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Subject Expert(s): George D. Kerr

Site Expert(s): N/A

Approval:	John M. Byrne Signature on File for William E. Murray, Document Owner	Approval Date:	05/15/2009
Concurrence:	Signature on File John M. Byrne, Objective 1 Manager	Concurrence Date:	05/15/2009
Concurrence:	Signature on File Edward F. Maher, Objective 3 Manager	Concurrence Date:	05/15/2009
Concurrence:	Signature on File Kate Kimpan, Project Director	Concurrence Date:	05/21/2009
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☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
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10/11/2005	00 PC-1	Page change initiated to incorporate the definition of U.S.C. on page 7 and details for the definition of a DOE facility on page 8. No sections were deleted. First approved page change revision for Revision 00. Retraining is not required. Initiated by William E. Murray. Approval: Signature on File 10/06/2005 William E. Murray, TBD Team Leader Signature on File 10/04/2005 Judson L. Kenoyer, Task 3 Manager Signature on File 10/04/2005 Richard E. Toohey, Project Director Signature on File 10/11/2005 James W. Neton, Associate Director for Science
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05/11/2006	00 PC-3	Approved page change revision. Updates required language in Introduction (Section 6.1) on page 9. Revised to incorporate additional directions for dose reconstruction for non-presumptive cancers that are excluded from the 1943 through 1947 Special Exposure Cohort on page 10 Section 6.1. Revised to delete text on page 10 in Section 6.1, specified in comments from OCAS review. No sections were deleted. Training required: As determined by the Task Manager. Initiated by William E. Murray. Approval: Signature on File 04/10/2006 William E. Murray, TBD Team Leader

		John M. Byrne Signature on File for 04/10/2006 Judson L. Kenoyer, Task 3 Manager Signature on File 05/10/2006 Edward F. Maher, Task 5 Manager Signature on File 05/02/2006 Kate Kimpan, Project Director Signature on File 05/11/2006 James W. Neton, Associate Director for Science
06/02/2009	01	Revision to add Section 6.1.3.3 and Attachment B to address the Special Exposure Cohort (SEC-00098). Updates required language and adds purpose and scope in Section 6.1. Revised to incorporate additional directions for dose reconstruction for nonpresumptive cancers that are excluded from the March 1, 1943, through December 31, 1947, Special Exposure Cohort in Section 6.1.3.3. Adds Attributions and Annotations Section. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by William E. Murray.

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ACRONYMS AND ABBREVIATIONS

ABRWH	Advisory Board on Radiation and Worker Health
AEC	U.S. Atomic Energy Commission
ANSI	American National Standards Institute
A-P	anterior-posterior
CATI	computer-assisted telephone interview
C.F.R.	Code of Federal Regulations
Ci	curie (a unit of radioactivity)
cm	centimeter
d	day
DCF	dose conversion factor
DHHS	U.S. Department of Health and Human Services
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
ft	foot
g	gram
hr	hour
HEU	highly enriched uranium
HPRR	Health Physics Research Reactor
IAEA	International Atomic Energy Agency
IARC	International Agency for Research on Cancer
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
in.	inch
IREP	Interactive RadioEpidemiological Program
ISO	International Standards Organization
K-25	Oak Ridge Gaseous Diffusion Plant
keV	kiloelectronvolt
kV	kilovolt
MDL	minimum detection limit
MED	Manhattan Engineer District
MeV	megaelectronvolt
mg	milligram
mm	millimeter
min	minute
MMES	Martin Marietta Energy Systems
mrem	millirem
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health

NTA	nuclear track emulsion, type A
OD	optical density
ORAU	Oak Ridge Associated Universities
ORGDP	Oak Ridge Gaseous Diffusion Plant (K-25)
ORISE	Oak Ridge Institute for Science and Education
ORNL	Oak Ridge National Laboratory
PIC	pocket ionization chamber
PNAD	personal neutron accident dosimeter
PNL	Pacific Northwest Laboratory
POC	probability of causation
R	roentgen
RADCAL	Radiation Calibration Laboratory at ORNL
REF	radiation effectiveness factor
RPG	radiation protection guideline
SD	standard deviation
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
SSN	Social Security Number
TBD	technical basis document
TEC	Tennessee Eastman Corporation
TLD	thermoluminescent dosimeter
TLND	thermoluminescent neutron dosimeter
UCC	Union Carbide Corporation
UCC-ND	Union Carbide Corporation-Nuclear Division
U.S.C.	United States Code
WB	whole body
wk	week
Y-12	Y-12 Plant or Y-12 National Security Complex
§	section or sections

6.1 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …" [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. pt. 82) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work (NIOSH 2007).

The statute also includes a definition of a DOE facility that excludes "buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program" [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled "Exposure in the Performance of Duty"] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2007):

- Background radiation, including radiation from naturally occurring radon present in conventional structures

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

- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

6.1.1 Purpose

The purpose of this technical basis document (TBD) is to describe the external dosimetry practices and systems at the Y-12 Plant (now the Y-12 National Security Complex). This document discusses historical and current practices in relation to the evaluation of external exposure data for monitored and unmonitored workers.

6.1.2 Scope

The Y-12 Plant was first conceived in the fall of 1942 by the Manhattan Engineer District (MED) of the U.S. Army Corps of Engineers. The construction of the first building was completed in 1943 (Wilcox 2001, pp. 8–13[2]). The Tennessee Eastman Corporation (TEC) operated Y-12 between June 1943 and May 1947. During this period, the operations at Y-12 primarily involved the use of the electromagnetic separation process to enrich uranium in ^{235}U; Y-12 shipped the enriched product to Los Alamos for production of atomic weapons. Until the latter part of 1945, Y-12 converted UO_3 to UCl_4, which was subsequently enriched in ^{235}U by the electromagnetic separation process using two calutron stages (termed "alpha" and "beta"). In late 1945, Y-12 discontinued the use of the alpha calutron stage, and processes at Y-12 were changed to receive UF_6 from the Oak Ridge Gaseous Diffusion Plant (ORGDP) or K-25 Plant. The UF_6 was then further enriched at Y-12 by the beta calutrons and shipped to Los Alamos. In these early days of Y-12, TEC relied entirely on facility monitoring to measure and control the radiation exposure to workers (Murray 1948a, p. 476; Murray 1948b, p. 469). The nature of the work at Y-12 in these early years resulted in internal occupational exposure to uranium dust being more important than occupational external exposure (Souleyrette 2003a, p. 1).

In May 1947, management of Y-12 was assigned to the Carbide and Chemicals Company, a division of the Union Carbide and Carbon Corporation, and emphasis was directed away from enrichment to the fabrication of nuclear weapon parts. Numerous changes have occurred over the years in the fabrication procedures, but the general features have remained the same. Typically, enriched uranium (EU) was received at Y-12 in the form of UF_6, converted to UF_4, reduced to a metal, and then fabricated into weapon parts. These fabrication processes involved metal casting, rolling and forming, machining, and recycling of the EU salvage. The fabrication of weapon parts was expanded over the years to include other radioactive and non-radioactive materials. In addition to facility monitoring to measure and control the radiation exposure to workers, an external dosimetry program was started in 1948 to monitor individual personnel working in the Assay Laboratories, Radiographic Shop, Spectrographic Shop, and the "Metal" Machine Shops (Murray 1948a, p. 3; Murray 1948b, p. 3; Struxness 1948, p. 8; Struxness 1949a, p. 9; 1949b, p. 6). This program, which monitored less than 25% of the total number of Y-12 employees, was continued through the time that the criticality accident occurred at Y-12 in 1958 (Watkins et al. 1993, p. 39). As a result of the 1958 criticality accident, a program was instituted in 1961 to individually monitor all Y-12 workers for external radiation exposure using a dosimeter system that was an integral part of the worker's identification badge and contained components for both routine and accident dosimetry (West 1993, p. 7). Therefore, Y-12 has used both facility monitoring and individual worker monitoring to measure and control radiation exposures to radiation workers since 1948. The percentages of Y-12 workers who

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[2] Page numbers provided in the text are those specific to the referenced report. Page numbers that may be useful in locating a specific report within a Site Research Database (SRDB) file are provided after the SRDB Ref ID number in the reference list.

were monitored for external radiation exposure from the start of the external dosimetry program in 1948 through the switch to monitoring nearly all workers in 1961 are shown in Figure 6-1. The external monitoring data for Y-12 workers from 1948 to 1950 are not readily available by Social Security Number (SSN) and are not being supplied by Y-12 in response to EEOICPA requests (Souleyrette 2003a, p.1). These 1948 to 1949 external monitoring data were previously made available to Oak Ridge Associated Universities (ORAU), along with other data for use in epidemiological studies of Y-12 and other DOE sites in Oak Ridge, Tennessee (Watkins et al. 1993, p. 15). The data have recently been analyzed and made available for use in dose reconstruction for Y-12 workers (ORAUT 2005, p. 5).

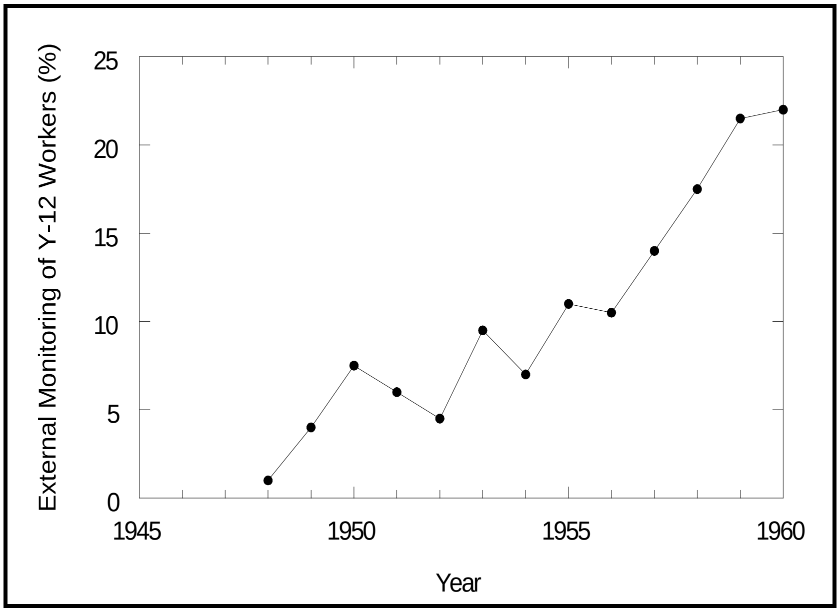

Figure 6-1. Percentage of Y-12 workers monitored for external radiation exposure from start of external monitoring program in 1948 through the switch to monitoring nearly all workers in 1961 (Watkins et al. 1993, p. 39).

There are numerous Y-12 records about facility monitoring, safety evaluations, investigations, and other radiation safety practices. It is time consuming to find and evaluate these records for all Y-12 facilities and processes since 1943. Evaluations of the extensive scope of facility, process, and worker information relevant to an individual worker's potential dose, many years or even decades after employment, are difficult or even impossible in some instances.

Records of radiation dose to individual workers from personnel dosimeters worn by the worker and coworkers are available for some employees from 1948 to 1950, for employees with the highest potential for external radiation exposure from 1950 to 1961, for all workers from 1961 to 1996, and to workers entering radiological areas after 1996 (West 1993; Souleyrette 2003a, p. 2; ORAUT 2005). The information in this section pertains to the analysis of these records but does not address parameters for skin and testicular or breast radiation dose that could result from acute exposure to beta particles in generally nonroutine workplace exposure situations.

Radiation dosimetry practices were initially based on experience during several decades of radium and X-ray usage in medical diagnostic and therapeutic applications. These methods were generally well advanced at the start of the MED project to develop nuclear weapons in the 1940s (Morgan 1961; Taylor 1971). The primary difficulties MED encountered in its efforts to measure worker doses to external radiation were (1) the large quantities of high-level radioactivity that were not encountered previously, (2) the mixed radiation fields of beta particles and photons (X-rays and gamma rays), and (3) neutrons with a broad spectrum of energies (Morgan 1961, p. 5).

6.1.3 Special Exposure Cohort Designations

6.1.3.1 March 1943 to December 1947

Based on the findings and recommendations of NIOSH and the Advisory Board on Radiation and Worker Health (ABRWH), the Secretary of the U.S. Department of Health and Human Services (DHHS) has designated a class of Y-12 employees as an addition to the Special Exposure Cohort (SEC):

DOE employees or contractor or subcontractor employees who worked in uranium enrichment operations or other radiological activities at the Y-12 facility in Oak Ridge, Tennessee, from March 1943 through December 1947, and who were employed for a number of work days aggregating at least 250 work days either solely under this employment or in combination with work days within the parameters (excluding aggregate work day requirements) established for other classes of employees included in the SEC (DHHS 2005).

NIOSH, however, has determined that, for the period from March 1943 to December 1947, the only radiological exposures at Y-12 for which doses can be reconstructed are those that resulted from occupational medical X-rays.

From 1948 onward, the internal dose from uranium, whether occupational or environmental, can be reconstructed using the available information, which includes environmental data, bioassay results, site technical information, and coworker data.

6.1.3.2 January 1948 to December 1957

Based on the findings and recommendations of NIOSH and the ABRWH, the Secretary of DHHS designated the following class of employees as an addition to the SEC:

DOE employees or contractor or subcontractor employees who were monitored or should have been monitored for:

1. Thorium exposures while working in Buildings 9201-3, 9202, 9204-1, 9204-3, 9206, or 9212 at Y-12 for a number of work days aggregating at least 250 work days from January 1948 through December 1957 or in combination with work days within the parameters (excluding aggregate work day requirements) established for one or more classes of employees in the SEC; or

2. Radionuclide exposures associated with cyclotron operations in Building 9201-2 at Y-12 for a number of work days aggregating at least 250 work days from January 1948 through December 1957 or in combination with work days within the parameters (excluding aggregate workday requirements) established for one or more classes of employees in the SEC (DHHS 2006).

