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## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AEC | U.S. Atomic Energy Commission |
| ANL | Argonne National Laboratory |
| AP | anterior-posterior (X-ray view) |
| AWE | Atomic Weapons Employee |
| BJC | Bechtel Jacobs Company |
| CDC | Centers for Disease Control and Prevention |
| CFR | Code of Federal Regulations |
| $cm^2$ | square centimeter |
| cpm | counts per minute |
| DDE | deep dose equivalent |
| DE | dose equivalent |
| DL | detection limit |
| DOD | U.S. Department of Defense |
| DOE | U.S. Department of Energy |
| DOELAP | DOE Laboratory Accreditation Program |
| DOL | U.S. Department of Labor |
| dpm | discintigrations per minute |
| DU | depleted uranium |
| EEOICPA | Energy Employees Occupational Illness Compensation Program Act of 2000 |
| ERP | extended range product |
| EU | enriched uranium |
| FBI | Federal Bureau of Investigation |
| GAT | Goodyear Atomic Corporation |
| GDP | gaseous diffusion plant |
| GM | Geiger mueller detector |
| HASA | High-Assay Storage Area |
| HEU | highly enriched uranium |
| HEW | Hanford Engineering Works |
| HFM | Hand and Foot monitor |
| HMPD | Hanford Multipurpose TLD |
| HP | health physics |
| Hp(d) | personal dose equivalent at depth d in tissue |
| hr | hour |
| IARC | International Agency for Research on Cancer |
| ICN | International Chemical and Nuclear Corporation |
| ICRP | International Commission on Radiological Protection |
| ICRU | International Commission on Radiation Units and Measurements |
| in. | inch |
| IREP | Interactive RadioEpidemiological Program |
| LANL | Los Alamos National Laboratory |

| | |
|---|---|
| LAT | lateral geometry |
| LAW | low-assay withdrawal |
| LLDR | lower limit of dose limit |
| LOD | limit of detection |
| LODR | limit of dose rate |
| | |
| MDL | Minimum Detection Level |
| MED | Manhattan Engineering District |
| mg | milligram |
| mm | millimeter |
| mR | milliroentgen |
| mrem | millirem |
| mSv | milliseivert |
| | |
| NBS | National Bureau of Standards |
| NCRP | National Council on Radiation Protection and Measurements |
| NIOSH | National Institute for Occupational Safety and Health |
| NRC | Nuclear Regulatory Commission |
| NVLAP | National Laboratory Accreditation Program |
| | |
| ORNL | Oak Ridge National Laboratory |
| OW | open window (i.e., no filter) nonpenetrating dose |
| | |
| PGDP | Paducah Gaseous Diffusion Plant |
| PNL | Pacific Northwest Laboratory |
| PORTS | Portsmouth Gaseous Diffusion Plant |
| PW | product withdrawal |
| | |
| QF | quality factor |
| | |
| R | Roentgen |
| rem | radiation equivalent man |
| rep | radiation equivalent physical |
| RN | radionuclide |
| ROT | rotational geometry |
| RU | recycled uranium |
| | |
| SDE | Shallow Dose equivalent |
| SNM | special nuclear material |
| SRS | Savannah River Site |
| | |
| TBD | technical basis document |
| TED | track-etch dosimetry |
| TEPC | Tissue Equivalent Proportional Counter |
| TLD | thermoluminescent dosimeter |
| TLND | thermoluminescent neutron dosimeter |
| | |
| U.S.C. | United States Code |
| USEC | United States Enrichment Corporation |
| | |
| WB | whole-body |

# 6.0 OCCUPATIONAL EXTERNAL DOSIMETRY

## 6.1 INTRODUCTION

Technical Basis Documents (TBDs) 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. § 7384l (5) and (12)).

PORTS operations, which involved several processes of the nuclear enrichment cycle, played a significant role in the U.S. energy program and the U.S. Department of Defense (DOD) nuclear fuel program. These processes included nuclear fuel enrichment; radiochemical separations; refining, finishing, and storing uranium; and handling the associated radioactive waste.

PORTS workers, especially those employed during the production decades of the 1950s and 1960s, have been exposed to radiation types and energies associated with nuclear energy development processes. PORTS used facility and individual worker monitoring methods to measure and control radiation exposures. Evaluations are difficult because the extensive scope of facility, process, and worker information relevant to an individual worker’s potential dose might involve many years or even decades after employment.

Records of radiation doses to individual workers from personnel dosimeters worn by the worker and coworkers are available for PORTS operations beginning in 1954. Doses received by these dosimeters were recorded at the time of measurement and routinely reviewed by the PORTS operations and radiation safety staff for compliance with radiation control limits. The *External Dose Reconstruction Implementation Guide* (NIOSH 2002) indicates that these records represent the highest quality records for retrospective dose assessments.

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

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

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

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

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

### 6.2 BASIS OF COMPARISON

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

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

## 6.3 DOSE RECONSTRUCTION PARAMETERS

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

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

- **Administrative practices** that facilities adopt to calculate and record personnel dose based on technical, administrative, and statutory compliance considerations.
- **Dosimetry technology**, which includes the physical capabilities of the dosimetry system, such as the response to different types and energies of radiation, in particular in mixed radiation fields.
- **Calibration** of the respective monitoring systems and similarity of the methods of calibration to sources of exposure in the workplace.
- **Workplace radiation fields,** which can include mixed types of radiation, variations in exposure geometries, and environmental conditions.

An evaluation of the 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 PORTS Historic Administrative Practices

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

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

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

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

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

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

Table 6-2. Security badge letter prefix.

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

Note: no prefix was used for PORTS employees.

In another administrative method, PORTS places badges that cannot be assigned to personnel, visitors, or contractors in the “bucket” file. In addition, since at least December 29, 1969, PORTS has processed badges at random for employees not among the “selected employees” at a rate of 100 per quarter for spot checks; the Plant records positive results only for readings greater than the limit of detection (LOD) of 30 mrem (GAT 1971). “Selected employees” were determined by department, job category, or job assignments. Anyone likely to exceed 10% of the contemporary regulatory limits probably received permanent badges. This policy continued from the film dosimetry program though the TLD program until it ended on January 1, 1999.

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

### 6.3.2 PORTS Dosimetry Technology

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

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

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

a. Maximum annual missed dose (NIOSH 2002). [For photon/beta missed dose = LOD/2 × n(frequency, p. 18), for neutron missed dose = LOD/2 × n(frequency), p. 29]

b. Kodak personnel type 2 film with gold sandwiched with cadmium for high-energy gamma, OW with aluminum for low-energy gamma and beta. LOD for SDE and DDE are the same, 0.03 rem, and the reporting level.

c. GAT Film badge procedure, May 5, 1963, reporting level for gamma and beta.

d. Personal communication (Wagner 2003)

e. Bassett (1986, p. 3)

f. ICN (2003).

g. Personal communication (Souleyrette 9/12/2003)

h. Personal communication (McMahon 9/17/2003)

i. GAT (undated)

Table 6-4. PORTS historic dosimetry events.

| Date | Description |
|---|---|
| 9/22/54 | PORTS Film 2-element system started |
| 8/1/60 | Film badge combined with security badge (picture) |
| 7/6/64-12/28/69 | All employees monitored |
| 1/1/1981 | Harshaw 2276 TLD system/4-element enacted (without window), Sr-90 calibration; depleted uranium beta, radium gamma (1985). In addition, company photo was placed over beta chip during 1981 preventing beta (skin) determinations for 1981. |
| 1/1/1983 | Window added to TLD badge |
| 4/20/1986 | DOELAP accreditation process begins |
| 1/1/97 | Neutron monitoring begins with ICN TLD 760; beta/gamma monitoring continues |
| 1/1/97 | BJC splits with USEC, utilizes Y-12 Panasonic TLND dosimetry |
| 1/1/99 | USEC ends PORTS TLD program, utilizes ICN TLD 760 or equivalent |
| 1/1/99 | BJC utilizes ORNL Panasonic 8805 (beta/gamma)/8806 (neutron) |

See Table 6.3 for references. Based on GAT procedures and USEC/BJC procedures.

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

### 6.3.2.1 Beta/Photon Dosimeters

#### 6.3.2.1.1 Film Dosimetry Two-Element, 1954-1980

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

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

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

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

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

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

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

If the shielded portion of the film read greater than zero, but the ratio of the open window reading to the shielded portion reading was greater than the ratio on the calibration graph, the formula:

$$OW - S(R) = \text{beta}$$

was used to determine the darkening due to the gamma exposure,

where

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

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

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

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

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

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

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

a. Ratio of recorded dose to $H_P(10)$
b. (Thierry-Chef et al. 2002).
c. (ORAU 2003a or ORAU 2003b)

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

| Beam (energy, keV) | AP exposure | | | Rotational exposure | | |
|---|---|---|---|---|---|---|
| | Film dosimeters | | | Film dosimeters | | |
| | Two-element 1944–56 | Multielement 1957–71 | TLD 1972–present | Two-element 1944–56 | Multielement 1957–71 | TLD 1972–93 |
| 16 | 0.1 | 0.9 | | | | |
| 59 | 0.5 | 1.1 | | | | |
| M150 (70) | 0.7 | 0.70 | 0.95 | 1.31 | 1.31 | 1.77 |
| H150 (120) | 1.6 | 0.64 | 0.87 | 3.00 | 1.20 | 1.64 |
| $^{137}$Cs (662) | 1.0 | 1.0 | 1.0 | 1.46 | 1.46 | 1.46 |

a. Divide recorded dose by table value to estimate $H_p(10)$.
b. Based on Wilson et al (1990).

#### 6.3.2.1.2 Thermoluminescent Dosimeters, 1981-present

**PORTS Four-Element TLD, January 1, 1981, to December 31, 1998**
The TLD program began in 1981 under Goodyear Atomic Corporation. Most of the following information is from *Thermoluminescent Dosimeter Response and Calibration* (Bassett 1986).

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

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

A TLD system similar to that used at the Hanford or Savannah River Site (SRS) (Panasonic 802D four-element) was evaluated. The TLD was irradiated by several photon energies in exposure orientations as a combination of AP, rotational, and isotropic geometries). As indicated in Table 6-5,

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

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

a. (Bassett 1986, p. 3).

the TLD dosimetry system closely estimated Hp(10) for all exposure orientations within ± 20% of the expected exposure. The 118-keV listed energy (AP and rotational orientation) is the most applicable energy for PORTS exposure potential. A similar French Harshaw TLD with a plastic filter of 1,000 $mg/cm^2$ was within ± 10% of the U.S. Panasonic TLD. Although the Panasonic system might have different filters, the overall response was similar (Thierry-Chef et al. 2002, Table 2, p. 106).

