ORAU Team **NIOSH Dose Reconstruction Project** Technical Basis Document for the Oak Ridge National Laboratory – Occupational External Dose	Document Number: ORAUT-TKBS-0012-6 Effective Date: 08/11/2004 Revision No.: 00 Controlled Copy No.: Page 1 of 82
Subject Experts: Robert E. Burns, Jr.; Joe Mohrbacher Document Owner Approval: Signature on File Date: 08/04/2004 Robert E. Burns, Jr., TBD Team Leader Approval: Signature on File Date: 08/10/2004 Judson L. Kenoyer, Task 3 Manager Concurrence: Signature on File Date: 08/04/2004 Richard E. Toohey, Project Director Approval: Signature on File Date: 08/11/2004 James W. Neton, OCAS Health Science Administrator	Supersedes: None

TABLE OF CONTENTS

Section | Page

LIST OF TABLES

LIST OF FIGURES

RECORD OF ISSUE/REVISION

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	02/08/2004	00-A	New technical basis document for the Oak Ridge National Laboratory – occupational external exposures. Initiated by Robert E. Burns, Jr.
Draft	03/30/2004	00-B	Revises draft in response to input from informal, internal review and additional information obtained from ORNL. Initiated by Robert E. Burns, Jr.
Draft	06/14/2004	00-C	Incorporates internal and NIOSH review comments. Initiated by Robert E. Burns, Jr.
Draft	07/21/2004	00-D	Revises the non-critical comments per NIOSH request. Initiated by Robert E. Burns, Jr.
08/11/2004	08/11/2004	00	First approved issue. Initiated by Robert E. Burns, Jr.

ACRONYMS AND ABBREVIATIONS

γ	gamma (photon)
AEC	U.S. Atomic Energy Commission
CEDS	Centralized External Dosimetry System
cm	centimeter
D-T	deuterium-tritium
DC	ORNL designation for critical organ dose, analogous to deep or penetrating dose
DL	ORNL designation for dose to the lens of the eye at a dose depth of 300 mg/cm^2
DM	ORNL designation for moderately penetrating dose at a dose depth of 130 mg/cm^2
DOE	U.S. Department of Energy
DOELAP	Department of Energy Laboratory Accreditation Program
DP	ORNL designation for penetrating dose at a dose depth greater than or equal to 1,000 mg/cm^2
DS	ORNL designation for skin or superficial (nonpenetrating) dose at a dose depth of 7 mg/cm^2
ft	foot
g	gram
GSD	geometric standard deviation
HFIR	High Flux Isotope Reactor
HPRR	Health Physics Research Reactor
Hp(d)	personal dose equivalent at depth *d* in tissue (*d* in centimeters)
hr	hour
ICRP	International Committee on Radiological Protection
ICRU	International Committee on Radiation Units and Measurements
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kilovolt-electron, 1,000 electron volts
LEN	ORNL designation for dose to the lens of the eye (see DL)
LOD	limit of detection
m	meter
MeV	megavolt-electron, 1 million electron volts
mg	milligram
mm	millimeter
MOP	ORNL designation for moderately penetrating dose (see DM)
mR	milliroentgen
mrep	millirep
MW	megawatt
^{1}n	neutron
NBS	National Bureau of Standards

NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NTA	nuclear track emulsion, type A
ORELA	Oak Ridge Electron Linear Accelerator
ORIC	Oak Ridge Isochronous Cyclotron
ORNL	Oak Ridge National Laboratory
OW	designation for film dosimeter response behind the open window with no filter present, just the inherent filtration of the dosimeter materials (see W)
PEN	ORNL designation for penetrating dose (see DP)
PME	probable maximum exposure
PNL	Pacific Northwest Laboratory
PTR	probable total reading
R	roentgen
RADCAL	Radiation Calibration Laboratory
RASCAL	Radiation Standards and Calibration Laboratory
REDC	Radiochemical Engineering Development Center
S	designation for penetrating dose behind 1-mm-thick cadmium filter
SWSA	solid waste storage area
TDF	Transuranium Decontamination Facility
TLD	thermoluminescent dosimeter
TLD-100	lithium-fluoride TLD chip with lithium at its natural enrichment
TLD-600	lithium-fluoride TLD chip with lithium enriched in ^{6}Li (to enhance neutron response)
TLD-700	lithium-fluoride TLD chip with lithium enriched in ^{7}Li (to minimize neutron response)
TRU	transuranic
TSR	Tower Shielding Reactor
UCC-ND	Union Carbide Corporation Nuclear Division
U.S.C.	United States Code)
W	alternate designation for film dosimeter response behind the open window (see OW).
W	watt
WEAF	Waste Examination and Assay Facility
WTA	Waste Transfer Area

6.1 INTRODUCTION

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

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

The Oak Ridge National Laboratory (ORNL) began operations in early 1943. The startup of the Graphite Reactor in November 1943 introduced the potential for personnel exposures to ionizing radiation. Operations with radioactive materials and radiation-generating devices at ORNL expanded over subsequent years with the Laboratory functioning in the role of technology development and pilot testing of new processes. These activities introduced increasing numbers of radiation sources and increasing potential for personnel exposure.

In early 1943, ORNL instituted external dose measurement techniques using paired pocket ionization chambers. At that time, the Laboratory used photographic film dosimeters experimentally, which became the primary dosimeter of record at ORNL on June 25, 1944. Film dosimeters, augmented with nuclear track emulsion, type A (NTA) film for neutron monitoring, were the principal means of personnel monitoring at ORNL until the advent of thermoluminescent dosimeter (TLD) technology in the mid-1970s. This technical basis document summarizes ORNL personnel monitoring practices for external dosimetry from late 1943 to the present.

6.2 BASIS OF COMPARISON

Since the initiation of the Manhattan Engineer District project in the early 1940s, various radiation dose concepts and quantities have been used to measure and record occupational dose. The basis of comparison for reconstruction of dose is the *personal dose equivalent*, $H_p(d)$, 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* is 0.07 mm and the dose equivalent is noted as *Hp(0.07)*. For penetrating radiation of significance to *whole-body dose*, *d* is 10 mm and the dose equivalent is noted as *Hp(10)*. Both *Hp(0.07)* and *Hp(10)* are the radiation quantities recommended by the International Commission on Radiation Units and Measurements (ICRU) for use as the operational quantities to be recorded for radiological protection purposes (ICRU 1993). In addition, *Hp(0.07)* and *Hp(10)* are the radiation quantities used since the 1980s in the U.S. Department of Energy (DOE) Laboratory Accreditation Program (DOELAP) to accredit personnel dosimetry systems (DOE 1986). The International Agency for Research on Cancer selected *Hp(10)* as the quantity to assess error in historical recorded whole-body dose for workers in Agency 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 based historically on different dose quantities such as first collision dose, multiple collision dose, dose equivalent index, and so forth. The numerical difference in using these dose quantities compared to the *Hp(10)* dose used in current DOELAP performance testing could be evaluated by using the relative values of the dose conversion factors for the respective dose quantities in conjunction with characteristics of the respective ORNL neutron dosimeter response characteristics and workplace radiation fields.

6.3 DOSE RECONSTRUCTION PARAMETERS

Examinations of the beta, photon (X-ray and gamma ray), and neutron radiation types, energies and exposure geometries, and their respective dosimeter response characteristics are crucial for the assessment of bias and uncertainty in the original recorded dose in relation to the radiation quantity *Hp(10)*. Bias and uncertainty for current dosimetry systems are typically well documented for *Hp(10)*. The performance of current dosimeters is often comparable to performance characteristics of historic dosimetry systems in the same, or highly similar, facilities or workplaces. In addition, the application of current performance testing techniques to earlier dosimetry systems can result in a consistent evaluation of all dosimetry systems. Dosimeter response characteristics for radiation types and energies in the workplace are crucial to the overall analysis of error in recorded dose.

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

- Administrative practices adopted by facilities to calculate and record personal dose based on technical, administrative, and statutory compliance considerations

- The use of dosimetry technology that includes the physical capabilities of the dosimetry system, such as the response to radiation type and energy, especially in mixed radiation fields

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

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

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

6.3.1 Historical Administrative Practices

Beginning in October 1943, ORNL issued paired pocket dosimeters to workers assigned to duties in the restricted areas associated with Graphite Reactor operations. Pocket dosimeters were used in pairs because of reliability issues associated with early designs. ORNL issued two dosimeters so there was a backup if one of them failed, went off the scale, or was otherwise unusable. An individual received these dual dosimeters if restricted area entries were likely more than three times per week. If entry into restricted areas was likely to be less than three times per week or on a random basis, workers received dosimeters for each entry. Workers received dosimeters at the beginning of the shift at the entry portal and deposited them at the same portal at the end of the shift.

The dosimeters were read, data were recorded, and the dosimeters were recharged before the beginning of the next shift. Both readings were recorded, but the lower was identified as primary and maintained as the dose of record. Each dosimeter was matched and assigned to an individual and replaced at the portal for the next day’s operation. ORNL investigated irregularities with the pocket dosimeters or their readings and made administrative notations on the dose record if a justification for the irregularity was evident. Follow-up was initiated if the dosimeter results exceeded 30 mR for the day. In addition, notations were made if a dosimeter was lost, damaged, returned late, or not worn.

In 1944, exposure controls became more complicated due to the introduction of the bismuth-phosphate process for plutonium separation and other operations involving short-decay irradiated fuel slugs from the Graphite Reactor. The accumulation of fresh fission products and process upsets started to produce significant dose rates in the new process areas. As a result, steady improvements in radiation exposure control and technological improvements in dosimetry systems were required.

ORNL experimentally used a beta-gamma film dosimeter based on the design developed at the Metallurgical Laboratory at the University of Chicago during the first half of 1944 (Mitchell et al. 1993). On June 25, 1944, ORNL introduced this dosimeter as the primary beta-gamma dosimeter of record. It used a DuPont Type 552 film packet consisting of two emulsions, one sensitive (Type 502) and the other insensitive (Type 510). The insensitive film was included in case there was a large exposure beyond the range of the sensitive film. The film dosimeter consisted of two windows (elements): One open (i.e., no filtration other than that of the film packet), and one under a 1-mm-thick cadmium filter with a density thickness of approximately 900 mg/cm^2. This two-element film dosimeter was used only for beta-gamma monitoring. It was assigned initially only to employees required to work in restricted areas more than 3 days per week.

Neutron personnel monitoring using NTA film was formally introduced at ORNL around 1949. However, NTA film was apparently used on a limited basis to supplement field measurements as early as February 1945 (Wirth, Morgan, and Curtis 1945) and, beginning on December 28, 1947, the cards used to record personnel exposures included a column for neutron dose. Neutron dose data were recorded for both the open window (OW) of the dosimeter and for the region behind the cadmium shield (S). The two readings recorded as a fraction of tolerance values for fast and thermal neutron exposures. The neutron exposures and tolerance were expressed in terms of number of tracks per field (of view) or, more specifically, number of tracks in a certain number of fields. For example, fast and thermal neutron tolerance levels were both 20 tracks per 12 fields in early 1949. The previous value (from late 1947 through the eleventh week of 1949) was 16, although it is not clear if this value was also based on 12 fields. It appears that the fast and thermal neutron tolerance level increased to 22 tracks as of the 38th week of 1949. Once it became available, NTA film was used in all assigned employee badges and exchanged on a weekly basis. However, the film was not processed unless the health physicist recommended it. The practice of recording beta-gamma and neutron film badge readings on the same data card was discontinued as of December 30, 1951, although separate neutron exposure records were in use before then.

The film dosimeters were exchanged and processed on a weekly basis, until a quarterly exchange was introduced in the third quarter of 1956. The use of paired pocket dosimeters continued for exposure control purposes. Film dosimeters were initially stored at the entrance portal to the restricted area for employees to pick up and return each day. There is evidence that strong administrative controls to ensure that workers wore dosimetry as prescribed were not in effect, and that wearing dosimetry was a matter of the honor system. There are no indications that film dosimeters were used at the portal storage locations for monitoring for or subtracting ambient exposure. Employees were not required to wear film dosimeters while working outside the restricted area. Those entering the restricted area on an occasional basis (less than three times per week) were considered visitors and were assigned *limited badges*. The limited badges were processed at the end of the assignment or weekly, whichever was shorter. Nonradiation workers who did not enter the restricted area did not receive film dosimetry. These included, but were not limited to, visitors, clerical personnel, accountants, and construction personnel.

As of November 20, 1951, the wearing of a combination security badge and film dosimeter, called a *photobadge*, was required of all regular workers on the ORNL site regardless of work area. This marked a transition period between the practice of storing personnel film dosimeters at the entry portals and the adoption of *take-home badges*. It appears the transition to the take-home photobadge was completed by late September 1953. It is important to note that take-home badges were issued mainly to ORNL employees and other workers with routine access to the site (i.e., that had a photobadge issued). Other personnel, such as temporary employees, some construction workers, services personnel, or visitors, did not receive take-home dosimeters. The weekly exchange of photobadge dosimeters continued until the third quarter of 1956, when the frequency was changed to quarterly.

A new four-element film dosimeter was introduced in late September 1953. This dosimeter had four shields (plastic, copper, lead, and cadmium). The intent of the design was to enable depth-dose measurements in accordance with National Bureau of Standards (NBS) Handbook 59 (NBS 1954); however, “no serious attempt was made to utilize the full capability of the multi-filter badge until the beginning of the second half of 1956” (Hart 1966). Beginning then, four depth-dose quantities were considered: A skin dose, a moderately penetrating dose, a lens-of-the-eye dose, and the penetrating dose. However, no routine determination of skin dose was made because the element was behind an effective density-thickness of approximately 80 mg/cm^2 (Hart 1966). ORNL did not report skin (or superficial) dose until the second half of 1961.

A film dosimeter designated ORNL Model I was introduced in the late 1950s, and ORNL Model II was introduced in July 1960. These dosimeters employed filters consisting of an OW, plastic, aluminum, and cadmium. Gold foils were included in the cadmium element for use in the event of an accident-level neutron exposure (Hart 1966). The ORNL Model II film dosimeter used a DuPont Type 544 film packet, which consisted of Type 555 sensitive emulsion and Type 834 insensitive emulsion. ORNL switched from the DuPont Type 552 film packet to the Type 544 sometime in 1959.

Beginning in 1961, only two depth dose assignments were made from personnel dosimetry data: The skin (or superficial) dose (designated DS) and the critical organ dose (DC). Techniques were developed to allow assignment of a skin dose based on extrapolation of the response of the film element under the OW of the dosimeter (Hart 1966).

As of 1968, different classes of personnel dosimetry were assigned to workers based on their duties and access requirements. Three classes of personnel dosimetry components were used: Classes A, B, and C. The class assigned to a given worker depended on the assigned photobadge.

Class A dosimetry consisted of the ORNL Model II four-element film dosimeter described above. This photobadge was assigned to workers with routine access to the site, including ORNL employees, loaned persons, consultants, and others who required frequent access. The combination badges for ORNL employees and loaned personnel were take-home. The Model II film dosimeter was also used in so-called *gate badges*, which were left at the entry portals (i.e., not taken home). Exchange frequencies for both the beta-gamma film packet and the NTA packet were quarterly for radiation workers. The film packets were renewed annually in the dosimeters issued to personnel for whom monitoring was not required (i.e., nonradiation workers) (Gupton 1968).

Class B dosimetry consisted of a DuPont Type 544 film packet, an NTA film packet, and components for accident dosimetry. The beta-gamma film packet was renewed monthly. The NTA film was renewed annually, or any time it was removed for processing. Class B dosimetry was issued to persons with casual access to ORNL facilities that included radiation zones, including U.S. Atomic Energy Commission (AEC) employees and other visitors with access to radiation zones. In addition, Class B dosimetry was used in temporary ORNL employee badges issued to employees as replacements for their photobadges (Gupton 1968).

Class C dosimetry consisted of a beta-gamma film packet, an NTA film packet, and accident dosimetry components. The beta-gamma film packet was renewed quarterly and the NTA film was renewed annually. Persons issued Class C dosimetry were not allowed to enter radiation zones. Class C dosimetry was issued to some construction workers, services visitors, tour groups, and other visitors for whom entry to radiation zones was prohibited. It is important to note that not all construction workers were issued Class C dosimetry. ORNL maintained a construction photobadge for workers who required frequent access to the site. This badge was the same as the Class A dosimetry issued to ORNL employees (Gupton 1968).

Gupton (1968) stated that as of 1968 the beta-gamma film packet used in the Class C dosimeter was an Eastman-Kodak Type 2. However, Gupton (1969) stated it was a DuPont Type 544, the same as the Class A and Class B dosimeters. ORNL switched from the DuPont Type 544 film packet to the Eastman Kodak Type 2 at some point, but the date is not clear. However, the Type 544 packet appears to have been in use through at least 1969. The Eastman Kodak Type 2 packet had sensitive and insensitive emulsions bonded to opposite sides of the film base. In a 1979 interview, Gupton indicated that the change from the DuPont Type 544 film packet to the Eastman Kodak Type 2 packet resulted in lowering the dosimeter limit of detection (LOD) from 30 mR to 20 mR (Beck, Stansbury, and Watson 1979). The Type 2 film packet was in use in 1975 as a backup to the newly adopted TLDs, but because the date when the Type 2 packet replaced the Type 544 is unknown, the LOD for all ORNL film badges has been assumed to be 30 mR.

Film dosimeter technology was used for beta-gamma and neutron personnel monitoring at ORNL until TLD technology replaced it in 1975. In fact, review of the claims files shows the use of TLDs actually began for at least some radiation workers at ORNL as early as 1974 to assign deep dose. The practice of combining the security badge and personnel dosimeter continued, and workers received a specific type of TLD depending on their duties. At first, each TLD contained a Type 2 film packet as a backup. NTA film was used as an integral part of the ORNL neutron dosimeter until elimination of routine use sometime in the mid-1980s.

The practice of assigning personnel dosimetry based on different classes (A, B, and C) of security badges apparently continued following the initial widespread use of TLDs at ORNL in January 1975. At first, TLDs were issued only to those with Class A photobadges. Individuals with Class B and C badges were issued film dosimetry as described above (Gupton 1974). Workers with Class A badges received different models of TLDs depending on their potential for radiation exposure and the types of radiation to which they were likely to be exposed. For instance, neutron-sensitive dosimeters were assigned only to individuals whose work involved potential neutron exposures.

As of January 1975, workers with Class A photobadges were assigned one of four different TLDs. The four versions were distinguished with color-coded labels: Red, yellow, green, and none. Red TLDs went to radiation workers with the potential for neutron exposure. Only 40 to 50 red dosimeters were available (Gupton 1974). This was a four-element dosimeter with TLD-600 and TLD-700 elements (two of each) and NTA film. Yellow dosimeters, which were assigned to radiation workers for which neutron exposure was not anticipated, contained one TLD-100 and one TLD-700. Approximately 130 of these dosimeters were available (Gupton 1974). Uncolored dosimeters were the same as the yellow version, but the two TLD chips were under different filters. In 1975 or 1976, the use of thin (0.015-in.) TLD-700 chips began in yellow and red dosimeters (or Type 2 and 3 dosimeters described below). It is not clear if the thin chips were used from the outset, but the uncolored (or Type 1) badge appears always to have used a thicker 0.035-in. TLD-700 chip. Uncolored dosimeters went to ORNL employees who were not classified as radiation workers. Approximately 4,500 of these dosimeters were available (Gupton 1974). Green dosimeters, which contained a single TLD-700 chip, were issued to non-ORNL employees. Approximately 700 of these dosimeters were available (Gupton 1974). These initial TLDs included beta-gamma film packets as backup in the event of a problem with the TLD elements.

