ORAU TEAM
Dose Reconstruction
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Oak Ridge Associated Universities I Dade Moeller I MJW Technical Services

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Subject Expert(s): George D. Kerr

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PUBLICATION RECORD

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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

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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.
12/18/2009	02	Attachment A in earlier versions of this document was deleted and the relevant data incorporated in the main body of the current document. Sections 6.3.1, 6.3.2, 6.3.3, 6.3.4, 6.4.2, 6.5.1, and 6.5.2 were revised to incorporate updated data from a number of Technical Information Bulletins. Updates references within the text and Reference 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 George D. Kerr.

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ACRONYMS AND ABBREVIATIONS

ABRWH	Advisory Board on Radiation and Worker Health
AEC	U.S. Atomic Energy Commission
A-P	anterior-posterior
CFR	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
D_2O	deuterium oxide or heavy water
E	energy
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
eV	electron volt (a unit of energy)
ft	foot
g	gram
GM	geometric mean
GSD	geometric standard deviation
hr	hour
HEU	highly enriched uranium
HP	Health Physics
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
K-25	Oak Ridge Gaseous Diffusion Plant
keV	kiloelectron volt
kV	kilovolt
LANL	Los Alamos National Laboratory
m	meter
MDL	minimum detection limit
MED	Manhattan Engineer District
MeV	megaelectron volt
mg	milligram
mm	millimeter
mrem	millirem

NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NTA	nuclear track emulsion, type A
NU	natural uranium
OD	optical density
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
PIC	pocket ionization chamber
PNL	Pacific Northwest Laboratory
POC	probability of causation
R	roentgen
RPG	radiation protection guideline
s	second
SD	standard deviation
SEC	Special Exposure Cohort
SRDB Ref ID	Site Research Database Reference Identification (number)
TBD	technical basis document
TEC	Tennessee Eastman Corporation
TLD	thermoluminescent dosimeter
TLND	thermoluminescent neutron dosimeter
UCCND	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 (TBDs) 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 historical 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 CFR Part 82) define “performance of duty” for DOE employees with a covered cancer or restrict the “duty” to nuclear weapons work (NIOSH 2007a).

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 2007a):

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

- Background radiation, including radiation from naturally occurring radon present in conventional structures
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

6.1.1 Purpose

The purpose of this 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 (ORAUT 2007a; Wilcox 2001). 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 National Laboratory (LANL) 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 (or K-25) Plant (ORAUT 2007a, pp. 9-10). The UF_6 was then further enriched at Y-12 by the beta calutrons and shipped to LANL. In these early days of Y-12, TEC relied primarily on facility monitoring to measure and control the radiation exposure to workers (Murray 1948a, pp. 1-6; Murray 1948b, pp. 1-6). 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 (ORAUT 2007a). 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 nonradioactive 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, Spectrograph Laboratory, and Machine Shops where uranium metals were handled (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, pp. 39-40). As a result of the 1958 criticality accident, a program was instituted in 1961 to monitor all Y-12 workers individually 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 1993a,

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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.

p. 5). 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 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 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, pp. 14-15). The data have recently been analyzed and made available for use in dose reconstruction for Y-12 workers (ORAUT 2005a, p. 13).

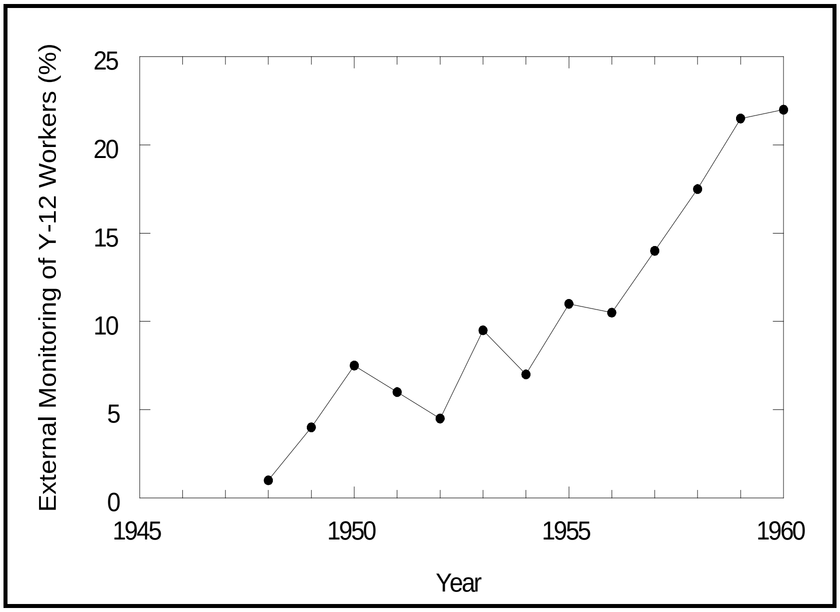

Figure 6-1. Percentage of Y-12 workers monitored for external radiation exposure, 1948 through 1961 (Watkins et al. 1993, p. 39).

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 the external whole-body (WB) dose to individual workers from personnel dosimeters are available for some employees from 1948 to 1950, for employees with the highest potential for external radiation exposure from 1950 to 1961, and for all employees from 1961 to 1996, and for employees entering radiological areas after 1996 (West 1993a; Souleyrette 2003a, p.2; ORAUT 2005b, pp. 12-14). Records of the external radiation dose to the extremities (hands and forearms) are also available for some individual workers who handled uranium metals. The so-called extremity personnel monitors for a worker's hands consisted of film patches from 1948 to 1952 (Murray 1948a, p. 3; Struxness 1949a, pp. 8 and 15; Larson 1949; Lister 1951), finger rings containing photographic film from 1952 to 1980 (Struxness 1951a, pp. 25-26; Struxness 1953, p. 13), and finger rings or wristbands containing thermoluminescent dosimeter (TLD) chips after 1980 (Bogard 1983; Henderson 1991, pp. 40-41; Oxley 2000; Veinot 2007).

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 (RU) contaminants (^{99}Tc, ^{237}Np, ^{238}Pu, and ^{239}Pu in lesser quantities); 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. Details on reconstructing external photon doses are contained in Attachment A.

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 millimeters and represents the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose, d = 0.07 mm and is noted as $H_p(0.07)$. For penetrating radiation of significance to 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).

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. 37–50). 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 (2007b) guidelines depend on:

- **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; and

- **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 dosimeter program was started in May 1948 to monitor individual external exposures of personnel working in the Assay Laboratories, Radiographic Shop, Spectograph Laboratory, and Machine Shops where uranium metals were handled (Struxness 1948, p. 8; Struxness 1949a, p.9). Other groups of workers were added in July 1948, January 1949, and July 1949 (ORAUT 2005a). At first, the monitoring of radiation workers was performed using pocket ionization chambers (PICs), typically exchanged on a daily basis (Souleyrette 2003a, p. 1). Early efforts were concerned with using a photographic film pad on the hands of the uranium metal workers and attempting to correlate the film pad readings with WB radiation exposures, which were first recorded using PICs and later with personnel WB film badge dosimeters (Souleyrette 2003a, p.1; Murray 1948a, pp. 3-4; Struxness

1948, pp. 8-9). In 1950, all Y-12 personnel working with depleted uranium (DU), discrete gamma or beta sources, X-ray machines, or fission-product contaminated materials were asked to wear a film badge dosimeter for purposes of estimating the external exposures to both photon and beta radiation (McLendon 1960, p. 3). There were only a few locations at the Y-12 Plant where exposures to neutron radiation were routinely possible (Emlet 1956), and in these instances personnel monitoring for a worker was provided by placing a nuclear track emulsion, Type A, or so-called NTA film, in a worker's film badge starting around 1950 (Souleyrette 2003a, p. 3). These NTA films were routinely exchanged biweekly because there was significant fading of the neutron-produced recoil ion tracks over longer periods (Thornton, Davis, and Gupton 1961, p. 28).

The first film badge used at Y-12 was the same badge used at the Oak Ridge National Laboratory (ORNL) (West 1993a, p. 2) and described by Thornton, Davis, and Gupton (1961, pp. 2-5). This film badge was a U.S. Atomic Energy Commission (AEC) Catalog Number PF-1B film badge manufactured by the A. M. Samples Machine Company in Knoxville, Tennessee (Patterson, West, and McLendon 1957, p. 21). The film in the PF-1B badge was encased in a protective packet with a clip for attachment to clothing or a lanyard (Handloser 1959, Figure 8-1). The film badge was worn on the front of the torso between the neck and the waist. A portion of the film was covered by a 1-mm-thick or 1,000-mg/cm^2 cadmium filter to determine the radiation dose from photons (gamma rays and high-energy X-rays), and the remaining uncovered portion of the film (open window) was used to determine the radiation dose from beta particles and low-energy X-rays (Handloser 1959). This type of film badge is commonly referred to as a two-element film badge dosimeter with one of the two elements being the reading from film behind the open window and the other element being the reading from the film behind the 1-mm cadmium shield (Patterson, West, and McLendon 1957, p. 21).

The above film badge was used until 1961 when a newer film badge was adopted for use at all Union Carbide Corporation – Nuclear Division (UCCND) facilities (Thornton, Davis, and Gupton 1961, p. 2; McLendon 1963, pp. 23-25; McRee, West, and McLendon 1965, pp. 19-21). This film badge was issued to all personnel at the Y-12 Plant because it served as a security badge and also provided for monitoring of both routine and accident-related exposures. As in the PF-1B film badge, a cadmium filter with a mass density of 1,000 mg/cm^2 was used to determine the penetrating WB dose from gamma radiation. In addition, the newer film badges continued to include an open window to measure beta radiation and to distinguish between beta and photon radiation. Plastic and aluminum filters were also incorporated into the UCCND badges; thus, they are commonly referred to as four-element film badge dosimeters. The film areas behind the plastic and aluminum were read, but the results were not used routinely in the evaluation of a worker's dose from photon and beta radiation at Y-12 (Sherrill and Tucker 1973, p. 7). All badges contained an NTA film but they were developed and read only if a worker had a potential for exposure to neutrons (McLendon 1963, p. 23). Thornton, Davis, and Gupton (1961, pp. 24-29) had experimented with moisture-proof pouch paper and found that the latent image of tracks from the neutron-produced recoil ions could be controlled while the NTA films remained in the film badge dosimeters for a calendar quarter, as shown in Figure 6-2.

In 1980, the film badge dosimeters were replaced by TLDs (McLendon et al. 1980, pp. 5-7; Howell and Batte 1982, p. 6). A TLD is referred to as a two-element dosimeter if it contains two TLD chips and a four-element dosimeter if it contains four TLD chips for discrimination of the responses of the chips as functions of the type and energy of various radiations. A two-element Harshaw TLD was provided to each Y-12 worker for monitoring their external exposure to photons (gamma rays and X-rays). In addition, radiation workers were provided a supplemental dosimeter, consisting of two four-element Panasonic TLDs (MMES 1992, Executive Summary, pp. 1-6). One of these Panasonic dosimeters was a photon-beta dosimeter, the other a neutron dosimeter (MMES 1992, Vol. 1, Executive Summary, pp. 1-6). In 1989, the Y-12 Plant began the so-called Centralized External Dosimetry System (CEDS) developed by Martin Marietta Energy Systems (MMES 1992, Vol. 1,

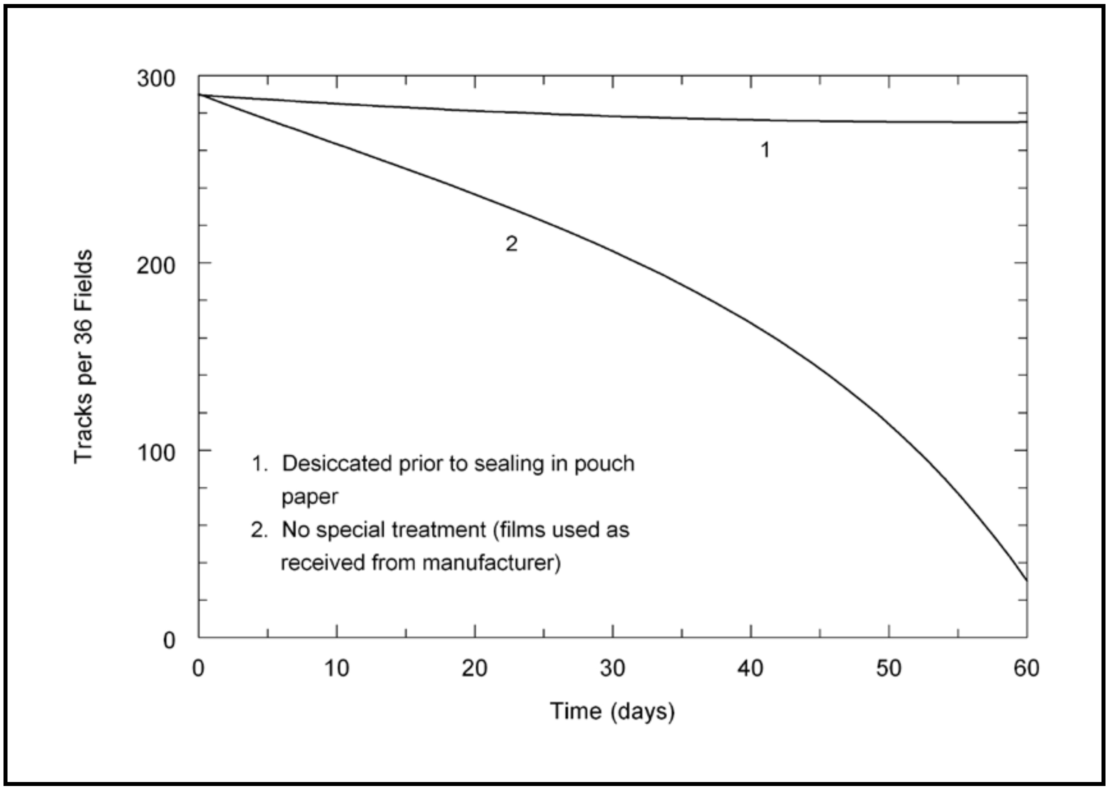

Figure 6-2. Effect of sealing NTA film in a moisture-proof pouch (Thornton, Davis, and Gupton 1961, p. 28; Morgan, Davis, and Hart 1963, p. 69).