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

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

### 6.3.2.2 Neutron Dosimeters

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

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

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

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

**Neutron Surveys**

Neutron dosimetry was not used at PORTS until January 1, 1992. In 1996, at the request of several union representatives, NIOSH prepared a hazard evaluation report (Cardarelli 1997). This report referred to a survey of neutron radiation levels around 5-in. cylinders of highly enriched uranium (HEU) in storage in 1985. This survey resulted in measurements of 3 mrem/hr at the surface and 0.5 mrem/hr at 1 meter. Lower enrichment 10-ton storage cylinders produced radiation levels of 0.5 mrem/hr neutron dose equivalent (DE) rate at the surface. Based on this information, a worker who spent about 3 hours a week exposed to 0.5 mrem/hr would receive 75 mrem of neutron dose per year. Because this is 1.5% of the regulatory limit of 5,000 mrem, the HP group deemed neutron personnel monitoring to be unnecessary (Cardarelli 1997). [Cylinder lot surveys measured an average gamma DE rate of about 22 mrem/hr at the surface and 3 to 12 mrem/hr at 1 meter (Cylinder lot-745D and Cylinder lot-745G)]

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

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

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

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

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

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

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

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

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

a. Cardarelli (1997), pp. 21 and 22.
b. Measurement period of 1 month

From November 1996 through February 1997, specific process areas and personnel were monitored, using the Landauer/Neutrak ER badge (Landauer 2003). This badge combines an albedo dosimeter with a CR-39 chip, which monitors neutrons of > 30-50 keV to 35 meV. The TLD albedo chip monitors neutrons in the energy range of 0.5 to 100 keV. Thermal neutrons were not monitored due to the assumption that most are intermediate to fast neutrons (Cardarelli 1997)

### 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 similarity between the radiation fields used for calibration and that in the workplace.

#### 6.3.3.1 Film Badges Beta/Photon

For PORTS film dosimeter calibration, eight badges were exposed to a 22.5-mCi $^{226}$Ra source AP at various distances free in air to yield 0 (control), 30, 60, 100, 200, 500, 750, and 1,000 mR of gamma exposure (GAT 1964). To correct for this free-in-air exposure [a conversion from roentgen (87.6 ergs/gram) to rad (95 ergs/gram)], ratios of the differences of rad to roentgen were taken, as follows:

$$\{87.6 - 95\}/87.6 = 8.45\ \% \qquad (6\text{-}1)$$

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

Because $^{226}Ra$ or $^{137}Cs$ was used for calibration, a 8.0 % or 3.9% under-response, respectively , due to higher energy calibration energies would result, compared to the actual lower exposure energies encountered in the workplace (ORAU 2003a). A correction of 16.5% for film dosimetry (1954-1980), 12.5 % for TLD dosimetry (1981-1986) and 4 % for TLD dosimetry (1987-present) is recommended.

Figure 6-1 shows a comparison of a two-element film dosimetry system to the other film dosimetry systems. Wilson et al. (1990) measured the AP photon energy response of the Hanford systems. As indicated in Figure 6-1, the dosimeter open-window response shows a significant over-response to lower energy photon radiation. Most of the photon energy spectrum and the dose equivalent will be from the 30 to 250 keV range at PORTS where there was an over-response from the 2 element film shielded portion of the dosimeter (silver shield). This indicates that recorded results for the PORTS two-element film would overall be claimant-favorable and no corrections are needed for the response of the dosimetry to the radiation work environment. The under-response of the 2 element film shielded portion to low energy of less than 50 kev should be of little consequence since the vast majority of the photon energy spectrum and dose equivalent will be from photons greater than 50 kev. This is due to the fact that most radiation work environments involve shielded or self shielded uranium sources which would allow little exposure from < 30 kev photons. In the case of open systems such as processing, recovery, and maintenance the exposure to low energy photons is more probable. In these situations, since the low energy photons will be monitored conservatively by the open window it may be claimant favorable to equate the shallow dose to the deep dose if the deep dose was found less than the shallow dose in the claimants' records.

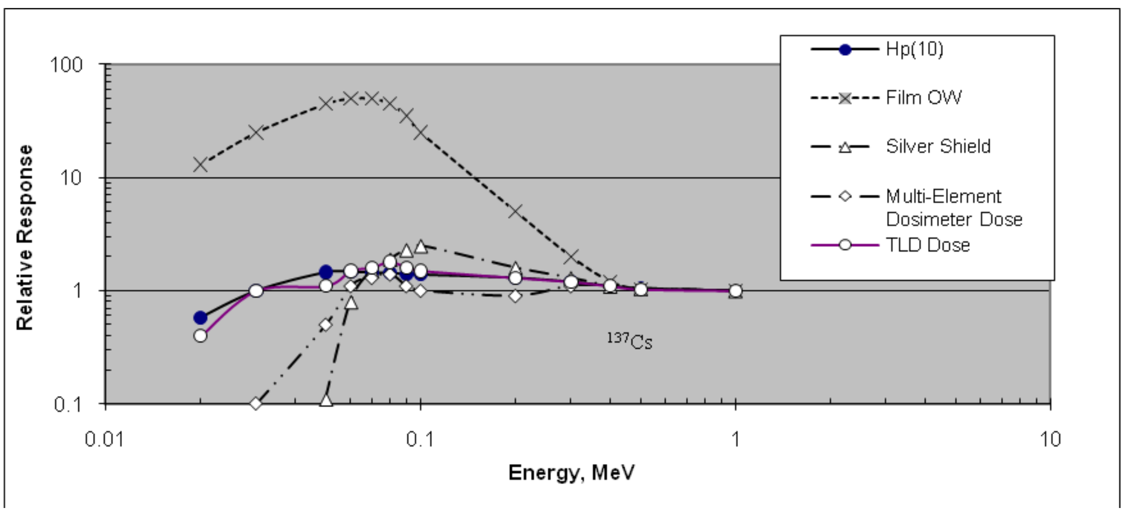


Figure 6-1. Measured Hanford two-element dosimeter photon response characteristics (Wilson et al. 1990).

The film nonpenetrating (i.e., open window) and penetrating (i.e., gold-cadmium filter) response was used to estimate skin dose from beta and photon radiation.

For beta calibration, six badges were exposed to depleted uranium slugs (240 mR/hr) in an AP geometry at various times to yield 0, 60-, 120-, 240-, 480-, and 960-mR beta radiation exposure. The slugs were placed directly on the badges (GAT 1964). This meant that the calibration closely matched the work environment and no corrections for beta response is necessary.

### 6.3.3.2 TLD Badges Beta/Gamma

TLD badges were calibrated in free air with a $^{137}$Cs source, NBS filtered X-rays, and depleted uranium (slab geometry). Because different energies were used for calibration, exposure to DE factors was used in the calibration algorithms, as listed in Table 6-10.

Table 6-10. Roentgen-to-rem conversion factors used for TLD dosimeter calibration[a].

| | Conversion factors (rem/R) shallow (0.007 cm) | Conversion factors (rem/R) deep(1.0 cm) |
|---|---|---|
| NBS filtered X-rays: | | |
| M30 (20 keV) | 1.08 | 0.45 |
| S60 (36 keV) | 1.15 | 1.07 |
| M150 (70 keV) | 1.41 | 1.47 |
| H150 (120 keV) | 1.41 | 1.41 |
| Cs-137 662 | ---- (b) | 1.03 |

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

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

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

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

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

where:

$D_i$ and $S_i$ are the deep and shallow multiple linear regression coefficients for their respective chip positions, and $CR_i$ are the nanocoulomb TL chip responses for chips 1 through 4.

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

Table 6-11 summarizes laboratory sources of uncertainty parameters in beta and photon calibrations. If uncertainty or bias is positive, make no change. In the case of negative bias, make the appropriate corrections using the values in Table 6-11. Uncertainty was estimated using PORTS procedure information or dosimetry performance information from similar dosimetry systems.

### 6.3.3.3 TLD Badges Albedo Neutron

In 1992, some workers in the PORTS Radiation Calibration Facility and the Applied Nuclear Technology Department were placed on a routine neutron dose monitoring program. In October 1994, only workers entering a Controlled or Restricted Area were monitored. From 1992 through 1994, the albedo dosimeter was calibrated with an unmoderated $^{252}$Cf source resulting in higher doses than expected. In 1995, the calibration procedure was modified to utilize a moderated $^{252}$Cf source that would result in lower doses than expected due to the fast neutron environs of PORTS (210 – 860 keV neutrons) (Cardarelli 1997).

The use of the ICN TLND, the Y-12 TLND, and the ORNL TLND from 1997 to the present should not present much variation because all are either NVLAP- or DOELAP-certified, and all use phantoms for calibration and similar geometries. ICN is a large commercial vendor

Table 6-11. Laboratory sources of uncertainty for beta/photon dosimeter calibration parameters.

| Parameter | Historical description | Uncertainty[a] | Comment |
|---|---|---|---|
| In-air calibration | Prior to 1/1/1987, calibrations were performed free in air. In 1997 for BJC and 1999 for USEC, use of commercial or outside vendors for dosimetry began. Phantom for calibration introducing increase of response from backscatter occurred relative to dosimetry used before 1997 and 1999. | ±10% | After 1997, recorded dose of record **too high.** Backscatter radiation from worker's body is highly dependent on dosimeter design (ORAU 2003b, Table 6-2). Before 1987, recorded dose of record too low. (See section 6.3.3.1.) |
| Radiation quantity | Before 1981, PORTS used Cs-137 and Ra-226 for beta and photon beam calibration. | ±5% | For higher energy Cs-137 and Ra-226 for beta and photon beam calibration, this caused about 4% or 8% **under-response** in recorded dose (Basset 1986). |
| Tissue depth of dose | Historically, PORTS used specified depth of 10 mm to estimate deep dose. | ±5% | Numerical effect of this for photon radiation is comparatively low. PORTS dosimeter designs had filtration density thickness of about 1,000 mg/cm$^2$ that would relate closely to 1-cm depth in tissue. |
| Angular response | PORTS dosimeter system is calibrated using AP laboratory irradiations. | ~ -25% 100 keV | Recorded dose of record likely **too low** because dosimeter response is usually lower at non-AP angles. Effect is highly dependent on radiation type and energy (Table 6-13 and 6-16) |
| Environmental effects | Workplace heat, humidity such as dosimeter fading impact dosimeter results. | ±5% | Fading should have been less than 2% for TLDs. Heat effect should have been much less than 1%. Recorded dose due to these effects likely **too low**. |

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

with references to its dosimetry program available at its Internet site (www.ICN.com). Table 6-12 lists possible dosimeter lab parameter uncertainty.