NIOSH has determined that it is possible to reconstruct or bound occupationally required medical doses and external gamma, beta, and neutron doses. NIOSH also found that sufficient bioassay data from January 1948 to December 1957 are available for reconstruction of:

1. Internal doses for workers with potential for exposure to uranium or recycled uranium contaminants (^{238}Pu (^{239}Pu in lesser quantities), ^{237}Np, and ^{99}Tc); and

2. Internal doses for workers involved in plutonium operations when plutonium was enriched with the calutrons.

6.1.3.3 March 1, 1943 to December 31, 1947

On August 15, 2008, as provided for under 42 U.S.C. § 7384q(b), the Secretary of DHHS designated the following class of employees as an addition to the SEC:

All employees of DOE, its predecessor agencies, and DOE contractors or subcontractors who worked at the Y-12 Plant in Oak Ridge, Tennessee, from March 1, 1943, through December 31, 1947, for a number of work days aggregating at least 250 work days occurring either solely under this employment or in combination with work days within the parameters established for one or more classes of employees in the SEC (DHHS 2008).

NIOSH found that it is not possible to reconstruct radiation doses for these employees for all potential radiation exposures. However, NIOSH has decided that the occupational medical dose and external exposures directly associated with the calutron uranium enrichment process can be estimated. NIOSH will use limited available external dose rate measurements that might be applicable to an individual claim (and can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals with nonpresumptive cancers or employment of fewer than 250 work days in the class period may be performed using these data as appropriate. (Attachment B contains details on reconstructing the external photon dose.)

6.2 BASIS OF COMPARISON

Since the creation of the MED in the early 1940s, various radiation dose concepts and quantities have been used to measure and record occupational dose. A basis of comparison for dose reconstruction is the personal dose equivalent, $H_p(d)$, 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 $H_p(0.07)$. For penetrating radiation of significance to whole body (WB) dose, d = 10 mm and is noted as $H_p(10)$. Both $H_p(0.07)$ and $H_p(10)$ are the radiation quantities recommended for use as the operational quantity to be recorded for radiological protection purposes by the International Commission on Radiation Units and Measurements (ICRU) in Report 51 (ICRU 1993). In addition, $H_p(0.07)$ and $H_p(10)$ are the radiation quantities that are used in the DOE Laboratory Accreditation Program (DOELAP) to accredit DOE personnel dosimetry systems since the 1980s (DOE 1986). The International Agency for Research on Cancer (IARC) also selected $H_p(10)$ as the quantity to assess error in historical, recorded WB dose for workers in the IARC nuclear worker epidemiologic studies (Thierry-Chef et al. 2002, p. 5).

The basis for comparison for neutron radiation is more complicated because the calibration of dosimeters to measure neutron dose was historically based on different dose quantities such as first collision dose, multiple collision dose, and dose equivalent index. The numerical difference in using these dose quantities in comparison with the $H_p(10)$ dose in current DOELAP performance testing could be evaluated by using the relative values of the dose conversion factors (DCFs) for the respective dose quantities in conjunction with characteristics of the respective Y-12 neutron dosimeter response characteristics and workplace radiation fields (DOE 1986).

6.3 DOSE RECONSTRUCTION PARAMETERS

Examinations of the beta, photon (X-ray and gamma ray), and neutron radiation types, energies, and exposure geometries, and the characteristics of the respective Y-12 dosimeter response, are crucial for the assessment of bias and uncertainty of the original recorded dose in relation to the radiation quantity $H_p(10)$. Bias and uncertainty for current dosimetry systems are typically well documented for $H_p(10)$ (Oxley 2001, pp. 37–49). The performance of current dosimeters can often be compared with performance characteristics of historical dosimetry systems in the same, or highly similar, facilities or workplaces. In addition, current performance testing techniques can be applied to earlier dosimetry systems to achieve a consistent evaluation of all dosimetry systems (Wilson et al. 1990). Dosimeter response characteristics for radiation types and energies in the workplace are crucial to the overall analysis of error in recorded dose.

Overall, the accuracy and precision of the original recorded individual worker doses and their comparability to be considered in using NIOSH (2002) guidelines depend on (Fix et al, 1997a):

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

- **Dosimetry technology** including the physical capabilities of the dosimetry system, such as the response to radiation type and energy, especially in mixed radiation fields.

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

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

An evaluation of the original recorded doses based on these parameters is expected to provide the best estimate of $H_p(0.07)$, as necessary, and $H_p(10)$ for individual workers with the least relative overall uncertainty.

6.3.1 Y-12 Historical Administrative Practices

A dosimetry program was started in 1948 to monitor individual external exposures of personnel working in the Assay Laboratories, Radiographic Shop, Spectrographic Shop, and “Metal” Machine Shops where uranium metals were handled (Murray 1948a, p. 3; Murray 1948b, p. 3; Struxness 1948, p. 8; Struxness 1949a, p. 9). At first, the external radiation monitoring was performed using pocket ionization chambers (PICs), typically exchanged on a weekly basis (Souleyrette 2003a, p. 1). Early efforts were concerned with using a photographic film pad on the hands of the uranium metalworkers and attempting to correlate the film pad reading with WB exposures, which were recorded first with PICs and later with personnel WB film badge dosimeters (Souleyrette 2003a, p. 1). The Y-12 Plant used film badges for external monitoring of exposures to beta particles and photons (X-rays and gamma rays) (Souleyrette 2003a, p. 2; West 1993, p. 11). In 1950, they began to use nuclear track emulsion, Type A (NTA) film for monitoring worker exposures to fast neutrons (Struxness 1954, p. 31).

In 1980, the film badge dosimeters for beta particles and photons were replaced with thermoluminescent dosimeters (TLDs) (McLendon et al. 1980, pp. 5–7; Howell and Batte 1982, p. 6). Thermoluminescent neutron dosimeters (TLNDs) were added to NTA badge dosimeters for neutrons (Gupton 1978, p. 3; Berger and Lane 1985, p. 3). The TLND was used to determine the doses from thermal and intermediate energy neutrons, and the NTA films were maintained to determine the doses from fast neutrons (Gupton 1978, pp. 8–9).

These combination TLND and NTA dosimeters were replaced in 1990 with newer dosimeters containing TLDs for the measurement of dose from beta particles and photons and TLNDs for the measurement of doses from neutrons of all energies (Oxley 2001, pp. 3–6; McMahan 1991). The frequency of exchange of these personnel dosimeters is summarized in Table 6-1 (Souleyrette 2003a; Watkins et al. 1993; West 1993).

Table 6-1. Monitoring technique and exchange frequency at Y-12 Plant for external WB exposures.[a]

Period	Monitoring technique	Exchange frequency	Monitored personnel
Beta/photon dosimeters			
1948–1950	PICs and two-element film dosimeters	Some daily, some weekly	Personnel expected to receive over 10% of the radiation protection guideline (RPG)
1950–04/07/58	Two-element film dosimeters	Weekly	Personnel expected to receive over 10% of the RPG
04/08/58–06/30/61	Two-element film dosimeters	Monthly	Personnel expected to receive over 10% of the RPG
06/30/61–10/01/80	Four-element film dosimeters	Quarterly	Nearly all personnel monitored
10/01/80–01/03/89	Two-element TLDs	Some quarterly, some annually, a very limited group on a monthly basis	Quarterly exchange for personnel expected to receive 500 mrem or more, annual exchange for personnel expected to receive less than 500 mrem
01/03/89–present	Four-element TLDs	Mostly quarterly, some monthly	Nearly all personnel monitored from 1989 to 1996. After 1996, only personnel entering radiological areas.
Neutron dosimeters			
1950–10/01/80	NTA film	Biweekly, monthly, and quarterly	Personnel exposed to neutron sources
10/01/80–01/03/89	NTA film for fast neutrons and TLND dosimeters for other energy neutrons	Quarterly	Personnel exposed to neutron sources
01/03/89–present	TLNDs for neutrons of all energies	Quarterly	Personnel exposed to neutron sources

a. Souleyrette (2003a), Watkins et al. (1993), West (1993).

The minimum detection level (MDL) of the various dosimeters used at Y-12 to monitor for beta/gamma and neutron exposures of the whole body is summarized in Table 6-2. The first film dosimeter that Y-12 used was identical to the badge used at the Oak Ridge National Laboratory (ORNL) in 1949 (West 1993, p. 2), which is described by Thornton, Davis, and Gupton (1961, pp. 2-4). This film badge was a U.S. Atomic Energy Commission (AEC) PF-1B film badge manufactured by the A. M. Samples Machine Company in Knoxville, Tennessee (Patterson, West, and McLendon 1957, p.21). It had an open window over a portion of the film to measure both beta and photon radiation, and a 1-mm cadmium filter over a portion of the film to measure only higher energy photon radiation (Handloser 1959, p. 152). This film dosimeter was used at Y-12 until 1961, when a newer film dosimeter was developed for use at all Union Carbide Corporation-Nuclear Division (UCC-ND) facilities (Thornton, Davis, and Gupton 1961, pp. 2–8). This newer film dosimeter served as an identification badge and also provided for both personnel routine and accident monitoring.

Some of the MDLs in Table 6-2 were difficult to estimate, particularly for the film dosimeters. For the current TLD dosimeters, the MDLs are precisely identified in *Technical Basis for the External Dosimetry Program at the Y-12 National Security Complex* (Oxley 2001, pp. 43–45), which is based on a DOELAP protocol (DOE 1986). For the film dosimeters, the MDLs are subject to additional uncertainty because factors that involve the radiation field, film type, processing, developing, and reading system cannot be tested. The MDLs for the film dosimeters in Table 6-2 were based on information from Souleyrette (2003a, pp. 5–6), Watkins et al. (1993, pp. 28–29), West (1993), and Wilson et al. (1990, p. 7.6).

Table 6-2. Dosimeter type, period of use, exchange frequency, laboratory MDL, and maximum annual missed dose.[a]

Dosimeter	Period	Exchange frequency	Laboratory MDL (mrem)	Maximum annual missed dose (mrem)
Beta/photon dosimeters				
Pocket ionization chamber	1948–1950	Daily	<5	1,300
		Weekly	<5	260
Two-element film badge	1948–1958	Weekly	40	2,080
	1958–1961	Monthly	40	480
Four-element film badge	1961–1980	Quarterly	40	160
Two-element TLD dosimeter	1980–1989	Quarterly	20	80
Four-element TLD dosimeter	1989–Present	Quarterly	10	40
Neutron dosimeters				
NTA film	1948–1980	Biweekly	<50	1,300[b]
		Monthly	<50	600[b]
		Quarterly	<50	200[b]
Combination NTA film and TLND dosimeter	1980–1989	Quarterly	<50	200[b]
TLND dosimeter	1989–Present	Quarterly	10	40

a. Souleyrette (2003a, pp. 5–6), Watkins et al. (1993. pp. 28–29), West (1993), and Wilson et al. (1990, p. 7.6).
b. Potential annual missed dose based on data from laboratory irradiations may not be directly applicable to workplace missed dose (Wilson et al. 1990, p. 7.6; IAEA 1985, pp. 63-71).

The Y-12 Plant administration practices that are important to dose reconstruction include the following policies for:

- Assigning dosimeters to workers,
- Exchanging dosimeters,
- Recording notional dose (i.e., some identified values for lower dose workers based on a small fraction of the regulatory limit),
- Estimating dose from lost or damaged dosimeters,
- Replacing destroyed or missing records,
- Evaluating and recording doses for incidents or accidents, and
- Obtaining and recording occupational dose to workers for other employer exposures.

Policies appear to have been in place at Y-12 for all of these parameters (BWXT Y-12 2000, Section 2.7, pp. 22–26; Patterson, West and McLendon 1957, pp. 7–8, 21–26; McLendon 1963, pp. 11–13, 23–25, 95–101; McRee, West, and McLendon 1965, pp. 10–11, 19–21, 87–93; McLendon et al. 1980; West 1993; Oxley 2001; Souleyrette 2003a, pp. 1–6). Their routine practices appear to have required assigning dosimeters to personnel who might receive an external radiation dose that was greater than 10% of the radiation protection guidelines (RPGs) in effect at that time. According to West (1993), all workers were monitored from 1961 to 1979, after which only those workers having an exposure potential were monitored quarterly and others annually. Souleyrette (2003a, p. 2) states that nearly all workers were monitored from 1989 to 1996 and all workers entering radiological areas were monitored after 1996.

Dosimeters were exchanged on a routine schedule. All beta/photon dosimeters were processed and the measured results were recorded and used for dose estimation to the individual workers. Unless the worker was actually or could have been exposed to neutrons, a TLND or NTA film was not issued to the worker (West 1955) or the NTA emulsion in the worker's film badge dosimeter was not processed (West 1993, p. 5). There appears to be no use of recorded notional doses, although there are issues of recording dose for low-dose exposures (see Table 6-2). There is also the problem of missing dose components from the total WB dose for a worker designated simply as "not available" or "damaged film" in the worker's records (West 1993). These missing dose components for workers could be estimated using a method described by Watson et al. (1994) and Watkins et al. (1994, pp. 22–26) based on examination of continuity in the worker's job and work activities.

6.3.2 Y-12 Dosimetry Technology

The Y-12 dosimetry methods evolved during the years as improved technology was developed and the complex radiation fields in the workplace were better understood. The adequacy of the respective dosimetry methods to accurately measure radiation dose as discussed in later sections depends on radiation type, energy, exposure geometry, etc. The dosimeter exchange frequency of the dosimeters was gradually lengthened and corresponded generally to downward reductions in the RPGs (Morgan 1961; Taylor 1971). During the early stages of the Y-12 program to monitor individual workers, a weekly dose control of 0.3 rem was in effect. This was changed to an annual limit of 5 rem in the latter part of the 1950s. A summary of the major historical events in the Y-12 dosimetry program for external radiation is provided in Table 6-3.