Executive Summary, pp. 1-6). The CEDS used a four-element Harshaw TLD (Model 8805) to monitor a worker's exposure to photon and beta radiation (MMES 1992, Vol. 1, Executive Summary, pp. 1-9). With the introduction of the Harshaw beta-gamma dosimeter, neutron dosimetry continued with the supplemental Panasonic dosimeter. However, work continued on the development of a new neutron dosimeter that was compatible with the automated processing equipment used for the four-element Harshaw beta-photon dosimeter (MMES 1992, Executive Summary, pp. 1-6). In January 1990, use of the supplemental Panasonic dosimeter was discontinued and the new four-element Harshaw TLD (Model 8806B) became the neutron dosimeter of record (MMES 1992, Vol. 1, Executive Summary, pp. 1-6). A summary of the typical exchange frequencies for the various personnel dosimeters used at the Y-12 Plant during different periods of time is provided in Table 6-1.

The MDL of the various film badge dosimeters used at Y-12 to monitor for beta/gamma exposures of the whole body is summarized in Table 6-2. It is important to note that the Y-12 estimates for the MDLs of the film badge dosimeters varied significantly with time (West 1993a). As noted in the footnotes to Table 6-2, the dates are approximate because the changes did not occur for all workers at the same time (West 1993a). In 1950 and 1951, dose from beta and gamma radiation were recorded as zero if they were less than the estimated MDL of 30 mrem (see Table 6-2). In other years prior to 1955, it was the practice to assign a gamma or beta dose to a worker equal to the MDL. If the portions of the film covered by the cadmium shield and open window gave readings less than the MDL, the assigned dose was attributed to gamma rays if the worker had a high potential for exposure to gamma rays and to beta particles if the worker had a high potential for exposure to beta particles (West 1993a). From 1955 to 1958, an assigned dose equal to one-half the MDL was attributed only to beta particles if the portions of the film covered by the cadmium shield and open window were both less than the estimated MDL (West 1993a). The MDL of the NTA film used in the film badge dosimeters is typically quoted in the literature as varying between 50 and 100 mrem

(Morgan 1961, p. 18; Parrish 1979, p. 48; Wilson et al. 1990, p. 160). However, the MDL depends on a number of parameters including the number of microscopic fields used in the evaluation of the neutron-produced recoil ion density of an exposed NTA film and the energy of the neutrons used in the calibration of the NTA film (Thomas, Horwood, and Taylor 1999; IAEA 1985, pp. 63-71). The MDL of NTA films based on calibrations at Y-12 using either Po-Be or Am-Be neutron sources has been estimated previously to be approximately 50 mrem (ORAUT 2005c, p. 8).

Table 6-1. Monitoring technique and exchange frequency used at the Y-12 facility for occupational WB radiation exposures.[a]

Period	Monitoring technique	Exchange frequency	Monitored personnel
Photon and beta radiation			
May 1948–December 1949	PIC for gamma radiation and two-element film dosimeter for photon radiation	PICs daily, film dosimeters weekly	Personnel expected to receive over 10% of the radiation protection guideline (RPG)
January 1950–September 1958	Two-element film dosimeter	Weekly	Personnel expected to receive over 10% of the RPG
October 1958–December 1960	Two-element film dosimeter	Monthly	Personnel expected to receive over 10% of the RPG
January 1961–December 1979	Four-element film dosimeter	Quarterly	Nearly all personnel monitored
January 1980–December 1988	Two-element Harshaw TLD for photon radiation	Some quarterly, some annually	Quarterly exchange for personnel expected to receive 500 mrem or more, annual exchange for personnel expected to receive less than 500 mrem
	Supplemental four-element Panasonic TLD for beta radiation	Quarterly	Personnel exposed to beta-particle sources
January 1989–Present	Four-element Harshaw TLD for photon and beta radiation	Some quarterly, some annually	Nearly all personnel monitored from 1989 to 1996. After 1996, only personnel entering radiological areas.
Neutron radiation			
January 1950–December 1960	NTA film	Biweekly	Personnel exposed to neutron sources
January 1961–December 1979	NTA film	Quarterly	Personnel exposed to neutron sources
January 1980–December 1989	Supplemental four-element Panasonic TLND	Quarterly	Personnel exposed to neutron sources
January 1990–Present	Supplemental four-element Harshaw TLND	Quarterly	Personnel exposed to neutron sources

a. Oxley (2007a), Souleyrette (2003a, p. 2), MMES (1992, Volume 1, Executive Summary, pp. 1-6); West (1993a), McMahan (1991), Y-12 Plant (1988a, 1980a), McLendon et al. (1980), Thornton, Davis, and Gupton (1961), McLendon (1958), Reavis (1958).

The exchange frequencies, MDLs, and potential annual missed dose for all external radiation dosimeters used at the Y-12 Plant are summarized in Table 6-3. The potential annual missed doses in the last column of Table 6-3 are based on the number of dosimeter exchanges per year times one-half of the MDL as recommended by NIOSH (2007, p. 16). Some of the MDLs in Table 6-3 were difficult to estimate, particularly for the film dosimeters. For the current TLDs, the MDLs are more precisely identified in the *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 a larger uncertainty because factors involving the radiation field, film type, processing, developing, and reading system cannot be retested. The MDLs for the various radiation dosimeters in Table 6-3 are based on information from the references in the footnotes to the table.

Table 6-2. MDLs and assigned doses (mrem) for film dosimeters used to measure photon and beta doses.[a]

Period[b]	MDL[c]	Assigned dose
May 1948–December 1949	30	30[d]
January 1950–December 1951	30	0
January 1952–September 1952	50	50[d]
October 1952–December 1952	43	43[d]
January 1953–June 1954	50	50[d]
July 1954–December 1954	30	30[d]
January 1955–December 1957	30	15[e]
January 1958–December 1979	30	Not applicable[f]

a. ORAUT (2005a); West (1993a), McLendon (1958), Reavis (1958).
b. Dates are approximate because the changes did not occur for all employees at the same time (West 1993a).
c. Film badge dosimeters typically exhibited about the same sensitivity to photon and beta radiation; i.e., a dose of 1 rem from beta particles yielded about the same response in the film as 1 rem of gamma rays (Auxier 1967).
d. Assigned to gamma dose for those workers with a high potential for exposure to gamma rays or to beta dose for those workers with a high potential for exposure to beta particles if both shielded and open-window film readings were less than the MDL (West 1993a).
e. Assigned to beta dose if shielded and open-window film readings were both less than the MDL (West 1993a).
f. The actual shielded and open-window film readings of the film dosimeter were used to calculate the gamma and beta doses even when the film readings were less than the MDL.

Table 6-3. Dosimeter type, exchange frequency, MDL, and potential annual missed dose.[a]

Period	Dosimeter type	Exchange frequency	Laboratory MDL (mrem)	Potential annual missed dose (mrem)
Photon and beta radiation				
May 1948–December 1949	PICs	Daily	5	650
May 1948–December 1951	Two-element film badge	Weekly	30[b]	780
January 1952–September 1952	Two-element film badge	Weekly	50[b]	1,300[c]
October 1952–December 1952	Two-element film badge	Weekly	43[b]	1,118[c]
January 1953–June 1954	Two-element film badge	Weekly	50[b]	1,300[c]
July 1954–September 1958	Two-element film badge	Weekly	30[b]	780[c]
October 1958–December 1960	Two-element film badge	Monthly	30[b]	180[c]
January 1961–December 1979	Four-element film badge	Quarterly	30[b]	60
January 1980–December 1988	Two-element Harshaw TLD for photon radiation	Quarterly	20[d]	40
	Supplemental four-element Panasonic TLD for beta radiation	Quarterly	20[d]	40
January 1989–	Four-element Harshaw	Quarterly	20	40

Period	Dosimeter type	Exchange frequency	Laboratory MDL (mrem)	Potential annual missed dose (mrem)
Present	TLD			
Neutron radiation				
June 1952–June 1960	NTA film	Biweekly	50	650[e]
July 1960–December 1979	NTA film	Quarterly	50	100[e]
January 1980–December 1989	Supplemental four-element Panasonic TLND	Quarterly	20	40
January 1990–Present	Supplemental four-element Harshaw TLND	Quarterly	20	40

a. Souleyrette (2003a), Oxley (2001), Ashley et al. (1995), West (1993a), Wilson et al. (1990), Thornton, Davis, and Gupton (1961).
b. Film badge dosimeters typically exhibited about the same sensitivity to photon and beta radiation; i.e., a dose of 1 rem from beta particles yielded about the same response in the film as 1 rem of gamma rays (Auxier 1967).
c. Different MDL values were used during portions of 1952, 1954, and 1958. Thus, the potential annual missed gamma and beta dose for these years is slightly different than the values listed in the table. For example, the potential annual missed annual gamma and beta doses for 1952 are calculated to be 25 mrem/wk times 39 wk plus 21.5 mrem/wk times 13 wk or 1255 mrem. The potential annual missed gamma and beta doses for 1954 and 1958 are calculated to be 1090 mrem and 630 mrem, respectively.
d. Gamma and beta doses were recorded as zero if TLD readings were less than 20 mrem (Howell and Batte 1982, p.48).
e. Potential annual missed dose based on data from laboratory irradiations might not be directly applicable to workplace missed neutron dose (Wilson et al. 1990, p. 7.6; IAEA 1985, pp. 63-71). Discussions of additional correction factors for missed neutron dose in various workplaces at the Y-12 facility can be found in ORAUT (2006a).

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 1993a; Oxley 2001; Souleyrette 2003a, pp. 1–6). Routine practices appear to have required assigning dosimeters to personnel who might have received an external radiation dose that was greater than 10% of the radiation protection guidelines (RPGs) in effect at that time (see, for example, ORAUT 2007b, pp. 33-34). According to West (1993a), 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, NTA film was not issued to the worker prior to 1960 (West 1955) or the NTA emulsion in the worker's film badge dosimeter was not processed from 1961 to 1979 (West 1993a, 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 6-3). There is also the problem of missing dose components for a worker designated simply as NOTAV for lost or not turned and "not available" or DAM for "damaged and not readable" in a worker's records (Watson 1990; West 1956; 1993a, p. 3; McLendon 1963). 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.

During certain time periods, it was the practice to assign either gamma or beta doses to workers that were equal to the MDL/2 or MDL of the film badge dosimeters if the readings of both the cadmium-shielded and open-window portions of the film were less than the MDL (see Table 6-2). It may be assumed, therefore, that there was no missed gamma or beta dose to Y-12 workers wearing film badge dosimeters during these periods (see Table 6-2). This assumption would only be true in the case of the shallow dose to skin which is the same for both gamma rays and beta particles (ORAUT 2007b). If a worker's recorded dose was assigned to beta particles and a recorded dose of zero for gamma rays was assigned to the worker, then the estimated radiation doses to more deeply seated organs of the body from gamma rays could be grossly underestimated. Thus, reconstruction of doses to a deeply seated organ of the body should use a probable missed gamma dose equal to one half of the MDL for each film-badge measurement that was equal to zero or less than the MDL divided by 2 as recommended in the external dose reconstruction guidelines issued by NIOSH (2007b).