Table 6-12. Laboratory sources of uncertainty for neutron dosimeter calibration parameters.

| Parameter | Historical description | Uncertainty[a] | Comment |
|---|---|---|---|
| Source energy spectra | PORTS used unmoderated and moderated Cf-252 source for calibration. Unmoderated source led to increased bias. | ±100% | Delivered dose used in calibrating neutron dosimeters is **uncertain.** Uncertainty listed is claimant-favorable. |
| Radiation quantity | QF or spectrum used for PORTS by outside vendors. | ±50% | This represents significant and complicated issue. |
| Angular response | PORTS TLND dosimeters calibrated using AP laboratory irradiations. | ±50% | Recorded dose of record likely **too high** because dosimeter |

| | | | |
|---|---|---|---|
| | | | response is often higher at angles other than AP. Effect is highly dependent on energy. |
| Environmental stability | TLD systems are subject to signal fade with time, heat, humidity, light, etc. | ±50% | Recorded dose of record likely **too low** because of fading; however, this effect depends strongly on such routine dosimetry practices as when calibration dosimeters were irradiated. |

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

### 6.3.4 Workplace Radiation Fields

The PORTS radiation work environment is comprised of a variety of combinations of complex beta, gamma (photon), X-ray, neutron, and *bremsstrahlung* (X-ray) radiations. The worker's job, location, and occurrences of incidents would affect the external exposure acquired. Worker location in relation to radiation sources during the course of the work history is one of the most challenging parameters to establish, as is estimation of actual radiation source levels. Assumptions related to these parameter definitions need to be based on available worker information and the work environment. The basis of the assumptions will be given. For the PORTS areas listed in Table 6-13, claimant-favorable assumptions were used. As suggested in NIOSH (2002), 30 to 250 keV for photons is used if a specific gamma spectrum is not known. Several exceptions occur in Building X-330, where a large amount of tails that would contain $^{238}$U in higher concentrations might be present. Because the daughter products of $^{238}$U, $^{234m}$Pa and $^{234}$Th in equilibrium with $^{238}$U, have photon energies greater than 250 keV, the energy bin of 30 to 250 keV was allocated 85% photon field percentage, and > 250 keV was allocated 15% photon field percentage. Buildings X-710 and X-720 have calibration source usage, $^{137}$Cs and $^{226}$Ra, which yield more than 250 keV photons. Therefore, 25% of the photon field percentage was allocated to these areas. This is claimant-favorable because exposure from or usage in areas other than calibration facilities or structural analysis areas is limited. *Bremsstrahlung* could be present in the ambient work area but would be of a lower level than the ambient gamma radiation.

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

PORTS processed recycled uranium (RU), which contained trace amounts of radioactive impurities not present in natural uranium feed material. Because these impurities were present at such minute concentrations, their radiological impact was usually negligible. However, some routine chemical processes would concentrate them. 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 2000a). In addition, $^{99}$Tc was concentrated for recovery and removal. The relatively low-energy beta particles (maximum 294 keV) from $^{99}$Tc pose minimal external exposure potential because of their limited range. Neither film nor TLD efficiently detect them, particularly in the presence of uranium. Clothing and gloves provide adequate shielding. Skin contamination is the most credible scenario in which a significant shallow dose could occur from $^{99}$Tc. Table 6-14 lists the principal locations and periods for which recovery operations at PORTS are believed to have occurred (DOE 2000b).

Table 6-15 summarizes the reactor returns (RU). (See also Section 2 and Table 2.4.2-1 of this TBD). RU comprised about 1,094 MTU of the 330,000 MTU fed to the PORTS cascade (BJC 2000, p. 22).

### 6.3.4.1 Workplace Beta/Photon Dosimeter Response

All PORTS radiological work areas involve beta/photon radiation covering a wide range of energies, which are characterized by radionuclides present in the work environment. Table 6-16 lists properties of the radionuclides and machine sources at PORTS. Information on the actual radiation environment can be reviewed in the Site Description TBD (ORAU 2004) and Attachment A of this PORTS Site Profile.

Table 6-13. PORTS workplace beta/photon dosimeter response.

| Process/ buildings | Description | Operations | | Radiation type | Energy selection | % |
|---|---|---|---|---|---|---|
| | | Begin | End | | | |
| X-326 | High assay withdrawal station (HASA) {HEU} | 1954 | 2001 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| | Purge system (top and side) {EU – HEU} | 1954 | 2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X – 330 | Tails withdrawal station (DU) | 1954 | 2001 | Beta[a]<br>Photon | > 15 keV<br>30 – 250 keV<br>>250 keV[b] | 100<br>85<br>15 |
| X-333 | Reactor grade LAW (EU) [when used as backup for tails withdrawal; see X-330] | 1954 | 2001 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X-342, 342A, & 343 | Fluorine generation (342) and fixed feed (EU) | 1954 | 2001 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X-344 | Feed manufacturing plant | 1958<br>1962 | 1962<br>2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X-345 | Special Nuclear Material (SNM) storage and HASA | 1978 | 2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X-700 | Maintenance | 1954 | 2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X-705[c,d] & 705E[c] | Decontamination, cleaning and recovery, oxide conversion plant | 1954 | 2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |
| X-710[e] | Analytical labs | 1954 | 2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV<br>>250 keV | 100<br>75<br>25 |
| X-720[d] | Compressor shop | 1954 | 2003 | Beta<br>Photon | > 15 keV<br>30 – 250 keV<br>> 250 keV | 100<br>75<br>25 |
| X-744 | Smelter for aluminum recovery | 1961 | 1983 | Beta<br>Photon | > 15 keV<br>30 – 250 keV | 100<br>100 |

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

Table 6-14. Major facilities at PORTS where $^{99}Tc$ might have accumulated.

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

Source: DOE (2000b), p. 16.

Table 6-17 summarizes the common beta/photon personnel dosimeter parameters important to Hp(10) performance in the workplace. PORTS dosimetry has made reasonable measurements of workplace radiation fields. The two-element dosimeter would have over-responded to the average photon field of about 100 keV by about 50% or more. No corrections to over-response are recommended. As of 1987, PORTS estimated shallow dose by adding the gamma (deep) and beta (shallow) doses together. This policy would be an overestimate of shallow dose. Extremity

Table 6-15. Reactor returns fed to cascade.

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

Source: Table 2.2.2.5-1, BJC (2000, p. 22).

exposures were monitored infrequently. BJC has not monitored for extremity exposure.

### 6.3.4.2 PORTS Workplace Neutron Response

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

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

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

#### 6.3.4.2.1 Calibration Laboratory in Building 710

Measurements were made of the bare $^{252}$Cf calibration source in the low scatter room, where it was used with a Tissue Equivalent Proportional Counter (TEPC) at 1 meter. On the day of the measurement (March 12, 1992), the source was rated at 27.54 μgm, corresponding to a dose equivalent rate of 64.9 mrem/hr at 1 meter. The TEPC measurements ranged from 48 to 57 mrem/hour (Soldat and Tanner 1992, p. 3.3).

Pacific Northwest Laboratory (PNL) measured the ORNL bare $^{252}$Cf calibration source for a neutron energy spectrum. The results are indicated by the solid lines in Figure 6-2. Table 6-18 lists the dose fractions for the neutron energy groups indicated by the dashed lines in Figure 6-2. The dose

Table 6-16. Properties of radionuclides that might be found at uranium facilities.[a]

| | | Energies (MeV) and abundances of major radiations | | |
|---|---|---|---|---|
| **Nuclide** | **Half-life** | **Alpha** | **Beta** | **Gamma** |
| Primary uranium isotopes | | | | |
| U-238 | $4.51 \times 10^9$ yr | 4.15 (25%) | -- | 0.013 (8.8%) |
| | | 4.20 (75%) | | |
| U-235 | $7.1 \times 10^8$ yr | 4.37 (18%) | -- | 0.144 (11%) |
| | | 4.40 (57%) | | 0.185 (54%) |
| | | | | 0.013 (31%) |
| U-234 | $2.47 \times 10^5$ yr | 4.58 ( 8%) | -- | 0.204 (5%) |
| | | 4.72 (28%) | | 0.053 (0.20%) |
| | | 4.77 (72%) | | 0.013 (10%) |
| U-236 | $2.34 \times 10^7$ yr | 4.49 (76%) | | 0.013 (10%) |
| | | 4.44 (24%) | | |
| U-232 | 72 yr | 5.26 (31%) | | 0.013 (12%) |
| | | 5.32 (69%) | | |
| Decay Products | | | | |
| | | | | |
| Th-234 (U-238 parent) | 24.1 d | -- | 0.103 (21%) | 0.063 (3.5%) |
| | | | 0.193 (79%) | 0.093 (4%) |
| Pa-234m (U-238 parent) | 1.17 m | | 2.29 (98%) | 0.765 (0.30%) |
| | | | | 1.001 (0.60%) |
| Th-231 (U-235 parent) | 25.5 hr | | 0.140 (45%) | 0.026 (2%) |
| | | | 0.220 (15%) | 0.084 (10%) |
| | | | 0.305 (40%) | |
| Th-230 (U-234 parent) | 77,000 yr | 4.62 (23.4%) | | 0.068 (0.4%) |
| | | 4.69 (76.3%) | | 0.012 (8.4%) |
| Th-228 (U-232 parent) | 1.913 yr | 5.34 (26.7%) | | 0.012 (9.6%) |
| | | 5.42 (72.7%) | | 0.084 (1.2%) |
| Impurities | | | | |
| Tc-99 | $2.12 \times 10^5$ yr | -- | 0.294 | -- |
| Np-237 | $2.14 \times 10^6$ yr | 4.78 (75%) | | 0.030 (14%) |
| | | 4.65 (12%) | | 0.086 (14%) |
| | | | | 0.145 ( 1%) |
| Pu-238 | 86.4 yr | 5.50 (72%) | | |
| | | 5.46 (28%) | | |
| Pu-239 | $24.4 \times 10^4$ yr | 5.16 (73%) | | |
| | | 5.14 (15%) | | |
| | | 5.10 (12%) | | |
| Pu-240 | $6.6 \times 10^3$ yr | 5.17 (76%) | | |
| | | 5.12 (24%) | | |
| Pu-241 | 13.2 yr | | 0.021 | |
| | | | | |
| Cs-137/calibration sources | 30.6 yr | | 0.514 (95%) | .662 (85%) |
| Ra-226 | 1,600 yr | 4.60 (6 %) | | .186 (3.6 %) |
| | | 4.78 (94%) | | |
| Machine generated X-rays | | | | 0.07 – 0.2 |

a. In most part from Bassett (1986).

fractions for the lower (< 10 keV) and intermediate (10 to 100 keV) energy neutron groups were less than 1% of the total dose from the measurements. Thus, combining the lower and intermediate energy groups into the fast group of 0.1 to 2 MeV is a reasonable simplification of the neutron dose calculation.