6.3.2.1 Beta-Photon Dosimeters

In 1948, Y-12 started an external dosimetry badge service with the assistance of ORNL (Murray 1948a,b). ORNL had earlier implemented the beta/photon film dosimeter design that was developed originally at the Metallurgical Laboratory of the University of Chicago (Pardue, Goldstein, and Wollan 1944). Several minor modifications had been made to this original design as discussed by Patterson, West, and McLendon (1957) and by Thornton, Davis, and Gupton (1961). The film badge was a so-called two-element badge because a portion of the film was covered with a comparatively "open window" and a portion of the film was covered by a cadmium shield or filter. In 1961, the two-element film badge was replaced by a multielement film badge with an open window over a portion of the film and three filters of plastic, aluminum, and cadmium over other portions of the film (Thornton, Davis, and Gupton 1961; McLendon 1963; McRee, West, and McLendon 1965). Cadmium filters have been consistently incorporated in all film badge designs at Y-12 since 1948 and have an approximate thickness of 1 mm or mass density of 1000 mg cm^{-2}. The external doses to Y-12 workers from photons were always determined from film readings behind the cadmium filter (Sherrill and Tucker 1973). In addition, the Y-12 film badges have always included a comparatively open window to measure significant beta radiation and to distinguish film exposures due to beta and photon radiation. The film areas behind the plastic and aluminum filters of the multielement film dosimeters were read at Y-12, but they were not used in the normal evaluation of worker doses (Sherrill and Tucker 1973).

Table 6-3. Y-12 historical dosimetry events.

Date	Event	Reference
1948	An external dosimetry program was started to monitor personnel in radiological areas using PICs and two-element film badges. Film pads were also used as an extremity monitor for beta particle exposure to the hands of uranium "metal" workers.	Murray 1948a,b; Struxness 1949a
1949	NTA film used to monitor personnel exposed to neutron sources.	Struxness 1949b
1951	Converted from film pads to film rings as an extremity monitor for beta particle exposure to the hands of uranium "metal" workers.	Struxness 1951
1955	Strips of indium foil with a mass of approximately 1 g each were included in the security badges of all employees at Y-12. The foils provided a quick means of identifying employees who might have been exposed to elevated radiation levels during a nuclear criticality accident.	McLendon 1959
June 16, 1958	A nuclear criticality accident occurred during recovery of enriched uranium in Building 9212 of the Y-12 Plant. The indium foils in the security badges of 31 employees indicated that they were exposed to elevated levels of thermal neutron radiation during the accident.	Callihan and Thomas 1959; McLendon 1959; Hurst, Ritchie and Emerson 1959
July 1, 1961	All UCC-ND employees were issued a badge meter that served as both a security pass and a routine and accident dosimeter. The badge meter contained a four-element film dosimeter for routine monitoring of beta particle, photon, and neutron exposures. The badge meter also contained indium, other neutron activation foils, and special photon dosimeters for nuclear accident dosimetry.	Thornton, Davis and Gupton 1961; McLendon 1963; McRee, West and McLendon 1965
October 1, 1980	All UCC-ND employees were issued a badge meter with (1) a security badge for identification and (2) a two-element TLD badge for personnel radiation monitoring of beta particles and photons. A 1-g foil of indium was included in the security badge to provide a quick means of identifying personnel exposed to elevated levels of radiation during a nuclear accident. An additional ORNL neutron badge containing NTA film and TLNDs was issued to Y-12 personnel who were exposed to neutrons.	McLendon et al. 1980; Howell and Batte 1982; Gupton 1978; Berger and Lane 1985
1980	The accident dosimetry components from the 1961 UCC-ND badge meter were consolidated and incorporated into Y-12 security badge covers. All persons were required to use these security badge covers when entering controlled areas that had an installed criticality alarm.	McLendon 1980; Y-12 Plant 1982a
January 3, 1989	All Martin Marietta Energy Systems (MMES) employees were issued a badge with (1) a security badge for identification and (2) a four-element TLD badge for personnel radiation monitoring of beta particles and photons. A 1-g foil of indium was included in the security badge for accident dosimetry purposes, and an additional four-element TLND badge was issued to Y-12 personnel potentially exposed to neutrons.	Y-12 Plant 1988a; Oxley 2001
1991	The beta/gamma TLD that was issued to all MMES employees became the official personnel nuclear accident dosimetry (PNAD) in the event of a criticality accident. All persons were required to use a beta/gamma dosimeter attached to their security badge before entering controlled areas with an installed criticality alarm.	Y-12 Plant 1991; Kerr and Mei 1993

The film badges Y-12 used from 1948 to 1963 contained DuPont type 552 film packets (Souleyrette 2003a). These packets consisted of (1) a sensitive 502 emulsion with an effective range from approximately 30 mrem to 10 rem and (2) an insensitive 510 emulsion with an effective range from approximately 500 mrem to 20 rem (Craft, Ledbetter, and Hart 1952; Thornton, Davis and Gupton 1961; Parrish 1979). In 1963, Y-12 switched to the use of DuPont 554 film packets (McLendon 1963; Souleyrette 2003a). These packets consisted of (1) a sensitive 555 emulsion with an effective range from approximately 30 mrem to 5 rem (McLendon 1963) and (2) an insensitive 834 emulsion with an effective range from approximately 5 rem to 150 rem (Thornton, Davis, and Gupton 1961; Davis and

Gupton 1963; Parrish 1979). In 1971, DuPont stopped manufacturing the 554 film packets and Y-12 switched to Eastman type 2 film (Jones 1971; Souleyrette 2003a). The Eastman type 2 packet contained one film with two emulsions bonded to opposite sides of the base film. The sensitive emulsion had an MDL of approximately 30 mrem (Jones 1971). During the switch to the Eastman type 2 film, some film to be evaluated was removed from cold storage, inserted in 16 pairs of badges, and the badges placed in racks with the office area to investigate onsite radiation background. A sample of film was developed on the day of the background study and used as a base point for other measurements. Every 2 weeks, film from a pair of badges was unloaded, developed, and read with a densitometer. On the first day of the study, the fresh film had an optical density (OD) of 0.205, and after 215 days, the OD had reached a level of 0.405. This increase in the OD of 0.2 represented an increase of only about 15 mrem or approximately 0.5 mrem/wk. It was noted that these OD readings did not have the OD readings of an unexposed blank subtracted from them and, therefore, represent the actual OD readings of the films (Jones 1971).

The response of the film badge to photon radiation of different energies is illustrated in Figure 6-2. This figure also shows the $H_p(10)$ response. Two responses are shown: one response is for a sensitive DuPont 502 emulsion in a two-element film badge (Pardue, Goldstein, and Wollan 1944), and one response is for a sensitive DuPont 555 emulsion in the a multielement film badge (Thornton, Davis, and Gupton 1961). The response of the sensitive Eastman type 2 film in a multielement film badge should be quite similar to that of the sensitive DuPont 555 emulsion. The film badges show an under-response at the lower photon energies and an over-response at photon energies around 100 keV. This over-response is due primarily to the silver and bromine in the film emulsions. The response of the newer TLD badges provided a much better match to the $H_p(10)$ response in the soft tissues of the body due to the lower atomic numbers of the lithium and fluorine in the TLD chips (Horowitz 1984; Cameron, Sunthanalingham, and Kennedy 1968). The two-element TLD badges Y-12 used from 1980 to 1989 had LiF chips covered by an open window and an aluminum filter for beta/photon discrimination (McLendon et al. 1980); the multielement TLD badges Y-12 adopted for use in 1989 had four LiF chips covered by an open window, a plastic filter, a copper filter, and a hemispherical Teflon button (Y-12 Plant 1988b; Oxley 2001). The photon doses were determined primarily from the readings of the LiF chips that were covered by the aluminum filter in the two-element TLD badge and the hemispherical Teflon button in the multielement TLD Badge. A value for average background radiation of 0.75 mrem/wk from photons was determined for the Oak Ridge area by storing a total of 1,680 TLDs in 70 houses for up to 1 year (Sonder and Ahmed 1991). The distribution of results indicated a rather large variation in background among houses, with a few locations exhibiting a background of double the average. It was suggested that the results from the high-background houses be ignored in determining values of the MDL to be used in routine personnel dosimetry monitoring.

IARC has conducted a recent dosimeter intercomparison study of ten commonly used historical dosimetry systems from around the world (Thierry-Chef et al. 2002). The results of this IARC study, for the U.S. dosimeters only, are presented in Table 6-4.

Three of the designs were from the United States. These included a two-element film dosimeter that was previously used at Hanford (identified as US-2), a multielement film dosimeter that was previously used at Hanford (identified as US-8), and the Panasonic TLD that is currently used at the Savannah River Site (identified as US-22). The IARC study considered that exposure to workers could be characterized as a combination of anterior-posterior (A-P), rotational, and isotropic irradiation geometries. Dosimeter responses for these various irradiation geometries were investigated using two different phantoms to represent the torso of the body. The first phantom was a water-filled slab phantom with polymethyl methacrylate walls, an overall width of 30 cm, an overall height of 30 cm, and an overall depth of 15 cm. This phantom is widely used for dosimeter calibration and

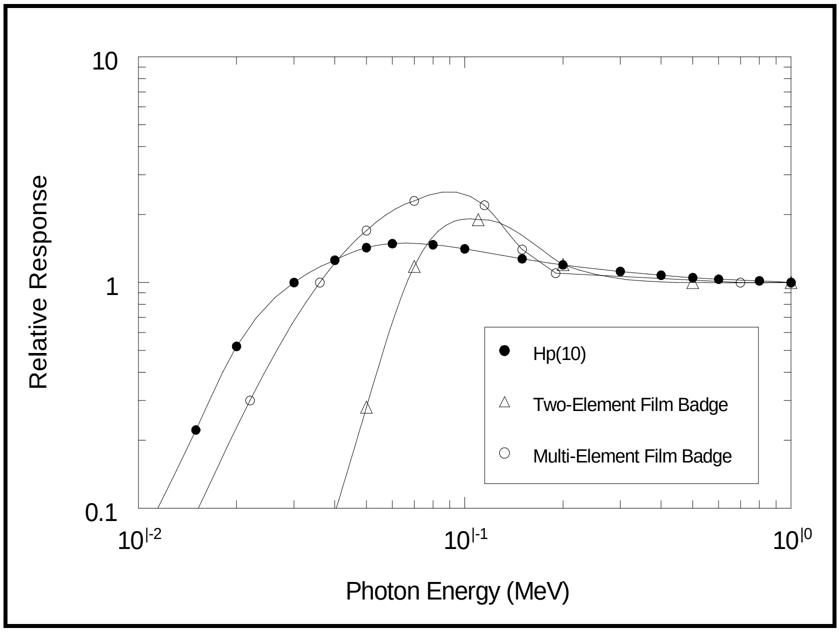

Figure 6-2. Comparison of $H_p(10)$ from a broad beam of normally incident photons (ICRP 1996) with the energy responses for a sensitive DuPont 502 emulsion in a MED two-element film badge (Pardue, Goldstein, and Wollan 1944) versus a sensitive DuPont 555 emulsion in an ORNL multielement film badge (Thornton, Davis, and Gupton 1961).

Table 6-4. IARC study results for U.S. beta/photon dosimeters (Thierry-Chef et al. 2002).

Geometry	Phantom	118 keV		208 keV		662 keV	
		Mean[a]	**SD/Mean**	**Mean[a]**	**SD/Mean**	**Mean[a]**	**SD/Mean**
US-2 (two-element film dosimeter)							
A-P	Slab	3.0	2.1	1.3	1.0	1.0	0.8
A-P	Anthropomorphic	3.0	4.2	1.2	1.9	1.0	1.8
Rotational	Anthropomorphic	2.2	2.0	1.4	3.0	1.2	3.2
Isotropic	Anthropomorphic	1.5	4.4	1.1	1.6	1.0	2.7
US-8 (multielement film dosimeter)							
A-P	Slab	1.0	1.5	1.0	0.8	0.8	1.7
A-P	Anthropomorphic	0.8	9.5	0.9	6.0	0.8	1.8
Rotational	Anthropomorphic	1.2	1.9	1.2	17	1.1	1.8
Isotropic	Anthropomorphic	1.0	3.0	1.2	9.0	1.0	2.3
US-22 (multichip TLD dosimeter)							
A-P	Slab	0.9	4.4	0.9	3.9	0.9	3.5
A-P	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 $H_p(10)$.

performance testing by the International Standards Organization (ISO). The second phantom was an anthropomorphic Alderson Rando phantom. This realistic phantom is constructed using a natural human skeleton cast inside material that has a tissue equivalent composition.

As noted previously, the multielement film badge was used at Y-12 in essentially the same manner as the two-element film badge. It should also be noted that the two-element film dosimeter can significantly overestimate $H_p(10)$ at the lower photon energies of 118 keV and 208 keV as shown in Table 6-4.

6.3.2.2 Neutron Dosimeters

Several general types of neutron dosimeters that have been used at Y-12 differ significantly in their response to neutrons of different energies as illustrated in Figure 6-3 (IAEA 1990). An NTA emulsion was included in the same holder used for the Y-12 beta-gamma dosimeter until 1980. Between 1980 and 1989, Y-12 workers who were exposed to neutrons were provided with a separate neutron dosimeter. This neutron dosimeter contained both an NTA film for measurement of the fast neutron dose and a TLND for measurement of the neutron dose from lower energy neutrons (Gupton 1978). From 1980 to 1985, the neutron doses to Y-12 workers were determined at ORNL using both the NTA and TLND dosimeters, as discussed by Gupton (1978) and Berger and Lane (1985). Since 1989, the neutron doses to Y-12 workers have been measured using a separate albedo TLND that is worn on the belt to keep it in close contact with the worker's body (Gunter 1994; Oxley 2001). In general, TLNDs have a response that increases with decreasing neutron energy, while NTA films have very little response to neutrons with energies less than its detection threshold of approximately 500 keV (Figure 6-3). Results reported at the first AEC neutron dosimetry workshop in 1969 indicated that laboratory dose measurements that were made with NTA film were about one-half to one-fourth of those that were measured with other methods including the TLND (Vallario, Hankins, and Unruh 1969). The response of both dosimeters is highly dependent on the neutron energy spectra, and both dosimeter types require matching the laboratory calibration neutron spectra to the workplace neutron spectra for reliable results.