As of August 1975, the system of Class A, B, and C photobadges had been redefined. There were still three classes of badges, but they were now referred to as:

- Employee take-home
- Nonemployee take-home
- Nonemployee gate or exchange (not take-home) (Gupton 1975)

Persons with photobadges were designated as monitored or nonmonitored, referring to whether they received personnel dosimetry. All ORNL employees were monitored, as were persons with take-home photobadges for whom monitoring was requested by their Divisional Radiation Control Officer. All others were designated as "monitoring not required" (Gupton 1975). Monitored persons were assigned one of three types of personnel dosimetry, depending on their work assignments and exposure potential. The three dosimeters were designated Type 1, Type 2, and Type 3.

The Type 1 dosimeter was the same as the uncolored dosimeter described above: A TLD-100 chip under cadmium and a TLD-700 chip under the OW. The Type 2 dosimeter contained a TLD-100 chip under an aluminum filter and two thin (0.015-in.) TLD-700 chips, one under the OW and the other under plastic (Gupton 1975). This dosimeter had the same role as the yellow dosimeter described above (i.e., issued to radiation workers for whom neutron exposure was unlikely). The Type 3 (or red) dosimeter was a four-element design with a TLD-600 and a TLD-700 under a cadmium filter, a TLD-600 and a TLD-700 under an aluminum filter, and two thin (0.015-in.) TLD-700 chips under the OW and plastic filters, respectively (Gupton 1975). A vapor-sealed NTA film monitored fast neutrons. The Type 3 dosimeter was issued to radiation workers for whom neutron monitoring was appropriate, as was the red dosimeter described above.

Gupton (1978) describes the different models of TLDs used for personnel monitoring at ORNL as being the same as the Type 1, 2, and 3 dosimeters above. However, by that time they were designated Class 1, Class 2, and Class 3 rather than Type 1, 2, and 3.

The Class 1 dosimeter was assigned to individuals for whom little or no radiation exposure was anticipated (Gupton 1978). This dosimeter was used for nonradiation workers, including but not limited to clerical, administrative, and construction workers as well as visitors. These were functions for which ORNL anticipated that an inconsequential fraction of the potential dose would be from neutrons, betas, or photons of energy below 30 keV. The exchange period for these dosimeters was annual. Positive readings, which were rare, were investigated, and a notational dose was made as necessary.

The Class 2 dosimeter was used for whole-body and skin dose, and for indicating neutron exposure (i.e., it was not used for neutron dosimetry but would indicate that an unexpected neutron exposure had occurred). This dosimeter was assigned to radiation workers where there was low anticipation of neutron exposure.

The Class 3 dosimeter was used for measurement of beta, gamma, and neutron dose. The analysis of results from Class 3 dosimeters in accordance with established algorithms (see Section 6.4.2) allowed ORNL dose evaluators to assign dose equivalents for fast, thermal, and intermediate neutron energy groups. The total neutron dose equivalent was the sum of the three components (Gupton 1978).

ORNL continued to use the three dosimeters described above after the two-element Union Carbide Corporation Nuclear Division (UCC-ND) TLD came into use at the four Oak Ridge Reservation sites (ORNL, Y-12, K-25, and the Paducah Gaseous Diffusion Plant) in 1981. ORNL TLDs were used either in lieu of or in addition to the UCC-ND TLD. The UCC-ND TLD had two LiF chips: one under an OW and the other under an aluminum filter. The unit provided beta-photon discrimination in determining skin and whole-body doses (McLendon 1980). A 1-g foil of indium was included in the UCC-ND badge for accident dosimetry.

Gupton (1981) describes the ORNL Class 1, 2, and 3 dosimeters as being called *green dot*, *yellow dot*, and *red dot*, respectively. He also describes an *H dosimeter* that was assigned to persons who did not have a UCC-ND dosimeter for short-term use where doses between 1% and 10% of DOE annual limits were likely during the period of use. By this time, the yellow dot (formerly yellow or Class 2) dosimeter had evolved to a four-element design with three thin TLD-700 chips and one TLD-100 chip of normal thickness. The green dot dosimeter was the same as the yellow dot design, but had an annual exchange frequency. The ORNL neutron dosimeter (Class 3 or red dot) remained unchanged. Yellow dot and red dot dosimeters were referred to collectively as *HP meters*. In addition, a blue dot dosimeter for visitors and other nonemployees was the same as the green dot dosimeter (Berger and Lane 1985).

Table 6-1 summarizes the various ORNL TLDs used from 1975 to 1979.

Table 6-1. TLDs used from 1975 to 1979.

Approximate date range	TLD designation	TLD elements (filters)	Usage
Jan. 1975 – Aug. 1975	Uncolored	TLD-100 (Cd) TLD-700 (OW)	ORNL employees, nonradiation workers
	Green	TLD-700 (OW)	Non-ORNL employees
	Yellow	TLD-100 (Al) TLD-700 (OW)	Radiation workers in non-neutron areas
	Red	TLD-600 (Al) TLD-600 (Cd) TLD-700 (OW) TLD-700 (plastic)	Radiation workers with potential for neutron exposure
Aug. 1975 – 1978	Type 1	TLD-100 (Cd) TLD-700 (OW)	Nonradiation workers
	Type 2	TLD-100 (Al) TLD-700 (OW) TLD-700 (plastic)	Radiation workers in non-neutron areas
	Type 3	TLD-700 (OW) TLD-700 (plastic) TLD-600 + TLD-700 (Al) TLD-600 + TLD-700 (Cd)	Radiation workers with potential for neutron exposure
1978 – 1981	Class 1	Same as Type 1	Nonradiation workers
	Class 2	Same as Type 2	Radiation workers in non-neutron areas
	Class 3	Same as Type 3	Radiation workers with potential for neutron exposure
1981 – 1989[a]	Green Dot	Same as Yellow Dot	Nonradiation workers
	Yellow Dot	TLD-100 (Al) TLD-700 (OW) TLD-700 (plastic) TLD-700 (Cd)	Radiation workers in non-neutron areas
	Red Dot	Same as Type 3	Radiation workers with potential for neutron exposure

a. UCC-ND and Panasonic TLDs were also used in this period.

Sometime around the mid-1980s, ORNL began using four-element Panasonic TLDs to augment the two-element UCC-ND dosimeters. Two Panasonic dosimeters were employed for this purpose, one for beta-gamma monitoring and the other for neutrons (MMES 1992).

In 1989, ORNL began implementing the Centralized External Dosimetry System (CEDS), which used two dosimeters for whole-body monitoring: A four-element beta-gamma dosimeter (blue holder) and a four-element neutron-sensitive dosimeter (red holder). The CEDS beta-gamma dosimeter became the dosimeter of record as of January 1989. However, the use of the Panasonic neutron dosimeter continued as the neutron dosimeter of record until January 1990. The exchange frequency for the CEDS dosimeters is quarterly; however, neutron dosimeters can be exchanged independently of the beta-gamma dosimeters if an individual's duties involve neutron exposure from different energy spectra. Workers may be issued more than one neutron dosimeter during a monitoring period so appropriate TLD response-to-neutron-dose factors can be applied. Only one neutron dosimeter is issued at a time, and workers are required to wear them even if they were not working in neutron areas. In addition, there are sometimes workers for whom dose evaluation is performed more frequently than quarterly, as ORNL health physics staff can request that an individual's dosimeter be processed at any time. These individuals include workers for whom higher doses are anticipated, and sometimes pregnant workers are monitored more frequently. If an interim dose evaluation is requested for a neutron worker, in general both the beta-gamma and neutron dosimeters are processed. From review of claims files it appears such cases will be apparent to the dose reconstructors, as results for each dosimeter issued to an individual appear to be provided.

The CEDS dosimetry system is DOELAP-accredited, so reported deep and shallow dose values can be treated as representative of *Hp(10)* and *Hp(0.07)* for dose reconstructions.

Table 6-2 summarizes personnel dosimeters worn by radiation workers at ORNL from 1944 to the present.

Table 6-2. Dosimeter characteristics from 1944 to present.

Period	Dosimeter type	Filters	Film packets used
1944 - 1953	Two-element film badge	OW, Cd	DuPont Type 552, NTA (in routine use around 1949)
1953 – June 1956	Four-element film badge	OW, plastic, Cu, Cd, Pb	DuPont Type 552, NTA
July 1956 - 1974	Four-element film badge	OW, plastic, Al, Cd	DuPont Type 552 (until 1959), DuPont Type 544 (until replaced with Eastman Type 2), Eastman Type 2 (early 1970s), NTA
1975 – 1988 (TLDs were in use on limited basis in 1974).	ORNL TLDs - different designs assigned to workers based on duties and employment status. Used LiF TLD chips. Panasonic TLDs used from mid-to-late 1980s; these used lithium-borate and calcium-sulfate elements.	OW, plastic, Al, Cd (not all elements used in every design).	Eastman Type 2, NTA (Type 2 film was used as backup, NTA was used in neutron dosimeters only. Use of film was discontinued in mid-1980s.)
1981 – 1988	Two-element UCC-ND TLD (some workers might have worn these rather than ORNL dosimeters)	OW, Al	N/A
1989 - present	Commercial Harshaw four-element TLD system (DOELAP accredited); Panasonic neutron dosimeters used from 1989 to 1990.	OW, plastic, Teflon, Cu	N/A

6.3.1.1 Interpretation of Reported Data

The following items are pertinent to the interpretation of personnel dosimetry data recorded for ORNL workers.

- Film dosimeter readings (results) encountered in ORNL personnel monitoring histories before June 25, 1944, could be unreliable because the use of dosimeters during this time was experimental (Hart 1966).

- It appears that the reporting limit for personnel exposures based on film dosimeter readings was 10 mR from 1944 through 1946, with exposures reported to the nearest 5 mR. It also appears that the reporting limit was revised up to 30 mR (in most cases) around 1947. NOTE: The 10 mR reporting limit apparently used in the 1944-to-1946 period was not taken to imply that the LOD for the film dosimeters was that low.

  Film density results less than 30 mR were originally recorded as 0 (zero) or less than 30 mR (Hart and Martin 1949). This same reference asserts that approximately 85% of all film badge readings at that time were in the range from zero to 30 mR(ep). (Readings from behind the shielded element were reported in mR, and those from the open window were reported in mrep.) It appears that ORNL maintained a consistent practice of differentiating dosimeter results that were zeros or less than LODs from blank fields in dosimetry records. The Laboratory's practice was to attempt always to mark results that were less than established reporting limits with a zero or some other indication to distinguish these entries from blank fields. Blanks apparently implied that monitoring of a given type was not performed for that period. However, the practice of recording zeros for values less than reporting levels did not necessarily apply to all fields for a given record (e.g., if the OW reading for a monitoring period was recorded as zero, the S field would typically be blank – zeros were not duplicated across all fields). In addition, the distinctions between blanks and zeros described here apply only to handwritten data on the various dosimetry data cards maintained by ORNL; they do not apply to summary-level information or information from databases such as the so-called IBM system installed in 1949. A review of claim files verifies that blanks on semiannual exposure summary records from the IBM system correspond to monitoring periods during which values of zero were recorded for weekly exposures on the handwritten data cards. Thus, blanks in the computer-generated summary records do not mean the same thing as they do on the handwritten data cards. Dose reconstructors should always use individual data cards if they are available rather than summary-level data.

- It appears that information on neutron and beta-gamma dose was always either differentiated on the data cards or reported on separate cards. Therefore, dose reconstructors should not have to attempt to separate reported dose values into different components as long as the data cards (and not summary-level information) are available. However, sometimes neutron dose was included in the deep and shallow dose totals reported on the Beta-Gamma Dosimetry Record cards, though annotations were apparently made in such cases to identify when neutron dose was added. Neutron Dosimetry Record cards were used in addition to beta-gamma cards. NOTE: It does not appear that neutron doses were included in the semiannual summary reports from the IBM System beginning in 1949. Review of claims data indicates that these reports include only beta-gamma exposures.

- Neutron exposure data were originally expressed as a fraction of tolerance, where both the exposure and the tolerance were expressed in terms of tracks per field, or tracks for a number of fields. Section 6.4.2 provides information on interpreting neutron exposure results expressed in this manner. Later, neutron exposure data were expressed in terms of tracks per scan, where a *scan* was a unit of volume (of emulsion). It appears conversion factors (in dose equivalent per tracks per scan) were recorded directly on the dosimetry data cards for neutron exposure data expressed in this way.

- From the mid-1970s through the late 1980s, ORNL issued different TLD models to workers, depending on their duties. The type of dosimeter can be inferred from the individual elements listed on the Beta-Gamma Dosimetry Record cards. NOTE: The September 1975 version of these cards for the yellow dosimeter incorrectly listed the TLD-100 element as 100C rather than 100A, which implied that it was under a cadmium filter, when in fact it was under aluminum. Dosimetry technicians would sometimes correct the label to read 100A by hand, but often it was left as 100C.

6.3.2 Dosimetry Technology

Dosimetry methods at ORNL evolved over the years with the development of improved technology and better understanding of the complex radiation fields in the workplace. The following sections discuss performance characteristics of the various personnel dosimetry methods employed.

6.3.2.1 Beta-Gamma Dosimeters

Figure 6-1 shows the response of the ORNL film badge to photon radiation of different energies as well as the *Hp(10)* response. The figure shows two responses for film badges: One for a sensitive DuPont 502 emulsion in a two-element film badge (Pardue, Goldstein, and Wollan 1944), and one for a sensitive DuPont 555 emulsion in a multielement film badge (Thornton, Davis, and Gupton 1961). The response of the sensitive Eastman Kodak Type 2 film in a multielement film badge should be similar to that of the sensitive DuPont 555 emulsion. The film badges show an under-response at 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 better match to the *Hp(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).

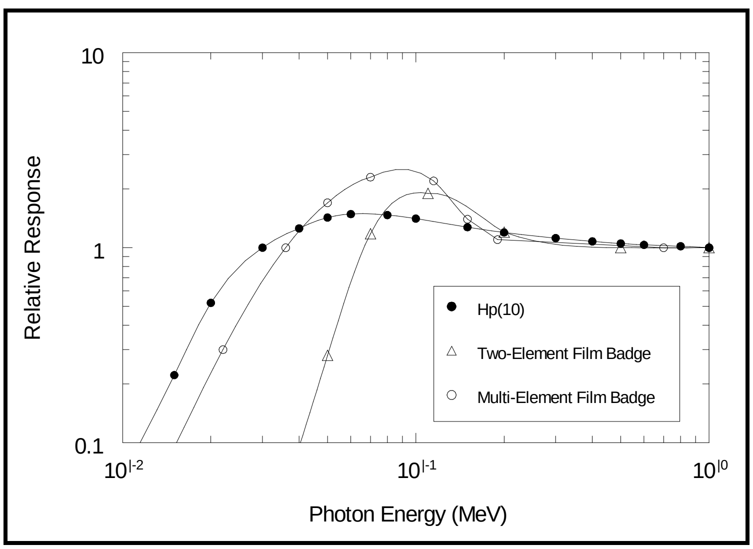

Figure 6-1. Comparison of *Hp(10)* for photons with the energy responses for a sensitive DuPont 502 emulsion in a two-element film badge (Pardue, Goldstein, and Wollan 1944) and a sensitive DuPont 555 emulsion in a multielement film badge (Thornton, Davis, and Gupton 1961).

6.3.2.2 Neutron Dosimeters

The two general types of neutron dosimeters that have been used at ORNL (NTA film and neutron-sensitive TLDs) differ significantly in their responses to neutrons of different energies. In general, the response of NTA film decreases with decreasing neutron energy that is greater than a threshold energy estimated to be about 500 keV (IAEA 1990), while TLD response increases with decreasing neutron energy. Results reported at the first AEC Neutron Dosimetry Workshop in 1969 indicated that laboratory dose measurements made with NTA film were about one-half to one-fourth of those measured with other methods, including neutron-sensitive TLDs (Vallario, Hankins, and Unruh 1969). The response of both dosimeters is highly dependent on the neutron energy spectra, and both dosimeter types require matching the laboratory calibration neutron spectra to the workplace neutron spectra for reliable results.

NTA film would not, in general, have been very sensitive to the leakage neutron spectra associated with ORNL reactor facilities.

6.3.3 Dosimeter Calibration Practices

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

6.3.3.1 Beta-Gamma Dosimeters

A ^{226}Ra photon source was used to calibrate film badges for response to photons for both the OW and cadmium-shielded elements. The radium source was used to develop calibration curves for the sensitive film for both the open and filtered elements of the dosimeters, and to develop a curve for the insensitive film under the shielded element. The use of ^{226}Ra for the calibration of personnel dosimetry at ORNL continued until the late 1980s, when it changed to ^{137}Cs. Such isotopic photon sources were not used exclusively for photon calibrations, as the Laboratory made efforts to understand the performance of various personnel dosimeters in use to different types of radiation. For instance, film dosimeters were calibrated to various X-ray spectra as of the mid-1950s to generate calibration curves used for evaluating special cases involving X-ray exposures (West 1992).

In the ninth week of 1947, natural uranium plate sources were introduced and used as beta calibration sources to augment calibrations from the radium gamma source (Hart 1966). At that time, the assumed dose rate on the surface of the uranium slab was 270 mrep/hr. On March 1, 1951, this value was revised to 240 mrep/hr (Hart 1966). The uranium slab was used to develop calibration curves for converting film density from under the OW of the dosimeters to dose in millirep. Before the ninth week of 1947, readings from under the OW were expressed in milliroentgen based on calibration to ^{226}Ra photons. Therefore, OW readings for this period must be adjusted if they are to be compared with results from later periods. Similarly, the OW results for the period between the ninth week of 1947 and March 1, 1951, require adjustment to enable consistent comparisons. Section 6.4.3.2 discusses these adjustments.

Initially, ORNL felt the OW of the film badge was sensitive to any beta energy able to penetrate the skin and “cause damage to underlying tissue,” and that “[a]ny exposure which the meters will not record is thought to be non-dangerous” (Roberson 1947). The disparity between the response of the OW to low-energy photons versus that for beta radiation was well understood, as was the difference in response between low- and high-energy photons. The ORNL choice to evaluate the OW readings using a calibration curve derived from natural uranium beta was based on taking advantage of the conservatism associated with evaluating the photon response of the OW using the uranium calibration curve.

Thornton, Davis, and Gupton (1961) discussed the results of experiments that showed film density per unit absorbed dose from natural uranium beta for the OW was equal to that for the cadmium-filtered element for ^{226}Ra photons. That document also discussed results that showed the film density per unit absorbed dose from beta sources varied slowly with the average beta energy over a wide range and concluded that film dosimeters were equally sensitive to beta and gamma radiation. NOTE: These results compare the beta response of the OW to the photon response of the cadmium-filtered element. Hart (1966) asserted that the film density per unit dose from uranium beta for the OW was 2.2 times lower than that from ^{226}Ra gamma radiation. By extension, this suggests the OW response of the dosimeter to ^{226}Ra photons was 2.2 times that of the cadmium-shielded element if the assertion that the uranium beta response of the OW and the ^{226}Ra photon response of the shielded element were equivalent is accurate.