The use of the ORNL $^{252}Cf$ calibration source is reasonable because the energy spectra would be the same. However, neutron distribution would differ from calibration facility to facility based on scatter of the neutrons or room rate return. This difference should be minimal because calibration facilities are

Table 6-17. Common workplace beta/photon dosimeter $H_p(10)$ performance.[a]

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

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

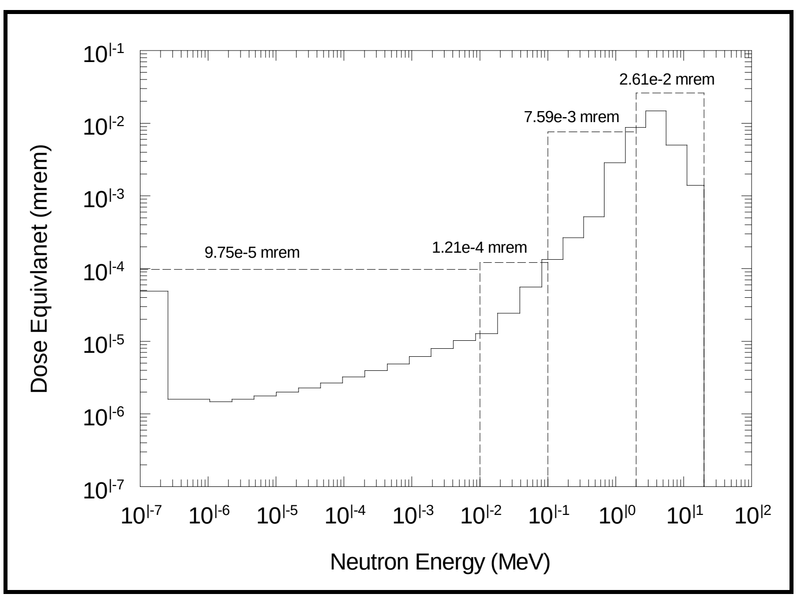

Figure 6-2. Results of neutron spectrum measurements made at 1 meter from bare $^{252}$Cf fission neutron source (Soldat and Tanner 1992). (See Y-12 TBD, Fig. 6.3.4.2-3)

Table 6-18. Dose fractions for PORTS calibration facility.

| Neutron energy group | Near unshielded Cf-252 source |
|---|---|
| < 10 keV | 0.003 |
| 10-100 keV | 0.004 |
| 0.1-2 MeV | 0.224 |
| 2-20 MeV | 0.769 |
| **Claimant-favorable dose fractions** | |
| 0.1-2 MeV | 0.23 |
| 2-20 MeV | 0.77 |

designed to minimize scatter and maintain consistency. This difference would probably not be more than 20% (NCRP 1991).

### 6.3.4.2.2 Nondestructive Assay Laboratory in Building 710

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

The solid lines in Figure 6-3 show the calculated energy spectrum from the multisphere detectors (Bonner Spheres). Table 6-19 lists the dose fractions for the neutron energy groups (indicated by the dashed lines in Figure 6-3). The dose fractions for the lower (< 10 keV) and intermediate (10 – 100 keV) energy neutron groups were about 20% of the total dose from the measurements. Combining the lower and intermediate energy groups into the 0.1 – 2 MeV group is a reasonable simplification of the neutron dose calculation.

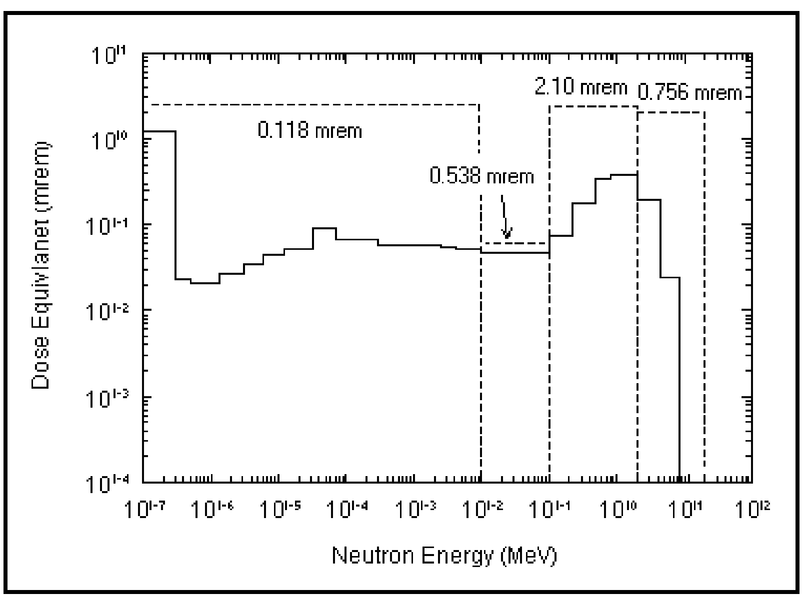

Figure 6-3. Results of neutron spectrum measurements made in the front of shuffler unit (Soldat and Tanner 1992).

Table 6-19. Dose fractions for PORTS nondestructive assay laboratory facility in Building 710.

| Neutron energy group | Near unshielded Cf-252 source |
|---|---|
| < 10 keV | 0.034 |
| 10-100 keV | 0.153 |
| 0.1-2 MeV | 0.598 |
| 2-20 MeV | 0.215 |
| **Claimant-favorable dose fractions** | |
| 0.01-2 MeV | 0.785 |
| 2-20 MeV | 0.215 |

### 6.3.4.2.3 HEU Storage Vault in Building 345

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

The solid lines in Figure 6-4 show the calculated energy spectrum from the multisphere detectors (Bonner Spheres). Table 6-20 lists the dose fractions for the neutron energy groups (indicated by the dashed lines in Figure 6-4). The dose fractions for the lower (< 10 keV) and intermediate (10 –100 keV) energy neutron groups were about 47 % of the total dose from the measurements.

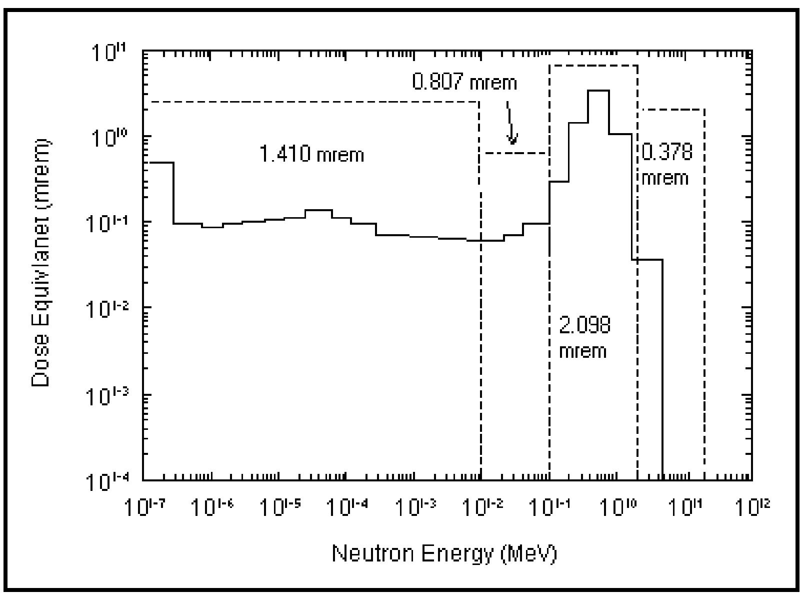


Figure 6-4. Results of neutron spectrum measurements made about 24 in. in front of 93 – 96% HEU cylinders (Soldat and Tanner 1992).

Table 6-20. Dose fractions for PORTS HEU storage vault in Building 345.

| Neutron energy group | Near unshielded Cf-252 source |
|---|---|

| | |
|---|---|
| < 10 keV | 0.300 |
| 10-100 keV | 0.172 |
| 0.1-2 MeV | 0.447 |
| 2-14 MeV | 0.081 |
| **Claimant-favorable dose fractions** | |
| < 10 keV | 0.300 |
| 0.01- 2 MeV | 0.610 |
| 0.1-2 MeV | 0.081 |

### 6.3.4.2.4 Neutron-to-Photon Ratio

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

A neutron dose can be calculated at all PORTS facilities with a potential for such a dose using the neutron-to-photon ratio of 0.125 applied to a fraction of the recorded photon dose. For example, a worker in a process facility with a recorded photon dose of 500 mrem in 1992 would be assigned a 62.5-mrem neutron dose (i.e., 500 × 0.125 = 62.5). The photon dose should be adjusted for missed dose before estimation of the neutron dose. Since the routine monitoring for neutron exposure began in 1997, this ratio method should be used before 1997. The LOD method can apply after 1997 for neutron exposures.

PORTS measurement of the process and storage areas was actually about 0.125:1. For the PORTS neutron-to-photon ratio, a default neutron-to-photon ratio of 0.20 obtained from a PGDP cylinder survey (Meiners 1999) would be reasonable. Table 6-21 lists these values for PORTS facilities.

Table 6-21. PORTS neutron-to-photon dose equivalent ratios.

| Facilities | Neutron-to-photon ratio |
|---|---|
| General areas | 0.20 |
| Calibration facility | 0.20 |
| Nondestructive lab | 0.20 |
| HEU/EU storage | 0.20 |
| Process facilities | 0.20 |

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

### 6.3.4.3 PORTS Workplace Neutron Dosimeter Response

Table 6-22 summarizes typical neutron personnel dosimeter parameters important to Hp(10) performance in the workplace. The most important parameter related to Hp(10) performance of neutron dosimeters is the difference between calibration and workplace neutron energy spectra.

Table 6-22. Typical workplace neutron dosimeter Hp(10) performance.[a]

| Parameter | Description | Potential workplace bias[b] |
|---|---|---|

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

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

### 6.3.5 PORTS Workplace Dose Uncertainty

Uncertainty in the recorded dose emanates primarily from two sources, laboratory bias and workplace radiation field composition and geometry. Sections 6.3.3 and 6.3.4 discuss potential effects of these parameters on the recorded dose. Table 6-23 lists the judged estimates of uncertainty in the recorded dose at PORTS, based on the combination of parameters.

Table 6-23. Estimates of uncertainty.

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

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

## 6.4 ADJUSTMENTS TO RECORDED DOSE

Adjustments to the PORTS reported dose are necessary considering the uncertainty associated primarily with the complex workplace fields and exposure orientations.

### 6.4.1 Beta Dose Adjustments

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

### 6.4.2 Photon Dose Adjustments

Utilizing only AP geometry, the only major dose adjustments necessary to use is a correction of 16.5% for film dosimetry (1954-1980), 12.5 % for TLD dosimetry (1981-1986) and 4 % for TLD dosimetry (1987-present) as listed in Table 6-24.

Table 6-24. Adjustments to reported PORTS deep photon dose.

| Period | Dosimeter | Facility | Adjustment to reported dose |
|---|---|---|---|
| Prior to 1/1/1981 | All beta/photon dosimeters | All facilities | Multiply reported film/TLD deep dose by factor of 1.165 . |

| | | | |
|---|---|---|---|
| 1/1/1981 -1986 | All beta/photon dosimeters | All facilities | Multiply reported film/TLD deep dose by factor of 1.125. |
| 1987 to the present | All beta/photon dosimeters | All facilities | Multiply reported film/TLD deep dose by factor of 1.04. |

### 6.4.3 Neutron Dose Adjustments

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

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

Table 6-25 summarizes historic changes in the quality factors and the average NCRP (1971) quality factor for the neutron energy groups used in dose reconstruction. In addition, Table 6-25 lists the average quality factor for the four neutron energy groups that encompass PORTS neutron exposures. The neutron dose equivalent correction factor can be calculated by dividing the dose fractions from Section 6.3.4.2 for each neutron energy group ($D_f(E_n)$) by the corresponding energy specific average (NCRP 1971) quality factor ($Q(E_n)$) and then multiplying by the ICRP (1991) radiation weighting factor ($w_R$), as shown in equation 6-5.

