.209
1951	489	232	0.132	0.605	1.93	0.386
1952	753	417	0.118	1.190	2.70	0.603
1953	594	579	0.150	1.510	2.70	0.765
1954	825	189	0.164	1.435	2.54	0.759
1955	929	129	0.093	0.825	2.55	0.435
1956	814	80	0.140	0.735	2.04	0.451
1957	596	163	0.278	3.025	2.79	1.499
1958	608	166	0.297	4.535	3.22	2.035
1959	373	443	0.192	2.475	2.99	1.168
1960	430	505	0.190	3.877	3.64	1.598
1961	549	676	0.089	2.350	4.07	0.898
1962	86	40	0.032	0.400	2.40	0.133
1963	71	42	0.036	0.151	1.92	0.106
1964	68	46	0.026	0.348	2.30	0.103
1965	79	71	0.065	0.544	2.18	0.236
1966	66	60	0.059	1.532	2.85	0.330
1967	66	59	0.068	1.398	2.69	0.345
1968	98	29	0.079	1.334	3.34	0.575
1969	79	35	0.070	0.797	3.01	0.429
1970	84	29	0.099	0.722	2.63	0.487
1971	68	46	0.066	0.622	2.73	0.343
1972	19	92	0.063***	0.693	0.102***	0.267
1973	93	38	0.045	0.331	2.84	0.249
1974	2	138	0.122	1.680	3.08	0.777
1975	466	4448	0.025	0.448	3.46	0.191
1976	3829	817	0.125	2.256	3.46	0.963
1977	4690	1266	0.326	1.960	2.16	1.157
1978	4554	1356	0.097	1.508	3.25	0.673
1979	5358	366	0.118	3.272	4.17	1.231
1980	1470	1653	0.126	7.932	5.92	2.346
1981	184	2921	0.070	2.545	4.68	0.884
1982	697	575	0.061	1.579	4.04	0.607
1983	866	1263	0.068	0.956	3.11	0.439
1984	887	1195	0.051	0.984	3.56	0.412
1985	889	1149	0.141	0.840	2.15	0.497
1986	803	1146	0.383	1.060	1.55	0.786
1987	988	1158	0.249	0.930	1.76	0.631
1988	1196	1166	0.243	1.080	1.90	0.696

* When the number of monitored workers receiving positive dose exceeded 100, the values in this column represent the arithmetic mean, rather than geometric mean (GM). Since the GM is actually less than the arithmetic mean for these cases, treatment of these values as GMs for purposes of dose reconstruction will produce claimant-favorable results.

** Only 2 data points exist for year 1945, therefore values for GSD and 95th percentile are not provided.
*** The data for year 1972 were normally distributed. The values for mean and GSD are the arithmetic mean and standard deviation.

Dose reconstructors may use any of the methods discussed in this section to assign unmonitored dose or assign the estimated "missed dose" values from Table 6-1, whichever is most claimant-favorable.

6.5.2 Estimating Missed and Unmonitored Shallow Dose

The method used to estimate missed deep photon dose is also applicable to missed shallow dose, i.e., MDL/2 multiplied by the number of zero-dose results. Table 6-1 lists the results of these calculations.

All of the methods discussed above for estimating unmonitored deep dose for photons are also applicable to unmonitored shallow dose. If the unmonitored shallow dose is to be estimated based on the K-25 population data, the approach should be identical to that described above for unmonitored deep dose and Table 6-5 should be used (Maisler 2004).

Table 6-5. Recorded shallow doses for workers by year at K-25

Year	Number of monitored workers receiving zero dose	Number of monitored workers receiving positive dose	Average Shallow Dose, GM (rem)*	Maximum Shallow Dose (rem)	GSD (rem)	95th Percentile Dose (rem)
1945	0	2	1.248	1.335	**	**
1946	15	37	0.234	4.150	4.74	3.021
1947	21	23	0.359	3.990	3.64	3.005
1948	164	93	0.252	6.000	3.43	1.913
1949	299	336	0.379	7.334	3.57	3.068
1950	492	198	0.222	6.205	4.17	2.331
1951	346	375	0.326	3.100	2.63	1.599
1952	521	649	0.563	7.970	3.12	3.656
1953	395	778	0.670	9.160	3.07	4.245
1954	594	420	0.544	7.290	3.04	3.399
1955	817	241	0.299	7.765	4.05	2.980
1956	725	169	0.512	2.985	2.13	1.777
1957	520	239	0.681	10.850	3.28	4.806
1958	508	266	0.843	10.955	3.01	5.153
1959	280	536	0.680	8.157	2.90	3.929
1960	379	556	0.565	10.569	3.52	4.467
1961	495	730	0.399	17.449	5.06	5.744
1962	51	75	0.128	1.386	3.37	0.944
1963	42	71	0.138	2.185	3.06	0.866
1964	38	76	0.121	0.720	2.60	0.581
1965	47	103	0.191	1.577	2.53	0.878
1966	58	68	0.345	2.790	2.95	2.045
1967	52	73	0.251	2.807	3.21	1.711
1968	69	58	0.177	1.774	2.90	1.023
1969	60	54	0.317	12.060	4.05	3.163