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 measure radiation dose accurately as discussed in later sections depends on radiation type, energy, exposure geometry, etc. The exchange frequency of the dosimeters was gradually lengthened and corresponded generally to downward reductions in the RPGs for the deep or penetrating WB dose (Morgan 1961; Taylor 1971; Inkret, Meinhold, and Tascher 1995). During the early stages of the Y-12 program to monitor individual workers, a weekly dose limit of 0.3 rem was in effect for gamma radiation (Wiley 2004; ORAUT 2007b, Table 3-4, p. 14). This was reduced to an annual limit of 5 rem in the latter part of the 1950s and has continued to be further reduced over time (Inkret, Meinhold, and Tascher 1995).

6.3.2.1 Beta-Photon Dosimeters

The film badges used at Y-12 from 1948 to 1961 had a completely open window over the film packet, and the film packet used to protect the film against light and humidity had a thickness of about 25 to 30 mg/cm^2 over the front surface of the film (Dudney 1956; ICRU 1997). This thickness stopped essentially all beta particles with energies less than approximately 150 keV from penetrating the film packet and producing a film response (ORAUT 2007b, Figure 3-1, p. 11). The film badges used at Y-12 from 1961 to 1980 had an open window covered by security credentials with a thickness of about 50 mg/cm^2, so the total thickness over the surface of the film was approximately 80 mg/cm^2 (Thornton, Davis, and Gupton 1961, p. 7). This thickness stopped essentially all beta particles with energies less than approximately 300 keV from producing a response in the film (ORAUT 2007b, Figure 3-1, p. 11). In the practical monitoring of beta particles, the exposure geometry is usually unknown and quite variable, but the source is usually known (the beta-emitting isotope, uranium metal, thorium metal, etc.). After filtration of the lowest energy beta particles by the paper jacket and security credentials over the film, the energy spectrum of the beta particles varies slowly with the source-to-film distance (i.e., the thickness of the intervening air). In addition, a 1-MeV beta particle can travel over 4 m in air (Berger et al. 2005). Therefore, a fairly good calibration can be achieved with a standardized source of the same beta emitter as the one whose beta radiation is to be monitored, especially in the case of radiation that is diffusely incident on the body (Ehrlich 1962). Thus, the beta radiation doses at a uranium facility are expected to be reliable for film badge dosimeters calibrated with a natural uranium (NU) source, and the MDLs for the beta radiation doses are expected to be approximately the same as those for gamma radiation doses (Auxier 1967; Thornton, Davis, and Gupton 1961, pp. 48-49).

The Y-12 film badges used from 1948 to 1963 contained DuPont type 552 film packets (Souleyrette 2003a, p. 3). 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 1953, p. 45; Thornton, Davis and Gupton 1961, pp. 32-35; Parrish 1979, p. 117). In 1963, Y-12 switched to the use of DuPont 544 film packets (McLendon 1963, p. 23; Souleyrette 2003a, p. 3). These packets consisted of (1) a sensitive 555 emulsion with an effective range from approximately 30 mrem to 5 rem (McLendon 1963, p. 25) and (2) an insensitive 834 emulsion with an effective range from approximately 5 rem to 150 rem (Thornton, Davis, and Gupton 1961, pp. 32-35; Davis and Gupton 1963, pp. 123-124; Parrish 1979, p. 117). In 1971, DuPont stopped manufacturing the 544 film packets and Y-12 switched to Eastman type 2 film (Jones 1971, p. 5; Souleyrette 2003a, p. 4). 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, p. 7). 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 in the office area to investigate onsite radiation background (Jones 1971, pp. 9-10). 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 (Jones 1971, pp. 9-10). 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, pp. 9-10).

The response of the film badge to photon radiation of different energies is illustrated in Figure 6-3. 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 a multielement film badge (Thornton, Davis, and Gupton 1961, p. 40). 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, pp. 5-7); the four-element TLD badges Y-12 adopted for use in 1989 had four LiF chips covered by an open window, plastic filter, copper filter, and hemispherical Teflon button (Y-12 Plant 1988a; Oxley 2001, pp. 7 and 11-12). 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 (McLendon et al. 1980, p. 7) and the hemispherical Teflon button in the four-element TLD badge (Oxley 2001, p. 12). 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 dosimeter intercomparison study of 10 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 the DOE Hanford Site (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 (Thierry-Chef et al. 2002). Dosimeter responses for these irradiation geometries were investigated using two different phantoms to represent the torso of the body (Tierry-Chef et al. 2002). 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 performance testing by the International Standards Organization. The second phantom was an anthropomorphic Alderson Rando phantom. This realistic phantom is constructed using a natural

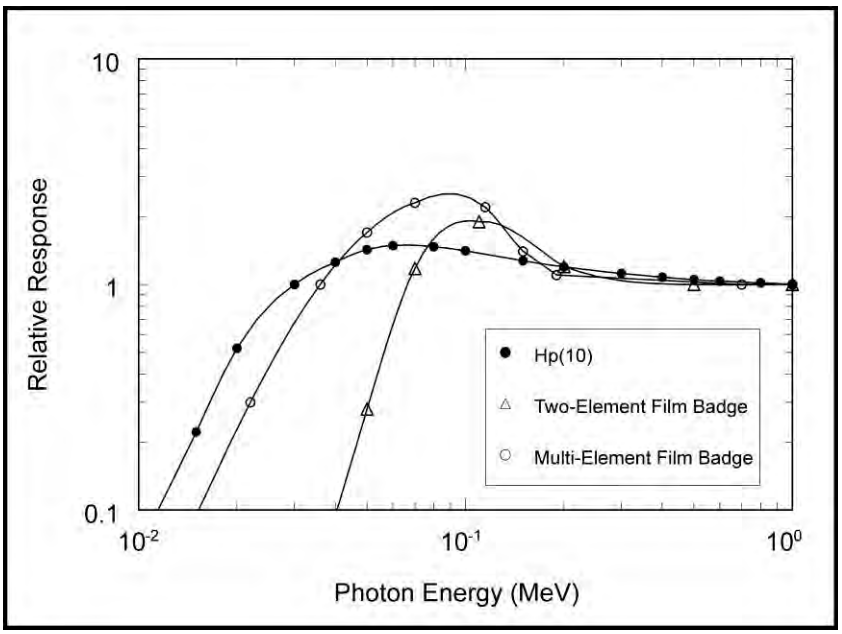

Figure 6-3. Comparison of $H_p(10)$ from a broad beam of normally incident photons (ICRP 1996) with the energy responses for a sensitive DuPont 502 emulsion in a MED two-element film badge (Pardue, Goldstein, and Wollan 1944) versus a sensitive DuPont 555 emulsion in an ORNL multielement film badge (Thornton, Davis, and Gupton 1961).

human skeleton cast inside material that has a tissue-equivalent composition. As noted above, 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 *Hp(10)* at the lower photon energies of 118 keV and 208 keV as shown in Table 6-4.

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)							
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 $H_p(10)$.

The TLDs currently used at Y-12 for the measurement of the radiation doses from beta particles, X-rays, and gamma rays are calibrated and tested using standards developed by DOE (see, for example, DOE 1986 and Oxley 2007b).

6.3.2.2 Neutron Dosimeters

The NTA film dosimeters and TLNDs that have been used at Y-12 differ significantly in their response to neutrons of different energies as illustrated in Figure 6-4 (IAEA 1990; 2001). An NTA film for neutron dosimetry was included in the same holder used at the Y-12 beta-gamma dosimeter until December 1979. Starting in January 1980, the neutron doses to Y-12 workers have been measured using a supplemental thermoluminescent neutron dosimeter (TLND) (see Table 6-3). 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 effective threshold energy of 500 to 700 keV (Drew and Thomas 1998, p. 2; ORAUT 2006a, Table 3-1, p. 10). Results reported at the first AEC neutron dosimetry workshop in 1969 indicated that laboratory measurements with NTA film were about one-half to one-fourth of those that were measured with other methods including TLNDs (Vallario, Hankins, and Unruh 1969, p. 49). The response of both dosimeters is highly dependent on the neutron energy spectra, and both dosimeter types require careful matching of laboratory calibration neutron energy spectra to the workplace neutron energy spectra for reliable results (Thomas, Horwood, and Taylor 1999; Oxley 2001).

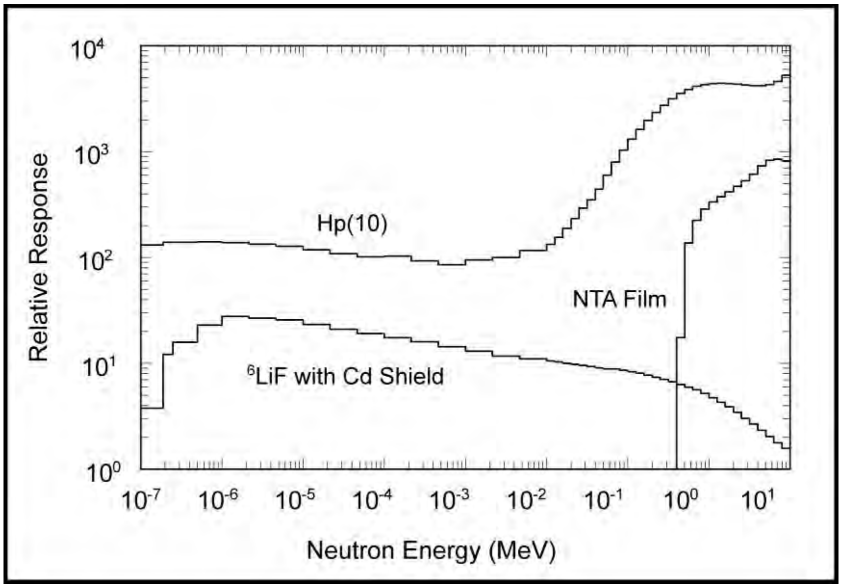

Figure 6-4. Comparison of $H_p(10)$ from normally incident neutrons (IAEA 2001) with the energy responses of an NTA film and a neutron albedo dosimeter with a neutron TLND chip made of ^{6}LiF shielded by cadmium (IAEA 1990, 2001).

6.3.3 Y-12 Dosimeter Calibration Procedures

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 (see Figure 6-3) and both NTA film dosimeters and TLNDs for neutron radiation (see Figure 6-4).

Beta/Photon Dosimeters

The Y-12 film dosimeters were calibrated using beta particles from a slab source of natural uranium (Souleyrette 2003a, pp. 3-4; ORAUT 2007b, pp. 9-12). A ^{226}Ra source enclosed in 0.5 mm of platinum was used initially as the calibration gamma-ray source and a ^{60}Co source was used starting in the early 1960s (ORAUT 2007b, p. 10). The dosimeters were exposed face down on the uranium slab and free in air (i.e., no phantom) facing the ^{226}Ra or ^{60}Co sources for preselected times to reproduce the beta particle and photon doses that were normally encountered in the workplace. These practices were similar to those used at other AEC sites. The current beta/photon TLDs have been calibrated using a variety of beta and gamma sources to meet rigid standards established by DOE (see, for example, DOE 1986 and Oxley 2007b).

Neutron Dosimeters

The NTA films were initially calibrated using a Po-Be neutron source (Struxness 1949a, 1953) and later an Am-Be neutron source (McLendon 1963; McRee, West, and McLendon 1965). The film badge 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. The newer TLNDs have also been calibrated and tested to meet standards established by DOE (see, for example, DOE 1986 and Oxley 2007b). The TLNDs are typically calibrated using a broad energy spectrum of spontaneous fission neutrons from a D_2O-moderated ^{252}Cf source (Oxley 2007b).

6.3.4 Y-12 Workplace Radiation Fields

The major workplace radiation fields at the Y-12 facility have been associated with the processing of uranium materials, principally EU (^{235}U) and DU (^{238}U). The external radiation exposures to workers from processes involving EU and DU materials have been discussed in recent reports by Henderson (1991) and Ashley et al. (1995). Other workplace radiation fields at Y-12 have involved industrial generating equipment (e.g., X-ray machines and particle accelerators) and isotopic gamma-ray and neutron sources for testing purposes (e.g., ^{60}Co and ^{252}Cf) (Souleyrette 2003a, p. 1). The Y-12 external dosimetry TBD provides a discussion of radiation fields from primary radionuclides and industrial test equipment in the workplace (Oxley 2007b, pp. 3-4).