As of the late 1980s, calibration and performance testing of CEDS dosimeters has been performed using both the ORNL Radiation Standards and Calibration Laboratory (RASCAL) and third-party facilities (ORNL 1994). RASCAL has provided irradiations for calculating element correction coefficients, developing algorithms, performing TLD response studies, ensuring reader reproducibility, and conducting blind audits. However, RASCAL was not the primary source of information for the development of neutron correction factors, which were determined from *in situ* characterization of workplace spectra via tissue-equivalent proportional counters and other instrumentation.

Facilities at RASCAL include a low-scatter room for performing neutron and high-energy photon irradiations, a beta irradiation facility, and an X-ray facility. Isotopic photon sources include ^{137}Cs and ^{241}Am. Neutron sources include bare and D_2O-moderated ^{252}Cf as well as a ^{238}Pu:Be source. Beta sources include ^{90}Sr/Y and ^{204}Tl, along with ^{147}Pm point sources and a depleted uranium slab source. The X-ray facility is capable of providing a number of standard spectra. Dosimeters can be calibrated using a variety of geometries, including free-in-air and DOELAP torso, finger, and extremity phantoms. RASCAL has participated in intercomparisons with the National Institute of Standards and Technology (ORNL 1994).

6.3.3.2 Neutron Dosimeters

The NTA film was calibrated initially using the Graphite Reactor and indium foil (Davis, Gupton, and Hart 1954). Thornton, Davis, and Gupton (1961) assert that a fast neutron exposure of 100 mrem (the weekly tolerance at that time) would produce approximately 1,000 recognizable tracks per square centimeter in the film, and that "[t]he figure for thermal neutrons is about the same." The small track lengths to be discerned in the emulsion required high (950 power) magnification, resulting in a field of view of 2×10^{-4}/cm^2. Thus, 100 mrem of fast or thermal neutron exposure gave a track density of 0.2 track per field (Thornton, Davis, and Gupton 1961).

As of the late 1960s, calibration of NTA film dosimeters used either unmoderated Po:Be or ^{238}Pu:Be sources, with the resultant delivered dose equivalent expressed in terms of equivalent exposure from a Po:Bo source. Specifically, the use of a fluence-to-dose-equivalent conversion factor of 2.23×10^{-7} n/cm^2/rem was based on cross-referencing between the calibration sources and a Po:Bo spectrum, because the latter was "more suited for calibration purposes" (Gupton 1968, 1969). NTA film dosimeters were exposed to 1 rem to obtain a calibration factor in terms of average number of tracks per scan per rem. A *scan* was a unit of volume defined by the depth of the emulsion, the width of the field of view, and a length of 2 mm.

NTA film was not capable of registering a response to neutrons having energies below about 500 keV. Nonetheless, when it first came into use, ORNL described a means of computing a *thermal* neutron dose based on results from NTA film; this dose was the difference between readings under the OW and cadmium-filtered element. Presumably, the calibration factors applied to get from track density to dose were determined from exposure of the dosimeters to thermal neutron beams from the Graphite Reactor. However, it is unclear if neutron dose was actually reported in this way (i.e., as separate fast and thermal components). Personnel neutron exposure data reviewed to date have been in terms of either OW and cadmium-filtered element readings or single dose equivalent values. NOTE: Early tolerance values for fast and thermal neutron exposure, which were expressed in terms of tracks per a number of fields, were equal (e.g., the tolerance values were both 20 tracks per 12 fields in mid-1949).

By the late 1960s, ORNL had apparently abandoned the idea of assigning a thermal neutron dose on the basis of results from NTA film dosimeters and, in fact, Gupton (1968) states "[p]ersonnel monitoring devices are inadequate for dosimetry of intermediate energy neutrons." However, he goes on to say that "relative blackening" of the region under the cadmium filter of the beta-gamma dosimeter provided an indication that "nominal" exposure to intermediate-energy neutrons had occurred. Dose estimations could therefore be made based on appropriate survey data if exposure to intermediate-energy neutrons was indicated (Gupton 1968).

Neutron calibration practices improved over time to where neutron doses were assigned based on characterizing the personnel neutron dosimetry to known neutron fields [e.g., from the Health Physics Research Reactor (HPRR)], and combining these response characteristics with knowledge of workplace neutron spectra. As such, reporting of neutron dose evolved from terms of OW and S (corresponding to response of the NTA film from under the OW and cadmium-filtered dosimeter elements, respectively) to single values representing the contribution from all energies. Isotopic sources (^{238}Pu:Be and ^{252}Cf) were used for routine performance testing once TLDs came into use (Gupton 1981).

In 1985, the neutron response of the red dot TLD (which was the dosimeter used at ORNL for neutron monitoring) was determined by irradiation of badges at the HPRR. The HPRR was a small, unmoderated, fast reactor fueled by enriched uranium. Different neutron energies were obtained by employing different shield materials, as summarized in Table 6-3.

Table 6-3. HPRR shield materials and corresponding neutron energies used to characterize TLDs (Berger and Lane 1985).

Shield material	**Median neutron energy (keV)**	**Mean neutron energy (keV)**
None (bare)	780.0	1,280
Steel	340.0	580
Concrete	3.3	560
Lucite	0.07	640

TLDs were irradiated to different neutron energies obtained from HPRR to determine response-to-dose calibration factors as well as relative energy response for development of neutron dose algorithms. Neutron dose algorithms and dose assignments were based on these characterizations and on characterizations of workplace neutron spectra. The same practices were applied to the CEDS program when it was adopted in 1989. A significant effort was begun at this time to further characterize the workplace neutron spectra at ORNL (see Section 6.3.4.2).

6.3.4 Workplace Radiation Fields

6.3.4.1 Beta-Gamma Exposures

To date, quantitative assessments of beta-gamma radiation fields encountered in ORNL facilities have not been identified. This is not surprising, because dosimetrists generally attempt to differentiate and account for different radiation qualities in beta-gamma personnel monitoring using the response of individual dosimeter elements or ratios between them. Further, ORNL's technical basis document for its current dosimetry system states, "Actual field characterizations of photon spectra have not been performed at facilities served by CEDS" (ORNL 1994). Indeed, attempting to develop defendable characterizations of photon and beta fields at ORNL facilities over its history could prove impractical. Therefore, dose reconstructors should apply claimant-favorable assumptions for radiation types and energies when converting deep and shallow beta-gamma dosimeter dose for ORNL workers to organ dose for input to the Interactive RadioEpidemiological Program (IREP). The method and energy-group assumptions to use for organ dose calculations from beta-gamma dosimeter dose readings are discussed below. NOTE: Dose reconstructors should use exposure-to-organ-dose conversion factors when computing organ dose from photon exposures measured before January 1989 for cases where the organ of interest is something other than skin.

For a given beta-gamma dosimeter result recorded before 1989, organ dose should be computed using the reported deep dose plus any *net* shallow dose for cases where exposure from transuranic (TRU) material was likely or the organ of interest is the skin. Net shallow dose is the difference between the reported shallow and deep (or OW and S) results. Net shallow dose should be considered in cases of potential TRU exposure to account for dose that would not have penetrated the shielded dosimeter elements. (It is assumed that TRU exposure would be accurately evaluated after 1989, when the ORNL external dosimetry system became DOELAP accredited.)

The recommended energy groups to be applied to the deep component are given in Attachment 6A, which provides a listing of principal ORNL facilities where workers could have received occupational beta-gamma exposure and their approximate dates of operation. For most facilities, the recommended energies are a mixture of 75% more-than-250-keV photons and 25% 30- to 250-keV photons. Exceptions are certain accelerator facilities and those where TRU materials were handled, but it is not known if fission products were also present. The net shallow component, if necessary, should be treated as either 100% less-than-30-keV photons or 100% greater-than-15-keV electrons. This choice should be made based on the worker's job function and exposure potential, e.g., the less-than-30-keV photon group should be selected for an individual that worked in a TRU facility, while the greater-than-15-keV electrons would be an appropriate choice for a uranium worker or one who worked with mixed fission products for whom dose to the skin is needed. The net shallow component is not used for deep organs unless exposure from TRU materials is likely. NOTE: Because the OW element of early film dosimeters was not capable of differentiating between beta and photon dose, ORNL employed a calibration curve based on calibration to a uranium beta spectrum, knowing this would provide conservative results for photon exposure. This, plus the over-response of the film to low-energy photons, resulted in what ORNL felt were conservative assessments of nonpenetrating dose. In fact, this conservatism led to the introduction of the quantity *probable total reading* in the second half of 1951, followed in subsequent years by the development of methods and algorithms for assigning shallow dose based on the response of the individual elements of the multielement dosimeters. The energy group recommendations given above and in Attachment 6A should be adjusted as appropriate if more specific information is available for a worker's reported dose for a given monitoring period or exposure incident.

Examples illustrating the method for computing organ dose from beta-gamma dosimeter results are given below. For the first two examples, consider an individual that worked in Building 3026-D and had a reported deep dose of 80 mR and a reported shallow dose of 100 mrep. For the last example, consider the same reported dose values for an individual that worked in Building 7920 instead (see Attachment 6A for energy group selections).

Example 1: Compute the dose to the stomach. The dose values and energy groups to be entered into IREP would be:

> 250 keV photons: (0.080)(0.75)(0.885) = 0.053 rem (acute)
30 keV – 250 keV photons: (0.080)(0.25)(1.251) = 0.025 rem (acute)

where the fractions 0.75 and 0.25 are those given in Attachment 6A for the corresponding energy groups, and the values 0.885 and 1.251 are the exposure-to-organ-dose conversion factors from the *External Dose Reconstruction Implementation Guidelines* (NIOSH 2002) (100% AP exposure geometry has been assumed). The net shallow component is not needed, since it would be treated as electron exposure for this facility, and the organ of interest is not the skin.

Example 2: Compute the dose to the skin. The dose values and energy groups to be entered into IREP would be:

> 250 keV photons: (0.080)(0.75) = 0.060 rem (acute)
30 keV – 250 keV photons: (0.080)(0.25) = 0.020 rem (acute)
> 15 keV electrons: (0.100 – 0.080) = 0.020 rem (acute)

where the net shallow component is treated as >15 keV electrons for the 3026-D facility.

Example 3: Compute the dose to the stomach. The dose values and energy groups to be entered into IREP would be:

> 250 keV photons: (0.080)(0.75)(0.885) = 0.053 rem (acute)
30 keV – 250 keV photons: (0.080)(0.25)(1.251) = 0.025 rem (acute)
< 30 keV photons: (0.100 – 0.080)(0.182) = 0.004 rem (acute)

where the net shallow component has been treated as <30 keV photons because Building 7920 is a TRU facility. The factor 0.182 is the exposure-to-organ-dose conversion factor, as in Example 1 above.

6.3.4.2 Neutron Exposures

With few exceptions, information on workplace neutron energy spectra or neutron exposure data for ORNL facilities before the late-1980s is sparse. In general, neutrons were not a large source of personnel exposures at ORNL (relative to photon sources), and this could be part of the reason that little information seems to be available on workplace neutron fields or exposures. Relatively few ORNL staff members were designated as *neutron workers* for a given monitoring period. Information is particularly lacking for many of the reactors that operated at ORNL early in its history. However, neutron dose rate information is available for the HPRR and the Graphite Reactor (see Section 6.3.4.2.1). Section 6.3.4.2.2 details measurements performed by Pacific Northwest Laboratory (PNL) to characterize the response of the CEDS neutron dosimeters to workplace neutron spectra at ORNL and Y-12. These characterizations took place between 1989 and 1991. Section 6.3.4.2.3 describes the use of ORNL personnel neutron dosimetry records from 1990 through early 2004 to extract distributions of personnel neutron dose and neutron-to-photon dose ratios for different groups of ORNL neutron workers. ORNL defines different groups of neutron workers for the purpose of assigning appropriate response-to-dose conversion factors for their TLD results. The different groups are defined based on the worker's duties and occupancy in different neutron fields.

Attachment 6B contains a list of principal ORNL facilities where workers could have received occupational neutron exposure, their approximate dates of operation, and the radiation energy selections that should be made when entering personnel exposure data in IREP. The energy selections are based on characterizations of ORNL workplace neutron spectra where available, as discussed in the following sections. The claimant-favorable default of 0.1 to 2.0 MeV neutron energy was assigned for facilities for which characterization data were unavailable. Dose reconstructors should adjust the neutron energy group data in Attachment 6B as appropriate if information that is more specific is available for a given worker or exposure incident.

Attachment 6C contains a summary of neutron-to-gamma dose ratio data derived from the workplace radiation field characterizations discussed in Sections 6.3.4.2.1 and 6.3.4.2.2. Dose reconstructors should use these ratios with caution, as they are highly dependent on location, shielding, and scattering characteristics and, in the case of freshly separated material, can be a function of material age relative to the time required for gamma-emitting decay products to reach equilibrium. The importance of ingrowth on neutron-to-gamma dose ratios depends on the proportion of photon dose that is due to radioactive decay versus that from neutron interactions with surrounding materials.

With the exception of those for the HPRR and the Graphite Reactor, all of the values given in Attachment 6C were derived from the PNL characterization measurements. These values reflect fixed measurements performed in conservatively chosen locations. In addition, all of the neutron-to-photon dose ratios discussed in Sections 6.3.4.2.1 and 6.3.4.2.2 and Attachment 6C reflect 100% occupancy (i.e., they do not account for photon dose not associated with the neutron field). Thus, applying these values to an individual's reported photon dose could result in a significant overestimation if the individual received photon exposure from other sources or geometries and relative occupancy is not considered. Section 6.3.4.2.3 addresses neutron-to-photon dose ratios for different groups of ORNL neutron workers where occupancy in all photon fields (i.e., total photon dose) is inherently included. These ratios were obtained from personnel neutron dosimetry records provided by ORNL and are summarized in Attachment 6D. Attachment 6E summarizes neutron dose data for ORNL neutron workers obtained from the neutron dosimetry records.

Attachment 6F contains a summary of available workplace neutron field characterization data in terms of the fraction of neutron dose equivalent from neutrons with energies above the 500-keV cutoff for NTA film. These data are included to provide some basis for estimating the sensitivity of NTA film for personnel monitoring in similar neutron fields, as necessary. The values were derived from the information in Sections 6.3.4.2.1 and 6.3.4.2.2.

The neutron energy spectra, doses, and neutron-to-gamma dose ratios discussed in the following sections and Attachments 6B through 6F are derived primarily from measurements performed since 1989. However, in general, significant differences between conditions at the time of these measurements and those before or after are not expected. Conditions at reactor, accelerator, and calibration facilities are not likely to have changed significantly over time, and operations at the Radiochemical Engineering Development Center (REDC) have remained consistent over its history. Changes in the TRU mix at the REDC are not expected to have a significant impact on the characterization and dosimetry data because the fission neutron and associated gamma spectra from the different neutron-emitting nuclides are essentially identical. Similarly, conditions in the TRU waste and storage areas are not expected to vary considerably over time.

6.3.4.2.1 Health Physics Research Reactor and Graphite Reactor

Kerr and Johnson (1968) reported radiation survey measurements performed for the HPRR control building. Their results were applicable to both steady-state and burst operation of the reactor (it was capable of both), and included dose equivalent rates (per unit reactor power) for fast and thermal neutrons as well as photons. The results showed neutron dose equivalent rates inside the control building (Building 7710) were dominated by fast neutrons, with approximately 90% of the total neutron dose equivalent rates due to fast neutrons and 10% to thermals (Kerr and Johnson 1968). The neutron quality factors applied in this study were 10 for fast neutrons and 2.5 for thermals. The Kerr and Johnson data showed the neutron-to-photon dose equivalent ratio for the HPRR control building to be approximately unity.

A study of the radiation attenuation properties of the Graphite Reactor’s concrete shield (ORNL 1958) provided insight on fast and thermal neutron dose equivalent rates at the surface of the shield and the corresponding neutron-to-photon dose equivalent ratio. Measurements of absorbed dose rates from fast neutrons and photons and thermal neutron flux were made as a function of position moving from the inner face to the outer face of the shield. The fast neutron and photon dose rates were stated in terms of ergs per gram of tissue per hour. Kerr (2003) converted these data to absorbed dose rate in rad per hour, then to dose equivalent employing a fast neutron quality factor of 10. The thermal neutron flux data were converted to dose equivalent rate using a neutron quality factor of 2.5. The Kerr (2003) reevaluation of the 1958 measurements shows the neutron dose equivalent rate at the surface of the shield to be approximately 1 mrem/hr, with essentially all of the neutron dose from thermal neutrons. The data also show a neutron-to-photon dose equivalent ratio of approximately 0.1 at the surface of the shield.

Dose rate measurements performed in the shield of the Graphite Reactor indicated the neutron spectrum at the exterior face of the shield was quite thermal, which is not surprising given its 7-ft (2.1-m) thickness. Thus, if a particular neutron exposure occurred in the general work area of the Graphite Reactor, the thermal (less than 0.01 MeV) neutron energy group would be a defendable choice. However, if the dose reconstructor is uncertain that *all* of the exposure was received in the general work area, or if exposure from experimental ports is possible, the claimant-favorable default 0.10- to 2.00-MeV neutron energy is advised.

6.3.4.2.2 Workplace Neutron Field Characterizations Performed by Pacific Northwest Laboratory

After the termination of most reactor operations at ORNL [about 1987, with the exception of the High Flux Isotope Reactor (HFIR)], the primary locations where workers could have been exposed to neutrons were:

- The REDC (Buildings 7920 and 7930), from operations involving separation and purification of TRU materials produced in the HFIR

- Accelerator facilities [the Oak Ridge Electron Linear Accelerator (ORELA) and Holifield Heavy Ion Research Facility, Buildings 6010 and 6000, respectively]

- The HFIR beam room (Building 7900)

- TRU waste storage facilities [Solid Waste Storage Area (SWSA) 5 and Building 3100, the Source and Special Nuclear Materials Vault]
- Calibration facilities: RASCAL, Building 2007; Radiation Calibration Laboratory (RADCAL), Building 7735
- The Isotope Materials Research Laboratory (Building 3038)

From 1989 through 1991, Pacific Northwest Laboratory (PNL) performed measurements to characterize the response of the CEDS neutron TLDs to various workplace neutron spectra at ORNL and Y-12. The measurements consisted of characterization of workplace neutron fields using tissue-equivalent proportional counters, Bonner spheres, ^{3}He detectors, and other instruments (MMES 1992). The characterization of the neutron field in each location generally consisted of measurements of dose equivalent rates, spectral measurements, and TLD exposures. Spectral measurements were not always performed, however, and in some cases TLD exposures were omitted. In general, both beta-gamma and albedo neutron TLDs were exposed on phantoms at each measurement location.

Neutron spectral data, where available, allow description of the neutron fields present at the time of the characterizations in terms of the relative contribution to the total dose equivalent from each neutron energy group used in IREP. In addition, the TLD data can be used to establish approximate neutron-to-gamma dose ratios. The following sections summarize characterization measurements performed for ORNL facilities and associated data. In addition, these sections contain information on worker exposure scenarios assumed by ORNL for determining personnel neutron dose for some of the facilities.

The fluence-to-dose-equivalent conversion factors and neutron quality factors used in the PNL measurements are similar to those from National Council on Radiation Protection and Measurements (NCRP) Report 38 (NCRP 1971) and International Commission on Radiological Protection (ICRP) Publication 21 (ICRP 1971). Sims and Killough (1983) contains a comparison of fluence-to-dose-equivalent conversion factors from NCRP Report 38, ICRP Publication 21, and several other commonly used information sources on fluence-to-dose equivalent conversion factors and neutron quality factors.