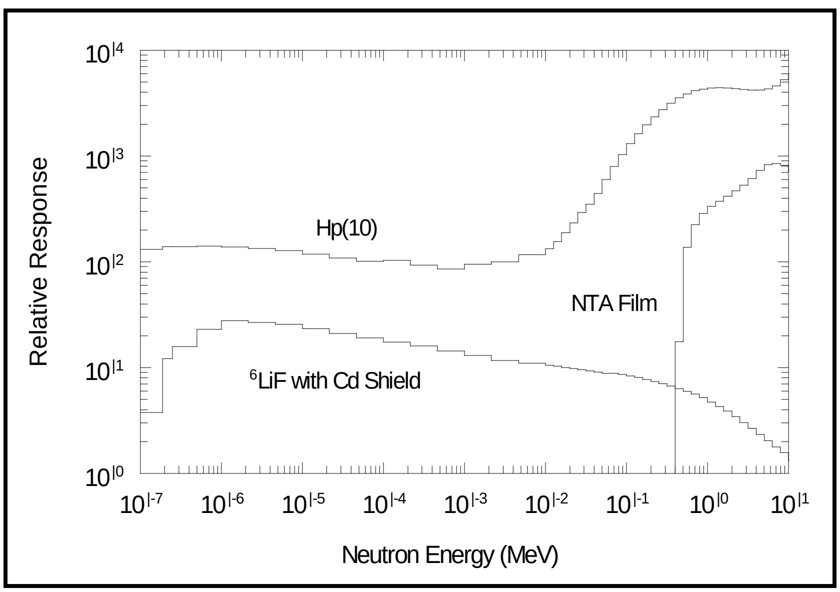

Figure 6-3. Comparison of $H_p(10)$ from normally incident neutrons (IAEA 2001) to the energy responses of an NTA film and a neutron albedo dosimeter with a neutron TLND chip made of ^{6}LiF shielded by cadmium (IAEA 1990, 2001).

6.3.3 <u>Y-12 Dosimeter Calibration Procedures</u>

Potential error in recorded dose is dependent on the calibration methodology and the extent of the similarity between the radiation fields for calibration and those in the workplace. The potential error is much greater for dosimeters with significant variations in response such as film dosimeters for low-energy photon radiation and both the NTA and TLND dosimeters for neutron radiation.

Beta/Photon Dosimeters
The Y-12 dosimeters were originally calibrated primarily using beta particles from a natural uranium slab and photons from a ^{226}Ra source (Souleyrette 2003a). The dosimeters were exposed face down on the uranium slab and free in air (i.e., no phantom) facing the ^{226}Ra source for preselected times to reproduce the beta particle and photon doses that were normally encountered in the workplace. This practice was similar to that at other AEC sites.

Neutron Dosimeters
A good account of the historical aspects of the calibration of the Y-12 neutron dosimeters is not available. It is known, however, that the NTA films were originally calibrated using a Po-Be neutron source (Struxness 1949a, 1952) and later an Am-Be neutron source (McLendon 1963; McRee, West, and McLendon 1965). The dosimeters containing the NTA films were exposed free in air (i.e., no phantom) to neutrons from the Po-Be and Am-Be sources for preselected times to reproduce the neutron doses that were normally encountered in the workplace. Some information on the calibration of Y-12 TLNDs can be found in Berger and Lane (1985) and Oxley (2001).

6.3.4 Y-12 Workplace Radiation Fields

The main workplace radiation fields at Y-12 are due to processes that involve either EU (^{235}U) or depleted uranium (DU) (^{238}U). Some other workplace radiation fields involve industrial radiation generating equipment (X-rays and electron accelerators) and isotopic gamma-ray and neutron sources for testing purposes (^{60}Co and ^{252}Cf). The current Y-12 external dosimetry TBD provides a discussion of the radiation fields due to different processes and primary nuclides in the Y-12 workplace (Oxley 2007, pp. 3–5).

6.3.4.1 Workplace Beta/Photon Dosimeter Response

Beta/photon radiation fields characteristic of the Y-12 facilities can be generally defined based on historical information as presented in Table 6-5. Because Y-12 is a nuclear weapons fabrication and disassembly facility, the most common materials are EU (^{235}U) and DU (^{238}U). Both ^{235}U and ^{238}U are primarily alpha particle emitters. However, ^{235}U does emit a 185-keV photon in 54% of its decays. Most of the external dose from ^{238}U comes from its short-lived ^{234}Th, ^{234m}Pa, and ^{234}Pa decay products. From an external dose standpoint, the most significant radiations from these decay products of ^{238}U are: (1) the 2.29-MeV beta particle from ^{234m}Pa, and (2) the photons emitted by ^{234}Pa with energies as large as 1.962 MeV. The various Y-12 dosimeters have filtration of about 1,000 mg cm^{-2} (i.e., nearly equivalent to 1-cm depth in tissue) for those regions of the dosimeter that are used to measure the WB dose. The response to beta radiation in Y-12 workplaces is limited because beta radiation usually cannot penetrate this much filtration.

The largest workplace exposures at Y-12 have historically occurred in the DU process areas (Struxness 1954; Henderson 1991). During casting operations, the decay products of ^{238}U float to the top surface of the molten metal and remain as surface residues. These surface residues result in an increased exposure potential because of the high beta and photon energies that are associated with the ^{234}Pa nuclide. The ^{234}Pa nuclide emits a number of high-energy photons and has a specific activity that is approximately 2 x 10^{15} times larger than the specific activity of its ^{238}U parent (Henderson 1991). For ^{234}Pa, the percentages of photons with energies of 30 to 250 keV and 250 keV or more are about 7% and 93%, respectively. For ^{238}U in equilibrium with its short-lived ^{234}Th, ^{234m}Pa, and ^{234}Pa, the percentages of photons with energies of 30 to 250 keV and 250 keV or more

Table 6-5. Selection of beta and photon radiation energies and percentages for Y-12 site processes.

Y-12 site processes	Building	Operations		Radiation type	Energy selection	Percent
		Begin	End			
Enriched uranium product recovery and salvage operations	9203	1947	1951	Beta	>15 keV	100
	9206[a]	1947	1959	Photon	30–250 keV	100
	9211	1947	1959			
	9201-1	1952	1963			
Uranium chemical operations and weapon production operations	9202	1947	1995	Beta	>15 keV	100
	9206[a]	1947	1995	Photon	30–250 keV	100
	9212[b]	1949	Ongoing			
Special nuclear material receiving and storage	9720-5	1949	Ongoing	Photon	30–250 keV	100
Uranium forming and machining for weapon component operations	9201-5	1949	Ongoing	Beta	>15 keV	100
	9204-4	1949	Ongoing	Photon	30–250 keV	100
	9215	1950	Ongoing			
Depleted uranium process operations	9201-5	1949	Ongoing	Beta	>15 keV	100
	9204-4	1949	Ongoing	Photon	30–250 keV	50
	9766	1949	?		>250 keV	50
	9998	1949	Ongoing			
Final weapon component assembly operations	9204-2	1952	Ongoing	Beta	>15 keV	100
	9204-2E	1952	Ongoing	Photon	30–250 keV	100
ORNL 86-in cyclotron	9201-2	1950	1983	Photon	30–250 keV >250 keV	50 50
Chemical assay and mass spectrometry laboratories	9203	1947	Ongoing	Photon	Specific to radiation source. Photon default values:	
Radiographic laboratory	9201-1	1947	Ongoing	Photon		
Calibration laboratory	9983	1949	Ongoing	Photon		
Weapon component assay laboratory	9995	1952	Ongoing	Photon	30–250 keV	50
Nondestructive assay laboratory	9720-5	1980	Ongoing	Photon	>250 keV	50
West End waste treatment facility	9616-7	1984	Ongoing	Beta Photon	>15 keV 30–250 keV >250 keV	100 50 50

a. Building 9206 Complex includes Buildings 9768, 9720-17, 9409-17, 9510-2, 9767-2, and the east and west tank farm pits.

b. Building 9212 Complex includes Buildings 9809, 9812, 9818, 9815, and 9980.

are about 82% and 18%, respectively. Therefore, an artificially high percentage of photons with energies greater than 250 keV was assumed in Table 6-5 for the normal and DU process areas. This produces doses that are favorable to the claimant because of the increased exposure potential to high-energy photons from the short-lived ^{234}Pa decay product of ^{238}U.

One serious problem with the workplace response of the Y-12 beta/gamma dosimeters involves workers who performed waist-level uranium handling jobs in the DU process areas (Henderson 1991). A personnel dosimeter worn at the collar might underestimate the $H_p(10)$ dose at the waist by rather significant factors. Y-12 now instructs these workers to wear the dosimeters at the waist, but many workers might have worn them on the collar in the past. Therefore, for all workers who performed waist-level uranium handling jobs, the recorded dose before 1991 should be multiplied by 1.34 (Henderson 1991). To determine when to make such adjustments, the dose reconstructor must depend on information about routine duties and work locations in the computer-assisted telephone interview (CATI) file for a claimant.

6.3.4.2 Workplace Neutron Dosimeter Response

Three main facilities at Y-12 with a potential for neutron exposure are: (1) the Calibration Laboratory in Building 9983, (2) the EU Storage Area in Building 9212, and (3) the Nondestructive Assay Laboratory in Building 9720-5. The following sections discuss the neutron exposure spectra and neutron-to-photon dose ratios in these three areas using data from recent measurements from the Pacific Northwest Laboratory (PNL) (Soldat et al. 1990; McMahan 1991; Oxley Y-12 2001).

6.3.4.2.1 Calibration Laboratory in Building 9983

The Calibration Laboratory has a highly shielded room used for storage of both photon and neutron sources. The walls of the room are all 3-in. steel, with a high-density concrete floor into which several source storage pits are sunk. The types of neutron sources stored in this room include twelve 2 to 4 Ci americium-boron (Am-B) sources, several americium-lithium (Am-Li) sources, and several americium-beryllium (Am-Be) sources. At the time of the PNL measurements, the neutron sources were stored in the room in shielded containers. The neutron shielding of the containers was either paraffin or high-density plastic depending on the container. Several sources were stored in containers inside a steel safe, others were in their containers on the floor of the room, and still others were in the storage pits below floor level. The neutron sources have not been used for routine calibration purposes since the early 1970s, when ORNL began calibrating all Y-12 neutron detection and survey instruments. Workers at Y-12 do access the source storage area for other purposes, and the PNL measurements were made outside the door to the source storage room to determine appropriate calibration factors for a worker's TLND.

6.3.4.2.1.1 Neutron Energy Spectrum

The PNL measurements of the neutron energy spectrum at a distance of 18 in. from the door to the source storage room are shown by the solid line in Figure 6-4. It should be noted that the PNL measurement data were provided as dose equivalent rates (Soldat et al. 1990); however, a 1-hour exposure was assumed here to show the results of the PNL measurements as dose equivalent. The fluence-to-dose equivalent conversion factors and neutron quality factors used in the PNL measurements were similar to those from National Council on Radiation Protection and Measurements (NCRP) Report 38 (NCRP 1971) and International Commission on Radiological Protection (ICRP) Publication 21 (ICRP 1973). A comparison of fluence-to-dose equivalent conversion factors from NCRP Report 38, ICRP Publication 21, and several other commonly used information sources on fluence-to-dose equivalent conversion factors and neutron quality factors can be found in a report by Sims and Killough (1983). The dashed line in Figure 6-4 shows the dose equivalent from the PNL measurements divided into the four energy groups in NIOSH (2002), and the dose fractions in each of these four energy groups are provided in Table 6-6. Although PNL measured some dose from lower (<10 keV) and intermediate energy (10 to 100 keV) neutrons, the contribution to the total dose was only about 6%. The radiation effectiveness factor (REF) used in the Interactive RadioEpidemiological Program (IREP) to calculate the POC for these two neutron energy groups is less than the fast (or fission) neutron energy group from 0.1 to 2 MeV (Kocher, Apostoaei, and Hoffman 2002). As a result, combining the lower and intermediate energy groups into the fast neutron group from 0.1 to 2 MeV is a reasonable simplification of the neutron dose calculation that is favorable to the claimant.

6.3.4.2.1.2 Neutron-to-Photon Dose Ratio

The neutron-to-photon dose ratio from the recent PNL measurements was approximately 8:1. For workers in the Calibration Laboratory, no other data on neutron-to-photon dose ratios have been

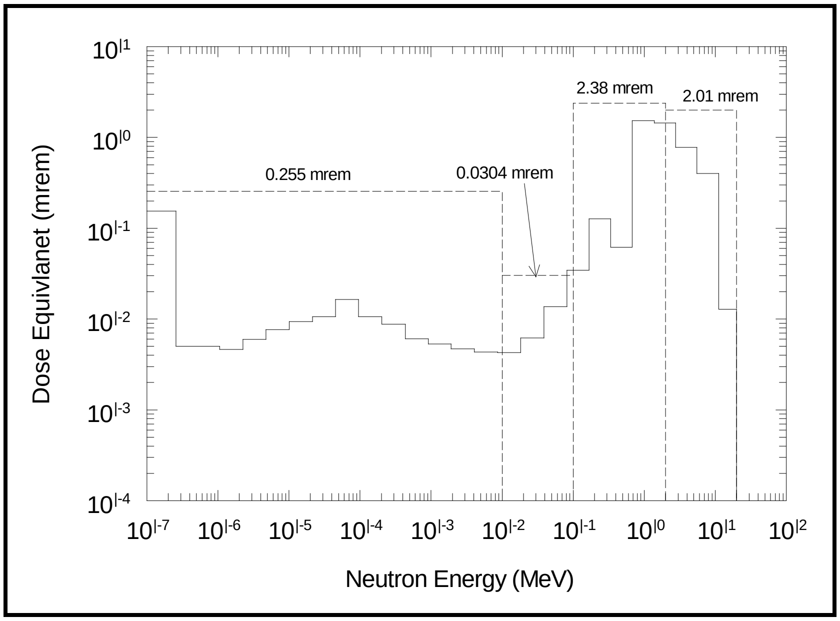

Figure 6-4. Results of PNL neutron spectrum measurements in the Calibration Laboratory of Building 9983 (solid line) and the PNL measurement results (dashed line) divided into the four neutron energy groups in the dose reconstruction for Y-12 workers (Soldat et al. 1990).