Table 6-5. Recorded shallow doses for workers by year at K-25 (continued)

Year	Number of monitored workers receiving zero dose	Number of monitored workers receiving positive dose	Average Shallow Dose, GM (rem)*	Maximum Shallow Dose (rem)	GSD (rem)	95^{th} Percentile Dose (rem)
1970	64	49	0.397	18.602	4.21	4.222
1971	55	59	0.234	6.327	3.91	2.207
1972	17	94	0.185	5.135	2.26	0.706
1973	80	51	0.196	2.381	3.04	1.219
1974	2	138	0.196	3.849	3.59	1.604
1975	209	4705	0.076	1.808	3.89	0.713
1976	3438	1208	0.149	3.392	3.82	1.353
1977	4328	1627	0.282	2.588	2.59	1.348
1978	3851	2059	0.108	5.632	5.45	1.763
1979	4930	794	0.136	6.972	5.41	2.193
1980	1306	1817	0.438	23.610	5.54	7.308
1981	128	2977	0.150	6.790	5.14	2.212
1982	576	696	0.187	4.540	3.93	1.778
1983	758	1371	0.241	4.000	3.34	1.750
1984	785	1297	0.274	4.960	3.46	2.117
1985	816	1222	0.353	7.720	3.76	3.116
1986	722	1227	0.321	1.570	1.98	0.984
1987	830	1316	0.288	6.370	3.78	2.563
1988	1026	1336	0.264	6.550	3.97	2.547

* When the number of monitored workers receiving positive dose exceeded 100, the values in this column represent the arithmetic mean, rather than geometric mean (GM). Since the GM is actually less than the arithmetic mean, treatment of these values as GMs for purposes of dose reconstruction will produce claimant-favorable results.

** Only 2 data points exist for year 1945, therefore values for GSD and 95^{th} percentile are not provided.

Specific incident reports might address significant nonroutine worker doses, such as skin contamination events. The dose assessments in such reports, based on investigations conducted at the time of the incident, should be the best estimates of dose received.

Potential doses from ^{99}Tc skin contamination have been evaluated using the VARSKIN computer code. The calculated 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. For this reason, the integrated dose resulting from ^{99}Tc skin contamination could be relatively high (e.g., 0.082 mrem per dpm/cm^2 of initial contamination, assuming a residence half-time on skin of 1.5 days).

In general, direct external beta dose from ^{99}Tc is minimal. The unshielded shallow dose rate to bare skin (no gloves or clothing) at a distance of 10 cm from a contaminated surface is about 10^{-4} mrem/hr per dpm/cm^2, as estimated with VARSKIN. Due in large part to shielding provided by the air, the dose rate at a distance of 30 cm would only be about 10^{-6} mrem/hr per dpm/cm^2. Table 6-5 summarizes these three benchmark values for shallow dose rate as determined from VARSKIN for skin contamination and for external exposure with intervening air.

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

Condition	Dose rate (mrem/hr per dpm/cm2)
Skin contamination	0.0016
External, 10 cm air	10^{-4}
External, 30 cm air	10^{-6}

Some skin contamination events involving ^{99}Tc could have gone undetected. In some cases, it might be appropriate to add an additional skin dose component to a reported shallow dose for a worker who might have had direct contact with ^{99}Tc. To estimate the annual missed dose in the absence of specific data, one must make assumptions regarding the number of times per year an affected skin region might have received contamination and the extent of each contamination. For example, one might assume a monthly contamination event at a specific location on the skin at an average level of 25,000 dpm/100 cm^2 (the action limit for ^{99}Tc contamination on work surfaces and hand tools at Portsmouth Gaseous Diffusion Plant). The annual dose for this example would be (12 × 250 dpm/cm^2 × 0.082 mrem per dpm/cm^2) = 240 mrem. The direct external dose rate at a distance of 10 cm from a surface contaminated at this same level would be (25,000 dpm/100 cm^2 × 10^{-6} mrem/hr per dpm/100 cm^2) = 0.025 mrem/hr at a distance of 10 cm and 0.00025 mrem/hr at a distance of 30 cm.

6.5.3 Estimating Missed and Unmonitored Neutron Dose

Neutron dosimeters were generally insensitive to the low neutron dose rates at K-25, which resulted in routine recordings of “zero” dose. A neutron component should be part of the annual dose of individuals who worked in the cylinder yard. However, dose reconstructors should give careful consideration to the claimant work history. In general, only workers who were near cylinders for extended periods have the potential for neutron exposure. Base an estimate on the neutron-to-photon ratio of 20% for dose equivalent, as determined from the survey conducted at Paducah Gaseous Diffusion Plant (Meiners 1999). Multiply the dose equivalent by 2 to adjust for the ICRP (1990) dose factor.