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. Material processing at Y-12 has included the recovery of both EU and RU from the Y-12 site and other sites operated by DOE (BWXT Y-12 2000). Material processing and fabrication at Y-12 has also included NU, DU, thorium, and ^{233}U (ORAUT 2007b, pp. 16-21). The Y-12 Plant was involved in the processing and fabrication of metal parts from ^{233}U for several years in the 1960s and 1970s (West and Roberts 1962; West 1974). Because of the occupational internal radiation hazards of working with ^{233}U, all operations were doubly contained (West and Roberts 1962; West 1974). The operations were short-term and involved limited amounts of ^{233}U due to its very small critical mass (Kang and von Hippel 2001). However, extensive area and individual monitoring for external radiation was provided by the Health Physics (HP) Department during these operations. The use of gloveboxes and remote handling equipment resulted in minimal beta doses to the skin and hands of the Y-12 workers who were involved in the processing and fabrication of the metal parts from ^{233}U (West and Roberts 1962; West 1974). The processing of thorium metals at Y-12 began in the late 1950s or early 1960s (West 1965; Wilcox 2001; ORAUT 2007a). In general, the same Y-12 facilities were involved in the processing and fabrication of both thorium and DU metals (ORAUT 2007a; McRee, West, and McLendon 1965; McLendon 1963). The external beta/photon doses to the thorium workers were found to be either lower than or of the same order of magnitude as those to depleted uranium workers (West 1965, p. 27).

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

Y-12 site processes[a]	Building[a]	Operations[a] Begin	Operations[a] End	Radiation type	Energy selection[b]	Percent[c]
Enriched uranium product recovery and salvage operations	9203	1947	1951	Beta Gamma	>15 keV 30–250 keV	100 100
	9206[d]	1947	1959			
	9211	1947	1959			
	9201-1	1952	1963			
Uranium chemical operations and weapon production operations	9202	1947	1995	Beta Gamma	>15 keV 30–250 keV	100 100
	9206[d]	1947	1995			
	9212[e]	1949	Ongoing			
Special nuclear material receiving and storage	9720-5	1949	Ongoing	Gamma	30–250 keV	100
Uranium forming and machining for weapon component operations	9201-5	1949	Ongoing	Beta Gamma	>15 keV 30–250 keV	100 100
	9204-4	1949	Ongoing			
	9215	1950	Ongoing			
Depleted uranium process operations	9201-5	1949	Ongoing	Beta Gamma	>15 keV 30–250 keV >250 keV	100 50 50
	9204-4	1949	Ongoing			
	9766	1949	?			
	9998	1949	Ongoing			
Final weapon component assembly operations	9204-2	1952	Ongoing	Beta Gamma	>15 keV 30–250 keV	100 100
	9204-2E	1952	Ongoing			
ORNL 86-in. cyclotron	9201-2	1950	1961	Gamma	30–250 keV >250 keV	50 50
Chemical assay and mass spectrometry laboratories	9203	1947	Ongoing	Beta Gamma X-ray	>15 keV 30-250 KeV Specific to X-ray source.	100 100 100
Radiographic Laboratory	9201-1	1947	Ongoing			
Weapon component assay Laboratory	9995	1952	Ongoing			
Nondestructive assay laboratory	9720-5	1980	Ongoing			
West End waste treatment facility	9616-7	1984	Ongoing	Beta Gamma	>15 keV 30–250 keV	100 100
Calibration laboratory	9983	1949	Ongoing	Gamma	30-250 keV	100

a. See ORAUT (2007a, p.13, Table 2-2).
b. See NIOSH (2002, p. 43, Table 1).
c. See DOE (2004, pp, 2-5 to 2-6, Tables 2-3 and 2-4) or Shleien, Slack, and Birky (1998, pp. 8-38 to 8-40, Tables 8.9 and 8.10)..
d. Building 9206 Complex includes Buildings 9768, 9720-17, 9409-17, 9510-2, 9767-2, and the east and west tank farm pits.
e. Building 9212 Complex includes Buildings 9809, 9812, 9818, 9815, and 9980.

Because Y-12 is a nuclear weapons fabrication and disassembly facility, the most common materials are EU (^{235}U) and DU (^{238}U) (Oxley 2007b, p. 3). 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 particles from ^{234m}Pa, and (2) the photons emitted by ^{234m}Pa with energies as large as 1.926 MeV (Oxley 2007b, p. 3). 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 (Oxley 2007b, p. 3). During casting operations, the decay products of ^{238}U float to the top surface of the molten metal and remain as surface residues (Struxness 1954, pp. 20-23). 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 × 10^{15} times larger than the specific activity of its ^{238}U parent (Henderson 1991, p. 4). 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 decay products, the percentages of photons with energies of 30 to 250 keV and 250 keV or more 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 DU process operations. This produces doses that are favorable to the claimant because of the added potential exposure to high-energy photons from the short-lived ^{234}Pa decay product of ^{238}U.

The largest workplace exposures at Y-12 have historically occurred in DU process areas (Struxness 1954; UCNC 1963a, p. 9-10; UCNC 1963b, p. 10-11; Henderson 1991; Ashley et al. 1995). In the foundry areas, forming areas, and machine shops, workers handle both large and small pieces of DU metal (Ashley et al. 1995, p. 5). The workers are not typically in close contact with the material. Large parts are lifted with mechanical assistance, and the workers usually remain 2 to 3 ft from the material. Smaller parts might be loaded by hand but a worker is rarely in close contact with the material for an extended period (Ashley et al. 1995, p. 5). Similar workplace geometries are found in the facilities of the Ultrasonic Testing Group of Y-12 Quality (Ashley et al. 1995, p. 6). Large parts are loaded mechanically onto the testing machines. Small parts might be loaded by hand but the operator then moves to a control center several feet away, where he remains for the majority of the time he is working with the material. However, workers in the Mechanical Properties Inspection Laboratories, Dimensional Inspection Group, and Radiographic Laboratories can spend a considerable part of their time working in close contact (less than 30 cm) for extended periods (Ashley et al. 1995, p. 6). One problem with the workplace response of the Y-12 beta/gamma dosimeters involves workers who performed waist-level uranium handling of small metal objects in the DU areas (Henderson 1991, p. 41). A personnel dosimeter worn at the collar might underestimate both the $H_p(0.07)$ and $H_p(10)$ doses at the waist by a rather significant factor. 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 $H_p(0.07)$ and $H_p(10)$ recorded doses before 1991 should be multiplied by a factor of 1.71 and 1.34, respectively (Henderson 1991, p. 42). 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 file for a claimant. Some of the Y-12 workers who handled small DU parts at waist level were scrap metal handlers in the H-2 Foundry (Henderson 1991) and testing operators in the Mechanical Properties Inspection Laboratories, Dimensional Inspection Group, and Radiography Laboratories (Ashley et al. 1995). A list of Y-12 departments with a significant number of workers assigned to the DU operation and testing areas is provided in Table 6-6. This list was compiled from several different sources of dosimetric data for Y-12 workers (UCNC 1963a,b; West 1991; Watkins et al. 1993; Markowitz et al. 2004). The number of annual shallow doses, $Hp(0.7)$, greater than 500 mrem for the workers in these departments was taken from Markowitz et al. (2004). The departments listed in Table 6-6 account for 75% of all annual shallow doses to workers at Y-12 that were greater than 500 mrem or 0.5 rem during the period from 1950 to 1988 (Markowitz et al., 2004, Attachment B, p. B1-3).

6.3.4.2 Workplace Neutron Dosimetry Response

The major sources for neutron exposures for Y-12 workers are characterized in this section using parameters of interest in the dose reconstructions for Y-12 workers (NIOSH 2007b). These parameters include (1) the energy spectrum of the neutrons incident on a worker's body from a workplace neutron source, (2) the neutron-to-gamma dose ratios for exposures to these various sources, and (3) missed neutron dose due to the reduced response of NTA films for neutrons at lower energy as shown in Figure 6-4. The newer TLNDs used at the Y-12 facility were carefully calibrated and corrections were applied to the neutron doses of record to account for the variation in response due to the angular and energy distributions of the workplace neutron fields (McMahan 1991; Oxley 2001, p. 23).

The energy spectrum used in the dose reconstruction of workers is typically divided into the following five energy groups: <0.01 MeV, 0.01-0.10 MeV, 0.10-2.00 MeV, 2.0-20.0 MeV, and >20 MeV (NIOSH 2007b, p. 19). The Radiation Effective Factors used in NIOSH-IREP (NIOSH 2002) for calculating the POC are less for the lower two energy groups of <0.01 MeV and 0.01-0.10 MeV than that for the 0.10-2.00 MeV energy group (NIOSH 2002, p. 31; Kocher, Apostoaei, and Hoffman 2005). Thus, the two lower energy groups of <0.01 MeV and 0.01-0.10 MeV are combined with the 0.10-2.00 MeV group throughout this TBD. The 0.10-2.00 group is a substantial neutron component in each workplace, and it alone can be used (i.e., assigned 100%) as an analysis that is favorable to claimants when calculating their POC using NIOSH-IREP (NIOSH 2002, p. 31).

Table 6-6. Y-12 departments with greatest potential for exposure to beta radiation and number of annual shallow doses, $H_p(0.7)$, greater than 500 mrem during the period from 1950 through 1988 (Markowitz et al. 2004).

Y-12 department	**Number > 500 mrem**
2233	461
2375	566
2617	525
2618	2,106
2619	1,060
2637	470
2638	1,069
2640	1,025
2701	1,062
2702	2,012
2703	3,448
2720	813
2722	417
2792	1,308
2793	544
Percent of total for all Y-12 departments	75%

NTA films were calibrated with neutrons from Po-Be or Am-Be sources that were perpendicularly incident on the face of the film badge and NTA film inside the film badge (Struxness 1949a, pp. 20-22; McLendon 1963, pp. 75-76; McRee, West and McLendon 1965, pp. 85-87). In most workplace environments, however, the neutrons are incident on a film badge and NTA film at widely variant angles. The angular dependence of NTA films has been studied experimentally by Kathren, Prevo, and Block (1965). They suggest multiplying the tracks produced by perpendicularly incident neutrons from a calibration source by a factor of 0.75 to avoid the underestimation of the neutron dose to workers in most typical workplace environments. This is equivalent to multiplying the recorded neutron dose from a worker's NTA film by a factor of 1.3 (1 divided by 0.75). It is recommended, therefore, that dose reconstructors apply this factor to all recorded NTA film dosimeter results for Y-12 workers and an additional factor to account for the missed neutron dose due to the so-called energy threshold of the NTA film (ORAUT 2006a). The threshold energy of the NTA film for fast neutron detection is typically quoted as having values ranging from 0.5 to 0.7 MeV (ORAUT 2006a, p. 10, Table 3-1). A favorable to claimant value of 0.7 MeV is assumed here for the energy threshold of the NTA film (see Figure 6-4).

6.3.4.2.1 **86-Inch Cyclotron**

The Oak Ridge 86-in. cyclotron is treated here as a major source of neutron exposure to Y-12 workers from 1950 to 1961 because several Y-12 departments were involved in the early operation of the cyclotron and the production of radioisotopes at the cyclotron (Struxness 1952, pp. 36-42). The need for a cyclotron in the Oak Ridge area was discussed as early as 1946 (Livingston and Boch 1952, pp. 7-8). The availability of large magnetic and vacuum components at the Y-12 Plant greatly simplified the planning of the 86-in. cyclotron and facilitated its fabrication (Johnson and Schaffer 1994, pp. 65-70). A study of available buildings at the Y-12 Plant resulted in the selection of the Alpha Process Building 9201-1, which had been in standby condition since 1945 (Livingston and Boch 1952, p.10). The early operation of the 86-in. cyclotron was a cooperative Y-12 and ORNL effort (Johnson and Schaffer 1994, p. 67; Kerr 2007a). Ground was broken for the magnet footings on September 21, 1949, and the machine was ready for test operations the following September (Livingston and Boch 1952, pp. 7-8). The 86-in. cyclotron became operational in November 1950, and the first proton beam was observed on November 11, 1950 (Livingston and Boch 1952, pp. 7-8). By the end of the year, a proton beam of a few microamperes had been obtained at proton energies of approximately 20 MeV, and by September 1951 proton currents above 1 milliampere at proton energies of 20 MeV were possible (Livingston and Boch 1952, pp. 7-8).

The first major use of the 86-in. cyclotron was the production of ^{208}Po (Livingston and Martin 1952, p. 5; Butler 1963, p. 17; Kerr 2007a). In 1952, internal revisions of the position and mounting of the ion source resulted in an increase in proton energy to 23 MeV (Livingston and Martin 1952, p. 50). At the higher energy, the ^{208}Po yield was more than doubled and a total of approximately 9 Ci of ^{208}Po was produced before the project was terminated in August 1952 (Livingston and Martin 1952, pp. 22-23; Butler 1963, p. 17). During the next few years, the groundwork was laid for the production of neutron-deficient radioisotopes (Butler 1963, p. 17). From 1952 to 1961, however, the 86-in. cyclotron was used mainly in nuclear physics studies by the ORNL Electromagnetic Research Division (Howard 1954; Livingston 1958), and isotope production time was made available only when it did not interfere with the primary program (Butler 1963, p. 17). Following completion of the construction and testing of the Oak Ridge Isochronous Cyclotron at ORNL in 1961, the responsibility for operation and maintenance of the 86-in. cyclotron was shifted on December 1 from the ORNL Electronuclear Research Division to the ORNL Isotope Division (Livingston and Zucker 1962, p.35). The 86-in. cyclotron was used primarily by ORNL for production of medical radioisotopes until it was shut down permanently in 1983 (Kerr et al. 1992, pp. 44-47; Terry 2007).