Table 6-6. Dose fractions for Y-12 calibration laboratory.

Neutron energy group	Near source storage safe
<10 keV	0.055
10–100 keV	0.007
0.1–2 MeV	0.509
2–14 MeV	0.429
Dose fractions favorable to the claimant	
0.1–2 MeV	0.57
2–14 MeV	0.43

found to use to estimate the missing neutron dose from measurements made with film dosimeters. However, the recent PNL studies indicate that more than 90% of the neutron dose is above the 500-keV threshold of the NTA films. In addition, neutron dose measurements for Calibration Laboratory workers with NTA dosimeters are expected to be reasonably accurate to within the parameters that are discussed in NIOSH (2002).

6.3.4.2.2 Enriched Uranium Storage Area in Building 9212

Building 9212 contains a secure storage area for containers of enriched uranium tetrafluoride (UF_4) and uranium trioxide (UO_3). Neutrons are produced by alpha particle reactions with the nucleus of the fluorine and oxygen atoms of the UF_4 and UO_3, respectively. Containers of these materials are placed on a rack of shelves and arranged in a matrix that is critically safe. The containers are spaced approximately 2 to 2.5 ft apart on a shelf, and can be placed one deep per shelf. There are four shelves per rack and 24 in. between shelves. The PNL measurements were made 39 in. above the floor and 24 in. from the shelf at a location near the center of a rack that was filled with 20 containers of UF_4.

6.3.4.2.2.1 Neutron Energy Spectrum

The PNL measurements of the neutron energy spectrum near the center of the UF_4 storage rack are shown by the solid line in Figure 6-5, and the dose fractions for the neutron energy groups shown by the dashed line in this figure are provided in Table 6-7. The dose fraction for the lower (<10 keV) and intermediate (10 to 100 keV) energy neutron groups were less than 2% of the total dose from these PNL measurements. As before, combining the lower and intermediate energy groups into the fast neutron group from 0.1–2 MeV is a reasonable simplification of the neutron dose calculation that is favorable to the claimant.

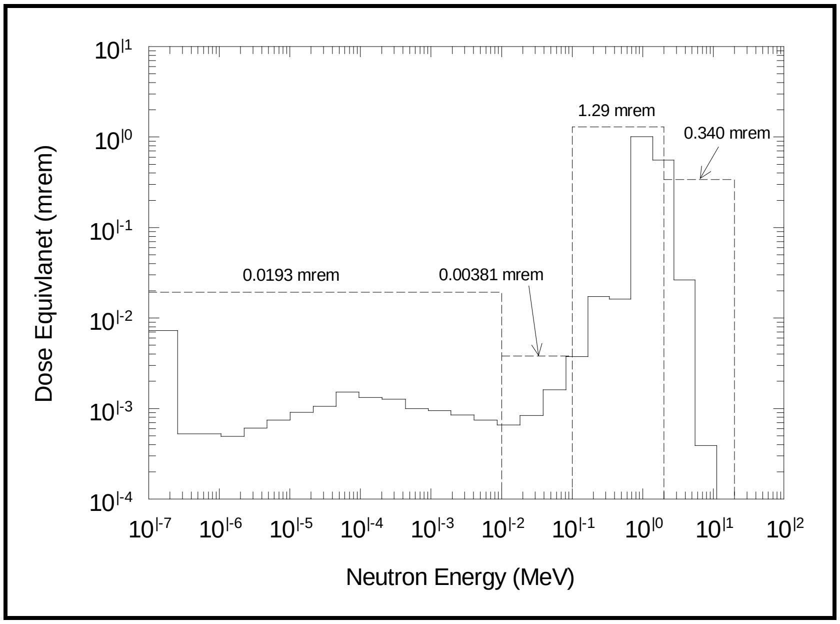

Figure 6-5. Results of PNL neutron spectrum measurements in the enriched uranium storage area of Building 9212 (solid line) and the PNL measurement results (dashed line) divided into the four neutron energy groups used in the dose reconstruction for Y-12 workers (Soldat et al. 1990).

Table 6-7. Neutron dose fractions for Y-12 EU storage area.

Neutron energy group	Near storage rack
<10 keV	0.012
10–100 keV	0.002
0.1–2 MeV	0.781
2–14 MeV	0.205
Dose fractions favorable to the claimant	
0.1–2 MeV	0.79
2–14 MeV	0.21

6.3.4.2.2.2 Neutron-to-Photon Dose Ratio

The neutron-to-photon dose ratio from the recent PNL measurements was approximately 1:1. For workers in the Enriched Uranium Storage Facility, no other data on neutron-to-photon dose ratios in Building 9212 have been found to use to estimate the missing dose from earlier measurements made with film dosimeters. However, the recent PNL studies indicate more than 95% of the neutron dose is above the 500-keV threshold of the NTA films. In addition, the neutron dose measurements for workers in the Enriched Uranium Storage Area of Building 9212 with NTA film dosimeters are expected to be reasonably accurate to within the parameters in NIOSH (2002).

6.3.4.2.3 Nondestructive Assay Laboratory in Building 9720-5

The Nondestructive Assay Laboratory in Building 9720-5 is used for recovery of highly enriched uranium (HEU) from manufacturing wastes (Hogue and Smith 1984). The laboratory contains instruments for gamma scanning and neutron interrogation of containers of solid wastes, gamma analysis of solution samples, and measurements of solution density. Because measurements of the neutron spectrum were made previously for a ^{252}Cf fission-neutron source at ORNL's Radiation Calibration Laboratory (RADCAL), it was not necessary to make additional neutron measurements to characterize the workplace radiation fields near the ^{252}Cf neutron source at the Nondestructive Assay Laboratory at Y-12. The neutron measurements at the RADCAL facility were made at a distance of 39 in. from the bare ^{252}Cf source and a height of approximately 39 in. above the floor.

6.3.4.2.3.1 Neutron Energy Spectrum

The results of the PNL neutron spectrum measurements made at 39 in. from the ^{252}Cf fission-neutron source at ORNL's RADCAL facility are shown by the solid lines in Figure 6-6. The dose fractions for the neutron energy groups (dashed lines) in this figure are provided in Table 6-8. The dose fractions for the lower (<10 keV) and intermediate (10–100 keV) energy neutron groups were less than 1% of the total dose from these PNL measurements. Therefore, combining the lower and intermediate energy groups into the fast neutron group of 0.1 to 2 MeV is a reasonable simplification of the neutron dose calculation that is favorable to the claimant.

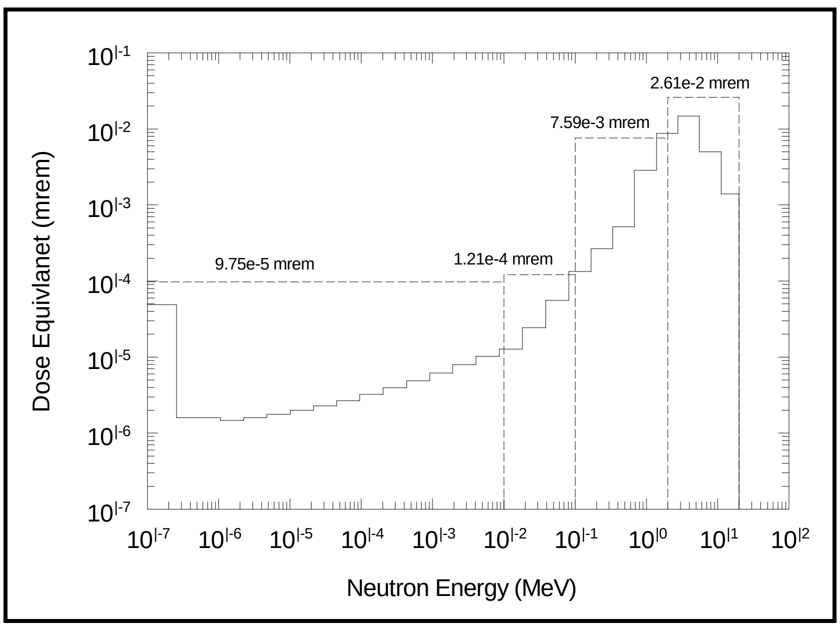

Figure 6-6. Results of PNL neutron spectrum measurements made at 39 in. from a bare ^{252}Cf fission neutron source (solid line) and the PNL measurement results (dashed lines) divided into the four neutron energy groups used in the dose reconstruction for Y-12 workers at the Nondestructive Analysis Laboratory in Building 9720-5 (Soldat et al. 1990).

Table 6-8. Dose fractions for Y-12 Nondestructive Analysis Laboratory.

Neutron energy group	Near unshielded Cf-252 source
<10 keV	0.003
10–100 keV	0.004
0.1–2 MeV	0.224
2–14 MeV	0.769
Dose fractions favorable to the claimant	
0.1–2 MeV	0.23
2–14 MeV	0.77

6.3.4.2.3.2 Neutron-to-Photon Dose Ratio

The neutron-to-photon dose ratio from the recent PNL measurements was approximately 25:1. For workers in the Nondestructive Assay Laboratory in Building 9212, no other data on neutron-to-dose ratios have been found to use to estimate missing dose from earlier measurements made with film dosimeters. However, the recent PNL studies indicate that more than 97% of the neutron dose is above the 500-keV threshold of the NTA films. In addition, the neutron dose measurements for workers in the Nondestructive Assay Laboratory with NTA film dosimeters are expected to be reasonably accurate to within the parameters in NIOSH (2002).

Typical Workplace Neutron Dosimeter $H_p(10)$ Performance

Typical neutron personnel dosimeter parameters important to $H_p(10)$ performance in the workplace are summarized in Table 6-9. The most important parameter related to $H_p(10)$ performance of the neutron dosimeters is the difference between calibration and workplace neutron energy spectra. Measurements made by PNL in the late 1980s and early 1990s could be used to correct the response of the Y-12 TLND dosimeters to workplace neutron energy spectra in several Y-12 areas. These measurements were discussed in the previous section of this report and the results of the corrected TLND workplace measurements over a 12-year period starting in 1990 are shown in Table 6-10.

Table 6-9. Typical workplace neutron dosimeter $H_p(10)$ performance.[a]

Parameter	Description	Potential workplace bias[b]
Workplace neutron energy spectra	NTA dosimeter response decreases and TLND response increases with decreasing neutron energy.	Depends on workplace neutron spectra. NTA recorded dose of record likely too low because of high 500-keV threshold for detection of neutrons.
Exposure geometry	NTA dosimeter response increases with increasing exposure angle and TLND response decreases with increasing exposure angle.	NTA recorded dose likely too high since dosimeter response is higher at angles other than A-P. TLD recorded dose is lower at angles other than A-P. Effect is highly dependent on neutron energy.
Missed dose	Doses less than the MDL are recorded as zero dose.	Recorded dose of record is likely too low. The impact of missed dose is greatest in earlier years because of the higher MDLs of the neutron dosimeters.
Environmental effects	Workplace environment (heat, humidity, etc.) fades the dosimeter signal.	Recorded dose of record is likely too low.

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

These data illustrate the low potential for routine exposure to neutrons at Y-12 both now and in the past.

Table 6-10. Number of neutron-monitored workers, cumulative annual neutron dose, and average annual neutron dose to Y-12 workers for 12-year period after introduction of the four-element TLND dosimeter in 1989 (Souleyrette 2003b).

Year	Number of neutron-monitored workers	Cumulative annual neutron dose (mrem)	Average annual neutron dose (mrem)
1990	82	1,085	13.2
1991	64	463	7.2
1992	86	200	2.3
1993	215	343	1.6
1994	301	1,289	4.3
1995	165	116	0.7
1996	203	470	2.3
1997	38	10	0.3
1998	47	57	1.2
1999	141	121	0.9
2000	49	35	0.7
2001	73	55	0.8

The recent PNL measurements also indicate that the past NTA film dosimeters worked reasonably well in the Y-12 workplace because the Ra-Be and Po-Be neutron spectra used to calibrate them were reasonably well matched to the workplace neutron spectra. These measurements suggest that the NTA film dosimeters missed less than 10% of the neutron dose equivalent at the Calibration Laboratory of Building 9983 and less than 5% of the neutron dose equivalent at the Enriched Uranium Storage Area of Building 9212 and the Nondestructive Assay Laboratory of Building 9720-5. There are many recorded zeros in the neutron dose data for Y-12 workers for two reasons: (1) a worker's NTA film was not developed and read, or (2) a worker's neutron dose equivalent was less than the MDL for the NTA film.

6.4 ADJUSTMENTS OF RECORDED DOSE

Adjustments to the Y-12 recorded doses are necessary to arrive at a dose that is favorable to the claimant because of the uncertainty that is primarily due to the complex workplace radiation fields and exposure geometries.

6.4.1 Photon Dose Adjustments

The average and maximum deep photon dose to Y-12 workers for the 10-year period from 1978 to 1987 is shown in Figure 6-7 (Y-12 Plant 1978, 1980, 1981, 1982b, 1983, 1984, 1985, 1986, 1987, 1988b). The UCC-ND policy at that time was to limit the maximum deep photon dose to workers to 500 mrem or less per quarter and 2,000 mrem or less per year. The average deep dose from photons to all Y-12 workers was approximately 20 mrem from 1978 to 1987. This period covers the change from film dosimeters to TLDs in 1980. No abrupt change occurred in the deep penetrating data for photon dose in 1980. Therefore, the recorded doses for the photon deep dose from both film and TLDs appear to be in very close agreement, and no adjustments are deemed necessary to the recorded deep photon doses for most Y-12 workers.

There is one group of Y-12 workers for which an adjustment in the recorded photon dose is recommended (see Section 6.3.4.1). These workers performed waist-level handing jobs in DU process areas (see Table 6-5). Examples of waist-level handling jobs are the unloading and sorting of DU scrap materials, shearing of larger pieces of scrap materials, cleaning of the scrap materials, crucible loading during the melting and casting operations, and materials sampling (Henderson 1991).