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

Table 6-26 summarizes the default neutron dose fractions by energy for PORTS work areas where field measurements of neutron spectra occurred, using the associated ICRP (1991) correction factors. The neutron dose equivalent is calculated by multiplying the recorded neutron dose by the area-specific correction factors. For example, consider a 50-mrem recorded neutron dose to a worker at the calibration facility. The corrected neutron dose is 22 mrem from neutrons between 0.1 MeV and 2.0 MeV, (i.e., 50 × 0.44 ) and 51 mrem from neutrons with energy between 2 MeV and 20 MeV, (i.e., 50 × 1.02 ). These adjustments should be applied to measured dose, missed dose, and dose determined based on a neutron-to-photon ratio.

Table 6-25. Historical neutron quality or weighting factors.

| Neutron energy (MeV) | Historical dosimetry guideline[a] | NCRP 38 quality factors[b] | Average quality factor used at PORTS $\bar{Q}(E_n)$ | ICRP 60 neutron weighting factor, $w_R$ [c] |
|---|---|---|---|---|
| $2.5 \times 10^{-8}$ | 3 | 2 | 2.35 | 5 |
| $1 \times 10^{-7}$ | 10 | 2 | | |
| $1 \times 10^{-6}$ | | 2 | | |

| | | | | |
|---|---|---|---|---|
| $1 \times 10^{-5}$ | | 2 | | |
| $1 \times 10^{-4}$ | | 2 | | |
| $1 \times 10^{-3}$ | | 2 | | |
| $1 \times 10^{-2}$ | | 2.5 | 5.38 | 10 |
| $1 \times 10^{-1}$ | | 7.5 | 10.49 | 20 |
| $5 \times 10^{-1}$ | | 11 | | |
| 1 | | 11 | | |
| 2 | | 10 | 7.56 | 10 |
| 2.5 | | 9 | | |
| 5 | | 8 | | |
| 7 | | 7 | | |
| 10 | | 6.5 | | |
| 14 | | 7.5 | | |
| 20 | | 8 | Not applicable | 5 |
| 40 | | 7 | | |
| 60 | | 5.5 | | |

a. Trilateral meeting in 1949 radiation protection guidelines (Fix, Wilson, and Baumgartner 1997).
b. Recommendations of NCRP Report 38 (NCRP 1971).
c. ICRP Publication 60 (ICRP1991)

Table 6-26. PORTS facility neutron dose fractions and associated ICRP (1991) correction factors.

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

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

## 6.5 MISSED AND UNMONITORED DOSE

There is undoubtedly missed recorded dose for PORTS workers. Missed dose applies to workers who were monitored but had results below the LOD of their personal radiation monitors. 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 may have occurred because workers may have the potential of receiving less than 10 % of the radiation protection guidelines or that they worked in uncontrolled areas and not considered radiation workers and therefore were not assigned dosimetry. The following sections discuss beta, photon and neutron missed or unmonitored dose. Dose reconstructors should apply adjustments to beta, photon, and neutron dosimetry discussed in Section 6.4 to the missed dose calculations discussed below. For these cases, dose reconstructors must rely on coworker data and/or population data to estimate a worker's potential unmonitored dose. 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 Missed and Unmonitored Shallow Dose

Missed shallow dose can occur if a zero dose was recorded for a dosimeter reading less than the LOD. Estimates of missed dose can be computed from personnel dose data from either before or after the missing dose period or from coworker data. NIOSH (2002) recommends calculating missed dose by multiplying LOD/2 by the number of zero dose results. The missed shallow dose for dosimeter results less than the LOD is particularly important for earlier years because LODs were higher and dosimeter exchange frequency was higher. The last column in Table 6-27 lists the resulting estimates of this annual missed dose for different years at PORTS. The LOD/2 method seems to be more conservative for most periods. Figure 6-5 shows the PORTS SDE average and the maximum dose by year.

Table 6-27. Missed beta dose according to dosimeter type.

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

For calculating unmonitored dose, the dose reconstructor might choose to utilize the departmental ratios listed in Table 6-29 to approximate doses to unmonitored personnel or unprocessed badges more closely. Take the results of applying the LOD/2 method or the Table 6-28 values and multiply by the corresponding department ratio from Table 6-29. The result should not exceed the cumulative dose that can be obtained by adding the maximum SDE column results for the years in question from Table 6-28. If more than one department is involved, those department ratios can be included by weighting the fraction of time worked in the department. For departments not in the list, choose the department with the closest function.

Table 6-29 is based on the average accumulated skin dose of PGDP employees from 1953 to 1988. The ratios were calculated from dividing the average cumulative dose column by the Mechanical Inspection group cumulative total. Since PORTS and PDGH had similar operations, the relative dose in each departmental area should be similar.

Another option for estimating unmonitored dose to personnel that should have been considered radiation workers is to use the average or maximum doses as listed in Table 6-28. In Table 6-28 the listed average dose is considered the geometric mean, the listed maximum dose the 99% confidence level dose with the assumed distribution log normal as indicated with the geometric standard deviation listed. The listed geometric mean and geometric standard deviation can be directly utilized in the IREP code.

The beta dose reported from 1981 through the end of 1982 may not be as reliable because there was no open window for the PORTS TLD badge for this period. However, the average SDE for 1981-1982 is close to those recorded from subsequent years as indicated in Table 6-28. Table 6-27 includes the potential missed beta dose for the respective periods of use, dosimeter types, LOD, and exchange frequency.

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

Table 6-28. Beta or SDE dose by year (rem).[a]

| Year | Number monitored | Total SDE | Maximum SDE | Average SDE (GeoMean) | Geometric Standard Deviation |
|---|---|---|---|---|---|
| 1954 | 172 | 57.374 | 1.640 | 0.334 | 1.98 |
| 1955 | 903 | 128.187 | 0.867 | 0.142 | 2.17 |
| 1956 | 610 | 53.579 | 0.655 | 0.088 | 2.37 |
| 1957 | 482 | 77.845 | 2.095 | 0.162 | 3.00 |
| 1958 | 873 | 229.719 | 5.770 | 0.263 | 3.76 |
| 1959 | 866 | 442.237 | 10.075 | 0.511 | 3.60 |
| 1960 | 488 | 106.240 | 3.420 | 0.218 | 3.26 |
| 1961 | 549 | 113.291 | 4.890 | 0.206 | 3.89 |
| 1962 | 693 | 94.380 | 1.340 | 0.136 | 2.67 |
| 1963 | 640 | 78.012 | 1.510 | 0.122 | 2.94 |
| 1964 | 621 | 49.253 | 1.388 | 0.117 | 2.89 |
| 1965 | 356 | 37.018 | 1.150 | 0.104 | 2.80 |
| 1966 | 236 | 24.590 | 0.790 | 0.104 | 2.39 |
| 1967 | 141 | 18.606 | 1.135 | 0.132 | 2.52 |
| 1968 | 611 | 90.098 | 1.040 | 0.147 | 2.32 |
| 1969 | 901 | 101.743 | 1.890 | 0.113 | 3.35 |
| 1970 | 414 | 70.396 | 2.028 | 0.170 | 2.90 |
| 1971 | 167 | 30.015 | 2.385 | 0.180 | 3.03 |
| 1972 | 234 | 35.116 | 1.440 | 0.150 | 2.64 |
| 1973 | 207 | 36.700 | 2.140 | 0.177 | 2.91 |
| 1974 | 189 | 27.925 | 2.185 | 0.148 | 3.18 |
| 1975 | 285 | 59.540 | 2.060 | 0.209 | 2.67 |
| 1976 | 363 | 63.585 | 1.350 | 0.175 | 2.40 |
| 1977 | 523 | 71.895 | 1.410 | 0.137 | 2.72 |
| 1978 | 618 | 127.100 | 1.840 | 0.206 | 2.56 |
| 1979 | 393 | 58.970 | 1.120 | 0.150 | 2.37 |
| 1980 | 1,004 | 122.872 | 1.500 | 0.122 | 2.94 |
| 1981 | 587 | 26.203 | 0.535 | 0.045 | 2.89 |
| 1982 | 784 | 37.638 | 1.709 | 0.048 | 4.63 |
| 1983 | 1,092 | 61.048 | 1.749 | 0.056 | 4.38 |
| 1984 | 1,002 | 46.394 | 0.958 | 0.046 | 3.68 |
| 1985 | 1,428 | 45.647 | 0.452 | 0.032 | 3.12 |
| 1986 | 1,308 | 49.436 | 2.198 | 0.038 | 5.71 |
| 1987 | 1,147 | 51.602 | 2.160 | 0.045 | 5.27 |

| | | | | | |
|---|---|---|---|---|---|
| 1988 | 1,189 | 42.922 | 0.608 | 0.036 | 3.36 |
| 1989 | 1,491 | 53.851 | 0.593 | 0.036 | 3.33 |
| 1990 | 1,862 | 81.447 | 0.694 | 0.044 | 3.27 |
| 1991 | 1,509 | 61.066 | 0.517 | 0.040 | 3.00 |
| 1992 | 889 | 50.365 | 0.910 | 0.057 | 3.28 |
| 1993 | 667 | 41.737 | 1.551 | 0.063 | 3.95 |
| 1994 | 583 | 24.610 | 0.507 | 0.042 | 2.91 |
| 1995 | 2,112 | 67.011 | 1.996 | 0.032 | 5.89 |
| 1996 | 3,461 | 100.426 | 1.951 | 0.029 | 6.09 |
| 1997 | 3,394 | 116.483 | 1.281 | 0.034 | 4.75 |
| 1998 | 355 | 14.238 | 0.300 | 0.040 | 2.37 |
| 1999 | 189 | 9.423 | 0.247 | 0.050 | 1.98 |
| 2000 | 703 | 19.103 | 0.338 | 0.027 | 2.96 |
| 2001 | 182 | 5.479 | 0.199 | 0.030 | 2.25 |
| 2002 | 224 | 10.710 | 0.353 | 0.048 | 2.35 |
| 2003 | 385 | 23.498 | 0.569 | 0.061 | 2.61 |

a. Data provided by PORTS HP department. (Litton 2004 and PORTS and PDGH spread sheet 2004)