6.6 UNCERTAINTY

A number of factors contribute to uncertainty in measured doses. Systematic errors can occur from calibration and processing as well as from extraneous conditions such as moisture, heat, and fading. Random errors also arise from variations among workers and the energy spectra and geometries of their exposures. No specific uncertainty assessments for K-25 dosimeter systems were found. However, the systems have much in common with ones used at other laboratories, such as Hanford, for which extensive studies have been carried out (Wilson et al. 1990; Fix, Gilbert, and Baumgartner 1994). The similarities enable reasonable comparisons for the K-25 systems, based on experience at Hanford and elsewhere. Descriptions of uncertainties at the Portsmouth Gaseous Diffusion Plant have also been based on the Hanford studies (PORTS 2004). Operations and dosimetry are similar at Portsmouth and K-25.

Guidance for estimating uncertainty in external dose reconstruction is provided in NIOSH (2002). Under good laboratory conditions, film-badge uncertainty can be at the level of 10% to 15%. The absolute uncertainty at 95% confidence should not be less than the MDL, which for K-25 was 0.03 rem for beta/gamma film (Table 6-1). Figure 2.1 of NIOSH (2002) shows the results from two methods of calculating the uncertainty factor. In the absence of any other site-specific data, it is recommended that numerical values used there be employed for film-badge dose reconstruction.

The uncertainty for TLDs is generally smaller than that for film and somewhat less dependent on energy. An uncertainty factor of 1.15 is suggested as claimant-favorable for doses of 0.10 rem and above and a factor of 1.5 for doses below 0.10 rem. [The factor 1.5 for doses throughout the range between the MDL and 0.10 rem is adequate to compensate for the actual increase to about 2.0 at the MDL, which for K-25 is 0.03 rem (Table 6-1).] These assessments appear to be in line with the data reported elsewhere.

Relatively little information is available on uncertainty for shallow dose. In view of the similar mechanisms between photon and beta film dosimetry, NIOSH (2002) recommends applying the methodology just described to the beta dose.

6.7 DOSE RECONSTRUCTION

As much as possible, the basis for dose to individuals should be the 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 contains appropriate information:

- Worker dosimetry records that provide nonzero beta-photon values for Hp(10) and Hp(0.07) are adequate. Dose reconstructors should consider beta energies to be greater than15 keV and, unless known to be otherwise, photon energies to be within the claimant-favorable energy range of 30 to 250 keV.

- Dose reconstructors should assign missed dose for workers for whom dosimetry records provide zero beta-photon values for Hp(10) and Hp(0.07), as identified in NIOSH (2002). This approach is conservative based on a review of historic data.

- For unmonitored workers, dose reconstructors should solicit and use coworker data. In general, assign the maximum reasonable coworker dose as a claimant-favorable estimate. As an alternative, use population data.

- For unmonitored workers whose exposure potential has been determined to be low, assign the environmental dose.

- Dose reconstructors should multiply neutron doses assigned since 1998 by 2 to adjust for the ICRP (1990) dose factor.

- Dose reconstructors should assign missed neutron dose for cylinder yard workers for whom there is no recorded neutron dose, based on a neutron-to-photon ratio of 20% for dose equivalent (Meiners 1999). Multiply the estimate by 2 to adjust for the ICRP (1990) dose factor 0.1 to 2 MeV.

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

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. For periods when calibrations were performed in free air, prior to 1986, dose reconstructors should use the exposure to organ DCF. For recorded doses from 1986 to the present, use the Hp(10)-to-organ DCF.

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GLOSSARY

beta radiation

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

curie

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

deep dose equivalent (H_d)

The dose equivalent at the respective 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 Gy, H is in Sieverts (Sv). (1 Sv = 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 the 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. The film when developed has an image caused by radiation that can be measured using an optical densitometer. (See *nuclear emulsion*.)

film dosimeter

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

flux density (n/cm^2-sec)

A measure of the intensity of neutron radiation in neutrons/cm^2-sec. It is the number of neutrons passing through 1 square centimeter of a given target in 1 second.

gamma rays (G or γ)

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.

minimum detection level (MDL)
The minimum quantifiable dose equivalent that can be reliably measured with a given dosimetry system.

missed dose
The potential dose equivalent that might not have been measured due to the limitation of the dosimeter, even though a worker was monitored.

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

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

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

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

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

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

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

radiation
Alpha, beta, neutron, and photon radiation.

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

roentgen (R)
A unit of exposure to gamma (or X-ray) radiation. It is defined precisely as the quantity of gamma (or X-) rays that will produce a total charge of 2.58×10^{-4} coulomb in 1 kg of dry air STP. 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

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 Sv = 100 rem.)

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

thermoluminescent dosimeter (TLD)
A device used to measure radiation dose. It is comprised 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.

unmonitored dose
The potential unrecorded dose equivalent that might have resulted because an exposed worker was not monitored.

whole-body dose
Commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1,000 mg/cm2); however, this term is also used to refer to the recorded dose.

X-ray
Ionizing electromagnetic radiation of external nuclear origin or a radiograph.