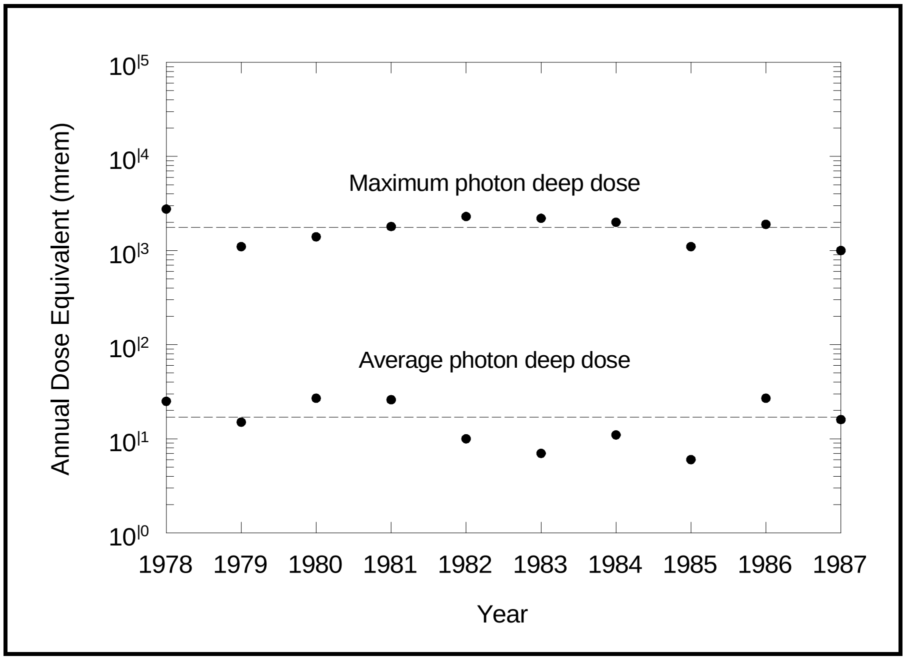

Figure 6-7. Maximum and average photon deep dose to Y-12 workers, 1978 to 1987.

Y-12 now instructs workers who perform these operations to wear their beta/photon dosimeters at the waist, but many workers might have worn their dosimeters at the collar in the past. The photon dose correction that is summarized in Table 6-11 is necessary to calculate an adjusted photon dose that is favorable to the claimant before 1991. From 1991 to the present, no correction is needed because the recorded photon dose is $H_p(10)$ equivalent.

Table 6-11. Adjustments to reported Y-12 deep photon dose.

Parameter	Description
Period	Before 1/1/1991
Dosimeters	All beta/photon dosimeters
Facilities	Depleted uranium process operations
Workers	Waist-level metal handling operators
Adjustment to recorded dose	Multiply reported deep photon dose by a factor of 1.34 to estimate $H_p(10)$

6.4.2 Neutron Dose Adjustments

The Y-12 facility incorporated the energy variation of the dose equivalent from neutrons into their calibration methodology. Therefore, the recorded dose equivalent is a combination of all neutron energies. To calculate the POC, the recorded neutron dose must be separated into neutron energy groups as discussed in Section 6.3.4.2 and then converted into ICRP Publication 60 (1991) methodology.

6.4.2.1 Neutron Weighting Factors

An adjustment to the neutron dose is necessary to account for the change in neutron quality factors between historical and current scientific guidance as is discussed by NIOSH (2002). At Y-12, the TLNDs were calibrated using PNL measurements based on fluence-to-dose conversion factors and quality factors similar to those from ICRP Publication 21 (1973) and NCRP Report 38 (NCRP 1971). These quality factors are point-wise data because they were calculated for a broad-parallel beam of monoenergetic neutrons incident on a 30-cm-diameter cylinder of tissue that represents the torso. The NCRP Report 38 (1971) quality factors are compared in Figure 6-8 with those used in the PNL measurements at Y-12. To convert from NCRP Report 38 quality factors to ICRP Publication 60 radiation weighting factors, a curve was fit that described the quality factors as a function of neutron energy. A group average quality factor was then calculated as shown in Figure 6-8 for each of the neutron energy groups to define the radiation weighting factors in ICRP Publication 60 (1991). A summary of the group-averaged NCRP Report 38 (1971) quality factors for dose reconstruction is provided in Table 6-12. This table also compares the group averaged NCRP Report 38 quality factors with historical dosimetry guidelines from the First Tripartite Conference at Chalk River in 1949 (Fix, Gilbert, and Baumgartner 1994).

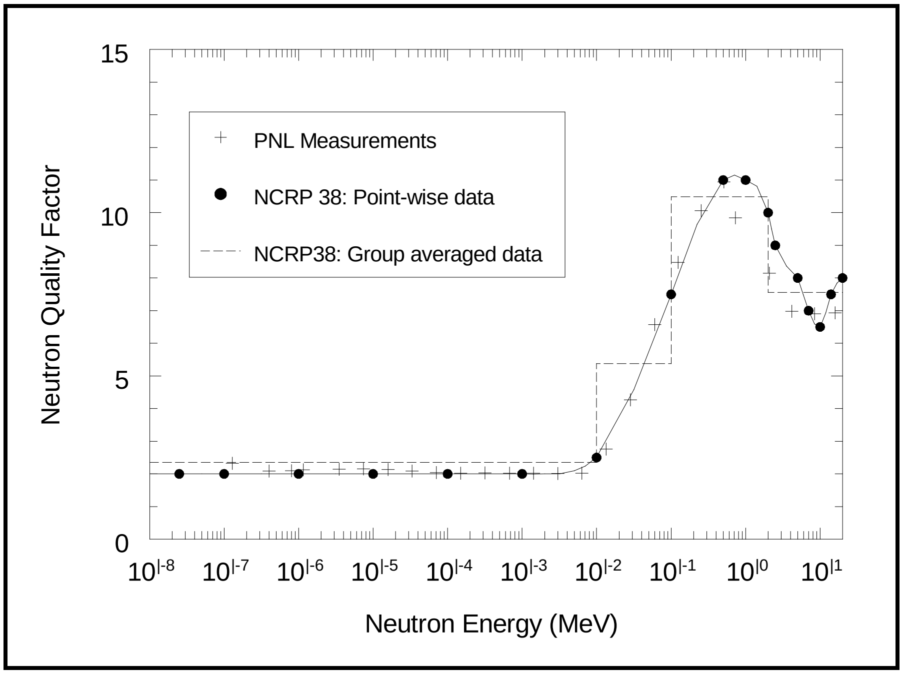

Figure 6-8. Comparison of the neutron quality factors in the PNL neutron spectrum measurements and the neutron quality factors from NCRP Report 38 (NCRP 1971) shown both as point-wise data and grouped averaged data over the four neutron energy groups for dose reconstruction for Y-12 workers.

Table 6-12. Neutron quality factor, Q, or weighting factor, w_r.

Neutron energy (MeV)	**Historical dosimetry guideline**[a]	**NCRP Report 38 (1971) group averaged quality factor**[b] (Q)	**ICRP Publication 60 (1991) neutron weighting factor** (w_r)
Thermal	3	2.35	5
0.5 eV–10 keV	10		
10 keV–100 keV		5.38	10
100 keV–2 MeV		10.49	20
2 MeV–14 MeV		7.56	10
14 MeV–60 MeV		Not applicable	5

a. First Tripartite Conference at Chalk River in 1949 (Fix, Gilbert, and Baumgartner 1994).
b. See Figure 4.6.2-1 in NCRP Report 38 (1971).

6.4.2.2 Neutron Dose Correction Factors

The average quality factor for the four energy groups that encompass the Y-12 neutron exposures are provided in Table 6-12. The neutron dose equivalent correction factor for each of these four energy groups $C_f(E_n)$ can be calculated by the use of the following equation:

$$C_f(E_n) = \frac{D_f(E_n)}{Q_{avg}(E_n)} \times w_R(E_n)$$

where

$D_f(E_n)$ is the dose fraction from Section 6.3.4.2 for the specific neutron energy group of interest
$Q_{avg}(E_n)$ is the group average NCRP 38 (1971) neutron quality factor for that specific energy group
$W_r(E_n)$ is the ICRP 60 (1991) neutron weighting factor for that specific energy group

The neutron dose distributions by energy for the various neutron exposure areas at Y-12 are summarized in Table 6-13. By multiplying the recorded neutron dose by the area-specific correction factors, the neutron dose equivalent is calculated as follows. Consider security personnel who inventory fissile material in the Enriched Uranium Storage Area of Building 9212. Assume that the worker receives a recorded annual neutron dose of 100 mrem. The corrected neutron dose is 151 mrem for neutrons with energies between 0.1 and 2 MeV, 28 mrem for neutron with energies between 2 and 14 MeV, and 179 mrem for neutrons of all energies. These corrections should be applied to both measured neutron dose and missed neutron dose. The dose fractions by energy and the associated ICRP Publication 60 (1991) correction factors for various neutron exposure areas at Y-12 are summarized in Table 6-13.

Table 6-13. Summary of neutron dose fractions and associated ICRP Publication 60 (1991) correction factors for Y-12 facilities.

Y-12 facilities	Building	Operations Begin	Operations End	Neutron energy	Neutron dose fraction	ICRP 60 correction factors
Calibration Laboratory	9983	1949	Ongoing	0.1–2 MeV 2–14 MeV	0.57 0.43	1.09 0.57
Enriched Uranium Storage Area	9212	1949	Ongoing	0.1–2 MeV 2–14 MeV	0.79 0.21	1.51 0.28
Nondestructive Analysis Laboratory	9720-5	1980	Ongoing	0.1–2 MeV 2–14 MeV	0.23 0.77	0.44 1.02

6.5 MISSED DOSE

There is undoubtedly missed dose for Y-12 workers. Analysis of the missed dose has been separated according to photon and neutron missed dose. The missed photon dose is discussed first and then the neutron missed dose.

6.5.1 Photon Missed Dose

Missed photon dose to Y-12 workers could have occurred for the following reasons: (1) the worker was not monitored before 1961, (2) there is no recorded dose for short periods of time after 1961, and (3) the worker's dose during a monitoring period was recorded as zero because the dosimeter response was less than the MDL. Before 1961, the policy at Y-12 was to issue a film badge only to those workers who might exceed 10% of the RPGs in effect at that time. This practice resulted in large numbers of workers not being monitored for external radiation exposure before 1961 (see Figure 6-1).

If a worker’s routine duties and work location remained essentially the same during the 1950s and early 1960s, it might be feasible to use the recorded annual doses in the early 1960s to estimate the missed dose before 1961. Methods are being investigated at present based on department numbers, job descriptions, and work locations that might be used to estimate annual doses for other workers who were not monitored for external radiation exposure before 1961 and did not remain in the same jobs during the 1950s and early 1960s. Methods to be considered when there is no recorded dose for short periods for normally monitored workers have been discussed by Watson et al. (1994). Estimates of the missed dose can be made using dose results for coworkers or using the nearby recorded dose for the specific worker of interest before and after a period of missed dose. Regardless of how the missing dose is estimated for unmonitored periods, these situations do require careful consideration.

The missed dose for dosimeter results less than the MDL is particularly important for earlier years when MDLs were higher and dosimeter exchange was more frequent. NIOSH (2002) describes several different options to calculate the missed dose in these situations. One option to estimate a maximum potential dose that is favorable to the claimant is to multiply the MDL by the number of zero dose results. This will provide an estimate of the maximum missed dose to the worker. The following sections consider missed photon dose for dosimeter results less than the MDL according to facility or location, dosimeter type, year, and energy range.

Facility or Location

Information has not been found that is adequate to describe the potential missing photon dose by facility or location within the Y-12 Plant. This is particularly true during the early years when the missed photon dose was most significant due to the frequent exchange of film dosimeters and the higher MDLs.

Dosimeter Type

The missed photon dose by dosimeter type is discussed in Section 6.3.1 and summarized in Table 6-2. The MDLs for the respective Y-12 beta/photon dosimeters are based on results of laboratory irradiations, and the actual MDLs for the workplace might be somewhat greater than these values because of additional uncertainty in actual field use. Nevertheless, the values in Table 6-2 are expected to provide reasonable estimates of the missed dose for Y-12 workers.

Year

Analysis of the missed photon dose by year and dosimeter exchange frequency is discussed in Section 6.3.1 and summarized in Table 6-2. The missed photon dose for workers who were unmonitored for external radiation exposures before 1961 is also discussed in Section 6.5.1.

Energy Range

An estimate of the missed dose by energy range might be possible based on the type of facility and predominant radiation sources or radionuclides at the facility. The recorded dose from the dosimeter does not typically provide information to estimate discrete energy ranges. It is possible to examine the energy response characteristics of the multielement film and TLD dosimeters, but this analysis does not recognize the substantial uncertainties present in the workplace associated with differing exposure geometries and mixed radiation fields.

6.5.2 Neutron Missed Dose

There can be significant missed neutron dose at Y-12 because of the very low potential for neutron exposure as illustrated by the data in Table 6-10. The neutron missed dose is divided into three periods in the following discussion. The first period is before 1980 when only NTA film dosimeters were used. The second period is from 1980 to 1989 when the switch from NTA film dosimeters to TLNDs was being completed. The third period is after 1989 when only TLNDs were used. The estimated MDLs for these neutron dosimeters are summarized in Table 6-2. It is possible to estimate the missed neutron dose using the MDLs because the neutron dosimeters were calibrated with neutron sources that had energies similar to those in the workplace and more than 90% of the neutrons to which workers were normally exposed had energies greater than the 500-keV threshold of the NTA film dosimeters. There was, of course, no threshold energy for the TLNDs as illustrated in Figure 6-3.