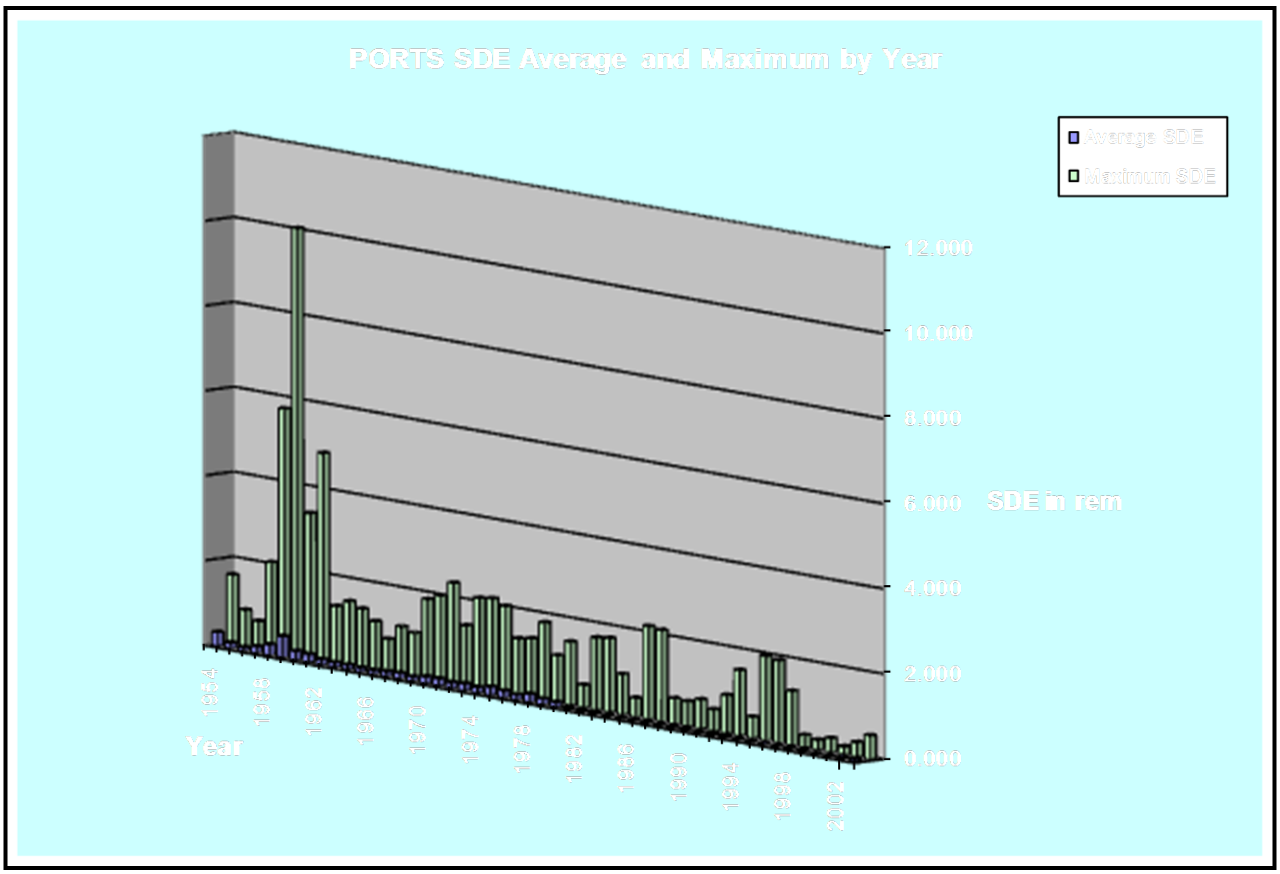


Figure 6-5. PORTS SDE average and maximum dose by year. Data provided by PORTS HP department (PORTS and PGDP spread sheet 2004).

**Energy Range**

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

**Skin Contamination**

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

Potential doses from $^{99}$Tc skin contamination have been evaluated by using the VARSKIN computer code (PNL-7913 DP). The calculated shallow dose rate from uniform $^{99}$Tc skin contamination is $1.6 \times 10^{-3}$ mrem/hr per dpm/cm$^2$ (Swinth 2004). Because the nuclide is difficult to remove from skin, the integrated shallow dose resulting from $^{99}$Tc skin contamination might be relatively large. For example, with a residence half-time of 1.5 days, the dose is $8.1 \times 10^{-2}$ mrem per dpm/cm$^2$.

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

Table 6-29. Departmental SDE dose ratios.

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

a. Adapted from PACE (2000, Table 7-3).

surface contaminated at this same level would be (250 dpm/cm$^2$ × 10$^{-4}$ mrem/hr per dpm/cm$^2$ ) = 0.025 mrem/hr. At 30 cm, the rate would be 0.00025 mrem/hr.

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

The limit of dose rate (LODR) appears to be between 0.07 and 0.04 mrem/hr (see Table 6-30) based on the PORTS monitoring limits. One method of skin contamination monitoring was for workers to wrap their hands around a GM tube. A typical GM tube is about 2% efficient for $^{99}$Tc beta particles. (A contemporary example is a Ludlum model 44-7 – end window GM)

For the routine meter used from at least the 1960s through the 1980s, the Samson meter appears to have had a LOD of 60 cpm, because this was the lowest count rate action limit. This would translate to a 3,000-dpm activity level, which corresponds to a 0.07 mR/hr $^{99}$Tc beta radiation level. Note in Table 6-30 that the lowest radiation field level utilized for an action level is 0.04 mR/hr. This is probably the LODR obtainable from a Cutie Pie ion chamber (used for beta and photon radiation field monitoring from 1965 to 1985).

**External Beta Dose Rates**

As a first method, dose reconstructors might estimate dose from probable external beta dose source terms. PORTS personnel have been exposed to some beta radiation levels, typically during maintenance. For example, with $^{99}$Tc trap maintenance, a typical Mg trap for $^{99}$Tc used for liquid $UF_6$ is a 10-in. diameter by 13.25-in. tall cylinder with about 5,650 grams of Mg trap material. About 1.7 μCi of $^{99}$Tc/gram of absorbent material is trapped, leading to about 10 mCi of $^{99}$Tc in the trap (gaseous

Table 6-30. Beta contamination plant limits for PORTS.[a]

| Area | Type of monitoring | Action limit 1963 | Action limit 1975 | Action limit 1980 and 1990 (Tc-beta) |
|---|---|---|---|---|
| Floor, work surfaces, etc. | Contamination<br>Radiation level | 500 cpm (Samson meter)<br>0.15 mR/hr @ 1 in. | 0.15 mR/hr @ 1 in. | Surface-25,000 dpm<br>500 cpm Samson<br>Removable-5000 dpm<br>100 cpm |
| Hands | Contamination<br>Radiation level | Hand and foot monitor as posted | | |
| Hands | Contamination<br>Radiation level | (Hands wrapped around OW GM tube) 100 cpm<br>0.04 mR/hr | 0.04 mR/hr @ 1 in. | |
| Body | Radiation level | | 0.08 mR/hr | 3,000 dpm/100 cm$^2$<br>(Samson – 60 cpm) |
| Shoes | Contamination or radiation level | Personal 500 cpm (HFM)<br>Issued 5,000 cpm | 0.15 mR/hr<br>1.5 mR/hr | Issued 400,000 dpm/100 cm$^2$ |
| Clothing, tables, equipment, tools, etc. | Radiation level | | 0.15 mR/hr | Coveralls, gloves-200,000 dpm/100 cm$^2$<br>Personal clothing – 9,000 dpm/100 cm$^2$ |
| Airborne activity | Contamination<br>Radiation level | 20 cpm/ft$^3$ (filter paper wrapped around GM tube) | 2000 dpm/ft$^3$ | |
| Hand tools and other equipment | Fixed<br>Removable | | | 25,000 dpm/100 cm$^2$<br>3,000 dpm/100 cm$^2$ |
| Process shop and | Fixed | | | 200,000 dpm/100 cm$^2$ |

| | | | | |
|---|---|---|---|---|
| purge equipment | Removable | | | 3,000 dpm/100 cm$^2$ |

a. Radiation Protection Manual, Portsmouth Facility

$UF_6$ $^{99}$Tc traps accumulate much less activity). Using the VARSKIN code, the beta dose rate result for 10 mCi in a trap at 30 cm (Swinth 2004) is 0.16 mrem/hr at the face. Because there are three traps for every changeout (PORTS) and 50 hr/yr, the resultant dose is 0.16 mrem/hr × 3 traps/changeout × 50 hours = 24 mrem/yr. This method assumes that the personnel were potentially exposed to these beta radiation fields even though most personnel did not perform $^{99}$Tc trap maintenance. In addition, this is a conservative estimate because it assumes there was no shielding of the Mg trap material, the accumulation of activity at the top of the cylinder occurred, and the personnel contact was very close.

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

A second method can be based on the LODR for an end GM of 0.08 mrem/hr (Table 6-30) for the whole body. With the assumption of 2,000 maximum hours of exposure, the exposure calculation would be: 0.08 mrem/hr × 2,000 hr exposure/ year = 160 mrem. Another approximation can be made using the conservative assumption that the technetium: uranium daughter ratio of about 0.40 (Basset 1986) along with the maximum exposure from ambient levels of about 0.2 mrem/hr (Basset 1986) would yield a result of: 0.20 × 0.40 × 2,000 hours/year = 160 mrem.

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

In general, direct external beta dose from $^{99}$Tc is minimal. The unshielded shallow dose rate to bare skin (no clothing) at a distance of 10 cm in air from a uniformly contaminated surface is about $10^{-4}$ mrem/hr per dpm/cm$^2$, as estimated with VARSKIN. The dose rate at 30 cm is only about $10^{-6}$ mrem/hr per dpm/cm$^2$. Table 6-31 summarizes these benchmark values for shallow dose equivalent rate as determined from VARSKIN for skin contamination and for external exposure with intervening air.

Table 6-31. Shallow dose-equivalent rates for $^{99}$Tc.

| **Condition** | **Dose rate (mrem/hr per dpm/cm$^2$)** | **Dose rate mrem/hr** |
|---|---|---|
| Skin contamination | $1.6 \times 10^{-3}$ | N/A |
| External, 10 cm air | $1 \times 10^{-4}$ | 0.08 mrem/hr LLDR whole body |
| External, 30 cm air | $1 \times 10^{-6}$ | 0.20 mrem/hr ambient<br>(X.40 $^{99}$Tc/U daughter ratio) |

For method 1, the total SDE would equal 240 mrem skin dose + 24 mrem external beta dose = 264 mrem/yr.

For method 2, the total SDE would equal 240 mrem skin dose + 160 mrem external beta dose = 400 mrem/yr.

### 6.5.2 Missed and Unmonitored Photon Dose

Missed photon dose can occur if a zero dose was recorded for a dosimeter reading less than the LOD. Estimates of missed dose can be computed from dose before or after the missing dose period or from coworker data. NIOSH (2002) recommends calculating missed dose by multiplying LOD/2 by the number of zero dose results. The missed photon dose for dosimeter results less than the LOD is particularly important for earlier years because LODs and dosimeter exchange frequency were higher. . The last column in Table 6-32 lists the resulting estimates of this annual missed dose for different years at PORTS. The LOD/2 method seems to be more conservative for most periods. Figure 6-6 shows PORTS DDE average and maximum dose by year.

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

For calculating unmonitored dose, the dose reconstructor might choose to utilize departmental ratios listed in Table 6-34 to approximate doses to unmonitored personnel or unprocessed badges. Take the results of applying the LOD/2 method or the Table 6-33 values and multiply by the corresponding department ratio from Table 6-34. The result should not exceed the cumulative dose that can be obtained by adding the maximum DDE column results for the years in question from Table 6-33. If more than one department is involved, include those department ratios by weighting the fraction of time worked in the department. For

Table 6-32. Missed photon dose adjustments to recorded deep dose according to PORTS facility.