6.3.4.2.2.1 Building 3038: Isotope Materials Research Laboratory

At the time of the characterization measurements, Building 3038 was used to store isotopic neutron sources such as ^{238}Pu:Be and ^{252}Cf before shipment. Two sets of measurements were performed: One with a ^{238}Pu:Be source in a 1-in.-thick Lucite glovebox, and one with both a ^{238}Pu:Be source and a ^{252}Cf source in a water-shielded drum. Measurements with the neutron sources in the water-shielded drum included a characterization of the neutron energy spectrum. Only TLD irradiations were performed with the ^{238}Pu:Be source in the glovebox.

Figure 6-2 shows the neutron energy spectrum determined for the two sources in the water-shielded drum in terms of dose equivalent rate. Table 6-4 lists the contribution to the total dose equivalent from the four IREP neutron energy groups of interest for ORNL workers. The table lists the data as measured as well as rebinned into two groups for convenience to the dose reconstructors. Combining the contributions from lower energy neutrons into the 0.1-to-2.0-MeV group is claimant-favorable given the difference in neutron quality factors for these energies.

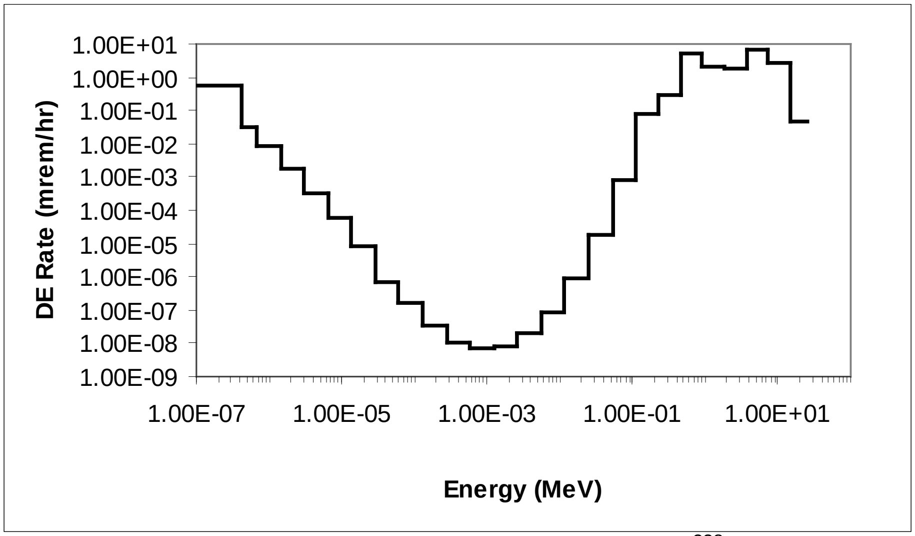

Figure 6-2. Neutron spectrum measured by PNL for ^{238}Pu:Be and ^{252}Cf sources in a water-shielded drum in Building 3038.

Table 6-4. Dose equivalent fractions for IREP neutron energy groups for ^{238}Pu:Be and ^{252}Cf sources in a water-shielded drum in Building 3038.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	3.0%
0.01 – 0.1	0.0%
0.1 – 2.0	39.1%
2.0 – 20.0	57.9%
Claimant-favorable values	
0.1 – 2.0	42%
2.0 – 20.0	58%

More than 92% of the neutron dose equivalent measured by PNL for the two sources in the water-shielded drum was from neutrons with energies above the 500-keV cutoff for NTA film. Thus, NTA film should have provided reliable results if used for personnel monitoring in similar spectra.

TLD irradiations performed with the two sources in the water-shielded drum indicated a neutron-to-gamma dose ratio of approximately 3.7. In contrast, TLD irradiations performed with the ^{238}Pu:Be source in the Lucite glovebox showed a neutron-to-gamma dose ratio of approximately 32.2. For choosing appropriate TLD correction factors for assigning neutron dose to workers in this area, ORNL assumed 80% of the exposure came from sources in the shielded configuration and 20% from unshielded sources (MMES 1992).

6.3.4.2.2.2 Building 3100: Source and Special Nuclear Materials Vault

At the time of the PNL measurements, the Building 3100 vault contained approximately 75 drums of ^{241}Am-oxide. The drums were stacked two high and two deep in two rows. Characterization measurements were performed at approximately the midpoint of the rows at a distance of approximately 84 cm.

Figure 6-3 shows the neutron spectrum determined for the ^{241}Am-oxide drums in terms of dose equivalent rate. Table 6-5 lists the contribution to the total dose equivalent from the four IREP neutron energy groups of interest for ORNL workers.

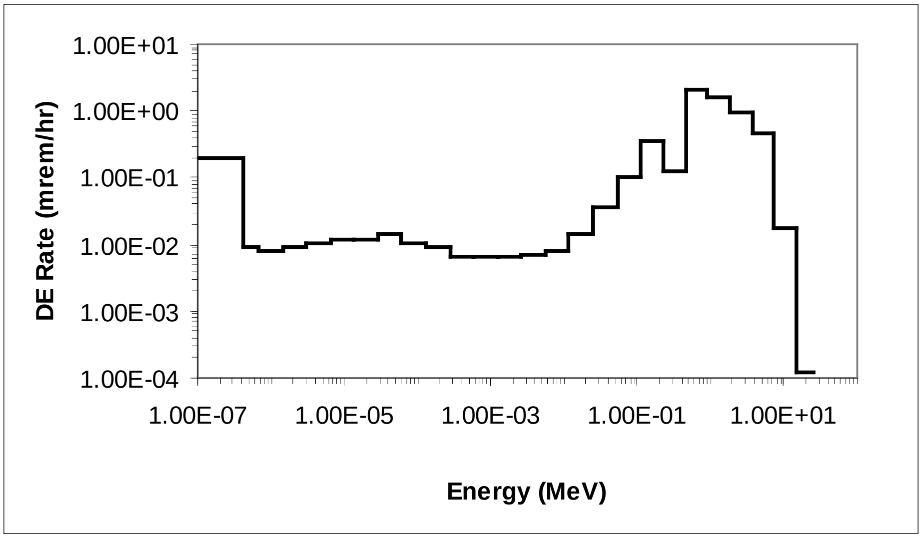

Figure 6-3. Neutron energy spectrum measured by PNL for drums of ^{241}Am-oxide in Building 3100.

Table 6-5. Dose equivalent fractions for IREP neutron energy groups for drums of ^{241}Am-oxide in Building 3100.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	5.4%
0.01 – 0.1	2.4%
0.1 – 2.0	70.4%
2.0 – 20.0	21.9%
Claimant-favorable values	
0.1 – 2.0	78%
2.0 – 20.0	22%

Approximately 80% of the neutron dose equivalent measured by PNL for the drums of ^{241}Am-oxide was from neutrons having energies above the 500-keV cutoff for NTA film. NTA film should, therefore, have provided reasonable results if used for personnel monitoring in similar spectra.

TLD irradiations performed at the same location as the spectral characterization indicated a neutron-to-gamma dose ratio of approximately 1.2.

6.3.4.2.2.3 Building 7910: High Flux Isotope Reactor

The only location at the HFIR where personnel could have been exposed to neutron radiation is the beam room. This room contains four stations (HB-1 through HB-4) where intense beams of nearly monoenergetic thermal neutrons can be directed. The thermal neutron beams are very well defined in terms of both energy and physical size. Neutron energies between approximately 0.02 and 0.08 eV can be selected. Personnel exposures to neutrons are the result of scatter from target materials being irradiated in the thermal neutron beams.

Neutron spectral characterizations in the HFIR beam room are unnecessary given the incident neutron energies. In terms of the IREP neutron energy groups, neutron exposures at HFIR should be assigned to the less-than-0.01-MeV group. NOTE: NTA film would show no response to neutrons in this energy range.

Two sets of TLD irradiations were performed with the HFIR operating at its maximum power level of 85 MW. One irradiation was at a distance of 54 cm from a ^{7}Li target in a 0.020-eV beam, and the other was at a distance of 67 cm from a nickel target in a 0.050-eV beam. Both indicated neutron-to-gamma dose ratios of approximately 0.2.

6.3.4.2.2.4 Building 6000: Holifield Heavy Ion Research Facility

The Holifield facility contains two accelerators. One of the accelerators is the Oak Ridge Isochronous Cyclotron (ORIC), which occupied Building 6000 before construction of the Holifield facility. The ORIC began operation in 1962. Following construction of the Holifield facility in the mid-1970s, the ORIC served as an ion source for the other accelerator, a 25-MV tandem Van de Graaff. The two accelerators can be operated in a number of different configurations that result in a variety of workplace exposure conditions. Nuclei are accelerated to very high energies and directed into target materials in any of several experiment rooms. However, personnel are not allowed into the experiment rooms unless dose rates (as determined by area monitors) are very low.

The operating conditions at the Holifield facility at the time PNL performed its neutron dosimetry characterizations did not allow the performance of neutron spectral measurements. Dose reconstructors should, therefore, assign neutron exposures at the Holifield facility to the claimant-favorable default 0.10-to-2.0-MeV energy group.

TLD exposure measurements were performed under conditions resulting in high-energy neutrons in one of the experiment rooms so that conservative estimates of TLD response per unit dose equivalent would be obtained. Specifically, exposures were made with the two accelerators running in tandem using a broad-range magnet and carbon target to produce ^{16}O and ^{17}O nucleons with energies of 375 and 353 MeV, respectively (MMES 1992). These irradiations indicated a neutron-to-gamma dose ratio of approximately 19.

6.3.4.2.2.5 Building 6010: Oak Ridge Electron Linear Accelerator

The ORELA is in the basement of Building 6010. It began operation in the late 1960s and is an intense, pulsed neutron source for time-of-flight experiments. Neutron energies produced by the ORELA range from thermal to beyond 50 MeV (MMES 1992). There is no personnel access to the target room or the accelerator room while the ORELA is in operation. Personnel neutron exposures result from neutron back-scattering from the target down the length of the beam line and out the two entry mazes. The PNL measurements were at the door to the west entry maze, which is the closest personnel can get to the accelerator during operation.

The measurements were made with the accelerator operating at approximately 500 W and a rate of approximately 13.5 pulses/second. The measurement location was approximately 23 cm from the grated steel door at the west entrance to the accelerator room at a height of about 1 m.

The neutron spectrum present at the measurement location was characterized using a ^{3}He proportional counter. The data from the proportional counter were subsequently rebinned to the same neutron energy group structure used in unfolding the Bonner sphere results for consistency.

Figure 6-4 shows these data in terms of dose equivalent rate. Table 6-6 lists the contribution to the total dose equivalent from the four IREP neutron energy groups of interest for ORNL workers.

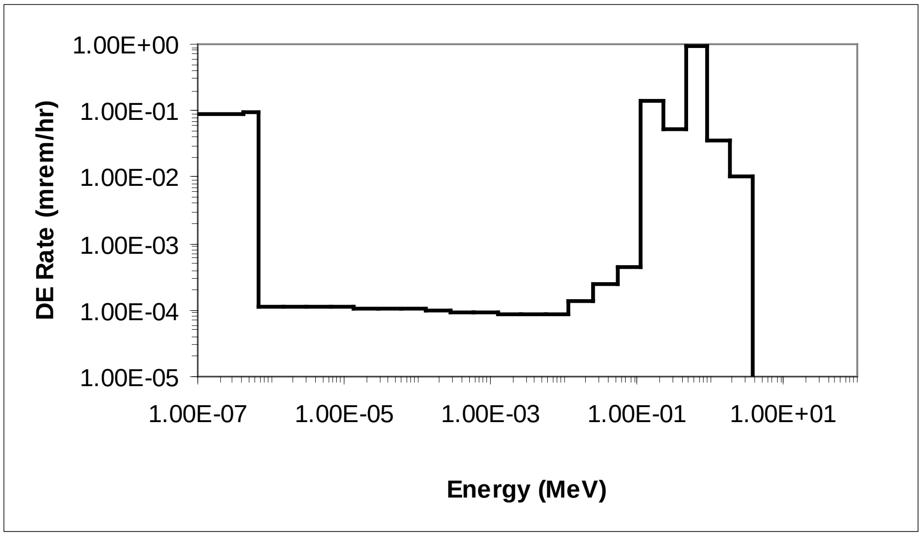

Figure 6-4. Neutron energy spectrum at the west entrance to the ORELA room.

Table 6-6. Dose equivalent fractions for IREP neutron energy groups for the west entrance to the ORELA room.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	13.6%
0.01 – 0.1	0.1%
0.1 – 2.0	85.6%
2.0 – 20.0	0.7%
Claimant-favorable values	
< 0.01	14%
0.1 – 2.0	86%

Approximately 65% of the neutron dose equivalent measured by PNL at the west entrance to the ORELA room was from neutrons with energies above the 500-keV cutoff for NTA film. NTA film would, therefore, have underestimated neutron dose if it was used for personnel monitoring in similar spectra.

TLD irradiations performed at the same location as the spectral characterization indicated a neutron-to-gamma dose ratio of approximately 2.5.

6.3.4.2.2.6 Building 7735: Radiation Calibration Laboratory

The ORNL RADCAL includes a neutron irradiation room that shares walls with other irradiation rooms and storage areas. Characterization measurements were performed in areas accessible to personnel with the ^{252}Cf neutron sources exposed. Personnel are not allowed in the neutron irradiation room when a source is in use. However, an additional characterization was performed with one of the neutron sources raised to near the surface of the source storage well. The sources are stored in a water-filled vessel 4 ft in diameter and 4 ft deep when not in use. There were two ^{252}Cf sources in use at RADCAL when the characterization measurements were performed. One source, of approximately 1,741 mg, was used primarily in a moderated configuration. The other, of approximately 573 mg, was used primarily in a bare configuration.

Five characterizations of workplace neutron exposure conditions were performed at RADCAL. Three of these included both spectral characterizations and TLD irradiations, one consisted of TLD irradiations only, and another of spectral characterization only.

Two sets of characterization measurements were made in the beta/X-ray irradiation room: One with the moderated ^{252}Cf source exposed and the other with the bare source exposed. Both sets of measurements were performed at a distance of approximately 62 cm from the wall shared between the beta/X-ray room and the neutron room. Figures 6-5 and 6-6 show the neutron spectra measured at this location for the bare and moderated sources, respectively.

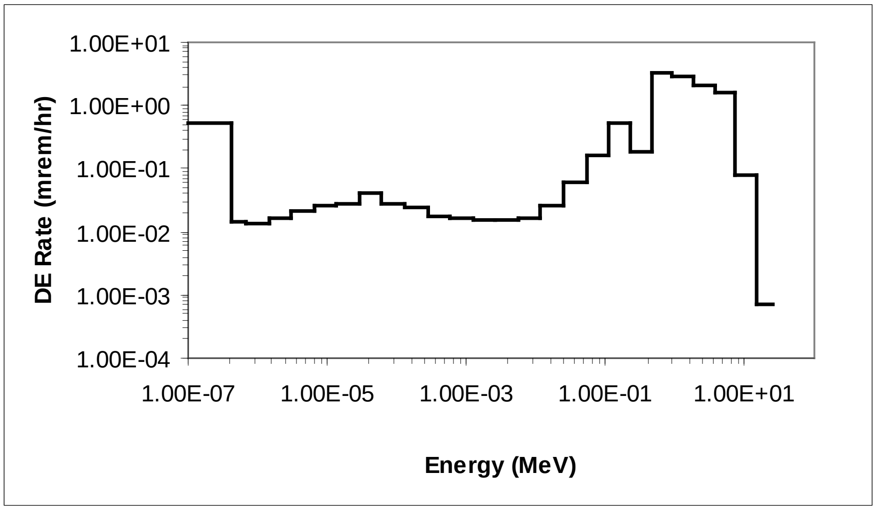

Figure 6-5. Neutron energy spectrum in the RADCAL beta/X-ray irradiation room with a bare ^{252}Cf source exposed in the adjacent neutron irradiation room.

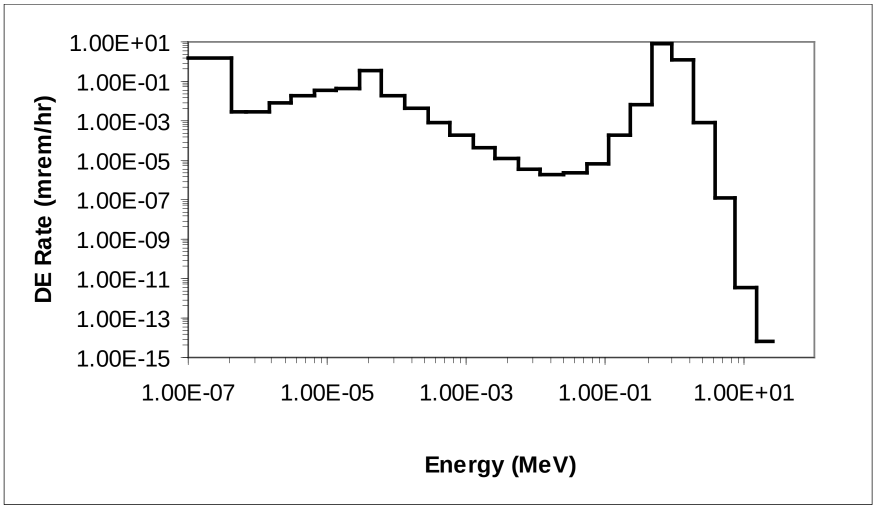

Figure 6-6. Neutron energy spectrum in the RADCAL beta/X-ray irradiation room with a moderated ^{252}Cf source exposed in the adjacent neutron irradiation room.

Tables 6-7 and 6-8 list the contributions to the total dose equivalent from the four IREP neutron energy groups of interest for ORNL workers for the bare and moderated sources, respectively.

Table 6-7. Dose equivalent fractions for IREP neutron energy groups for the RADCAL beta/X-ray irradiation room with a bare ^{252}Cf source exposed in the adjacent neutron irradiation room.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	7.0%
0.01 – 0.1	2.0%
0.1 – 2.0	60.3%
2.0 – 20.0	30.7%
Claimant-favorable values	
0.1 – 2.0	69%
2.0 – 20.0	31%

Table 6-8. Dose equivalent fractions for IREP neutron energy groups for the RADCAL beta/X-ray irradiation room with a moderated ^{252}Cf source exposed in the adjacent neutron irradiation room.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	16.4%
0.01 – 0.1	0.0%
0.1 – 2.0	83.6%
2.0 – 20.0	0.0%
Claimant-favorable values	
< 0.01	16%
0.1 – 2.0	84%

A spectral characterization was performed in the hallway outside the neutron irradiation room at a distance of approximately 30 cm from the shielded entry door. These measurements were performed with the moderated ^{252}Cf source only. Figure 6-7 and Table 6-9 indicate the spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest, respectively.

A spectral characterization was performed in the RADCAL equipment room approximately 66 cm from the wall it shares with the neutron irradiation room. These measurements were performed with the moderated ^{252}Cf source only. Figure 6-8 and Table 6-10 indicate the spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest, respectively.