Before 1980

The use of NTA films for neutron dosimetry before 1980 is well documented in various Y-12 reports. As noted above, it is possible to estimate the missed dose using the MDLs. It was also noted previously that there are many recorded zeros in the neutron dose data for Y-12 workers for two reasons: (1) a worker's NTA film was not developed and read, or (2) a worker's NTA film indicated a neutron dose equivalent that was less than the film's MDL (approximately 50 mrem). If the MDL for NTA film is used in estimating the missed neutron dose, it should be multiplied by 1.10 for workers in the Calibration Laboratory and by 1.05 for workers in the Enriched Uranium Storage Area of Building 9212 and the Nondestructive Analysis Laboratory. It is also possible to estimate the missed neutron dose in some facilities by use of neutron-to-photon dose ratios (NIOSH 2002). However, the only Y-12 facility where this dose ratio is expected to provide a reasonably reliable estimate of the missed neutron dose is the Enriched Uranium Storage Area of Building 9212. For this area, the dose ratio was determined by recent PNL measurements to be approximately 1:1. The dose ratios for other neutron exposure areas at Y-12 that were determined from recent PNL measurements were quite large, and the use of these data could lead to gross overestimates of missed neutron dose if a worker was also exposed to pure photon sources at the Calibration Laboratory or the Nondestructive Analysis Laboratory.

From 1980 to 1989

Y-12 became increasingly dependent over the years on ORNL to process the NTA films and to determine neutron doses because of the small numbers of workers who were exposed to neutrons. All workers at both ORNL and Y-12 were provided with a two-element TLD dosimeter for beta-particle and photon dosimetry (McLendon et al. 1980), and workers who were exposed to neutrons were provided with a separate neutron dosimeter (Gupton 1978). This dosimeter was a modification of the film badge dosimeter that was previously used at both ORNL and Y-12. The MDL of this neutron dosimetry is assumed to be about the same as that of the NTA film alone because most of the neutron dose at Y-12 comes from neutrons above the 500-keV threshold of the NTA film.

After 1989

Since 1989, the neutron dose has been measured using a newly developed albedo-type TLND that is worn on the belt to keep it in close contact with the body. The characteristics of this dosimeter are well documented (Oxley 2001), and the MDL to be used in estimating missed dose is 10 mrem (see Table 6-2).

6.6 UNCERTAINTY IN PHOTON AND NEUTRON DOSE

For film badges, the MDLs that are quoted in the literature range from about 30 to 50 mrem for beta/photon irradiation (Morgan 1961; Parrish 1979; West 1993; Souleyrette 2003a) and from 50 to 100 mrem for neutrons (Morgan 1961; Parrish 1979; Wilson et al. 1990). These are not the expected uncertainties at larger photon and neutron dose readings. For example, it was possible to read a photon dose of 100 mrem to within ±15 mrem if the exposure involved photons with energies between several hundred keV and several MeV (Morgan 1961). If the exposure involved photons with energies less than several hundred keV, the uncertainty was at least twice that for the more energetic photons. Therefore, the standard error in the recorded film badge doses from photons of any energy is estimated here to be ±30%. The standard error for the recorded dose from beta irradiation was essentially the same as that for photon irradiation, but when an unknown mixture of beta and photon irradiation was involved the standard error for the dose from beta irradiation was somewhat larger than 30% (Morgan 1961). The situation for neutrons was not as favorable as that for photons. With NTA films, the estimated standard error was much larger and varied significantly with the energy of the neutrons. Therefore, the standard error for a neutron dose reading of approximately 100 mrem is estimated here to be ±50%. For the TLDs that were used at Y-12 after 1980 and the TLNDs after 1985, the standard errors for a recorded dose reading of 100 mrem or more are estimated here to be approximately ±15% for photons, beta particles, and neutrons. The standard errors for TLD and TLND dose measurements less than 100 mrem and for TLD and TLND dose measurements in mixed radiation fields would be expected to be somewhat larger.

6.7 ORGAN DOSE

Once the photon and neutron doses and their associated standard errors have been calculated for each year, the values are then used to calculate organ doses of interest using the NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2002). There are many complexities and uncertainties when applying organ DCFs to the adjusted doses of record. Many of the factors that affect the recorded dose have already been discussed in this section. Some factors, such as backscattering (phantom calibration) and the over-response to low-energy photons, would indicate the recorded dose was too high. Other factors, such as calibration methodology, angular response, low energy threshold, and film fading, would result in a recorded dose that was too low. As a result, differences in film badge design (filtration) and calibration can have both positive and negative effects on the overall dose comparison to $H_p(10)$. ICRU (1988) indicates that film badge dosimeters, while not tissue equivalent, can be used for personnel dosimetry. The report emphasizes, however, that it is difficult to establish the variations in film response as functions of photon energy and angle of photon incidence on the film badge at very low photon energies. Given these broad uncertainties, especially with film badge dosimetry in the 1950 to 1980s, an approach is used to estimate organ dose that is favorable to the claimant. Because use of exposure-to-organ DCFs results in a higher organ dose and higher POC, and given the REFs of the intermediate-energy photons, these DCFs should be used to convert recorded film badge doses to organ dose. In the conversion of all recorded photon and neutron doses to organ doses, the exposure geometry must also be given careful consideration. Some of the more common exposure geometries in the workplace are: (1) an A-P exposure is typical for an individual who works in a directional radiation field and faces the radiation source, (2) rotational exposure is typical of an individual who is constantly turning in a directional radiation field, and (3) an isotropic exposure is typical of an individual who is working in a highly nondirectional or omnidirectional radiation field. The proposed default options based on exposure geometries for long-term workers that are favorable to the claimant are listed in Table 6-14. Attachment A discusses the use of the parameters in Section 6 to aid the dose reconstructor in preparing dose reconstructions for long-term Y-12 workers.

Table 6-14. Default exposure geometries favorable to the claimant for calculating organ doses.

Claim status	Job category	Exposure geometry	Percentage
Likely non-compensable	All	A-P	100
Compensable worker	All	A-P Rotational	50 50
Compensable supervisor	All	A-P Isotropic	50 50

6.8 ATTRIBUTIONS AND ANNOTATIONS

All information requiring identification was addressed via references integrated into the reference section of this document.

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A.1 OCCUPATIONAL EXTERNAL DOSE

The information necessary to evaluate claims is directed to the technical parameters of the annual estimates of the primary organ dose that is calculated from the dosimeter and interpreted as personal dose equivalents $H_p(10)$ for organ doses and $H_p(0.07)$ in the case of skin, testicular, and breast cancer. These are used as a consistent basis of comparison for all years of Y-12 occupational external dose starting in 1950.

The primary IREP screen used to input dose parameters is illustrated in Table A-1. The input to these fields is obtained from the Y-12 dose of record. The claim provides the primary organ of interest and other worker information necessary to run the IREP computer program. Guidance to the dose reconstruction analysis in selecting the external dose parameters to complete the respective fields in Table A-1 is provided in the following sections.

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Table A-1. IREP dose parameter input screen.

Exposure				Distribution parameters			
#	Year	Rate	Radiation type	Type	1	2	3
1	1960	Acute	Photon, 30-250 keV	Normal	2	2	0
2	1961						
3	1962						

A.2 YEARS OF EXPOSURE

The years of exposure should be identified from the Y-12 radiation dose reports. Missed dose is calculated for each of the following reasons: (1) the worker was not monitored for external radiation exposure before 1961, (2) there is no recorded dose for short periods after 1961, and (3) the recorded dose during a monitoring period was zero because the dosimeter response was less than the MDL. The missed dose should be calculated for a claimant for each year of record as an employee unless there are valid reasons for years in which there are no records.

A.3 EXPOSURE RATE

Acute is selected for all types of external beta and photon dose, and chronic is selected for neutron dose (NIOSH 2002).

A.4 RADIATION TYPE

A.4.1 Beta and Photon Radiation

Assumptions favorable to the claimant should be made using guidance in Table A-2 for beta particles and for photons (X-rays and gamma rays) to ensure that dose is not underestimated. The values in the table are intended to provide a reasonable estimate of parameters used to calculate the organ dose without significant numerical error for long-term Y-12 workers in the respective facilities. There is no direct guidance for selecting the specific values other than considerations of the radiation sources and usual work tasks. In those cases where there is some doubt in the values, a range of realistic values and the option that is favorable to the claimant should be selected.

Table A-2. Selection of beta and photon radiation energies and percentages for Y-12 site processes.

Y-12 Site Processes	Building	Operations Begin	Operations End	Radiation type	Energy selection (keV)	Percent
Enriched uranium product recovery and salvage operations	9203	1947	1951	Beta Photon	>15 30-250	100 100
	9206[a]	1947	1959			
	9211	1947	1959			
	9201-1	1952	1963			
Uranium chemical operations and weapon production operations	9202	1947	1995	Beta Photon	>15 30-250	100 100
	9206[a]	1947	1995			
	9212[b]	1949	Ongoing			
Special nuclear material receiving and storage	9720-5	1949	Ongoing	Photon	30-250	100
Uranium forming and machining for weapon component operations	9201-5	1949	Ongoing	Beta Photon	>15 30-250	100 100
	9204-4	1949	Ongoing			
	9215	1950	Ongoing			

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Y-12 Site Processes	Building	Operations Begin	Operations End	Radiation type	Energy selection (keV)	Percent
Depleted uranium process operations	9201-5	1949	Ongoing	Beta		
	9204-4	1949	Ongoing	Photon	>15	100
	9766	1949	?		30-250	50
	9998	1949	Ongoing		>250	50
Final weapon component assembly operations	9204-2	1952	Ongoing	Beta	>15	100
	9204-2E	1952	Ongoing	Photon	30-250	100
ORNL 86-in cyclotron	9201-2	1950	1986	Photon	30-250 >250	50 50
Chemical Assay and Mass Spectrometry Laboratories	9203	1947	Ongoing	Photon	Specific to radiation source	
Radiographic Laboratory	9201-1	1947	Ongoing	Photon	Photon default	
Calibration Laboratory	9983	1949	Ongoing	Photon	values:	
Weapon Component Assay Laboratory	9995	1952	Ongoing	Photon	20-250	50 50
Nondestructive Assay Laboratory	9720-5	1980	Ongoing	Photon	>250	
West End Waste Treatment Facility	9616-7	1984	Ongoing	Beta Photon	>15 30-250 >250	100 50 50

a. Building 9206 Complex includes Buildings 9768, 9720-17, 9409-17, 9510-2, 9767-2, and the east and west tank farm pits.
b. Building 9212 Complex includes Buildings 9809, 9812, 9818, 9815, and 9980.

A.4.2 Neutron Radiation

The default neutron dose distributions by energy for each of the neutron exposure areas at Y-12 are summarized in Table A-3.

Table A-3. Selection of neutron energies and percentages for Y-12 site facilities.

Y-12 site facility	Building	Operations Begin	Operations End	Neutron energy (MeV)	Default dose fraction (%)
Calibration Laboratory	9983	1949	Ongoing	0.1-2 2-14	57 43
Enriched Uranium Storage Area	9212	1949	Ongoing	0.1-2 2-14	79 21
Nondestructive Assay Laboratory	9720-5	1980	Ongoing	0.1-2 2-14	23 77

A.5 ADJUSTMENTS TO RECORDED DOSE

A.5.1 Parameter #1

Selection of the distribution parameters in Table A-1 involves the adjustments to the dose of record for missed dose before entry into IREP. The selection of a normal distribution for the “Type” determines the definition of Parameters #1 and #2. For a normal distribution, Parameter #3 is not used and Parameter #1 is the mean of the distribution of recorded dose for each year of monitoring.

A.5.2 Adjustment to Recorded Photon Dose

Adjustments to the Y-12 reported photon dose are necessary to arrive at a dose that is favorable to the claimant considering uncertainties that are associated primarily with the complex workplace radiation fields and exposure geometries. Henderson (1991) identified such problems for workers who perform waist-level handling jobs in the DU process areas of Y-12. Examples of waist-level handling jobs are unloading and sorting of DU scrap materials, shearing of larger pieces of scrap materials, cleaning of the scrap materials, crucible loading during the melting and casting operations, and materials sampling. Y-12 now instructs workers who perform these operations to wear beta/photon dosimeters at the waist, but many of these workers might have worn their dosimeters on the collar in the past. The photon dose correction summarized in Table A-4 is necessary to calculate an adjusted photon dose that is favorable to the claimant for years before 1991. From 1991 to the present, no correction is needed because the recorded dose is $H_p(10)$ equivalent. To determine when to make such adjustments, the dose reconstructor must depend on information about routine duties and work locations in the CATI file for a claimant.

Table A-4. Adjustments to reported Y-12 deep photon dose.

Parameter	Description
Period	Prior to 1/1/1991
Dosimeters	All beta/photon dosimeters
Facilities	Depleted uranium process operations
Workers	Waist-level metal handling operators
Adjustment to recorded dose	Multiply reported deep photon dose by a factor of 1.34 to estimate $H_p(10)$

A.5.3 Adjustments to Recorded Neutron Dose

The Y-12 facility incorporated the energy variation of the dose equivalent from neutrons into their calibration methodology. Therefore, the recorded dose equivalent is a combination of all neutron energies. To calculate the neutron dose input to IREP (see Table A-1), the recorded neutron dose must be separated into neutron energy groups as shown in Table A-3 and subsequently converted into ICRP Publication 60 methodology (ICRP 1991). The dose fractions by neutron energy group and the associated Publication 60 correction factors for the various neutron exposure areas at Y-12 are summarized in Table A-5. As an example, consider security personnel who inventory fissile material in the Enriched Uranium Storage Area of Building 9212 and assume that the worker receives a neutron dose of 100 mrem. The corrected neutron dose is 151 mrem for neutrons with energies between 0.1 and 2 MeV and 28 mrem for neutrons with energies between 2 and 14 MeV. Therefore, the total corrected neutron dose is 179 mrem. These corrections should be applied to both recorded and missed doses.

A.5.4 Unmonitored Photon and Neutron Dose

Missed photon and neutron dose would occur where there is no recorded dose because the worker was not monitored or the dose is unavailable for a short period because a film was either lost or damaged while being processed. Before 1961, the policy at Y-12 was to issue film badges only to those workers who might exceed 10% of the RPGs in effect at that time. This practice resulted in

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Table A-5. Neutron dose fractions and associated ICRP Publication 60 (1991) correction factors for Y-12 site facilities.