Fast neutrons are the radiation of most concern in occupied areas near cyclotrons operated at proton beam energies between 15 MeV and 50 MeV (Shleien, Slack, and Birky 1998, p. 11-71, Table 11.4.11). The shielding for the 86-in. cyclotron consisted of 5-ft-thick concrete walls supporting a 5-ft-thick concrete ceiling (Livingston and Boch 1952, pp. 88-89). Two mazes were the only openings into the cyclotron vault (Livingston and Boch 1952, p. 90). Measurements made in early 1950 indicated the presence of an excessive stray neutron flux outside the maze entrance and the emergency maze exit to the 86-in. cyclotron (Struxness 1951a, pp. 23-24; Ballenger 1952, pp. 37-38; Livingston and Boch 1952, p. 91). A completely enclosed cyclotron vault with a minimum of apertures and movable doors with overlapping side panels became the standard shielding design for other early cyclotrons producing high-energy proton beams (Patterson and Thomas 1973, p. 380). A maximum permissible flux of 250 n/cm^2-s was established to limit the neutron exposure of personnel at the 86-in. cyclotron to less than 300 mrem per 40-hr workweek, which was the radiation protection guideline at that time (Wiley 2004; ORAUT 2007b, p. 14). The measured neutron flux in most occupied areas outside the shielding for the cyclotron was typically smaller than the maximum permissible flux of 250 n/cm^2-s by 1 to 2 orders of magnitude (Livingston and Boch 1952, p. 91; Ballenger 1952, p. 12).

The neutron energy spectra in work areas near the Oak Ridge 86-in. cyclotron, like the neutron energy spectra near other early proton accelerators, is not well known (IAEA 1988). To simulate the stray neutron fields near the 86-in. cyclotron, a Maxwellian thermal neutron spectrum was used at energies less than 0.125 eV, and an E^{-1} slowing down spectrum was used at energies from 0.125 eV to 20 MeV, where E is the neutron energy. Table 6-7 lists the dose fractions in the neutron energy groups used in the dose reconstruction for Y-12 workers; the results appear to be consistent with observations by others (IAEA 1988). For example, it has been found that most of the dose at early proton accelerators came from neutrons with energies between 0.1 and 10 MeV, and the neutron dose contributions from both thermal neutrons and fast neutrons with energies above 10 MeV were quite small. The results in Table 6-7 suggest that neutrons with energies less than 0.1 MeV and neutrons with energies from 14 to 20 MeV contribute only 12% and 8.7%, respectively, to the total neutron dose equivalent in the stray radiation fields at the 86-in. cyclotron. Therefore, combining the dose from these neutron energy groups with other nearby neutron energy groups is a reasonable and favorable to claimant simplification of the neutron dose reconstruction for Y-12 workers (see Section 6.3.4.2).

Table 6-7. Neutron dose fractions for the 86-inch cyclotron (Building 9201-1).

Neutron energy group	Dose fraction
<10 keV	0.076
10–100 keV	0.044
0.1–2 MeV	0.336
2–14 MeV	0.457
14–20 MeV	0.087
Favorable to Claimant dose fractions	
0.1–2 MeV	0.46
2–20 MeV	0.54

The fraction of the dose equivalent from neutrons below the effective threshold energy of 0.7 MeV for the NTA film was calculated to be approximately 25% (ORAUT 2006a, p. 12, Figure 4-3). Thus, an estimated correction factor to the recorded neutron dose for missed neutron dose due to the threshold energy of the NTA film should be approximately 1.3 (1 divided by 0.75) and the total correction factor for both the estimated missed neutron dose due to threshold energy and angular response of the NTA film should be approximately 1.7 (1.3 multiplied by 1.3).

6.3.4.2.2 **Encapsulated Neutron Sources**

Encapsulated sources that produced neutrons by spontaneous fission in ^{252}Cf or by alpha particle reactions with boron or beryllium provide convenient sources of neutrons for a variety of applications. Known locations of such sources producing neutrons by alpha particle reactions in boron or beryllium are Buildings 9201-2, 9203, 9204-3, 9737, and 9983 (ORAUT 2007a, p. 21). These sources were stored in accordance with accepted practices (Struxness 1951b, p. 7). They were surrounded by hydrogenous material, such as paraffin or water, which degraded the energetic fast neutron component by proton scattering, and the resulting slow (thermal) neutrons were then captured by various chemical elements in the scattering medium. For many of the larger sources, it was customary to enhance the probability of neutron capture in the scattering medium by addition of borax (Struxness 1951b, p. 7). As a consequence, once the handling procedures for an encapsulated neutron source were established, the services provided by the HP Department evolved into routine periodic checks. Instrumentation calibrations with Po-Be, Pu-Be, Am-Be, and Am-B sources were also a standardized operation in that the majority of calibrations were made at one location and usually by the same operator. Any unusual operations or transfers involving encapsulated neutron sources were done under the supervision of the HP Department (Struxness 1951b, p. 7).

One unusual situation at the Y-12 Plant was the use of five large Ra-Be neutron sources in the Assay Laboratories of the Development Facility (Building 9203) and Development Laboratories (Building 9205) (Struxness 1949a, p. 6; Struxness 1951b, p. 8). These sources were contained in four so-called "Dog Houses" in Room 8 of Building 9203 (Dahl 1946; Struxness 1949a, p. 6; Struxness 1951b, p. 8; Emerson 1952) and the so-called "Birdbath Moderator" in Building 9205 (Struxness 1949a, p. 6; Sanders 1956, p. 3). In 1955, NTA films were included in the film badge dosimeters in all Department 2159 personnel (Alloy Maintenance), who were assigned to the maintenance of assay monitors containing Ra-Be sources (West 1955). These NTA film badges were exchanged every other week or biweekly (West 1955). Very little is known about the construction of the Birdbath Moderator in Building 9205, but there is detailed information available on the Dog Houses in Building 9203 (Dahl 1946). The Ra-Be source was at the center of a cylinder that provided about 12 in. of water shielding in the radial direction. The cylinder had a 17-in. outer radius, a 17-in. outer height, and an outer wall thickness of 0.125 in. of iron (Dahl 1946). There were six ports in each cylinder so that the assay samples could be placed near the Ra-Be source at its center. The Dog Houses were placed near a wall in Room 6 of Building 9205 with a distance of about 80 in. between the centers of the water-filled metal cylinders (Dahl 1946). Available records indicate that Ra-Be neutron sources from these Assay Laboratories were placed in permanent storage in the late 1950s (McRee 1960, 1961; Roberts 1964, Attachment I). The permanent storage area for excess neutron and photon sources was the Radioactive Source Storage Facility in Building 9987 (TAPP 2003, pp. 133-134).

Information on the neutron spectrum in work areas near the Dog Houses and Birdbath Moderator is not available. Thus, the neutron fields from these sources are simulated by the use of Pacific Northwest Laboratory (PNL) measurements made at a distance of 24 in. from the surface of a water-filled drum containing both a ^{238}Pu-Be and a ^{252}Cf encapsulated neutron source (Soldat et al. 1990, p. 3.12; MMES 1992). The characterization of the neutron field consisted of measurements of the energy spectrum of the neutrons and the personal dose equivalent from both neutrons and gamma rays. The neutron-to-gamma dose ratio was approximately 3.7, and the neutron spectrum is provided in Table 6-8 using the following four neutron energy groups: <10 keV, 10 to 100 keV, 0.1 to 2 MeV, and 2 to 20 MeV (NIOSH 2007b, p. 19). The results in Table 6-8 suggest that neutrons with energies less than 0.1 MeV contribute only 2.1% to the total neutron dose equivalent. Therefore, combining the dose from these lower neutron energy groups with the nearby neutron energy group of 0.1 to 2.0 MeV provides a reasonable and favorable to claimant simplification of the dose reconstruction for Y-12 workers (see Section 6.3.4.2). The fraction of the dose equivalent from neutrons with energies less than the assumed effective threshold energy of 0.7 MeV for NTA film was also estimated to be approximately 20% based on the neutron spectrum data for the water-shielded ^{238}Pu-Be and ^{252}Cf sources (Soldat et al. 1990, p. 3.12; MMES 1992). Thus, the estimated correction to the recorded neutron dose for missed neutron dose due to the threshold energy of the NTA film should be approximately 1.3 (1 divided by 0.8) and the total correction factor for missed neutron dose due to both the threshold energy and angular response of the NTA film should be approximately 1.7 (1.3 multiplied by 1.3). It is important to note here that the ^{238}Pu-Be and ^{252}Cf sources were also shielded by 12 in. of water (Soldat et al. 1990, p. A.5). In addition, the neutron output of the Cf source was 2×10^7 n/sec, while that of the Pu-Be source was an order of magnitude larger or 2×10^8 n/sec (Soldat et al. 1990, p. 3.12). Thus, these measurement results should provide a realistic simulation for use in the dose reconstruction for Y-12 workers who were exposed to neutrons from the Ra-Be sources in Buildings 9203 and 9205. The unmoderated neutron spectra coming directly from all neutron sources using the (α,n) reaction in beryllium are quite similar; the energies of the source neutrons range from about 1 MeV to a maximum of about 12 MeV (Kiefer and Maushart 1972, pp. 70-71), and the average energy of the source neutrons is about 4 MeV (Nachtigall 1967; Kerr, Jones, and Hwang 1978).

A storage facility for encapsulated neutron and photon sources at Y-12 was located at the HP Calibration Laboratory in Building 9983 (Soldat et al. 1990, p. 3.6; MMES 1992). The neutron field in this storage area was simulated by the use of PNL measurements made at a distance of 18 in. from the storage safe in one corner of the Calibration Laboratory. At the time of the measurements, a number of neutron sources were being stored in the room in shielded containers, and the safe contained a ^{252}Cf source (Soldat el al. 1990, p. 3.22). The characterization of the neutron field near the safe included measurements of the energy spectrum of the neutrons and the personal dose equivalent from both neutrons and gamma rays (MMES 1992). The neutron-to-gamma dose ratio was approximately 8 and the neutron spectrum is summarized in Table 6-9 using the following four

Table 6-8. Neutron dose fractions for water-shielded Ra-Be neutron sources (Buildings 9203 and 9205).

Neutron energy group	Dose fraction
<10 keV	0.013
10–100 keV	0.008
0.1–2 MeV	0.264
2–20 MeV	0.715
Favorable to Claimant dose fractions	
0.1–2 MeV	0.29
2.0–20 MeV	0.71

energy groups: <10 keV, 10 to 100 keV, 0.1 to 2 MeV, and 2 to 20 MeV. The results in Table 6-9 suggest that neutrons with energies less than 0.1 MeV contribute only 6.2% to the total neutron dose equivalent. Therefore, combining the dose from these lower energy groups with the nearby neutron energy group of 0.1 to 2 MeV is a reasonable and favorable to claimant simplification of the dose reconstruction for Y-12 workers (see Section 6.3.4.2). The fraction of the dose equivalent from neutrons with energies less than the assumed effective threshold energy of 0.7 MeV for NTA film was estimated to be approximately 35% (ORAUT 2006a, p. 13, Figure 4-4). Thus, the estimated correction to the recorded neutron dose for missed neutron dose due to the threshold energy of the NTA film should be approximately 1.5 (1 divided by 0.65) and the total correction factor for missed neutron dose due to both the threshold energy and angular response of the NTA film should be approximately 2.0 (1.5 multiplied by 1.3).

Table 6-9. Neutron dose fractions for HP Calibration Laboratory (Building 9983).