TLD irradiations were performed in the beta/X-ray irradiation room in the same locations as the spectral characterizations discussed above using both the bare and moderated ^{252}Cf sources. TLD irradiations were also performed in the hallway outside the neutron irradiation room with the moderated source exposed. TLD irradiations were not performed in the equipment room. A fourth set

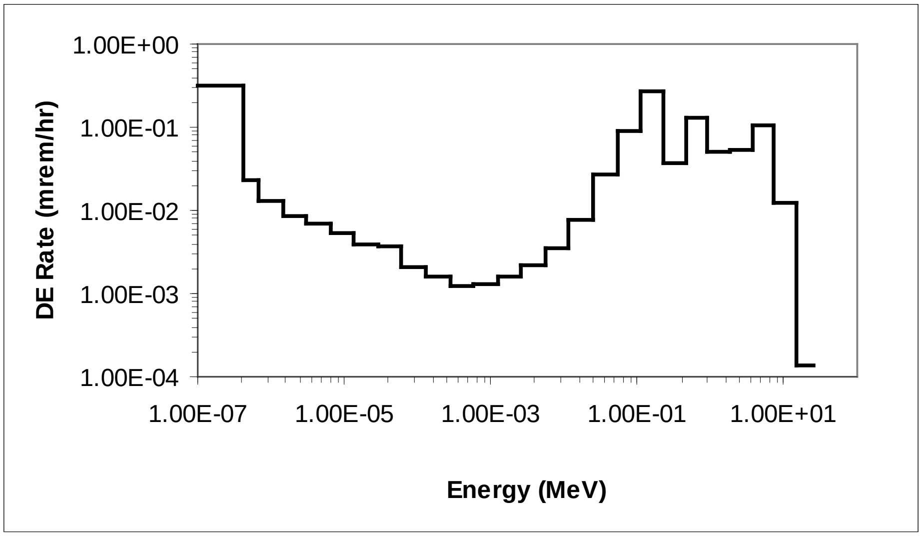

Figure 6-7. Neutron energy spectrum in the hallway outside the RADCAL neutron irradiation room with a moderated ^{252}Cf source exposed.

Table 6-9. Dose equivalent fractions for IREP neutron energy groups for the hallway outside the RADCAL neutron irradiation room with a moderated ^{252}Cf source exposed.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	33.2%
0.01 – 0.1	9.9%
0.1 – 2.0	42.6%
2.0 – 20.0	14.3%
Claimant-favorable values	
< 0.01	33%
0.1 – 2.0	53%
2.0 – 20.0	14%

of TLD irradiations was performed in the neutron irradiation room itself at the edge of the neutron source storage well. TLDs were exposed at a height of approximately 56 cm above the surface of the water with one of the ^{252}Cf sources raised to approximately 25 cm below the surface. Thus, four sets of TLD irradiation data were available from which approximate neutron-to-gamma dose ratios could be obtained. Table 6-11 summarizes these values.

The data in Table 6-12 show that NTA film might have provided a reasonable estimation of personnel neutron exposure in fields similar to that near the wall of the beta/X-ray irradiation room with the bare source in use, but would have underestimated exposures in fields similar to those resulting from the moderated source—substantially so in one case.

Table 6-12 summarizes the fraction of the total dose equivalent measured at locations in RADCAL as an element of neutron spectral characterizations that was due to neutrons with energies above the 500-keV cutoff for NTA film. This information provides a basis for estimating the performance of NTA film in similar neutron fields. NTA film was never used for personnel dosimetry at RADCAL.

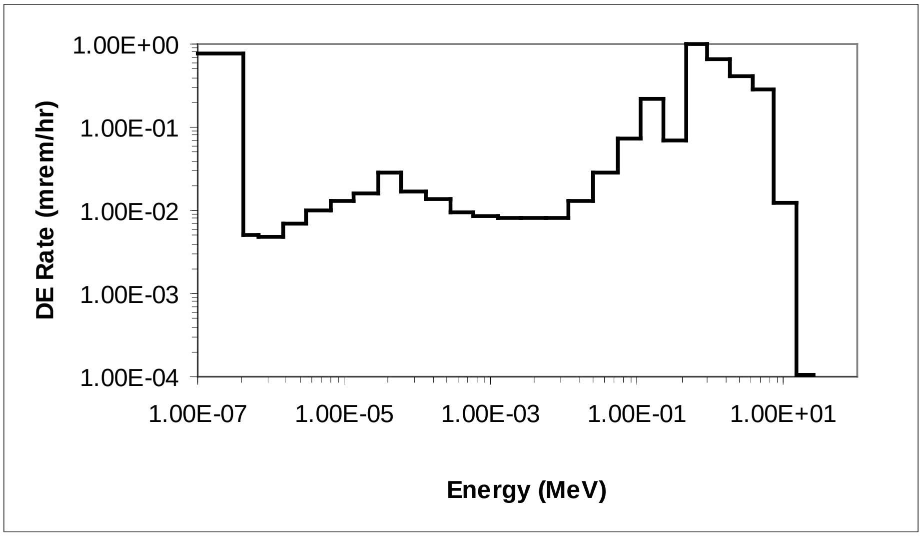

Figure 6-8. Neutron energy spectrum in the RADCAL equipment room with a moderated ^{252}Cf source exposed in the adjacent neutron irradiation room.

Table 6-10. Dose equivalent fractions for IREP neutron energy groups for the RADCAL equipment room with a moderated ^{252}Cf source exposed in the adjacent neutron irradiation room.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	25.0%
0.01 – 0.1	3.0%
0.1 – 2.0	53.9%
2.0 – 20.0	18.2%
Claimant-favorable values	
< 0.01	25%
0.1 – 2.0	57%
2.0 – 20.0	18%

Table 6-11. Neutron-to-gamma dose ratios for several locations in RADCAL.

Location	^{252}Cf source configuration	Approx. $^{1}n:\gamma$ dose ratio
Beta/X-ray irradiation room at wall shared with neutron room	Bare	15.6
Beta/X-ray irradiation room at wall shared with neutron room	Moderated	3.6
Hallway outside neutron room 30 cm from entry door	Moderated	2.1
Edge of source storage well, 56 cm above water	25 cm below surface of storage well	1.2

Table 6-12. Fraction of dose equivalent above NTA cutoff for several locations in RADCAL.

Location	^{252}Cf source configuration	Fraction of dose equivalent above 500 keV
Beta/X-ray irradiation room at wall shared with neutron room	Bare	82%
Beta/X-ray irradiation room at wall shared with neutron room	Moderated	76%
Hallway outside neutron room 30 cm from entry door	Moderated	29%
Equipment room at wall shared with neutron room	Moderated	61%

6.3.4.2.2.7 Building 2007: Radiation Standards and Calibration Laboratory

At the time of the PNL characterization measurements (October 1989), the RASCAL modern, low-scatter facility was not yet in use. Measurements were performed to characterize the neutron field experienced by instrument calibration technicians as they performed calibrations using ^{238}Pu:Be neutron sources. Specifically, the measurements were in Room 102 at 130 cm from the source.

Figure 6-9 and Table 6-13 indicate the neutron spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest, respectively. These data were determined for the ^{238}Pu:Be neutron source used for instrument calibrations at RASCAL before use of the new low-scatter facility about 1991.

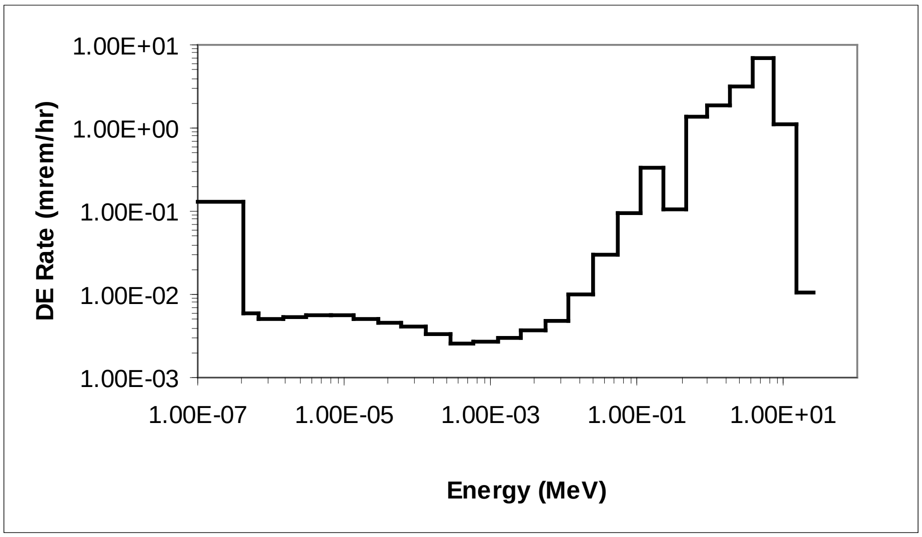

Figure 6-9. Neutron energy spectrum in RASCAL Room 102 at 130 cm from the ^{238}Pu:Be neutron calibration source.

Table 6-13. Dose equivalent fractions for IREP neutron energy groups for RASCAL Room 102 at 130 cm from the ^{238}Pu:Be neutron calibration source.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	1.3%
0.01 – 0.1	0.8%
0.1 – 2.0	26.4%
2.0 – 20.0	71.5%
Claimant-favorable values	
0.1 – 2.0	29%
2.0 – 20.0	71%

Approximately 94% of the neutron dose equivalent measured by PNL for the RASCAL Pu:Be source was due to neutrons with energies above the 500-keV cutoff for NTA film. NTA film should, therefore, have provided reliable results if used for personnel monitoring in similar spectra.

TLD irradiations performed at the same location as the spectral characterization showed a neutron-to-gamma dose ratio of approximately 28.9.

6.3.4.2.2.8 Buildings 7920 and 7930: Radiochemical Engineering Development Center

The mission of the REDC is to separate and purify transuranic isotopes (primarily ^{252}Cf) from ^{244}Cm target material that has been irradiated in the HFIR. The REDC consists of two facilities: Building 7920 and Building 7930. Both buildings contain hot cells and gloveboxes for working with irradiated targets and separated products. The separation and purification processes occur in Building 7920.

Products are prepared to customer specifications and packaged in Building 7930. Elements handled in the REDC include californium, berkelium, einsteinium, and fermium.

Given its importance as a source of personnel neutron exposures, a number of characterization measurements were performed in the REDC. ORNL subsequently defined different groups of neutron workers for assigning appropriate TLD correction factors based on their duties. Some worker groups were defined by determining occupancy factors for individuals for different areas in the REDC and using these in conjunction with characterization data to establish weighted average TLD correction factors. For other groups, conservative assumptions were made to essentially bound the radiation fields to which they could be exposed, and then a 100% occupancy factor was used to compute any neutron dose.

This section presents neutron spectra determined for five locations in the REDC, followed by a summary of neutron-to-gamma dose ratios estimated from TLD irradiation data from 11 locations. The fraction of the total dose equivalent from neutrons having energies above the 500-keV cutoff for NTA film are given for the five spectral measurements.

Neutron spectra were measured for five locations in REDC Building 7920:

- At the face of a glovebox in Room 111
- At the side of a glovebox in Room 211
- At the viewing window of the Transuranium Decontamination Facility (TDF)
- Next to a waste cask in the Limited Access Area
- At the exit of the Waste Transfer Area (WTA) transfer tunnel

At the time of the PNL measurements, Room 111 (the Shielded Cave Area) contained two shielded hot cells and two gloveboxes. Most of the neutron dose to personnel in this area came from the gloveboxes. It was estimated that less than 10% of neutron dose to workers in this area came from hot cell operations (MMES 1992). The neutron spectrum was measured at about 37 cm from the face of a glovebox containing waste and general contamination consisting primarily of ^{252}Cf. The spectrum was, therefore, expected to be representative of that when ^{252}Cf sources were present. The measurement was performed at a height of 1.25 m above the floor. Figure 6-10 and Table 6-14 indicate the neutron spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest for this location, respectively.

REDC Room 211 (the Alpha Laboratory) contained several banks of gloveboxes used for radiochemical procedures. The neutron spectrum was measured in a walkway at the end of the glovebox banks between the side of one of the boxes and the wall. The location was approximately 32 cm from the side of the box. None of the gloveboxes was in use at the time, but all were contaminated with transuranic material to some degree. The box nearest the measurement location held a small container of waste, thought to be primarily einsteinium. Figure 6-11 and Table 6-15 indicate the neutron spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest for this location, respectively.

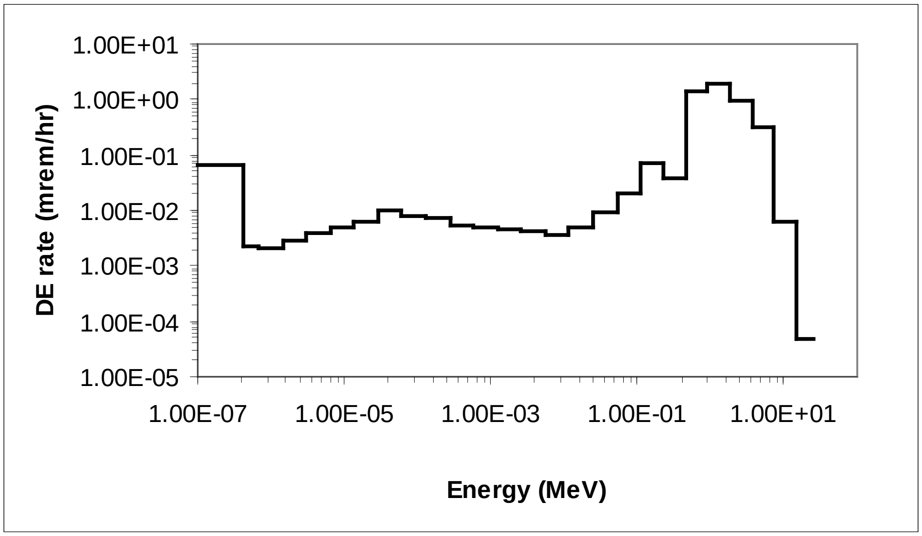

Figure 6-10. Neutron energy spectrum measured by PNL in REDC Room 111 at 37 cm from a glovebox containing ^{252}Cf.

Table 6-14. Dose equivalent fractions for IREP neutron energy groups for REDC Room 111 at 37 cm from a glovebox containing ^{252}Cf.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	2.9%
0.01 – 0.1	0.7%
0.1 – 2.0	72.3%
2.0 – 20.0	24.1%
Claimant-favorable values	
0.1 – 2.0	76%
2.0 – 20.0	24%

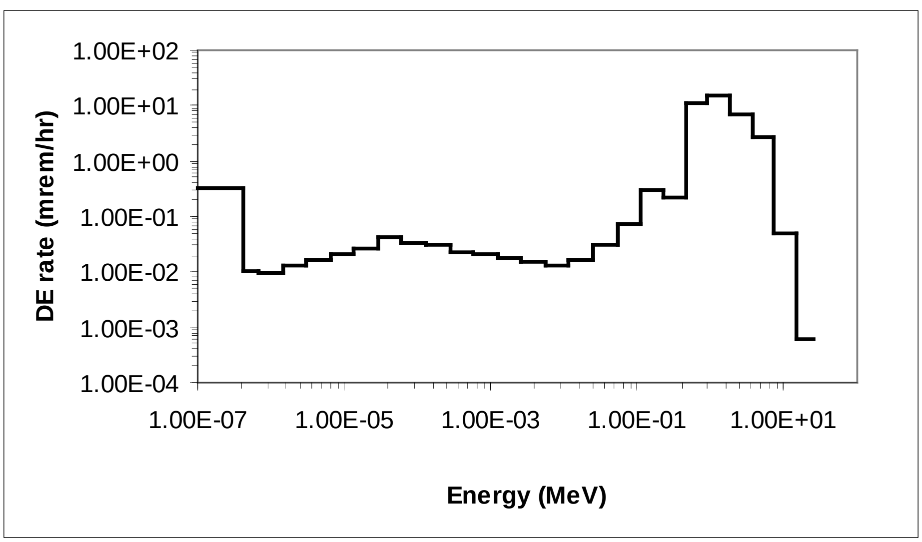

Figure 6-11. Neutron energy spectrum in REDC Room 211 at 32 cm from the end of a glovebox bank.

Table 6-15. Dose equivalent fractions for IREP neutron energy groups for REDC Room 211 at 32 cm from the end of a glovebox bank.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	1.6%
0.01 – 0.1	0.3%
0.1 – 2.0	73.2%
2.0 – 20.0	24.8%
Claimant-favorable values	
0.1 – 2.0	75%
2.0 – 20.0	25%

Measurements were made to characterize the neutron field experienced by personnel using the TDF, which is a movable hot cell on top of the Building 7920 hot cell bank. The TDF is a means of moving materials out of hot cells and into other containers. Sources or waste can be transferred into the TDF, when it is in position, through its bottom access port. Items are decontaminated as necessary in the TDF before being moved to other containers, and measurements to verify source strength can be performed in the TDF. The inner cavity of the TDF is 3 ft by 4 ft by 4 ft high. It is shielded on all sides by 3 ft of water. One face of the TDF contains a viewing window, which is where the PNL characterization measurements occurred. The measurement location was 33 cm from the window. A 12-mg ^{252}Cf source was placed on a shelf in the center of the cavity to facilitate the measurement. Figure 6-12 and Table 6-16 indicate the neutron spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest for this location, respectively.

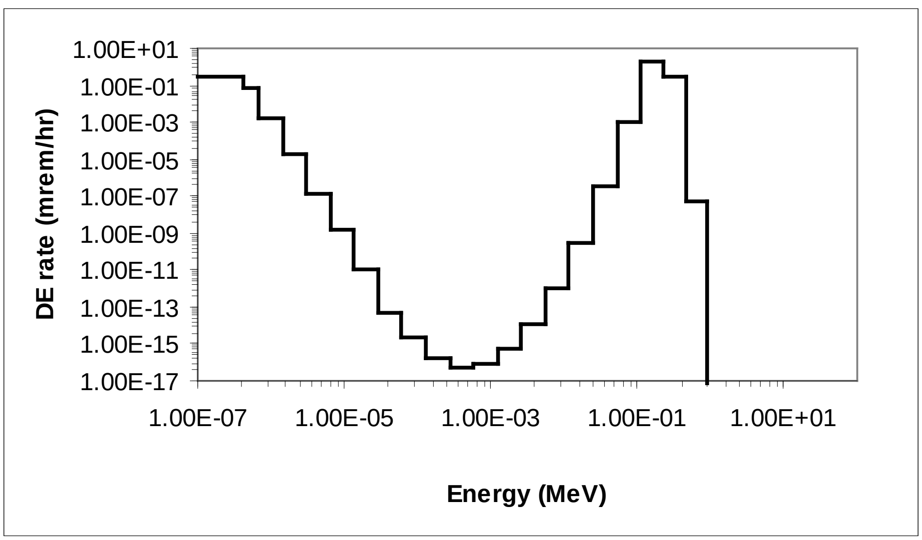

Figure 6-12. Neutron energy spectrum for the REDC TDF containing 12 mg ^{252}Cf.