Y-12 facilities	Building	Operations Begin	Operations End	Neutron energy	Neutron dose fraction	ICRP 60 correction factors
Calibration Laboratory	9983	1949	Ongoing	0.1-2 MeV 2-14 MeV	0.57 0.43	1.09 0.57
Enriched Uranium Storage Area	9212	1949	Ongoing	0.1-2MeV 2-14 MeV	0.79 0.21	1.51 0.28
Nondestructive Analysis Laboratory	9720-5	1980	Ongoing	0.1-2 MeV 2-14 MeV	0.23 0.77	0.44 1.02

large numbers of workers not being monitored for external radiation exposure before 1961 (see Figure 6-1).

If a worker's routine duties and work location remained essentially the same during the 1950s and early 1960s, it might be feasible to use the recorded annual doses in the early 1960s to estimate the missed dose before 1961. Methods are being investigated at present using department numbers, job descriptions, and work locations that might be used to estimate annual doses for other workers who were not monitored for external radiation exposure before 1961 and did not remain in the same jobs during the 1950s and early 1960s.

Methods to be considered when there is no recorded dose for short periods for normally monitored workers have been discussed by Watson et al. (1994). Estimates of the missed dose can be made using dose results for coworkers or using nearby recorded dose for the specific worker of interest before and after a period of missed dose due to a film that was either lost or damaged while being processed. Regardless of how the missing dose is estimated for unmonitored periods of time, these situations require careful consideration.

A.5.5 Missing Photon Dose

Missing photon dose also occurs when the recorded dose is zero because the dosimeter response was less than the MDL. This kind of missed dose is most important for earlier years when MDLs were higher and dosimeter exchange was more frequent. NIOSH (2002) guidance should be followed to calculate the missing photon dose by using a missed dose that is favorable to the claimant. This is calculated by multiplying the MDL by the number of zero dose results to estimate the maximum potential missed dose. The following sections discuss the missed photon dose corrections according to facility or location, dosimeter type, year, and energy range.

A.5.5.1 Facility or Location

Information has not been found that is adequate to describe the potential missing photon dose by facility or location within Y-12. This is particularly true during the early years when the missed photon dose was most significant due to the frequent exchange of film dosimeters and their higher MDLs.

A.5.5.2 Dosimeter Type

The missed photon dose by dosimeter type is summarized in Table A-6. The MDLs for the respective Y-12 beta/photon dosimeters are based on results of laboratory irradiations. The actual MDLs for the workplace could be somewhat greater than these values because of additional uncertainty in actual field use. Nevertheless, the values in Table A-6 are expected to provide reasonable estimates of the missed dose for Y-12 workers.

Table A-6. Dosimeter type, period of use, exchange frequency, laboratory MDL, and maximum annual missed dose.

Dosimeter	Period	Exchange frequency	Laboratory MDL (mrem)	Maximum annual missed dose (mrem)
Beta/photon dosimeters				
Pocket ionization chamber	1948–1950	Daily	<5	1,300
		Weekly	<5	260
Two-element film badge	1948–1958	Weekly	40	2,080
	1958–1961	Monthly	40	480
Four-element film badge	1961–1980	Quarterly	40	160
Two-element TLD dosimeter	1980–1989	Quarterly	20	80
Four-element TLD dosimeter	1989–present	Quarterly	10	40
Neutron dosimeters				
NTA film	1948–1980	Biweekly	<50	1,300
		Monthly	<50	600
		Quarterly	<50	200
Combination NTA film and TLND dosimeter	1980–1989	Quarterly	<50	200
TLND dosimeter	1989–present	Quarterly	10	40

A.5.5.3 Year

Analysis of the missed photon dose by year and dosimeter exchange frequency is summarized in Table A-6. The missed photon dose for workers who were unmonitored for external occupational radiation exposure before 1961 is also discussed in Section A.5.4.

A.5.5.4 Energy Range

An estimate of the missed dose by energy range might be possible based on the type of facility and predominant radiation sources or radionuclides at the facility. The recorded dose from the dosimeter response does not typically provide information to estimate discrete energy ranges. It is possible to examine the energy response characteristics of the multielement film and TLD dosimeters, but this analysis does not recognize the substantial uncertainties in the workplace that are associated with differing exposure geometries and mixed radiation fields.

A.5.6 Neutron Missed Dose

The estimated MDLs for the neutron dosimeters that were used at Y-12 are summarized in Table A-6. It is possible to calculate the missed neutron dose at Y-12 using the MDLs because the neutron dosimeters were calibrated with neutron sources that had energies similar to those in the workplace and more than 90% of the neutrons to which workers were normally exposed had energies greater than the 500-keV threshold of the NTA film dosimeters. If the MDL for NTA film is used in estimating the missed neutron dose, it should be multiplied by 1.10 for workers in the Calibration Laboratory and by 1.05 for workers in the Enriched Uranium Storage Area of Building 9212 and the Nondestructive Analysis Laboratory. It is also possible to estimate missed neutron dose in some facilities by use of neutron-to-photon dose ratios (NIOSH 2002). However, the only Y-12 facility where a neutron-to-photon dose ratio is expected to provide a reasonably reliable estimate of the missed dose is for workers in the Enriched Uranium Storage Area of Building 9212. For this area, the neutron-to-dose ratio was approximately 1:1. The neutron-to-dose ratios for other neutron exposure areas at Y-12 were quite large and the use of these data could lead to gross overestimates of missed neutron dose if an individual was also exposed to pure photon sources during the course of work at either the Calibration Laboratory or the Nondestructive Analysis Laboratory.

A.5.7 Organ Dose Equivalent

Once the adjusted photon and neutron doses have been calculated for each year, the values are used to calculate organ doses of interest using the NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2002). There are many complexities and uncertainties when applying organ dose conversion factors to the adjusted doses of record. Many of the factors that affect the recorded dose have already been discussed in the various tables throughout this section. Some factors such as backscattering (phantom calibration) and the over-response of low-energy photons would indicate the recorded dose was too high. Other factors such as calibration methodology, angular response, low-energy threshold, and film fading would result in a recorded dose that was too low. As a result, differences in film badge design (filtration) and calibration can have both positive and negative effects on the overall dose comparison to $H_p(10)$. ICRU (1988) indicates that film badge dosimeters, while not tissue equivalent, can be used for personnel dosimetry. The report emphasizes, however, that it is difficult to establish the variations in film response as functions of photon energy and angle of photon incidence on the film badge at very low photon energies. Given these broad uncertainties, especially with film badge dosimetry in the 1950 to 1980s, an approach is used to estimate organ dose that is favorable to the claimant. Because use of exposure-to-organ DCFs results in a higher organ dose and higher probability of causation, given the REFs of the intermediate-energy photons, these DCFs should be used to convert recorded film badge doses to organ dose. In the conversion of all recorded photon and neutron doses to organ doses, the exposure geometry must also be given careful consideration. The proposed default options based on exposure geometries for long-term workers that are favorable to the claimant are listed in Table A-7.

Table A-7. Default exposure geometries favorable to the claimant for calculating organ dose.

Claim status	Job category	Exposure geometry	Percentage[a]
Non-compensable	All	A-P	100
Compensable worker	All	A-P Rotational	50 50
Compensable supervisor	All	A-P Isotropic	50 50

a. Apply this percentage to the dose conversion factor in Appendix B of NIOSH (2002) to arrive at the total organ dose equivalent from the adjusted recorded dose.

A.5.8 Parameter #2

Parameter #2 is the standard deviation of the normal distribution for the organ dose. The individual dose result for each dosimeter exchange period is normally available to calculate the mean and standard deviation for each year. If it is not available, the adjusted organ dose can be used for each year, and a default standard deviation value can be used for parameter #2.

A.5.9 Organ Dose Conversion Factors

A detailed discussion of the conversion of measured dose to organ dose equivalent is provided in Appendix A of NIOSH (2002). Appendix B of NIOSH (2002) contains the appropriate DCFs for each organ, radiation type, and energy range based on the type of monitoring. In some cases, simplifying assumptions are appropriate.

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B.1 CALUTRON RELATED URANIUM ENRICHMENT ACTIVITIES FROM MARCH 1, 1943 THROUGH DECEMBER 31, 1947

The Y-12 calutrons used an electromagnetic process to separate ^{235}U from natural uranium. This enrichment process began in Building 9731 in November 1943 when the operations supervisors and the first crews were trained on the use of the alpha and beta calutrons. These calutrons were in operation until December 23, 1946 (Wilcox 2001).

The first run of a production (alpha) calutron occurred in Building 9201-1 on January 27, 1944. There were 864 alpha calutrons operated in Buildings 9201-1, 2, 3, 4, and 5. These operations were shut down by September 22, 1945. The beta calutrons began operations in March 1944. There were 288 beta calutrons operated in Buildings 9204-1, 2, 3 and 4. These operations ceased by December 23, 1946, marking the end of the calutron enrichment process (Wilcox 2001).

Each calutron was connected to a separate control panel called a cubicle, located about 30 ft from the calutron track. There were many dials, gauges, switches, and other controls on the front of the cubicle. The operator (control or cubicle operator) had to sit or stand in front of the cubicle and watch instrument readouts, make adjustments to maximize calutron performance, and record the calutron's performance. If the operator could not adjust the calutron to run properly, the operator would call a maintenance person to make additional adjustments. In addition to the cubicle operator, electricians and other maintenance employees worked in the calutron area (Tankersley 2008a, 2008b).

Both the alpha and beta cubicles and calutrons had high-voltage rectifier tubes, known as kenetrons, which could generate appreciable X-ray exposure levels. These tubes usually operated in the 50 to 100 kV range, but the voltage could go as high as 150 kV or more with a sudden drop in potential across the tube (Sterner and Riley date unknown, pp. 11–12).

Smith (1944a) made measurements of X-rays around the cubicles in Buildings 9731 and 9201-1 and the calutrons in Building 9731 using Eastman Type K Industrial X-ray film in dental packages. The response of the film to photon energy was determined, and the film was calibrated by Eastman Kodak using an ionization chamber (Dahl 1944). The operating voltage of the rectifier tubes was assumed to be 150 kV. Unless a specific measurement distance is specified in the report, it is assumed that the film was fixed to the cubicle or calutron.

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The measurements were made on all sides of the cubicle and in the immediate area around the cubicles and the calutrons. However only the measurements made on the front of the cubicle are used in this analysis because that was the position where the operator worked. The data shown in Table B-1 were used to bound the operators' external radiation exposure.

Table B-1. Cubicle x-ray radiation measurements (Smith 1944a).

Film position during test	Cubicle number	Exposure duration (h)	Exposure (R/8-h day)	Annual dose equivalent (rem)
XAX Cubicle in Bldg. 9731				
Front window over instrument panel	1 2	53 42.3	0.0003 0.0008	0.08 0.20
XBX Cubicle in Bldg. 9731				
Front window over instrument panel in direct line with regulator tube	1	11.5 11.5 12.2	0.001 0.001 0.008	0.25 0.25 2.00
Front window over instrument panel in direct line with regulator tube	2	20.6 32	0.011 0.005	 2.75 1.25

Complementary X-ray exposure measurements were made using a condenser R-meter chamber by the General Electric Company (Schmidt, Jr. 1944a, pp. 5–6; 1944b, pp. 13–15). These measurements showed that the two sets of measurements were comparable and represented the X-ray radiation levels that the control operators were exposed to.

Unshielded X-ray radiation levels from some equipment and to certain workers could have exceeded 0.1 R/d. The potential for exposure to these levels was recognized by TEC management. Shielding in combination with limiting the time employees worked near the equipment was used to keep exposures below the limit of 0.1 R/d established by the National Bureau of Standards (Hull 1944, p. 12; Smith 1944b, p. 7). The accepted value of 0.1 R/d was exceeded only on rare occasions (Sterner and Riley, date unknown, p. 11).

B.2 OPERATORS (CONTROL OR CUBICLE OPERATORS)

The film data can be used for an overestimate of X-ray exposures to cubicle operators that is favorable to the claimant. The highest measured exposure rate at the front of a cubicle was 0.011 R per 8-hour day, assuming that the rectifier was operating at 150 kV (Table V, Smith 1944a), which best represents the high end of the voltage range typically observed at the cubicle (Dahl 1944). This exposure rate applies to all control or cubicle operators.

B.3 OTHER WORKERS

Maintenance and/or trades workers were in the calutron enrichment area to monitor, adjust, and repair the equipment. They routinely looked through the windows (viewing ports) in the rear of the cubicle to ensure that the rectifier tubes were working properly.

Electricians likely spent more time in and around the cubicles and the calutrons because these were electrical equipment. They also would have worked in the cable and tube test operation. Here, overvoltages were applied to this electrical equipment to test it before putting it into service. Sterner

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and Riley (date unknown, p. 12) said that testing operators could have been exposed to levels of 1-2 R/d before the equipment being tested was enclosed with lead shields and leaded glass. This action reduced the exposure levels to 0.1 R/d. This is the only reference found where this operation is mentioned. Based on this limited information, NIOSH assumes that electricians were exposed at levels equal to the daily exposure limit of 0.1 R.

Other maintenance and trades workers presumably had other duties and did not spend as much time near the cubicles and calutrons as the electricians did. Therefore, NIOSH assumes that daily exposure is one-half of the daily limit which is 0.05 R.

For all of the exposures to control operators and other calutron related workers, it is assumed that the listed doses are WB rather than partial-body doses and that the worker was exposed at the location where the exposure was measured rather than the worker's actual physical location. The resultant measured exposure rates are used to bound the worker's annual, WB dose equivalent. Table B-2 is a summary of the recommended maximum annual dose assignments for various occupations. If an Energy Employee worked for only part of a year, the dose reconstructor can adjust the annual doses accordingly.

Table B-2. Bounded WB annual dose equivalent for calutron-related uranium enrichment workers, March 1, 1943 to December 31, 1947.

Occupation	Bounded annual WB annual dose equivalent (rem)	Basis for bounded dose equivalent	Reference
All control (cubicle) operators	2.75	Film measurements	Smith (1944a)
Electricians	25	Exposure limit	Sterner and Riley (date unknown)
All other maintenance and trades workers	12.5	Exposure limit	Sterner and Riley (date unknown)