Neutron energy group	Dose fraction
<10 keV	0.055
10–100 keV	0.007
0.1–2 MeV	0.509
2–20 MeV	0.429
Favorable to Claimant dose fractions	
0.1–2 MeV	0.57
2.0–20 MeV	0.43

Spontaneous fission neutrons from ^{252}Cf sources have recently been used for testing materials and recovery of highly enriched uranium (HEU) from waste products at the Nondestructive Assay Laboratory in Building 9720-5 (Hogue and Smith 1984, p. 6; Oxley 2001, p. 11). This laboratory contains instruments for the gamma scanning and neutron interrogation of containers of solid wastes, gamma analysis of solution samples, and measurements of solution density. The neutron spectrum from a ^{252}Cf source is characterized here using results of PNL measurements made at ORNL's Radiation Calibration Laboratory at a distance of 39 in. from a bare unshielded ^{252}Cf source (Soldat et al. 1990, pp. 3.18-3.21). The neutron-to-gamma dose ratio was approximately 25, and the measured dose fractions for the four neutron energy groups used in dose reconstruction are summarized in Table 6-10. The dose fractions for the lower (<10 keV) and intermediate (10 to 100 keV) neutron energy groups contributed less than 1% of the total dose. Therefore, combining these two energy groups with the fast neutron group having energies of 0.1 to 2 MeV is a reasonable and favorable to claimant simplification of the dose reconstruction for Y-12 workers (see Section 6.3.4.2). The Nondestructive Assay Laboratory became operational in 1980 (see Table 6-5) and the newer TLNDs used to measure neutron doses were carefully calibrated and corrected to account for the variation in response due to the angular or energy distributions of the workplace neutron fields (see Section 6.3.4.2).

Table 6-10. Neutron dose fractions for the Nondestructive Testing Laboratory (Building 9720-5).

Neutron energy group	Dose fraction
<10 keV	0.003
10–100 keV	0.004
0.1–2 MeV	0.224
2–20 MeV	0.769
Favorable to Claimant dose fractions	
0.1–2 MeV	0.23
2.0–20 MeV	0.77

6.3.4.2.3 **Chemical Operation Areas**

From about 1949 to 1964, the Y-12 Plant received cylinders of uranium hexafluoride (UF_6) containing HEU as feed material for the manufacture of nuclear weapons components (ORAUT 2007a, p. 15). After 1964, the majority of EU processed at Y-12 was RU materials from nuclear weapons stockpiles (Owings 1995, pp. 22-23). These operations were confined primarily to Buildings 9202, 9206, and 9212 (ORAUT 2007a, p. 15). The RU contained transuranic material (e.g., plutonium and ^{237}Np), fission products (e.g., ^{99}Tc), and reactor-generated products (e.g., ^{236}U) (BWXT Y-12 2000, p. 21). RU process streams involved the processing of other material forms including uranium metal, uranium alloys, and other uranium compounds such as UO_2, UO_3, and UF_4 (ORAUT 2007a, p. 15). The interaction of alpha particles from uranium with nuclei of fluorine, oxygen, and other low-atomic-weight atoms generates neutrons with energies of approximately 2 MeV (DOE 2004, Section 6, pp. 4-10). The magnitude of the neutron flux varies based on total activity of the uranium (a function of enrichment) and chemical compound in question (mixing of uranium with fluorine or oxygen). In the case of UF_6, the typically measured neutron dose equivalent rates for storage containers are as follows (DOE 2004, Section 2, p. 19):

Natural to 5% enrichment:	0.01 to 0.2 mrem/hr
Very high enrichment (97%+):	2 to 4 mrem/hr (contact) 1 to 2 mrem/hr (3 ft)

The potential for worker exposure to neutrons generated by (α,n) reactions in uranium compounds is not very high unless workers spend a significant amount of time near containers of uranium-fluoride or -oxide compounds, or near storage or processing areas for large quantities of those materials (DOE

2004, Section 2, p. 19). At very high ^{235}U enrichments, the neutron-to-gamma dose ratio can be as much as 2:1 and neutrons can be the limiting radiation source for WB exposure (DOE 2004, Section 6, p. 8). The neutron doses from lowly enriched ^{235}U compounds or from uranium metals are considerably lower than the gamma ray component and consequently are not limiting doses (DOE 2004, Section 6, p. 8).

The neutron fields in chemical processing areas of the Y-12 facility were simulated by the use of PNL measurements made in Building 9212 (Soldat et al. 1990, p. 3.6; MMES 1992). At the time of the measurements, this storage area in Building 9212 held storage containers of both enriched UF_4 and UO_3. Containers of these materials were placed on a rack of shelves and arranged in a matrix that was critically safe (Soldat et al. 1990, p. 3.6; MMES 1992). The measurements were made at a height of 39 in. (1 m) above the floor and 27 in. from the nearest container. The location of the measurements was also selected so that it was near only containers of UF_4 (Soldat et al. 1990, p. 3.6; MMES 1992). The characterization of the neutron field consisted of measurements of the energy spectrum of the neutrons and personal dose equivalent from both neutrons and gamma rays. The measured values for neutron dose equivalent and neutron-to-gamma dose ratio were 1.65 mrem/hr and 1, respectively, which are in reasonable agreement with the above-quoted values from the *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities* (DOE 2004, Section 2, p. 19). The results of the neutron spectrum are summarized in Table 6-11 and suggest that the neutrons with energies less than 0.1 MeV and energies greater than 2 MeV contribute only 3% to the total neutron dose equivalent. Therefore, assigning all of the neutrons to the 0.1- to 2-MeV energy group is a reasonable and favorable to claimant simplification of the dose reconstruction for Y-12 workers (see Section 6.3.4.2). The fraction of the dose equivalent from neutrons with energies less than the assumed 0.7-MeV threshold energy of the NTA film is estimated to be about 40% (ORAUT 2006a, p. 13, Figure 4-5). Thus, an estimated correction factor to the recorded dose for missed neutron dose due to the threshold energy of the NTA film should be approximately 1.7 (1 divided by 0.6) and the total correction factor for the missed neutron dose due to both the threshold energy and angular response of the NTA film should be approximately 2.2 (1.7 multiplied by 1.3).

Table 6-11. Neutron dose fractions for HEU storage areas (Buildings 9202, 9206, 9212).

Neutron energy group	Dose fraction
<10 keV	0.012
10–100 keV	0.003
0.1–2 MeV	0.970
2–20 MeV	0.015
Favorable to Claimant dose fractions	
0.1–2 MeV	1.00

The monitoring of workers in the UF_4 (fluid-bed) processing areas of the Y-12 Plant has been cited as an example of a group of workers who were involved in hands-on operations of neutron-producing materials and should have been monitored for fast- and thermal-neutron exposures during the film badge era (SCA 2005, p. 75). It was speculated that their exposures might have been due mainly to thermal and epithermal neutrons, and therefore, their neutron doses would not have been measured using NTA films (SCA 2005, p. 75). In the 1990s, however, workers in UF_4 (fluid-bed) processing areas were monitored for neutron exposure using four-element Harshaw TLNDs (SCA 2006, p. 8). These TLNDs were extremely sensitive to thermal and epithermal neutrons (Oxley 2001, p. 17) and had an MDL of the order of 20 mrem for exposures involving neutrons of all energies (Souleyrette 2003a, p. 6). No positive doses were observed for these individuals, so the neutron monitoring for this group of workers was discontinued in the mid- to late 1990s (SCA 2006, p. 8). Data on all Y-12 employees monitored for neutrons over the 12-year period following the introduction of the four-element Harshaw TLND in January 1990 is provided in Table 6-12 (Souleyrette 2003b). It should be noted from the data in Table 6-12 that the cumulative annual neutron doses and average annual neutron doses were quite small for the Y-12 workers who were monitored for exposure to neutron radiation during the 12-year period from 1990 to 2001.

Table 6-12. Number of neutron-monitored Y-12 workers, cumulative annual neutron dose, and average annual neutron dose for 12-year period following the introduction of the four-element Harshaw TLND in January 1990 (Souleyrette 2003b).

Year	Neutron monitored workers	Cumulative neutron dose (mrem)	Average 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

6.3.4.2.4 **Neutron-to-Gamma Dose Ratios**

A total of 25 Y-12 departments had workers with recorded quarterly neutron doses during the film badge period prior to 1980 (ORAUT 2005c, Appendix A, Table A-2, pp. 60-74). Only four of these departments had sufficient positive quarterly neutron dose data for calculating a neutron-to-gamma dose ratio for use during the film badge era. The lognormal predictive densities in Table 6-13 for workers in these four departments and the default lognormal predictive density in Table 6-13 for workers in other departments can be used to estimate neutron dose when only the gamma dose data for an individual worker is available.

If a worker's recorded gamma dose for a quarter was zero, a neutron-to-gamma dose ratio for the worker was calculated using an MDL of 30 mrem for the gamma dose so the derived predictive densities for the neutron-to-gamma dose ratios would be favorable to claimants (ORAUT 2005c, pp. 30-31). The lognormal prediction densities for the neutron-to-gamma ratios in Table 6-13 should

Table 6-13. Parameter estimates for a lognormal prediction density of neutron-to-gamma dose ratios (ORAUT 2005c, p. 38).[a]

Department	μ	σ	GM	GSD	ER
2108	–0.0629	1.2306	0.9391	3.4235	2.0025
2159	–0.7377	0.8984	0.4782	2.4557	0.7159
2160	–0.8151	1.4038	0.4426	4.0705	1.1855
2301	0.0430	1.4084	1.0439	4.0895	2.8146
Default value	–0.3421	1.6911	0.7103	5.4254	2.9678

a. GM = geometric mean; GSD = geometric standard deviation; ER = expected neutron-to-gamma dose ratio (mean) of the distribution.

provide more realistic estimates of missed neutron dose during the Y-12 film badge era than measurements at a fixed distance relative to a neutron source as discussed in Sections 6.3.4.2.2 and 6.3.4.2.3.

The default lognormal predictive density was derived based on the pooled quarterly neutron dose data for all Y-12 departments with the exception of Departments 2616 and 2617 (ORAUT 2005c, pp. 30-31). Departments 2616 and 2617 have recorded quarterly neutron data for 32 workers, but their neutron doses occurred primarily in 1954 and 1955 and include 25 quarterly doses that were multiples of 50 mrem. These positive neutron doses appear to consist of assigned biweekly neutron doses of 50 mrem substituted for below-MDL readings of 50 mrem (see Table 6-3).

The potential for large occupational exposures to neutron radiation was largely confined to research activities at the Oak Ridge 86-in. cyclotron and to workers in certain departments during the period from 1952 to 1962. During this period, however, only 375 positive quarterly film badge readings for exposure to neutron radiation were found among 143 workers. Therefore, only a small fraction of the Y-12 workers had a significant potential for exposure to neutron radiation, and those with a notable potential appear to have been monitored for neutron radiation. If a worker had no positive neutron doses before 1962, it is unlikely that the worker experienced exposure to neutron radiation; therefore, the best estimate of the individual's neutron dose would be zero (ORAUT 2008a, p. 6).

While the potential for neutron exposure was confined to a small area of the Y-12 site, it is important to include neutron doses in the exposure records of workers for whom neutrons were a relevant source of radiation. There are, however, large variations in neutron doses and neutron-to-gamma dose ratios for individual workers from one quarter to the next. Therefore, dose reconstructors should be wary of assigning neutron doses to other individuals in the same departments if they have no record of neutron exposure, unless such exposure is clearly indicated by an individual's work records.

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 6-5 (Y-12 Plant 1979, 1980b, 1981, 1982b, 1983, 1984, 1985, 1986, 1987, 1988b). The UCCND 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 close agreement, and no adjustments are deemed necessary to the recorded deep photon doses for most Y-12 workers.

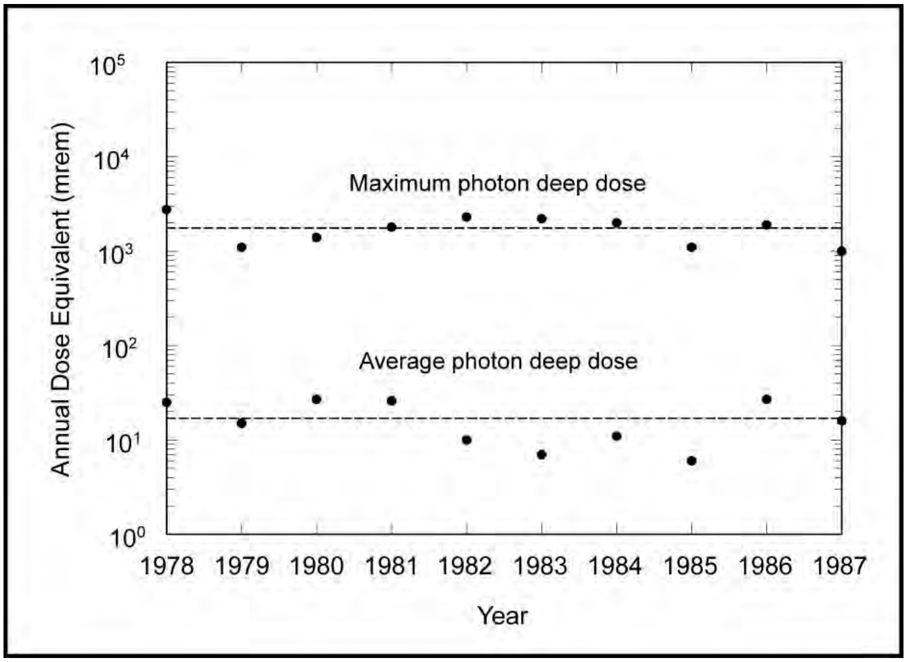

Figure 6-5. Maximum and average photon deep dose to Y-12 workers, 1978 to 1987 (Y-12 Plant 1979, 1980b, 1981, 1982b, 1983, 1984, 1985, 1986, 1987, 1988b).