A waste cask on a concrete pad in the Building 7920 Limited Access Area is a receptacle for waste generated in the process cubicles. Waste is placed in the cask as necessary by removing the cask cover using an overhead crane. The waste casks have walls of 24 cm of reinforced concrete plus 10 cm of armor-plate steel. Characterization measurements were performed approximately 18 cm from the wall of the cask at a location where the neutron dose rate was fairly uniform. The location was approximately 1 m above the support pad. Figure 6-13 and Table 6-17 indicate the neutron

Table 6-16. Dose equivalent fractions for IREP neutron energy groups for the REDC TDF containing 12 mg ^{252}Cf.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	12.2%
0.01 – 0.1	0.0%
0.1 – 2.0	87.8%
2.0 – 20.0	0.0%
Claimant-favorable values	
< 0.01	12%
0.1 – 2.0	88%

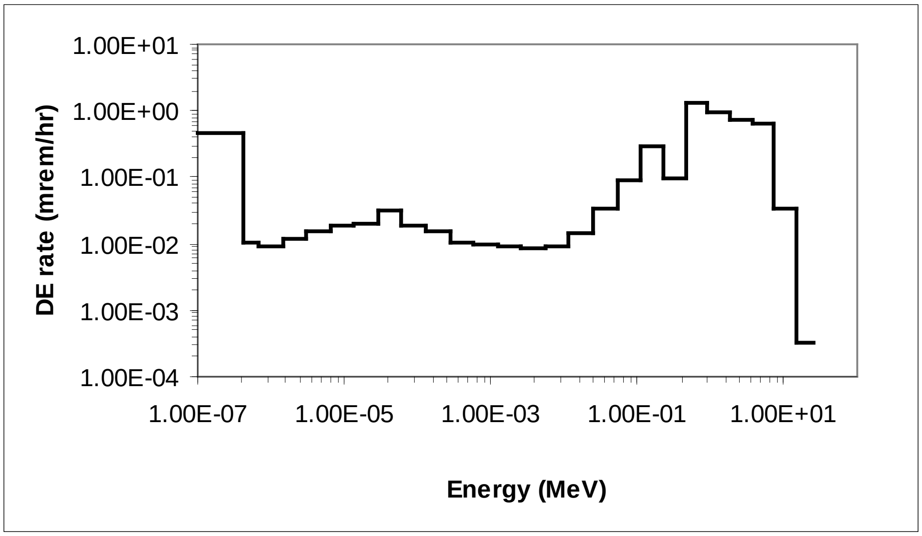

Figure 6-13. Neutron energy spectrum for a waste cask in the REDC Limited Access Area.

Table 6-17. Dose equivalent fractions for IREP neutron energy groups for a waste cask in the REDC Limited Access Area.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	13.8%
0.01 – 0.1	2.7%
0.1 – 2.0	56.6%
2.0 – 20.0	26.9%
Claimant-favorable values	
< 0.01	14%
0.1 – 2.0	59%
2.0 – 20.0	27%

spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest for this location, respectively.

Waste moves into the REDC WTA through a square tunnel that passes through the wall of process cubicle Number 10. The dimensions of the tunnel are 18 by 18 in. A metal table is attached to the outer wall of the cubicle just below the exit of the tunnel. Measurements were made directly in front of the tunnel exit at the edge of this table. This placed the measurement location approximately 58 cm from the tunnel exit and 83 cm above the floor. There were no sources present in the tunnel or in the WTA at the time of the measurements. The intent was to characterize the neutron field from contamination present on the tunnel walls. Figure 6-14 and Table 6-18 indicate the neutron spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest for this location, respectively.

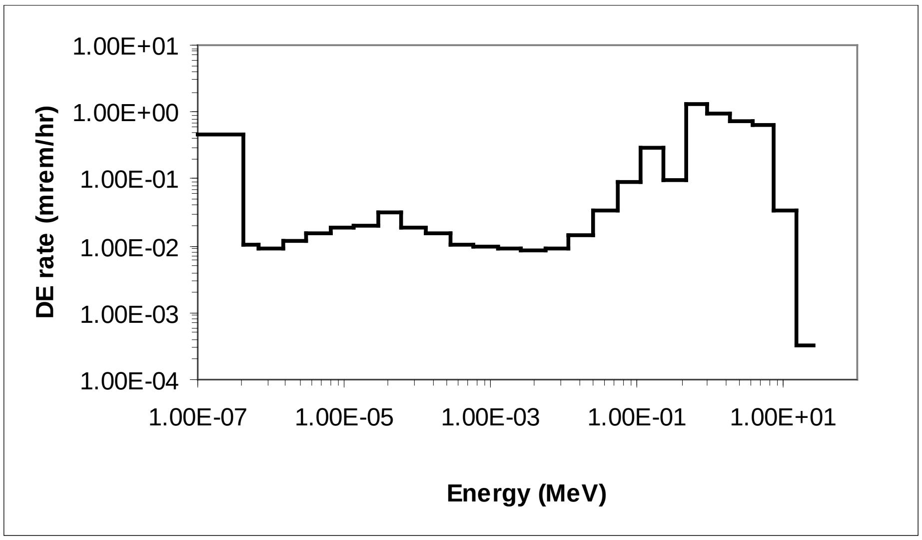

Figure 6-14. Neutron energy spectrum for contamination on the walls of the REDC WTA transfer tunnel.

Table 6-18. Dose equivalent fractions for IREP neutron energy groups for contamination on the walls of the REDC WTA transfer tunnel.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	2.5%
0.01 – 0.1	1.0%
0.1 – 2.0	90.9%
2.0 – 20.0	5.6%
Claimant-favorable values	
0.1 – 2.0	100%

TLD irradiations were performed at all five of the locations discussed above where neutron spectral characterizations were made and at an additional six locations in the REDC where spectral characterizations were not performed:

- The face of a glovebox in Building 7920, Room 211

- Approximately 4.6 m behind the glovebox bank in Building 7920, Room 211
- The face of the WTA glovebox
- The face of a glovebox in Building 7930 in a low-scatter geometry
- The face of a glovebox in Building 7920, Room 208
- The low-dose waiting area in Building 7920, Room 208

TLD irradiations in Room 211 were performed roughly 2 years after the measurements to characterize the neutron spectrum in 1989. These follow-up characterizations were in July 1991. The gloveboxes still contained transcalifornium waste, however. One irradiation was performed approximately 22 cm from the face of the same glovebox where the neutron spectrum was measured from the side in 1989. This irradiation was performed at a height of approximately 1 m off the floor. A second irradiation was performed at a location approximately 4.6 m behind this glovebox, also at a height of approximately 1 m.

An irradiation was performed 28 cm from the face of the glovebox in the WTA. This glovebox contains a rotary table for moving items into it from a small hot cell. A ^{252}Cf source was placed in the glovebox to facilitate the irradiations.

TLDs were irradiated at a distance of 76 cm from a glovebox in an open floor area in Building 7930. The location was selected because of its low-scatter geometry. The glovebox was in an area with no structure or other objects in its vicinity. The nearest wall was 3 m away, the roof was 4.6 m above, and the bottom of the box was more than 1 m off the floor. The glovebox was internally contaminated and read about 4 mrem/hr with no sources present. The TLD irradiations were performed with two ^{252}Cf sources in the glovebox (in addition to the contamination mix).

Room 208 of the 7920 building is used to prepare samples for analysis. At the time of the PNL measurements, it contained a row of gloveboxes, plus bench tops and a fume hood. Glovebox faces were shielded with a layer of Lucite in addition to their inherent shielding. TLDs were irradiated at 10 cm from the Lucite shield at the face of a glovebox containing californium and curium. This exposure was performed at a height of 1.1 m above the floor. TLDs were exposed in the same room in a low-dose area where workers were asked to stand during waiting periods.

Table 6-19 summarizes neutron-to-gamma dose ratios indicated by the TLD irradiations performed in various workplace radiation fields in the REDC.

Table 6-19. Neutron-to-gamma dose ratios for several locations in the REDC.

Location	Approx. [1]n:γ dose ratio
Face of glovebox in Building 7920, Room 111	5.8
Side of glovebox in Building 7920, Room 211	11.7
Face of glovebox in Building 7920, Room 211	5.3
4.6 m behind glovebox in Building 7920, Room 211	1.3
TDF viewing window (Building 7920)	0.2
Limited Access Area waste cask (Building 7920)	4.4
WTA transfer tunnel exit (Building 7920)	6.7
WTA glovebox (Building 7920)	1.2
Low-scatter glovebox in Building 7930	16.7
Face of glovebox in Building 7920, Room 208	1.3
Low-dose waiting area in Building 7920, Room 208	1.3

Table 6-20 summarizes the fraction of the total dose equivalent measured at locations in the REDC as an element of neutron spectral characterizations that was from neutrons having energies above the 500-keV cutoff for NTA film. The data in Table 6-20 show that NTA film should have provided a reasonable estimate of personnel neutron exposure in fields similar to those for glovebox workers in Rooms 111 and 211 and for the exit of the WTA transfer tunnel. Exposures from fields similar to that near the wall of the waste cask would have been underestimated. NTA film would not have shown any response in fields comparable to that associated with the TDF.

Table 6-20. Fraction of dose equivalent above NTA cutoff for several locations in the REDC.

Location	Fraction of dose equivalent above 500 keV
Face of glovebox in Building 7920, Room 111	91%
Side of glovebox in Building 7920, Room 211	94%
TDF viewing window (Building 7920)	0%
Limited Access Area waste cask (Building 7920)	72%
WTA transfer tunnel exit (Building 7920)	87%

6.3.4.2.2.9 Solid Waste Storage Area 5

In the late 1980s, there were two areas in SWSA 5 where personnel neutron exposures could occur. The first was the Waste Examination and Assay Facility (WEAF; Building 7824) where the waste assay equipment in use included a small deuterium-tritium (D-T) accelerator. The other was the TRU waste storage facility, which is a series of four concrete bunkers approximately 30 m behind the WEAF. At the time of the PNL measurements, three of the four bunkers had been filled and sealed. The fourth (northernmost) bunker was still in use and contained three casks of TRU waste. The casks contained californium and transcalifornium wastes, including berkelium and einsteinium, presumably from the REDC.

Characterization measurements were not attempted for the WEAF D-T accelerator, because the operating time required to get a reliable measurement would have significantly depleted the tritium target. Instead, WEAF workers were assumed to spend 10% of their time in the neutron field from the D-T accelerator. These exposures were evaluated using the TLD correction factors determined for the Holifield facility, a very conservative choice (MMES 1992). WEAF workers were assumed to spend the other 90% of their time in the field associated with the three waste casks in the storage bunker.

Two neutron spectral characterization measurements were made in the SWSA 5 TRU storage bunker. One was at 82 cm from the center cask in a row of three at a height of 1 m. The other was at approximately 41 cm from the wall shared with the adjacent bunker. This measurement was performed 4 m along the wall from the bunker entrance at a height of just over 1 m. This location was 5.7 m from the nearest of the three waste casks. Figure 6-15 and Table 6-21 indicate the neutron spectrum and contribution to the total dose equivalent from the four IREP neutron energy groups of interest for the waste casks, respectively. Figure 6-16 and Table 6-22 indicate spectrum and contribution results for the measurement along the bunker wall, respectively.

TLD irradiations performed at the same location as the spectral characterization for the three waste casks indicated a neutron-to-gamma dose ratio of approximately 2.8. TLD irradiations were not performed at the location near the shared wall.

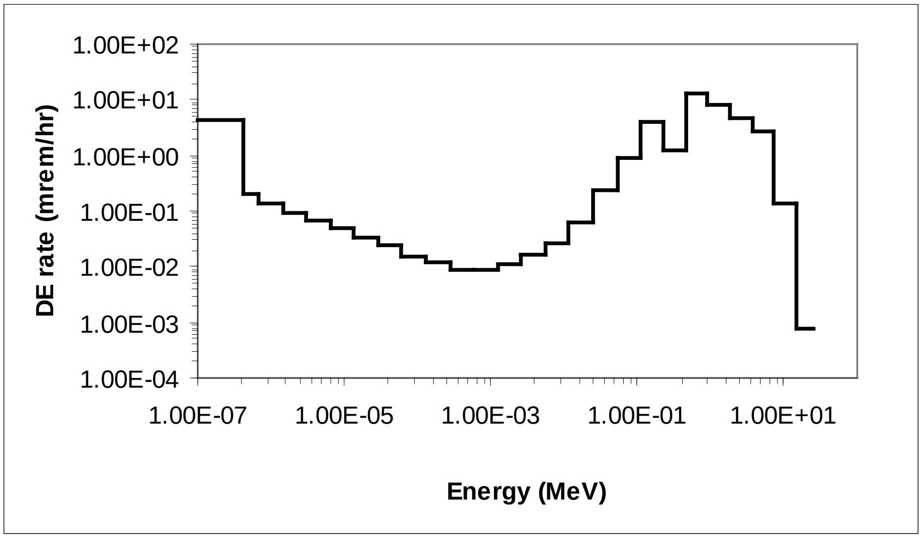

Figure 6-15. Neutron energy spectrum for waste casks in the SWSA 5 TRU storage bunker.

Table 6-21. Dose equivalent fractions for IREP neutron energy groups for waste casks in the SWSA 5 TRU storage bunker.

IREP neutron energy group (MeV)	**Dose equivalent fraction**
Measured values	
< 0.01	12.4%
0.01 – 0.1	2.9%
0.1 – 2.0	66.5%
2.0 – 20.0	18.2%
Claimant-favorable values	
< 0.01	12%
0.1 – 2.0	70%
2.0 – 20.0	18%

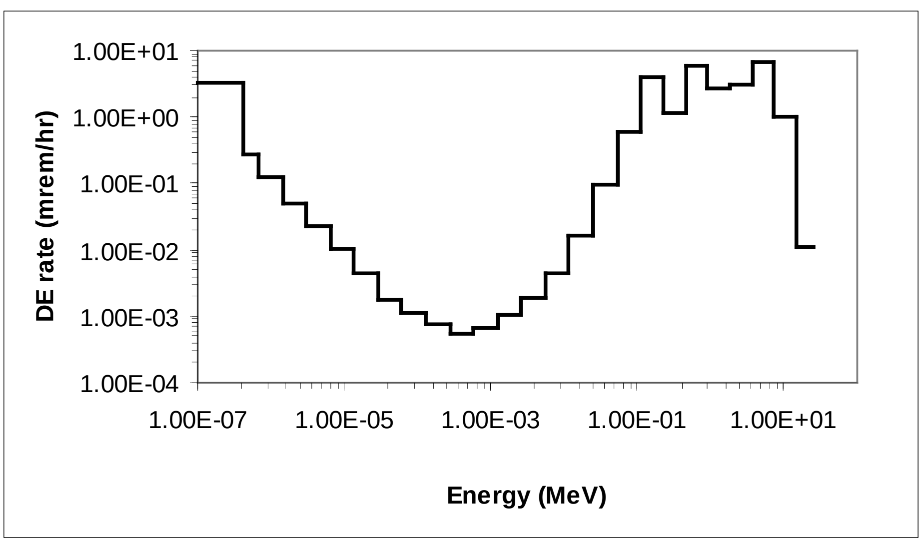

Figure 6-16. Neutron energy spectrum at a wall shared between two adjacent TRU storage bunkers at SWSA 5.

Table 6-22. Dose equivalent fractions for IREP neutron energy groups at a wall shared between two adjacent TRU storage bunkers at SWSA 5.

IREP neutron energy group (MeV)	Dose equivalent fraction
Measured values	
< 0.01	13.2%
0.01 – 0.1	2.3%
0.1 – 2.0	48.6%
2.0 – 20.0	35.9%
Claimant-favorable values	
< 0.01	13%
0.1 – 2.0	51%
2.0 – 20.0	36%

The fraction of the total dose equivalents measured for the waste casks and along the shared wall in the SWSA 5 TRU storage bunker that were from neutrons with energies above the 500-keV cutoff for NTA film were 68% and 65%, respectively. Thus, NTA film would have underestimated neutron exposures if used for personnel monitoring in similar fields.

6.3.4.2.3 Data from Neutron Dosimetry Records

The purpose of the characterization measurements performed by PNL at ORNL between 1989 and 1991 was to establish correction factors needed to compute neutron dose equivalent from the response of the neutron-sensitive elements of the CEDS albedo neutron dosimeters. The individual characterization measurements resulted in individual values of these TLD correction factors. These factors, and their associated neutron-to-photon dose ratios, are conservative and consider only photon exposure associated with the neutron-photon field. They are single measurements and do not provide information on the distribution of neutron dose or neutron-to-photon dose ratios. Therefore, neutron dosimetry results were obtained from ORNL for its various neutron workers groups to get additional information on neutron exposures and neutron-to-photon dose ratios for different facilities and job duties.

ORNL defines a number of different neutron groups, so appropriate TLD response-to-dose-equivalent correction factors can be applied when evaluating an individual's neutron dose. Correction factors for each group are determined based on worker's occupancy in different workplace neutron fields. There are currently 22 groups defined, 19 of which can be used to evaluate dose for ORNL workers. Some groups are used for ORNL facilities exclusively, and others are used for workers performing duties at ORNL or Y-12. There are also groups defined exclusively for other sites, such as Portsmouth and Y-12 (ORNL 1994). Some groups are defined based on an assumption of 100% occupancy in a specific neutron field. Others use weighted averages of occupancy in two or more fields. Occupancy in this context refers to time spent in the neutron field and not necessarily total time spent in a radiation area.

The ORNL external dosimetry group queried its database of personnel dosimetry records to produce a set of quarterly neutron and photon dose results for all of the various neutron worker groups. The query returned 20,037 records covering the period from 1990 through the second quarter of 2004. Each record consisted of five fields: Calendar year, calendar quarter, neutron group, neutron dose, and gamma dose. ORNL determined the gamma dose values from the response of the TLD-700 chip below 300 mg/cm^2 of plastic. The gamma dose values were determined from the neutron dosimeter's response rather than the gamma dose result from the worker's beta-gamma dosimetry because workers are sometimes issued more than one neutron dosimeter during a quarterly monitoring period. This occurs when a neutron worker's duties involve neutron exposure corresponding to more than one of the defined neutron worker groups. It can also occur when workers have interim dose evaluations performed at the request of health physics staff because higher dose is anticipated. The photon dose results determined from the neutron dosimeters should still represent the individual's total gamma dose (and not just that associated with neutron fields) because ORNL requires both dosimeters be worn regardless of the work being done. Therefore, the neutron-to-photon dose ratios derived from the dosimetry records (see Attachment 6D) are relative to total photon dose rather than just that received in mixed neutron-photon fields. These values differ from those derived from the PNL characterization data (see Attachment 6C) in that they inherently account for the workers' occupancy and thus can be applied directly to an individual's reported photon dose.

For neutron groups for which there were an adequate number of positive results, the dosimetry data obtained from ORNL were used to establish distributions of neutron dose and neutron-to-photon dose ratios. Typical values (rather than distributions) were determined from the dosimetry data for groups for which there were only a few results. Summary statistics, such as the fraction of total results that showed a positive neutron dose, minimum and maximum dose values, and so forth were determined for each group for which data were available. This analysis used measured TLD results for dosimetry data. The results do not include doses assigned through such other means as investigations or estimations. The reporting limit for quarterly neutron dose is 10 mrem. Photon dose values down to 1 mrem (net) were computed, but values below 10 mrem (above background) were not used.

It should be noted that of the 20,037 neutron dosimetry records obtained for the period from 1990 through the second quarter of 2004, 16,785 of them (84%) showed a reported neutron dose of zero, which means the individual's neutron dose evaluated to less than 10 mrem. Researchers at HFIR account for 40% of the records. For these individuals, only 5 out of 7992 (less than 0.1%) had an assigned neutron dose in excess of the reporting limit (the highest of the five was 14 mrem).

The sections that follow discuss the results obtained from the dosimetry data provided by ORNL for the various neutron groups. Comparisons between the neutron-to-photon dose ratios derived from the personnel dosimetry records and those from the PNL characterization measurements are discussed where applicable. For the ratios derived from the dosimetry records, in general the comparisons are made relative to the geometric mean of the distribution of these values.