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

departments not in the list, choose the department with the closest function. Table 6-33 is based on the average accumulated deep dose of PGDP employees from 1953 to 1988. The ratios were calculated from dividing the average cumulative dose column by the Steam Plant group cumulative total. Since PORTS and PDGH had similar operations, the relative dose in each departmental area should be similar.

Another option for estimating unmonitored dose to personnel that should have been considered radiation workers is to use the average or maximum doses as listed in Table 6-28. In Table 6-33 the listed average dose is considered the geometric mean, the listed maximum dose the 99% confidence level dose and the assumed distribution log normal as indicated with the geometric standard deviation listed. The listed geometric mean and geometric standard deviation can be directly utilized in the IREP code.

**Facility/Location**

Table 6-32 lists the missed photon dose at PORTS facilities. The same information is listed in Table 6-3, which contains the table references. The maximum annual missed dose is the LOD/2 multiplied by the exchange frequency. The LOD is based on laboratory irradiations or from citations from dosimetry vendors.

Table 6-33. Photon or DDE dose by year (rem).[a]

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

| | | | | | |
|---|---|---|---|---|---|
| 1992 | 889 | 19.173 | 0.446 | 0.022 | 3.64 |
| 1993 | 667 | 18.992 | 0.428 | 0.028 | 3.22 |
| 1994 | 583 | 8.852 | 0.259 | 0.015 | 3.40 |
| 1995 | 2,112 | 24.181 | 0.821 | 0.011 | 6.37 |
| 1996 | 3,461 | 35.346 | 0.317 | 0.010 | 4.41 |
| 1997 | 3,394 | 30.374 | 0.292 | 0.009 | 4.45 |
| 1998 | 355 | 10.182 | 0.285 | 0.029 | 2.67 |
| 1999 | 189 | 6.406 | 0.183 | 0.034 | 2.06 |
| 2000 | 703 | 15.404 | 0.173 | 0.022 | 2.42 |
| 2001 | 182 | 4.259 | 0.176 | 0.023 | 2.40 |
| 2002 | 224 | 7.168 | 0.316 | 0.032 | 2.67 |
| 2003 | 385 | 17.590 | 0.496 | 0.046 | 2.77 |

a. Data provided by PORTS HP department. (Litton 2004 and PORTS and PDGH spread sheet 2004)

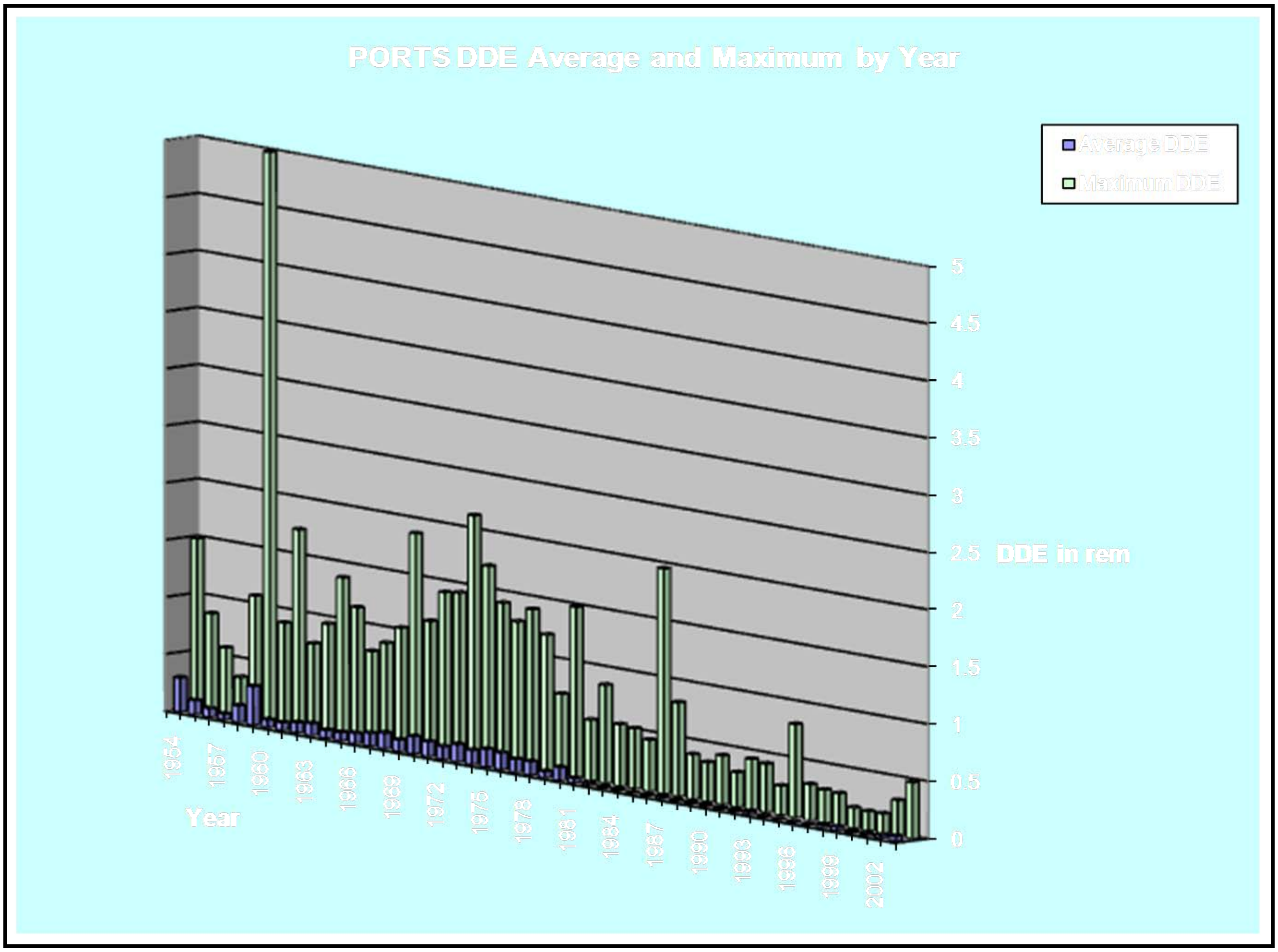


Figure 6-6. PORTS DDE average and maximum dose by year. Data provided by PORTS HP department. (PORTS and PGDP spread sheet 2004)

Table 6-34. Departmental DDE dose ratios.

| Description | Average cumulative dose in mrem[a] | Ratio (using steam plant employees as the basis) |
|---|---|---|
| Plant feed operators | 3,814 | 34.67 |
| Decontamination | 2,788 | 25.35 |

| | | |
|---|---|---|
| Feed Plant Mechanics | 2,587 | 23.52 |
| Cascade Operators | 627 | 5.70 |
| Chemical Operators | 595 | 5.41 |
| Instrument | 538 | 4.89 |
| Transportation Pool | 371 | 3.37 |
| Process Maintenance | 364 | 3.31 |
| Environ Control | 338 | 3.07 |
| Electricians | 298 | 2.71 |
| Mat. Term Mgr | 295 | 2.68 |
| PEMU Decontamination | 253 | 2.30 |
| Mechanical Inspection | 170 | 1.55 |
| Plant services | 147 | 1.34 |
| Converter Test | 145 | 1.32 |
| Feed Plant Mechanics | 143 | 1.30 |
| Nitrogen Plant | 142 | 1.29 |
| Metals Building | 132 | 1.20 |
| Fabrication Shops | 127 | 1.15 |
| Steam Plant | 110 | 1.00 |

a. Adapted from PACE (2000, Table 7-2).

### 6.5.3 Missed and Unmonitored Neutron Dose

Table 6-35 summarizes missed neutron dose for PORTS; see Table 6-3 for references. For the PORTS neutron-to-photon ratio is 0.20. Table 6-21 lists these values for different PORTS facilities. The photon dose should be adjusted for missed dose before estimation of the neutron dose. Since the routine monitoring for neutron exposure began in 1997, the neutron to photon ratio method should be used before 1997. The LOD method can apply after 1997 for neutron exposures. Multiply this dose equivalent by the neutron dose correction factors listed in Table 6-26 for different processes or buildings.

Unmonitored neutron dose may be assigned or be calculated with photon co-worker data or population data as presented in Tables 6-33 and 6-34. The same method as described in the above paragraph can than be applied to the unmonitored neutron dose calculation.

Table 6-35. PORTS dosimeter type, period of use, exchange frequency, LOD, and potential annual dose missed.

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

## 6.6 ORGAN DOSE

### 6.6.1 Organ Dose Conversion Factors

NIOSH (2002) describes the methodology to calculate the organ dose distribution for the respective radiation types using identified exposure geometries. The missed or unmonitored dose is to be estimated as described in Section 6.5. A correction of 16.5% for film dosimetry (1954-1980), 12.5 % for TLD dosimetry (1981-1986) and 4 % for TLD dosimetry (1987-present) is recommended. After the corrections are applied, the R to organ DCFs should be applied for the years 1954-1986 and the $H_P(10)$ to organ DCFs should be applied for the years from 1987 to the present. The DCFs can be found in appendix B of the External dose reconstruction implementation guideline, rev 1. (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 non-penetrating and penetrating doses and the actual Hp(0.07) and Hp(10). The following list summarizes appropriate information:

- Dosimetry records that provide nonzero beta-photon values for Hp(10) and Hp(0.07) are considered adequate. A correction of 16.5% for film dosimetry (1954-1980), 12.5 % for TLD dosimetry (1981-1986) and 4 % for TLD dosimetry (1987-present) is recommended. 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 LOD/2 times the number of zeros, as described in Sections 6.5.1 and 6.5.2 (NIOSH (2002).

- Individuals with no dose recorded and if it is definitely established that the individual was not a radiation worker, then the assigned dose is the environmental dose discussed in the Occupational Environmental Dose portion of this PORTS Site Profile.

- The missed dose is to be estimated as described in Section 6.5. A correction of 16.5% for film dosimetry (1954-1980), 12.5 % for TLD dosimetry (1981-1986) and 4 % for TLD dosimetry (1987-present) is recommended. After the corrections are applied, the R to organ DCFs should be applied for the years 1954-1986 and the $H_P(10)$ to organ DCFs should be applied for the years from 1987 to the present. The DCFs can be found in appendix B of the External dose reconstruction implementation guideline, rev 1. (NIOSH 2002)

- Reported and missed neutron dose equivalents should be adjusted according to 6.4.3 to adjust for ICRP (1990).

- Cylinder yard workers, security guards and general workers for whom no neutron dose is recorded should have missed neutron dose equivalent estimate assigned based on a neutron-to-photon ratio of 0.20 for dose equivalent (Meiners 1999).

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

- Uncertainty is discussed in Table 6.11, 6.12,6.17 and Table 6.23.

## REFERENCES

Bassett, A. C., 1986, *Thermoluminescent Dosimeter Response and Calibration*, GAT-S-56, Piketon, Ohio, April 20.