There is one group of Y-12 workers for which an adjustment in the recorded photon dose is recommended (Section 6.3.4.1). These workers performed waist-level handling jobs in DU process and testing areas (see Table 6-5). Some of the Y-12 workers who handled small pieces of DU at waist level were scrap metal handlers in the H-2 Foundry (Henderson 1991, pp. 3-8) or testing operators in the Mechanical Properties Inspection Laboratories, Dimensional Inspection Group, and Radiographic Laboratories (Henderson 1991, pp. 3-8; Ashley et al. 1995, p. 6). Y-12 now instructs workers who perform such 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 factor of 1.34 that is shown in Table 6-14 is necessary to calculate an adjusted photon dose that is favorable to the claimant before 1991 (Henderson 1991, p. 42). From 1991 to the present, no correction is needed because the recorded photon dose is $H_p(10)$ equivalent.

Table 6-14. Adjustments to reported Y-12 deep photon dose for waist-level metal handling operators at the Y-12 facility (Henderson 1991, p. 42)

Parameter	Description
Period	Before January 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 Beta Dose Adjustments

An adjustment to the recorded beta dose is also recommended for workers who performed waist-level handling jobs in DU process and testing areas (see Section 6.3.4.1). As noted above, some of the Y-12 workers who handled small pieces of DU at waist level were scrap metal handlers in the H-2 Foundry (Henderson 1991, pp. 3-8) or testing operators in the Mechanical Properties Inspection Laboratories, Dimensional Inspection Group, and Radiographic Laboratories (Ashley et al. 1995, p. 6).

Y-12 now instructs workers who perform such operations to wear their beta/photon dosimeters at the waist, but many workers might have worn their dosimeters at the collar in the past. If it is assumed that the shallow dose is due primarily to beta particles from the DU, the correction factor of 1.71 as shown in Table 6-15 is necessary to calculate an adjusted shallow beta-particle dose that is favorable to the claimant before 1991 (Henderson 1991, p. 42). From 1991 to the present, no correction is needed because the recorded shallow beta-particle dose is $H_p(0.07)$ equivalent.

Table 6-15. Adjustments to reported Y-12 shallow beta-particle dose for waist-level metal handling operators at the Y-12 facility (Henderson 1991, p. 42).

Parameter	Description
Period	Before January 1, 1991
Dosimeters	All beta/photon dosimeters
Facilities	Depleted uranium process operations
Workers	Waist-level metal handling operators
Adjustment to recorded dose	Multiply reported shallow beta-particle dose by 1.71 to estimate $H_p(0.07)$

For dose reconstruction purposes, the beta doses to the body, face, and hands must be corrected for attenuation of beta particles by protective clothing items worn by a worker (ORAUT 2005d). The transmission factors of a variety of protective clothing items for the very energetic beta particles from DU are summarized in Table 6-16. These data were taken from reports on radiological protection for workers at uranium facilities (Rich et al. 1988, p. 7-30; DOE 2004, p. 6-23).

Table 6-16. Transmission factors of protective clothing for beta particles from DU (Rich et al. 1988, p. 7-30; DOE 2004, p. 6-23).

Clothing	Transmission factor
Face shield	0.41
Vinyl surgeon's gloves	0.95
Latex surgeon's gloves	0.87
Two pairs of gloves plus liner	0.60
Lead-loaded, 10-mil lead equivalent	0.77
Lead-loaded, 30-mil lead equivalent	0.13
Pylox (vinyl) gloves	0.62
Leather, medium weight	0.62
White cotton gloves	0.89
Two pairs of coveralls plus paper liner	0.80
Tyvek coveralls	0.98
Durafab paper laboratory coat	0.96
Laboratory coat (65% Dacron and 35% cotton)	0.91

6.4.3 Neutron Dose Adjustments

Two adjustments must be made for recorded neutron doses or missed neutron doses based on NTA film measurements made prior to 1980. The first adjustment uses NTA film correction factors to account for the dose from neutrons with energies less than the effective threshold of approximately 700 keV for the NTA films (see Section 6.3.2.2) and for the decreased response of the NTA film in workplace radiation fields from neutrons that are incident at widely varying angles on the NTA film contained within the film badge dosimeters (see Section 6.3.4.2). The second adjustment is required to account for the difference between the neutron quality factors, Q, from NCRP Report 38 (NCRP 1971, p. 16, Table 2) and the neutron weighting factors, w_R, from ICRP Publication 60 (ICRP 1991). The newer neutron weighting factors, w_R, from ICRP Publication 60 must be use to obtain neutron doses required in the POC calculations using NIOSH-IREP (NIOSH 2002). Only the latter adjustment is required for recorded neutron doses or missed neutron doses based on recorded neutron doses based on TLND measurements starting in 1980 (see Section 6.3.4.2).

6.4.3.1 NTA Film Correction Factors

NTA films were calibrated with neutrons from Po-Be or Am-Be sources that were perpendicularly incident on the face of the film badge and NTA film inside the film badge (Section 6.3.4.2). In most workplace environments, however, the neutrons are incident on a film badge and NTA film at widely variant angles. The angular dependence of NTA films has been studied experimentally by Kathren, Prevo, and Block (1965) and they suggest that multiplication of the tracks produced by perpendicularly incident neutrons from a calibration source should be multiplied by a factor of 0.75 to avoid the underestimation of the neutron dose to workers in most typical workplace environments. This is equivalent to multiplying the recorded neutron dose from a worker's NTA film by a factor of 1.3 (1 divided by 0.75). This factor should be applied to all recorded NTA film dosimeter results for Y-12 workers plus an additional factor to account for the missed neutron dose due to the energy threshold of the NTA film (see Figure 6-4). A favorable to claimant value of 0.7 MeV is assumed here for the energy threshold of the NTA film (see Section 6.3.4.2). Table 6-17 gives correction factors for NTA film measurements made in the workplace fields for the neutron sources of most interest. The factors for correcting NTA film badge readings for underestimation of neutron dose due to the angular response and threshold energy have a range from 1.7 to 2.2 and a mean value of 1.9.

Table 6-17. Correction factors for NTA film badge measurements of neutron dose during the period from 1950 to 1980.[a]

	Correction factor		
Neutron source and location	**Threshold energy**	**Angular response**	**Total**
86-in. cyclotron (Building 9201-1)	1.3	1.3	1.7
Water-shielded Ra-Be neutron sources (Buildings 9203 and 9205)	1.3	1.3	1.7
Shielded sources in HP Calibration Laboratory (Building 9983)	1.5	1.3	2.0
EU storage areas (Buildings 9202, 9206, and 9212)	1.7	1.3	2.2

a. Neutron film badge measurements should be multiplied by the total correction factor to adjust neutron doses for reduced response of NTA film in the workplace fields of the above sources.

6.4.3.2 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 discussed in NIOSH (2007b). 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 (ICRP 1973) and the National Council on Radiation Protection and Measurements (NCRP) Report 38 (NCRP 1971, p. 16, Table 2). These quality factors are point-wise data because they were calculated for broad-parallel beams of monoenergetic neutrons incident on a 30-cm-diameter cylinder of tissue that represents the torso (NCRP 1971, p. 47, Figure 2). The NCRP Report 38 quality factors are compared in Figure 6-6 with those used in the PNL measurements at Y-12 (Soldat et al. 1990; MMES 1992). To convert from NCRP Report 38 quality factors to ICRP Publication 60 radiation weighting factors (ICRP 1991, p. 82, Table A-2), 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 6-6 for each of the neutron energy groups used to define the radiation weighting factors in ICRP Publication 60 (ICRP 1991). A summary of the group-averaged NCRP Report 38 quality factors for dose reconstruction is provided in

Table 6-18. 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 (Warren et al. 1949, Morgan 1949; Struxness 1949c) and NCRP Report 17 (NCRP 1954, pp. 45-46; Taylor 1971, pp. 24-28).

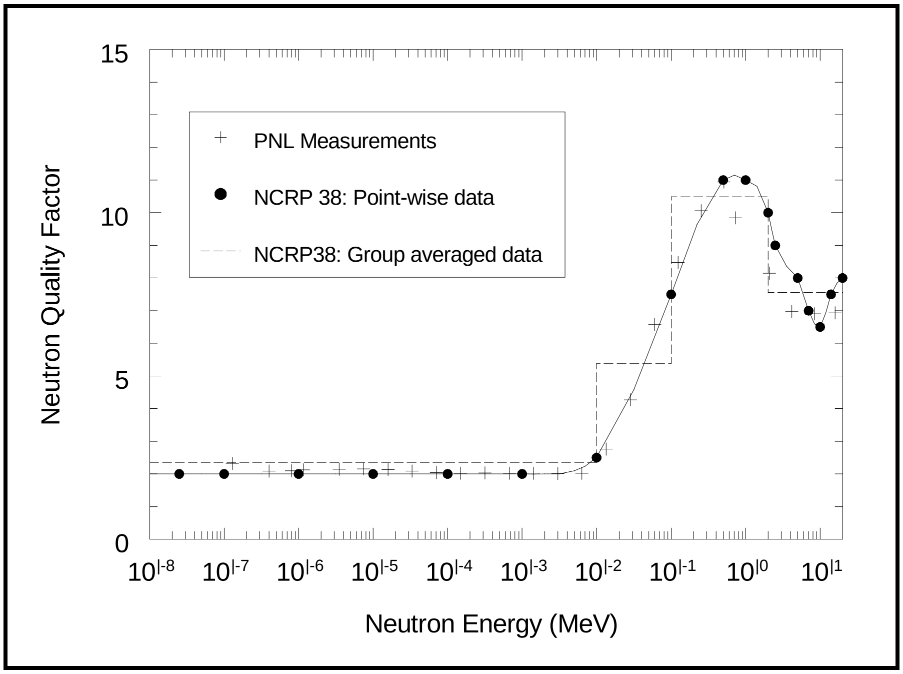

Figure 6-6. Comparison of the neutron quality factors of the PNL neutron spectrum measurements (Soldat et al. 1990; MMES 1992) with the neutron quality factors from NCRP Report 38 shown both as point-wise data (NCRP 1971, p. 16, Table 2) and grouped averaged data over the four neutron energy groups used in dose reconstruction for Y-12 workers (see Section 6.4.3.2).

Table 6-18. Neutron quality factor, Q, and neutron weighting factor, w_R.

<table>
<tr><th>Neutron energy</th><th>Historical dosimetry guidelines[a]</th><th>NCRP Report 38 group averaged neutron quality factor[b]</th><th>ICRP Publication 60 neutron weighting factor</th><th>Ratio[c]</th></tr>
<tr><td>Thermal</td><td rowspan="2">5</td><td rowspan="2">2.35</td><td rowspan="2">5</td><td rowspan="2">2.13</td></tr>
<tr><td>0.5 ev–10 keV</td></tr>
<tr><td>10 keV–100 keV</td><td rowspan="3">10</td><td>5.38</td><td>10</td><td>1.86</td></tr>
<tr><td>100 keV–2 MeV</td><td>10.49</td><td>20</td><td>1.91</td></tr>
<tr><td>2 MeV–20 MeV</td><td>7.56</td><td>10</td><td>1.32</td></tr>
</table>

a. First Tripartite Conference at Chalk River in 1949 (Warren et al. 1949, Morgan 1949, Struxness 1949c) and NCRP Report 17 (NCRP 1954, pp. 45-46; Taylor 1971, pp. 24-28).

b. See Figure 6-6.

c. Ratio of the ICRP Publication 60 weighting factor (ICRP 1991) to the NCRP Report 38 quality factor (NCRP 1971) for each energy group.

6.4.3.3 Neutron Dose Correction Factors

The average quality factor for the four energy groups that encompass the Y-12 neutron exposures is provided in Table 6-18. The neutron dose equivalent correction factor or so-called ICRP Publication

60 (ICRP 1991) correction factor for 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 Report 38 neutron quality factor (NCRP 1971) for that specific energy group (see Table 6-18), and

$w_R(E_n)$ is the ICRP Publication 60 neutron weighting factor (ICRP 1991) for that specific energy group (see Table 6-8).