Attachment 6D summarizes the neutron-to-photon dose ratios for each group, giving the geometric mean and geometric standard deviation (GSD) for the groups for which there were a sufficient number of results to establish a distribution. Minimum and maximum values are given. Attachment 6E summarizes neutron dose information, also giving the geometric mean and GSD for the groups for which there were a sufficient number of results to establish a distribution. The maximum neutron dose and the fraction of the total number of results that were greater than or equal to the 10-mrem reporting level are also given. In general, dose reconstructors should use the information provided in Attachment 6D instead of that in Attachment 6C, as practical, because the data in Attachment 6D are expressed as distributions and account for occupancy in all photon fields. In most cases, the neutron-to-photon dose ratios derived from the personnel dosimetry records are significantly lower than those implied by the characterization measurements performed by PNL. Exceptions are ratios for the TDF used by REDC workers (Group 17 below) and the HFIR (Group 5). The neutron-to-photon dose ratios indicated by the dosimetry records for TDF users were significantly higher than that implied by the PNL characterization, and the values for the HFIR were largely comparable. However, there were very few positive dose results among the records for HFIR personnel to use to compute a neutron-to-photon dose ratio.

6.3.4.2.3.1 REDC Workers: Groups 2, 3, 4, and 17

ORNL defines the following neutron groups for REDC workers:

- Group 2: Analytical chemistry (Building 7920 glovebox workers)
- Group 3: Packaging and shipping workers in Building 7930
- Group 4: REDC general area
- Group 17: TDF workers

REDC analytical chemistry (Group 2) workers prepare and analyze samples of materials at various stages of the production process. Most of this work takes place in gloveboxes, so ORNL makes the assumption that 100% of any neutron exposure for these individuals is from glovebox work. To be conservative, the TLD correction factors used are those determined for the glovebox in Room 111 rather than those for the box in Room 211.

Group 2 workers accounted for 1,535 (8%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 1,535 records, 1,197 (78%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 1,104 records for which both the neutron and photon dose were positive. Figure 6-17 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 1,104 results. The ratios are approximately lognormally distributed with a geometric mean of 1.6 and a GSD of 2.2. They range from 0.1 to 8.8.

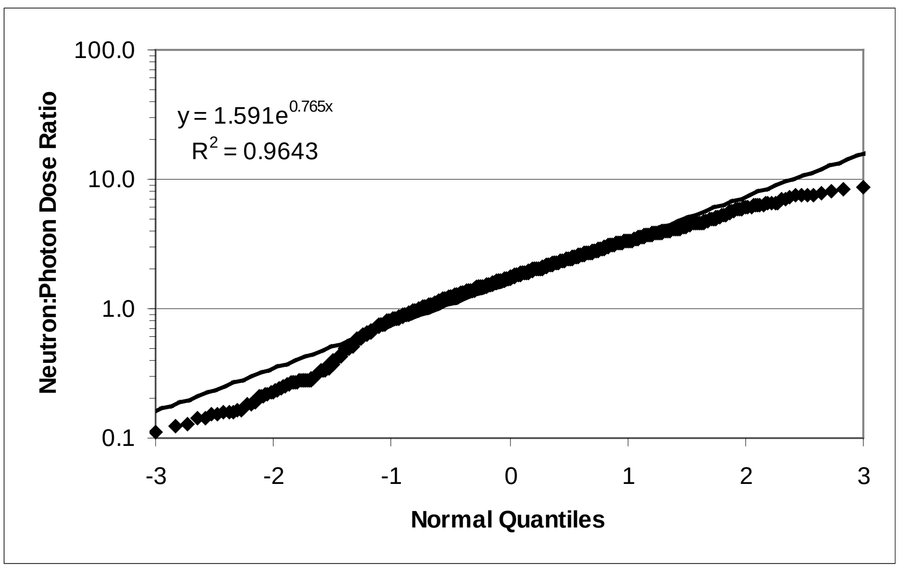

Figure 6-17. Normal probability plot for Group 2 neutron-to-photon dose ratios.

Figure 6-18 is a normal probability plot of the 1,197 positive neutron dose values for Group 2 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 50 mrem and a GSD of 2.2. They range from 10 to 584 mrem.

Workers in the general area of the REDC include control room staff and office and support personnel who normally do not work with product materials. These individuals are assigned to Neutron Group 4. In terms of occupancy in neutron fields, Group 4 workers are assumed to spend 60% of their time in the control room, 20% in the Limited Access Area behind the hot cells, and 20% in the WTA (MMES

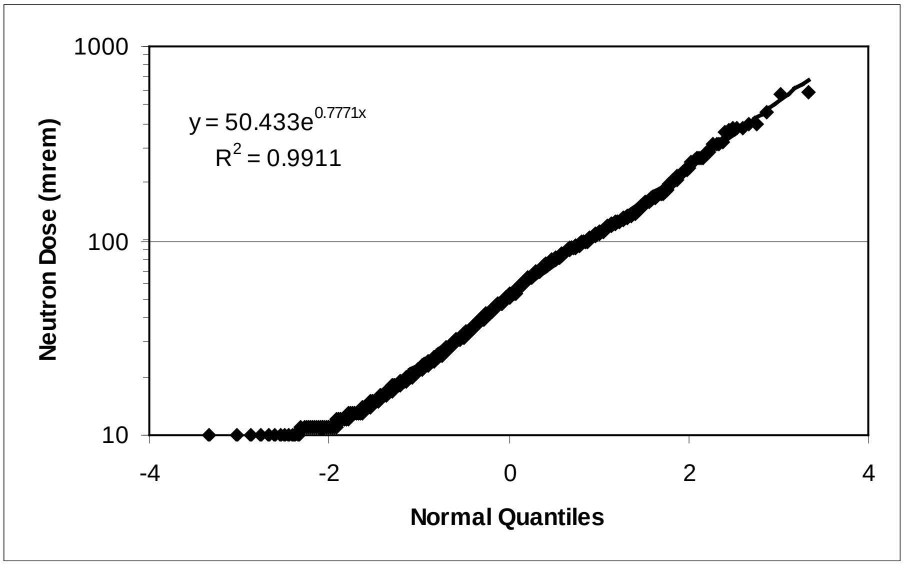

Figure 6-18. Normal probability plot for Group 2 neutron doses.

1992). The neutron spectrum assumed for the control room was a 50:50 combination of that determined for the TDF (as a proxy for the shielded hot cells) and that for the glovebox in Room 111 (as a proxy for the gloveboxes in Room 109). No spectral measurements were possible in the REDC control room because the neutron dose rate is so low (about 0.1 mrem/hr).

Group 4 workers accounted for 2,730 (14%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 2,730 records, 1,070 (39%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 982 records for which both the neutron and photon dose were positive. Figure 6-19 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 982 results. The ratios are approximately lognormally distributed with a geometric mean of 1.1 and a GSD of 1.7. They range from 0.1 to 3.7.

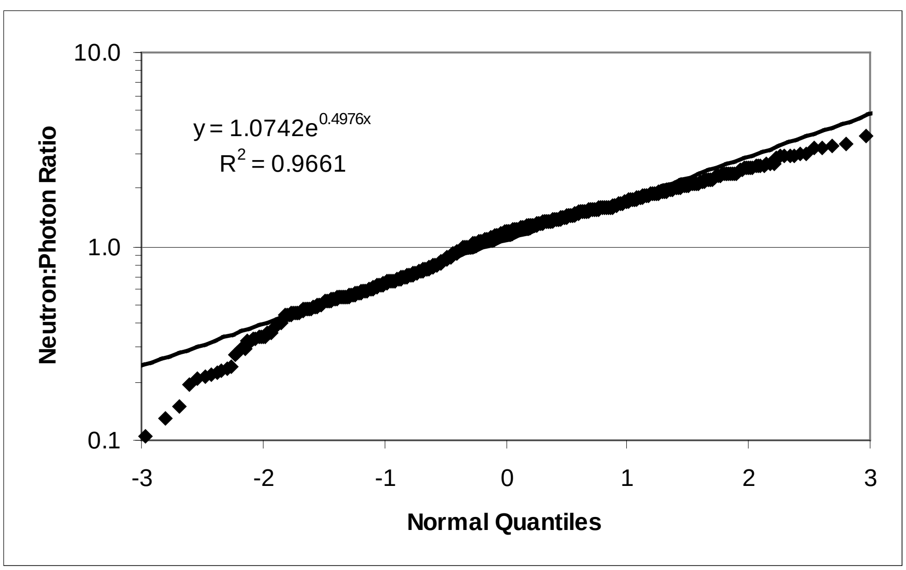

Figure 6-19. Normal probability plot for Group 4 neutron-to-photon dose ratios.

Figure 6-20 is a normal probability plot of the 1,070 positive neutron dose values for Group 4 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 23 mrem and a GSD of 1.6. They range from 10 to 91 mrem.

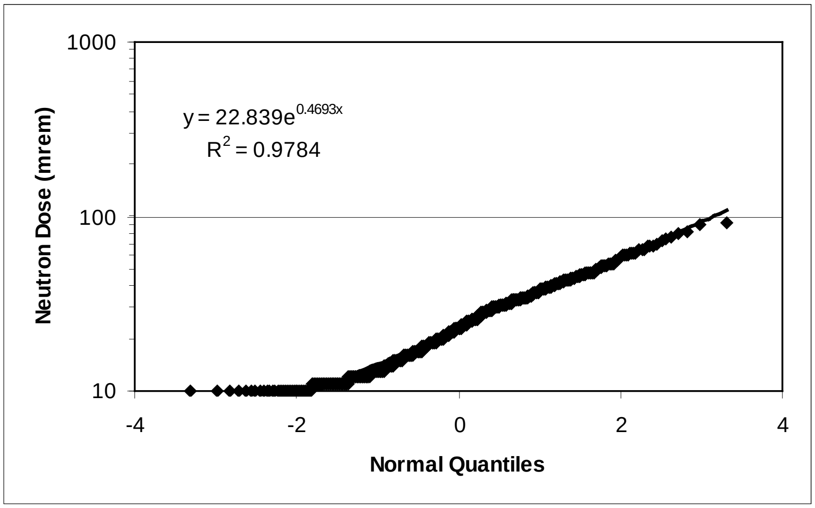

Figure 6-20. Normal probability plot for Group 4 neutron doses.

All of the neutron dose for workers in Building 7930 (Group 3) is assumed to be from the field determined for the glovebox in the low-scatter geometry. Similarly, 100% of the neutron dose for Group 17 workers using the REDC TDF is assumed to be from the spectrum determined for that unit.

Group 3 workers accounted for 81 (0.4%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 81 records, 79 (98%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 65 records for which both the neutron and photon dose were positive. Figure 6-21 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 65 results. The ratios are approximately lognormally distributed with a geometric mean of 2.5 and a GSD of 1.6. They range from 1.0 to 8.1.

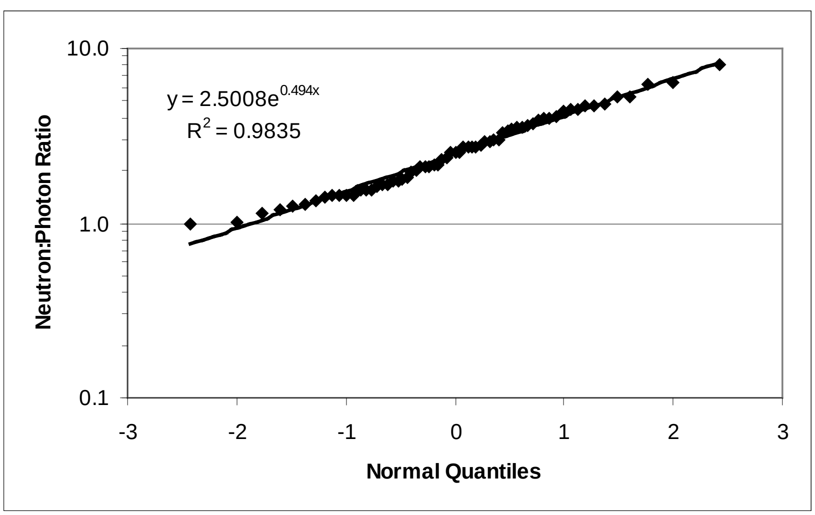

Figure 6-21. Normal probability plot for Group 3 neutron-to-photon dose ratios.

Figure 6-22 is a normal probability plot of the 79 positive neutron dose values for Group 3 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 42 mrem and a GSD of 1.8. They range from 11 to 137 mrem.

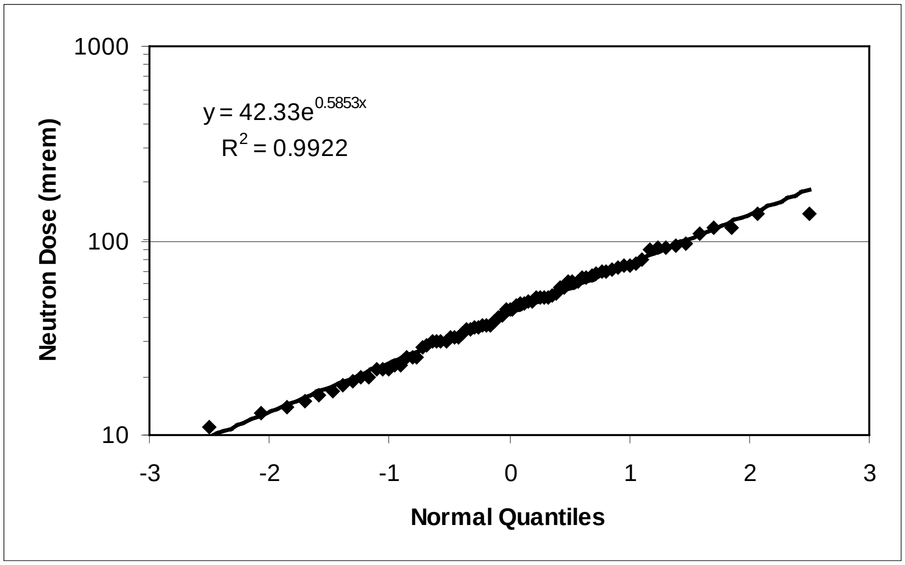

Figure 6-22. Normal probability plot for Group 3 neutron doses.

Group 17 workers accounted for 264 (1%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 264 records, 111 (42%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 86 records for which both the neutron and photon dose were positive. Figure 6-23 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 86 results. The ratios are approximately lognormally distributed, having a geometric mean of 2.1 and a GSD of 2.3. They range from 0.4 to 8.0.

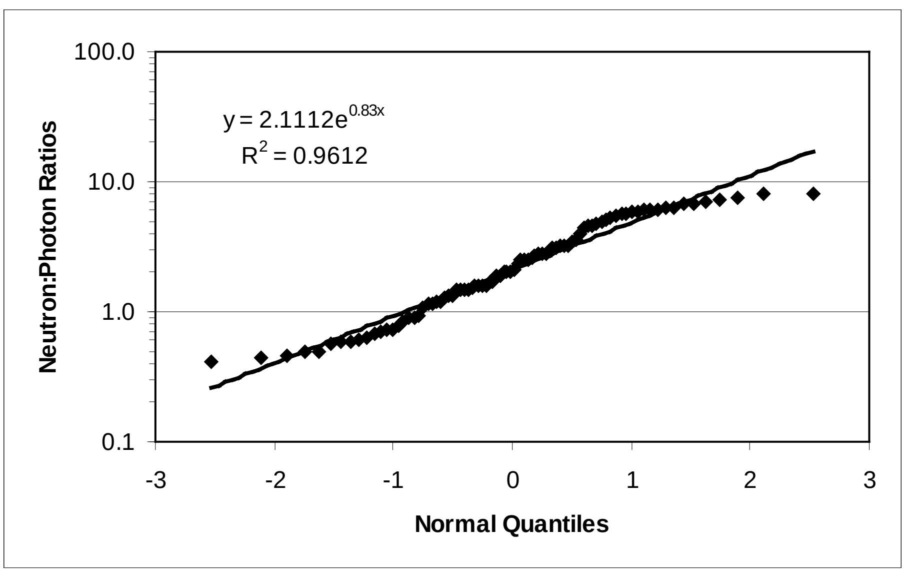

Figure 6-23. Normal probability plot for Group 17 neutron-to-photon dose ratios.

Figure 6-24 is a normal probability plot of the 111 positive neutron dose values for Group 17 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 37 mrem and a GSD of 2.3. They range from 10 to 230 mrem.

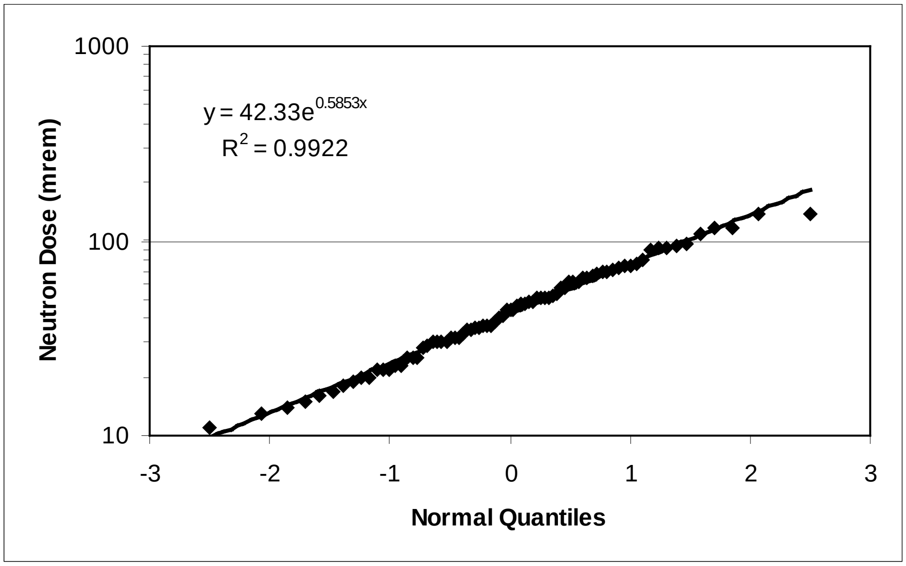

Figure 6-24. Normal probability plot for Group 17 neutron doses.

Comparison of the neutron-to-photon dose ratios computed from the personnel dosimetry records with those derived from the PNL characterization measurements shows the values from the dosimetry data are significantly lower for Groups 2, 3, and 4 and significantly higher for Group 17. The lower values are expected because the PNL measurements were performed in conservatively chosen locations with high neutron dose rates. Under real work conditions, workers do not spend all of their time in these locations and the neutron-to-photon dose ratios derived from the dosimetry data inherently account for occupancy in all photon fields and not just those associated with neutron sources. However, the neutron dose rate at the location where PNL performed its characterization measurements for the TDF (Group 17) was much lower than the corresponding photon dose rate. That the neutron-to-photon ratio determined from the personnel dosimetry records is much higher than that implied by the PNL measurements suggests that workers receive neutron exposures from other sources or other geometries where the neutron and photon dose rates are more comparable. Workers assigned to Group 17 are assumed to spend 100% of their time in the neutron spectrum determined for the TDF for the purpose of neutron dose evaluation.

6.3.4.2.3.2 Group 7

Group 7 is a generic group used for personnel working in neutron fields for which specific characterization information is not available and exposure from high-energy neutrons is not expected. The TLD calibration factors used for dose evaluation for personnel assigned to Group 7 are based on a D_2O-moderated ^{252}Cf spectrum.