BJC (Bechtel Jacobs Company LLC), 2000, *Recycled Uranium Mass Balance Project, Portsmouth, Ohio Site Report*, BJC/PORTS-139/R1, U. S. Department of Energy, Office of Environmental Management, Washington, D.C.

Cardarelli, J. J., 1997, *NIOSH Health Hazard Evaluation Report*, HETA 96-1298-2651, prepared for the U.S. Department of Labor, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health, Cincinnati, Ohio.

Cylinder lot-745D, Special Purpose Radiological Survey, 6/23/1998, p 1

Cylinder lot-745G, Special Purpose Radiological Survey, 7/9/1998, p 8

DOE (U.S. Department of Energy), 1986, *Department of Energy Standard for the Performance Testing of Personnel Dosimetry Systems*, DOE/EH-0027, Washington, D.C.

DOE (U.S. Department of Energy), 2000, *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities*, DOE-STD-1136-2000, Washington, D.C.

DOE (U.S. Department of Energy), 2002a, *Independent Investigation of the Portsmouth Gaseous Diffusion Plant, Volume 1: Past Environment, Safety, and Health Practices*, Office of Oversight Environment, Safety, and Health, Washington, D.C.

DOE (U.S. Department of Energy), 2002b, *Independent Investigation of the Portsmouth Gaseous Diffusion Plant, Volume 2: Current Environment, Safety and Health Practices,* Office of Oversight Environment, Safety, and Health, Washington, D.C.

Fix, J. J., R. H. Wilson, and W. V. Baumgartner, 1996, *Retrospective Assessment of Personnel Neutron Dosimetry for Workers at the Hanford Site,* PNNL-11196, Pacific Northwest National Laboratory, Richland, Washington.

Fix, J. J., L. Salmon, G. Cowper, and E. Cardis, 1997. "A Retrospective Evaluation of the Dosimetry Employed in an International Combined Epidemiological Study," Radiation Protection Dosimetry 74, 39-53.

GAT (Goodyear Atomic Corporation), 1963, *Film Badge Procedure*, Piketon, Ohio, (original dated May, 3, 1963).

GAT (Goodyear Atomic Corporation), 1964, *GAT Film Badge Program*, Piketon, Ohio, February 26.

GAT (Goodyear Atomic Corporation), 1971, *Film Badge Program*, Piketon, Ohio.

ICN (International Chemical and Nuclear Corporation), 2003, Internet website, http://www.ICN.com, Dosimetry section, accessed October 2003.

ICRP (International Commission on Radiological Protection), 1983, "Radionuclide Transformations: Energy and Intensity of Emissions," Publication 38, *Annals of the ICRP*, Vol. 21, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1991, "1990 Recommendations of the International Commission on Radiological Protection," Publication 60, *Annals of the ICRP*, Vol. 11, Pergamon Press, Oxford, England.

Landauer (Landauer Inc.), 2003, Internet site, http://www.landauerinc.com/neutron.htm, accessed October 2.

Litton, R., personal communication. February 2004. "PORTS Historical Film/TLD Summary – DDE>0, SDE > 0 or SDEWB > 0"

Meiners, S., 1999, *Paducah UF6 Cylinder Painting Project*, Bechtel Jacobs Company LLC., Paducah, Kentucky.

McMahon, 9/17/2003 Personal communication, "ORNL Panasonic 8805/8806 TLD and TLND LOD"

NCRP (National Council on Radiation Protection and Measurements), 1971, *Protection Against Neutron Radiation*, Report 38, Bethesda, Maryland.

NCRP (National Council on Radiation Protection and Measurements), 1991, *Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination*, Report 112, Bethesda, Maryland.

NIOSH (National Institute for Occupational Safety and Health), 2002, *External Dose Reconstruction Implementation Guideline*, Rev. 1, OCAS-IG-001, Office of Compensation Analysis and Support, Cincinnati, Ohio.

ORAU (Oak Ridge Associated Universities), 2003a, *Technical Basis Document for the Savannah River Site*, ORAUT-TKBS-0003, Oak Ridge, Tennessee.

ORAU (Oak Ridge Associated Universities), 2003b, *Technical Basis Document for the Hanford Site*, Draft, ORAUT-TKBS-0006, Oak Ridge, Tennessee.

ORAU (Oak Ridge Associated Universities), 2004, *Technical Basis Document for the Portsmouth Site*, Draft, ORAUT-TKBS-0015-2, Oak Ridge, Tennessee.

PACE (Paper, Allied Industrial, Chemical and Energy Workers International Union) and University of Utah, 2000, *Exposure Assessment Project at the Paducah Gaseous Diffusion Plant*, Paducah, Kentucky.

PNL-7913 DP, NUREG/CR-5873, "Varskin Mod 2 and SADDE MOD 2: Computer Codes for Assessing Skin Dose from Skin Contamination."

PORTS and PDGH spread sheet 2004. Developed by Demopoulos, P. J. May 2004

Radiation Protection Manual, Portsmouth Facility, "Acceptable Plant Limits for radiation and Radioactive Contamination(4/1/1963), Section 243.5 (7/1/1975) and GAT/GDP-1073, vol 2 pg 4.3-35 (5/10/1990).

Soldat, K. L., and J. E. Tanner, 1992, *Neutron Dose Equivalent and Energy Spectra Measurements at the Portsmouth Gaseous Diffusion Plant*, Pacific Northwest Laboratory, Richland, Washington.

Souleyrette, 9/12/2003 Personal communication, "Y-12 TLND LOD"

Stewart, D. N., 2004, *A Standard Complex-wide Correction Factor for Over-estimating External Doses Measured with Film Badge Dosimeters*, ORAUT-OTIB-0010.

Swinth, K. L., 200 4, E-mail correspondence dated March 5 and March 19, External Dosimetry Task Group.

Thierry-Chef, I., F. Pernicka, M. Marshall, E. Cardis, and P. Andreo, 2002, “Study of a Selection of 10 Historical Types of Dosimeter: Variation of the Response to Hp(10) with Photon Energy and Geometry of Exposure,” *Radiat Prot Dos*, 102(2): 101-113.

USEC (United States Enrichment Corporation), 2003, “The History of PORTS,” Internet site, http://www.usec.com/v2001_02/html/facilities_portsoverview.asp, accessed August 21.

Wagner, 9/2003 Personal communication, “LOD of PORTS Harshaw 2276 4-element TLD”

Wilson, R. H., J. J. Fix, W. V. Baumgartner, and L. L. Nichols, 1990, *Description and Evaluation of the Hanford Personnel Dosimeter Program From 1944 Through 1989*, PNL-7447, Pacific Northwest Laboratory, Richland, Washington.

Wooldridge, F., 1964, *A Description of Co-Operative Work Assignments in Industrial Hygiene and Health Physics,* Goodyear Atomic Corporation, Portsmouth, Ohio, June 1.

**Bibliography**

Bassett, A. C., T. H. Maggard, and W. S. Terry, 1985, *TLD Program Processing Procedures Harshaw 2276 Version BGXI*, GAT-S-54, Goodyear Atomic Corporation, Piketon, Ohio.

Cochran, S., 2002, *Radiation Calibration Facility Normal Operating Procedure*, XP4-GP-RI401, p. 5, December 31.

Fix, J. J., L. Salmon, G. Cowper, and E. Cards, 1997, “A Retrospective Evaluation of the Dosimetry Employed in an International Combined Epidemiologic Study,” *Radia. Prot. Dosi.*, Vol. 74, pp. 39-53.

GAT (Goodyear Atomic Corporation), 1963, *Transportation of RAM*, pp. 2-6, Piketon, Ohio, May 14.

GAT (Goodyear Atomic Corporation), 1963, *Film Calibration*, pp. 1-4 and appendix, Piketon, Ohio, June.

GAT (Goodyear Atomic Corporation), 1964, *Procedure for Extending Film Badge Range*, pp. 1-3, Piketon, Ohio, May 12.

GAT (Goodyear Atomic Corporation), 1971, *General Procedure for X-raying Film Badges*, Piketon, Ohio, April.

Hill, L. R. L., and D. J. Strom, 1993, *Internal Dosimetry Technical Basis Manual for Portsmouth and Paducah Gaseous Diffusion Plants*, PNL-8723, Batelle Memorial Institute, Pacific Northwest Laboratories, Richland, Washington, June.

ICRP (International Commission on Radiological Protection), 1987, “Data for Use in Protection Against External Radiation,” Publication 51, *Annals of the ICRP*, Volume 17, Pergamon Press, Oxford, England.

Ruggles, D. J., 1986, *TLD Error Analysis*, GAT-S-58, Goodyear Atomic Corporation, Piketon, Ohio, January 22.

## GLOSSARY

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

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

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

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

**algorithm**
A computational procedure.

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

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

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

**Bonner Sphere**
See *Multisphere Neutron Spectrometer*

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

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

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

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

**curie**
A special unit of activity. One curie exactly equals $3.7 \times 10^{10}$ nuclear transitions per second.

**deep absorbed dose (Dd)**
The absorbed dose at the depth of 1.0 cm in a material of specified geometry and composition.

**deep dose equivalent (Hd)**
The dose equivalent at the respective depth of 1.0 cm in tissue.

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

**density reading**
See *optical density*.

**dose equivalent (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.)

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

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

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

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

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

**exchange period (frequency)**
Time period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

**exposure**
As used in the technical sense, a measure expressed in roentgens of the ionization produced by gamma (or X) rays in air.

**extremity**
That portion of the arm extending from and including the elbow through the fingertips, and that portion of the leg extending from and including the knee and patella through the tips of the toes.

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

**field calibration**
Dosimeter calibration based on radiation types, intensity and energies present in the work environment.

**film**
Generally means a "film packet" that contains one or more pieces of film in a light-tight wrapping. The film when developed has an image caused by radiation that can be measured using an optical densitometer.

**film density**
See *optical density*.

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

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

**gamma rays**
Electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Physically, gamma rays are identical to X-rays of high energy, the only essential difference being that X-rays do not originate in the nucleus.

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

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

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

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

**hurst dosimeter**
Film-based criticality dosimeter.

**intermediate energy neutron**
Neutron of energy between 0.5 ev (assumed to be 0.4 ev because of cadmium cutoff in neutron response) and 10 keV.

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

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

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

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

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

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

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

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

**neutron**
A basic particle that is electrically neutral weighing nearly the same as the hydrogen atom.

**neutron, fast**
Neutrons with energy equal or greater than 10 keV.

**neutron, intermediate**
Neutrons with energy between 0.4 eV and 10 keV.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

**roentgen**
A unit of exposure to gamma (or X-ray) radiation. It is defined precisely as the quantity of gamma (or X) rays that will produce a total charge of $2.58 \times 10^{-4}$ coulomb in 1 kg of dry air. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

**thermoluminescent dosimeter (TLD)**
A holder containing solid chips of material that when heated will release the stored energy as light. The measurement of this light provides a measurement of absorbed dose. The solid chips are sometimes called crystals.

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

**X-ray**
Ionizing electromagnetic radiation of extranuclear origin.