The neutron dose distributions by energy for the various neutron exposure areas at Y-12 are summarized in Table 6-19. By multiplying the recorded neutron dose by the area-specific ICRP Publication 60 correction factors, the neutron dose equivalent is calculated as follows. Consider a Y-12 worker at the 86-in. cyclotron in Building 9201-1. Assume that this worker received a recorded annual neutron dose of 100 mrem in 1955. The worker's ICRP Publication 60 corrected neutron dose equivalent for 1955 is 88 mrem for neutrons with energies between 0.1 and 2 MeV, 72 mrem for neutrons with energies between 2 and 20 MeV, and 160 mrem for neutrons of all energies (ICRP 1991). The ICRP Publication 60 correction factors in Table 6-19 should be applied to both recorded neutron dose and missed neutron dose.

Table 6-19. Summary of neutron dose fractions and associated ICRP Publication 60 correction factors for Y-12 facilities.

Y-12 facilities	Building	Operations Begin	Operations End	Neutron energy	Neutron dose fraction	ICRP Publication 60 correction factor
86-Inch Cyclotron[a]	9201-1	1950	1961	0.1–2 MeV 2–20 MeV	0.46 0.54	0.88 0.72
Development Laboratories–Room 8[b]	9203	1949	1956	0.1–2 MeV 2–20 MeV	0.29 0.71	0.55 0.94
Assay Laboratories[b]	9205	1949	1956	0.1–2 MeV 2–20 MeV	0.29 0.71	0.55 0.94
Enriched Uranium Storage Areas	9212	1949	Ongoing	0.1–2 MeV	1.00	1.91
Nondestructive Analysis Laboratory	9720-5	1980	Ongoing	0.1–2 MeV 2–20 MeV	0.23 0.77	0.44 1.02
Calibration Laboratory	9983	1949	Ongoing	0.1–2 MeV 2–20 MeV	0.57 0.43	1.09 0.57

a. The operation of the 86-inch cyclotron was transferred to the Isotope Division of ORNL in December 1961 (see Section 6.3.4.2.1 and ORAUT 2005c, p. 23).

b. The Ra-Be sources used in these laboratories were placed in permanent storage in the late 1950s, probably 1956 according to neutron monitoring data for Department 2159 (see Section 6.3.4.2.2 and ORAUT 2005c, Appendix A, Table A.2, pp. 64-65)

6.5 MISSED DOSE

The potential for missed dose exists when workers are exposed to radiation at levels below the MDL of their personal dosimeters (NIOSH 2007). In the early years of radiation monitoring, when relatively high MDLs are combined with short monitoring durations, missed doses can be significant. The estimates of missed dose for a worker should be adjusted in the same manner as the worker's recorded doses (see Section 6.4).

6.5.1 Missed Photon and Beta Dose

In the assignment of missed gamma or beta dose for dose reconstruction, it is recommended that a value equal to one-half of the MDL be substituted for each dosimetry measurement (film badge, PIC, or TLD) that is recorded as zero or if it is less than MDL divided by 2 (NIOSH 2007, p. 16). Dosimetry readings for gamma or beta doses greater than or equal to the MDL divided by 2 are to be used as recorded (NIOSH 2007b, p. 16). When the number of zeros contributing to a recorded dose cannot be determined, the missed dose becomes more complicated (NIOSH 2007, p. 16). When only the quarterly or annual dose is known, the number of zero doses should be estimated based on the recorded dose and the quarterly or annual dose limits as discussed in the external dose reconstruction implementations issued by NIOSH (2007, p. 16). The historical radiation dose limits in effect at the Y-12 facility are provided in the following references (Wiley 2004; ORAUT 2007b, p. 14, Table 3-4).

6.5.2 Missed Neutron Dose

The missed neutron dose is divided into two periods in the following discussion. The first period is prior to 1980 when NTA film dosimeters were used, and the second period starts in 1980 when TLNDs were used (see Table 6-1). The MDLs for these neutron dosimeters are summarized in Table 6-3. The NTA film dosimeters had an energy threshold for detection of neutrons that is usually quoted as varying from 500 to 700 keV (ORAUT 2006a). A favorable to claimant value of 700 keV was assumed (see Section 6.3.4.2). There was, of course, no threshold energy for detection of neutrons by TLNDs, as illustrated in Figure 6-4 (see Section 6.3.2.2).

Prior to 1980

Lognormal predictive densities of neutron-to-gamma ratios for estimating missed neutron dose during the NTA film badge dosimetry period are provided in Table 6-13. These predictive densities should provide more realistic estimates of missed neutron dose than neutron-to-gamma dose ratios based on measurements made at a fixed distance from a neutron source (see Sections 6.3.4.2.2 and 6.3.4.2.3). Only four Y-12 departments had sufficient positive neutron doses for use in these calculations of neutron-to-gamma dose ratios. However, a default lognormal predictive density for the neutron-to-gamma dose ratio is recommended for application to workers in other Y-12 departments (see Table 6-13).

The potential for occupational exposure to neutrons was largely confined to the research activities at the Oak Ridge 86-in. cyclotron and to workers in certain departments during 1952 to 1962. During this period, only 375 positive neutron doses were found among 143 workers (ORAUT 2005c, Appendix A, Table A1, pp. 46-59). Thus, only a small fraction of the Y-12 workers had a significant potential for exposure to neutron radiation, and those with a notable potential appear to have been monitored for neutron exposure. If a worker had no positive neutron doses before 1962, it is unlikely that the worker experienced any neutron exposure; therefore, the best estimated neutron dose would be zero.

While the potential for neutron exposure was confined to small areas at the Y-12 site, it is important to include neutron doses in the exposure records of workers for whom neutrons were a relevant source of radiation. There are, however, large variations in neutron doses and neutron-to-gamma dose ratios for individual workers from one quarter to the next (ORAUT 2005c, Appendix A, Table A1, pp. 46-59). Therefore, dose reconstructors should be wary of assigning missed neutron dose to other individuals in the same departments if they have no record of neutron exposure, unless such exposure is clearly indicated by an individual's work records (ORAUT 2008a).

Starting in 1980

Starting in 1980, the neutron dose has been measured using an albedo-type TLND (see Table 6-1). The characteristics of these dosimeters are well documented (Oxley 2001) and estimated missed dose (or MDL/2) is 10 mrem per calendar quarter of 13 weeks or 40 mrem per calendar year of 52 weeks (see Table 6-3).

6.6 INCOMPLETE, MISSING, OR NO MONITORING DATA

Incomplete, missing, or no monitoring data usually occurs between two periods of monitoring data or at the beginning or end of a monitoring period (NIOSH 2007, p. 24). When personal monitoring data is missing between two periods of monitoring, interpolation between the two time periods may be reasonable (NIOSH 2007, p. 25). When the incomplete data is either before or after a monitoring period, extrapolation may be reasonable; however, caution should be used to properly account for any trends that may exist (NIOSH 2007, p. 25). When no personal monitoring data are available, the estimated doses for workers should be based on (1) coworker data, (2) radiation survey data, or (3) source term data (NIOSH 2007, p. 25).

6.6.1 Missing Monitoring Data

If the worker has sufficient monitoring data prior to and after the missing data, the dose can be usually be estimated by interpolation using a simple average between two nearby monitoring periods (Watson et al. 1994; Watkins et al. 1994, pp. 22-26). The interpolation is considered reasonable providing the work practices, radiological protection measures, engineering controls, and administration practices did not change (NIOSH 2007, p. 25). Interpolation should only be used if there is no indication from the claimant or site radiological records that a radiological incident resulting in a higher exposure occurred during the time period of the missing data (NIOSH 2007, p. 25).

Some workers are concerned that their dose records are not accurate because they were encouraged or instructed by a supervisor not to wear their badges (dosimeters), or they were not given badges while doing jobs that could have resulted in exposures sufficient to exceed an administrative or regulatory dose limit (ORAUT 2008b). If this concern is expressed by a claimant verbally in an interview or in written correspondence, the dose reconstructor should try to determine if this could have happened by examining the dose records and considering workplace conditions, potential source terms, and incident reports. In cases in which the dose reconstructor believes this could have happened, it might be necessary to perform additional research and to modify the dose reconstruction (ORAUT 2008b).

6.6.2 Incomplete Monitoring Data

The full scale personnel monitoring program for exposure to external radiation was first initiated at the Y-12 Plant in 1950 (McLendon 1960); thus, few monitoring data for external radiation exposure are available prior to this date (ORAUT 2005a). In 1950, all personnel working with depleted uranium, discrete gamma or beta sources, X-ray machines, or fission contaminated materials were asked to wear a film badge dosimeter (McLendon 1960). In order to reconstruct a worker's gamma and beta dose to these sources during earlier time periods, extrapolation using dose data from adjacent (nearby) time periods may be necessary as discussed in ORAUT-RPRT-0032 (ORAUT 2005b) and ORAUT-OTIB-0046 (ORAUT 2007b). Caution must be exercised, however, to account for trends in the exposure data resulting from work, radiological administration, or engineering controls that might have changed the exposure pattern with time (NIOSH 2007, p. 25).

6.6.3 No Monitoring Data

Only a subset of workers at the Y-12 Plant was monitored at the Y-12 Plant during very early periods of time to demonstrate compliance with radiation dose limits as shown in Figure 6-1 (see Section 6.1.2). Thus, coworker data have been developed for use in estimating photon and beta doses to workers with no monitoring data during the use of film badge dosimeters at the Y-12 Plant (ORAUT 2009a).

6.6.4 Unmonitored Doses to Special Groups

Unmonitored doses have been estimated for several special groups of individuals, and dose reconstructors should refer to the following references for information on the estimated doses for the following special groups: (1) unmonitored individuals near the 1958 criticality accident in Building 9212 (ORAUT 2006d); (2) unmonitored construction trade workers at the Y-12 facility (ORAUT 2007c, pp. 17-18; 2009a, pp. 16-17); and (3) unmonitored nuclear weapon assemblers in Building 9204-2 during the period from 1958 to 1990 (ORAUT 2009b).

The weapon assemblers wore airtight ventilated suits similar to those worn by astronauts to prevent body moisture from damaging components during assembly of the nuclear weapons (ORAUT 2009b). Film badge dosimeters and TLDs were not worn during these assembly operations because they would be damaged by the profuse sweating by personnel while suited. A health physicist stayed in the room with the weapon assemblers and made constant surveys of their radiation exposure rates. Measurements of the largest exposure rates to assemblers were used in estimating their unmonitored doses (Kerr 2007b).

Attenuation factors for the beta dose to the skin of the hands and face are not usually applied unless the use of gloves and face shields are provided in the records for exposed individuals. However, the use of gloves and hard-plastic head pieces were an integral part of the airtight ventilated suits worn by the weapon assemblers while working in the dry room at the Y-12 facility.

6.7 UNCERTAINTY IN PHOTON, BETA PARTICLE, 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, p. 18; Parrish 1979, p. 48; West 1993a, p. 4; Souleyrette 2003a, p. 8; Wilson et al. 1990, p. 160) and from 50 to 100 mrem for neutrons (Morgan 1961, p. 18; Parrish 1979, p. 48; Wilson et al. 1990, p. 160). These are not the expected uncertainties at larger photon, beta, 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, p. 18). If the exposure involved photons with energies less than several hundred keV, the uncertainty was at least twice that for the more energetic photons (Morgan, p. 18). Therefore, the standard error in the recorded film badge doses from photons of any energy is estimated 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, p. 18). The situation for neutrons was not as favorable as that for photons or beta particles. With NTA films, the estimated standard error was much larger and varied significantly with the energy of the neutrons (see Figure 6-4). Therefore, the standard error for a recorded neutron dose reading from an NTA film is estimated to be as much as ±50% (see Table 6-17) . For the TLDs and TLNDs used at Y-12 after 1980, the standard errors for a recorded dose reading of 100 mrem or more have been estimated to be approximately ±15% for photons, beta particles, and neutrons based on data summarized in Oxley (2007b). The standard errors for TLD and TLND dose measurements less than 100 mrem would be somewhat larger (Oxley 2007b, p.2; Souleyrette 2003a, pp. 5-6).

6.8 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 2007b). 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 radiation effectiveness factors of the intermediate-energy photons, these DCFs should be used to convert recorded film badge doses to organ dose. When converting recorded neutron, photon, and beta-particle doses to organ doses, the 100% A-P exposure geometry is used (NISOH 2007b, p.33).

6.9 ATTRIBUTIONS AND ANNOTATIONS

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

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ATTACHMENT A
OCCUPATIONAL EXTERNAL DOSE FOR UNMONITORED CALUTRON WORKERS

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A.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,b).

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 A-1 were used to bound the operators' external radiation exposure.

Table A-1. Cubicle X-ray radiation measurements (Smith 1944a).

Film position during test	Cubicle number	Exposure duration (hr)	Exposure (R/8-hr 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 l.25

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

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).

A.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 /8-hr 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.

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

Occupation	Bounded annual WB 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)