Group 7 workers accounted for 1,329 (7%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 1,329 records, 37 (3%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 24 records for which both the neutron and photon dose were positive. Figure 6-25 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 24 results. The ratios have a geometric mean of 0.3 and a GSD of 3.2. They

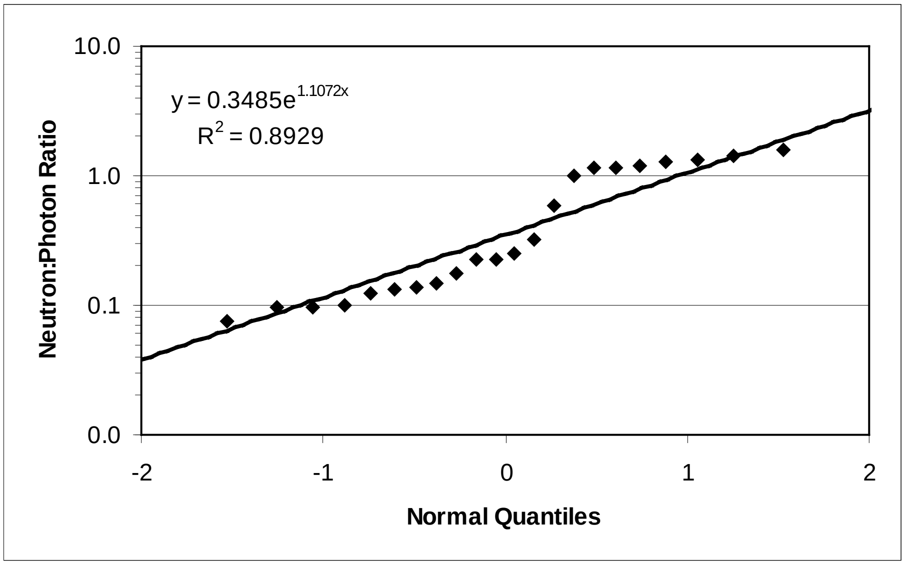

Figure 6-25. Normal probability plot for Group 7 neutron-to-photon dose ratios.

range from 0.1 to 2.2. The distribution of the Group 7 neutron-to-photon dose ratios appears bimodal, but this could be an artifact of the small number of values. The neutron doses associated with the 24 ratios range from 11 to 27 with the higher ratios associated with the lower doses. The lower ratios are associated with statistically better data—the photon doses in particular—so using the geometric mean of 0.3 with the large GSD should be valid.

Figure 6-26 is a normal probability plot of the 37 positive neutron dose values for Group 7 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 14 mrem and a GSD of 1.3. They range from 10 to 27 mrem.

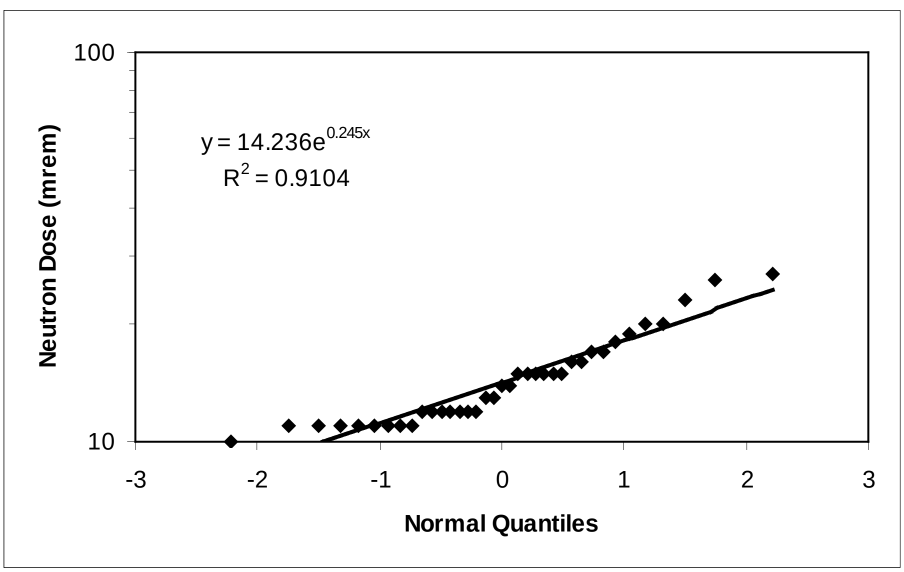

Figure 6-26. Normal probability plot for Group 7 neutron doses.

6.3.4.2.3.3 Unshielded Source Handlers: Group 9

Group 9 is a generic group defined for workers whose duties involve transfers or other operations with unshielded isotopic neutron sources. The TLD response-to-neutron-dose conversion factors for Group 9 reflect a weighted average of bare Pu:Be and D_2O-moderated ^{252}Cf fission spectra in an 80:20 ratio (ORNL 1994).

Group 9 workers accounted for 1,687 (8%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 1,687 records, 405 (24%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 163 records for which both the neutron and photon dose were positive. Figure 6-27 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 163 results. The ratios are approximately lognormally distributed with a geometric mean of 1.0 and a GSD of 2.8. They range from 0.1 to 14.1.

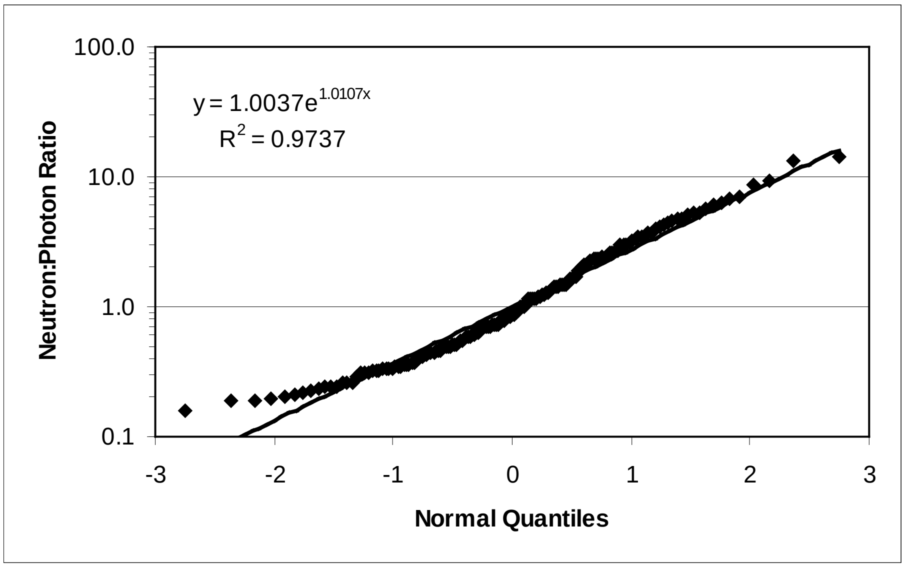

Figure 6-27. Normal probability plot for Group 9 neutron-to-photon dose ratios.

Figure 6-28 is a normal probability plot of the 405 positive neutron dose values for Group 9 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 24 mrem and a GSD of 1.8. They range from 10 to 212 mrem.

6.3.4.2.3.4 SWSA 5 workers: Groups 11, 12, and 21

ORNL defines three groups for workers exposed to neutrons from TRU wastes handled at SWSA 5:

- Group 11: Operators and other SWSA 5 personnel whose duties do not include moving wastes into the area
- Group 12: Personnel who transfer TRU wastes from other areas to SWSA 5
- Group 21: Operators and health physics staff associated with the transfer of drums into Building 7879.

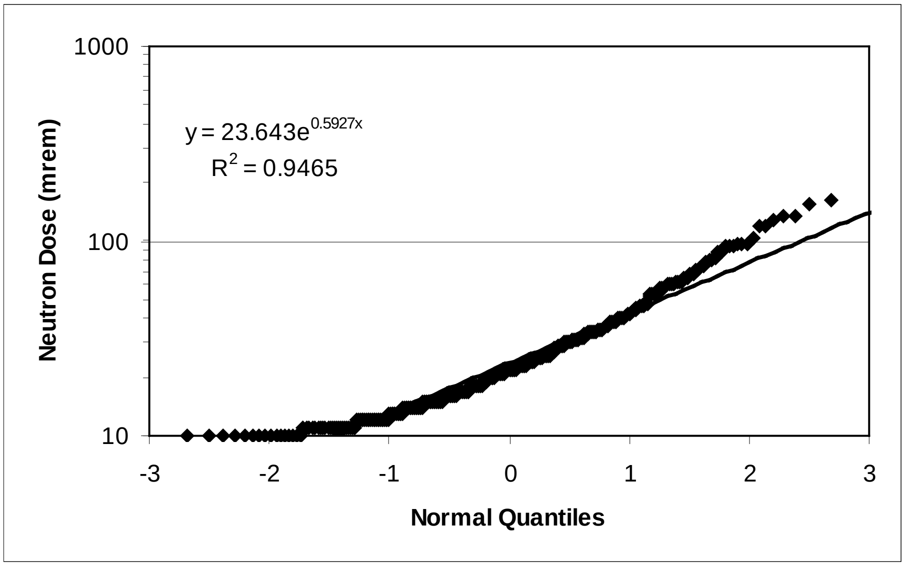

Figure 6-28. Normal probability plot for Group 9 neutron doses.

Personnel assigned to Group 11 for the purpose of neutron dose evaluation could be exposed to neutrons from the SWSA 5 TRU waste storage bunker or the intermittent neutron field associated with the small D-T accelerator housed at the WEAF. The TLD response-to-neutron-dose conversion factors for Group 11 reflect a weighted average of the spectrum measured for the TRU storage bunker and that for the Holifield Heavy Ion Facility in a 90:10 ratio (ORNL 1994). The high-energy component associated with the Holifield facility was included for conservatism to account for potential exposure from the D-T accelerator.

Personnel assigned to Group 12 are assumed to receive 100% of their neutron exposure from the spectrum measured for the TRU storage bunker, and those assigned to Group 21 are assumed to receive 100% of their exposure from the spectrum determined for waste drums handled in Building 7879.

Group 11 workers accounted for 448 (2%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 448 records, 38 (8%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were nine records for which both the neutron and photon dose were positive. The neutron-to-photon dose ratios associated with these nine records show a geometric mean of 1.1 and a GSD of 1.9. They range from 0.3 to 2.4. A normal probability plot was not created showing the distribution of the neutron-to-photon dose ratios given the small number of data points.

Figure 6-29 is a normal probability plot of the 38 positive neutron dose values for Group 11 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 14 mrem and a GSD of 1.3. They range from 10 to 26 mrem.

Group 12 workers accounted for 501 (3%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 501 records, 3 (1%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 3 records for which both the neutron and photon dose were positive. The neutron-to-photon dose ratios associated with these 3 records are 0.2, 0.7, and 1.1. The measured neutron doses associated with these three ratios were 17, 30, and 16 mrem, respectively.

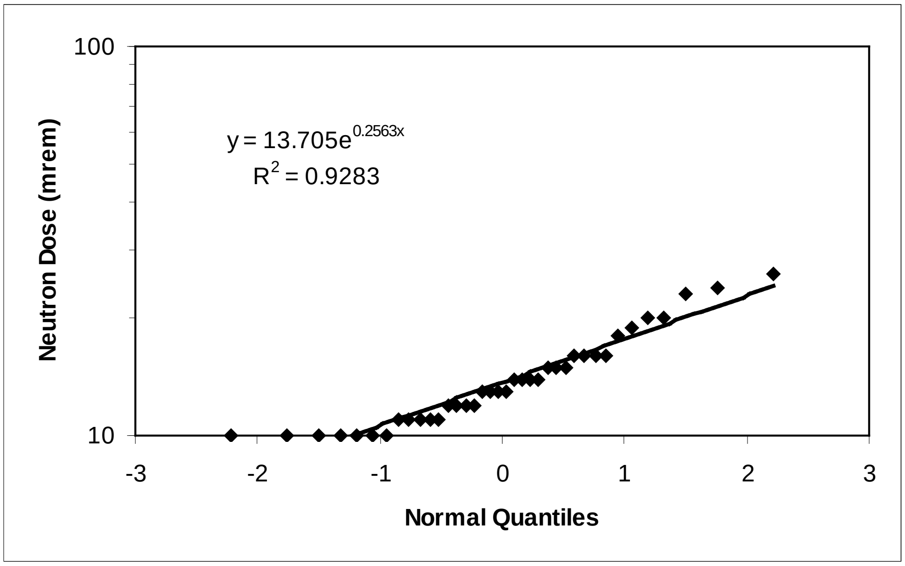

Figure 6-29. Normal probability plot for Group 11 neutron doses.

Group 21 workers accounted for 1,148 (6%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 1,148 records, 160 (14%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 148 records for which both the neutron and photon dose were positive. Figure 6-30 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 148 results. The ratios are approximately lognormally distributed with a geometric mean of 0.5 and a GSD of 1.9. They range from 0.1 to 2.8.

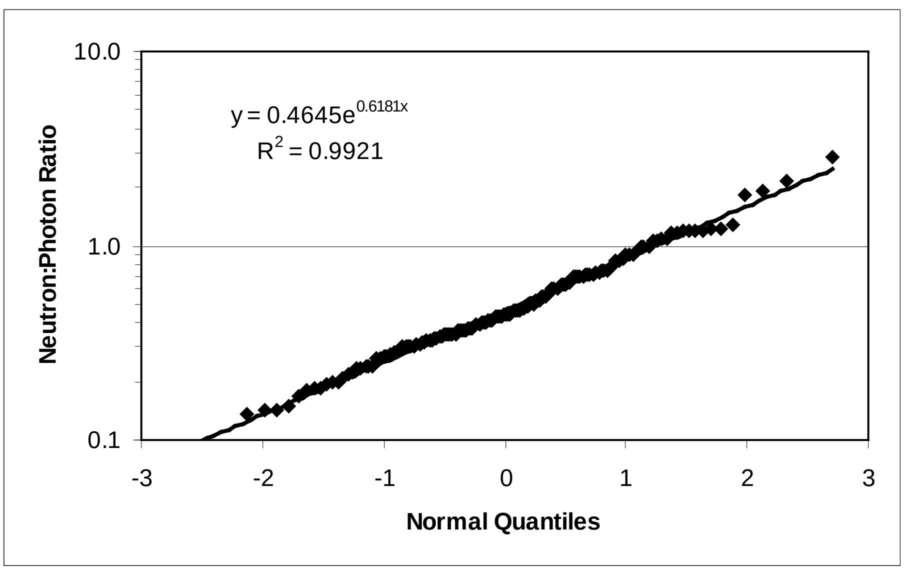

Figure 6-30. Normal probability plot for Group 21 neutron-to-photon dose ratios.

Figure 6-31 is a normal probability plot of the 160 positive neutron dose values for Group 21 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 17 mrem and a GSD of 1.5. They range from 10 to 58 mrem.

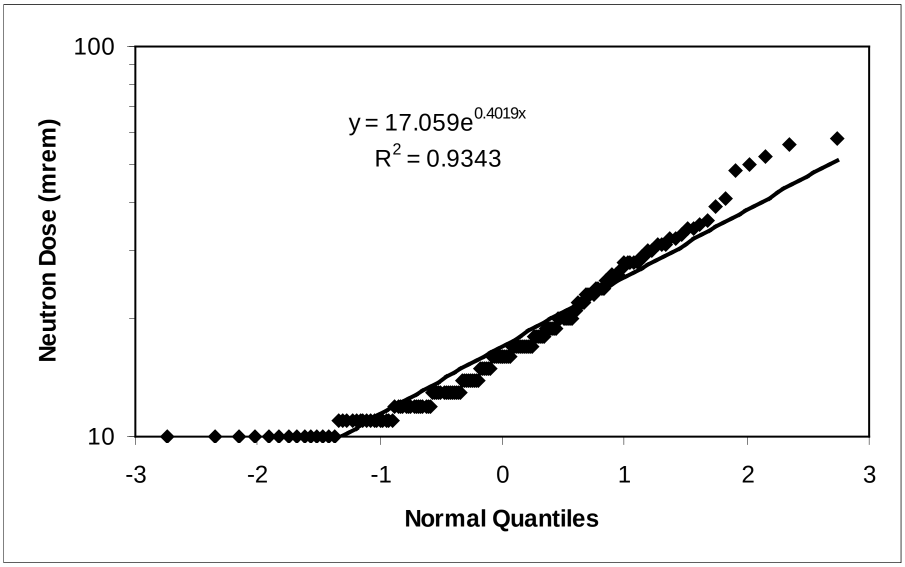

Figure 6-31. Normal probability plot for Group 21 neutron doses.

Comparison between the neutron-to-photon dose ratio determined from the personnel dosimetry records and that from the PNL characterization measurements for Group 11 show the value from the dosimetry data to be about a factor of 3 lower. However, the distribution of neutron-to-photon dose ratios determined for Group 11 is based on a limited number of results, as few people in this group receive neutron exposures in excess of the reporting limit. There were not enough records to establish a ratio for Group 12 workers, and workplace characterization data are not currently available for Building 7879 to allow a comparison for Group 21.

6.3.4.2.3.5 Holifield and ORELA Facilities: Groups 6, 18, and 19

ORNL defines three groups for workers exposed to neutrons from the Holifield and ORELA accelerator facilities:

- Group 6: Personnel that work exclusively at the Holifield facility
- Group 18: Personnel that work exclusively at the ORELA facility
- Group 19: Personnel that work at both facilities

The TLD response-to-neutron-dose conversion factors for Groups 6 and 18 reflect the spectra determined for those the two respective facilities. The factors for Group 19 reflect a weighted average of the spectra measured for the ORELA and Holifield facilities in a 90:10 ratio (ORNL 1994).

Group 6 workers accounted for 407 (2%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 407 records, 8 (2%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were four records for which both the neutron and photon dose were positive. The neutron-to-photon dose ratios associated with these 4 records are 0.7, 1.1, 1.2, and 1.5. The measured neutron doses associated with these three ratios were 17, 18, 28, and 16 mrem, respectively.

Group 18 workers accounted for 191 (1%) of the 20,037 neutron dosimetry records obtained from ORNL. None of the 191 records showed a neutron dose greater than or equal to the 10-mrem reporting limit.

Group 19 workers accounted for 290 (1%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 290 records, 4 (1%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were three records for which both the neutron and photon dose were positive. The neutron-to-photon dose ratios associated with these 3 records are 0.7, 0.9, and 1.9. The measured neutron doses associated with these three ratios were 20, 13, and 47 mrem, respectively.

There were not enough positive results in the dosimetry records to establish distributions of neutron-to-photon dose ratios for any of the three accelerator groups, but the few records available suggest that the actual ratios for Holifield facility workers are much lower than the value indicated by the conservative characterization performed by PNL. The neutron-to-photon dose ratio derived from the PNL characterization data was 19, whereas the personnel dosimetry results indicate values between 1.0 and 2.0.

6.3.4.2.3.6 Shielded Source Users: Group 13

Group 13 is used to evaluate neutron dose for personnel who work with shielded sources, such as well-loggers and some Instrumentation and Controls technicians. The TLD response-to-neutron-dose conversion factors for Group 13 reflect a weighted average of D_2O-moderated ^{252}Cf and bare Pu:Be spectra in a 95:5 ratio (ORNL 1994).

Group 13 workers accounted for 675 (3%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 675 records, 44 (7%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 37 records for which both the neutron and photon dose were positive. Figure 6-32 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 37 results. The ratios are approximately lognormally distributed with a geometric mean of 0.5 and a GSD of 2.4. They range from 0.1 to 3.5.

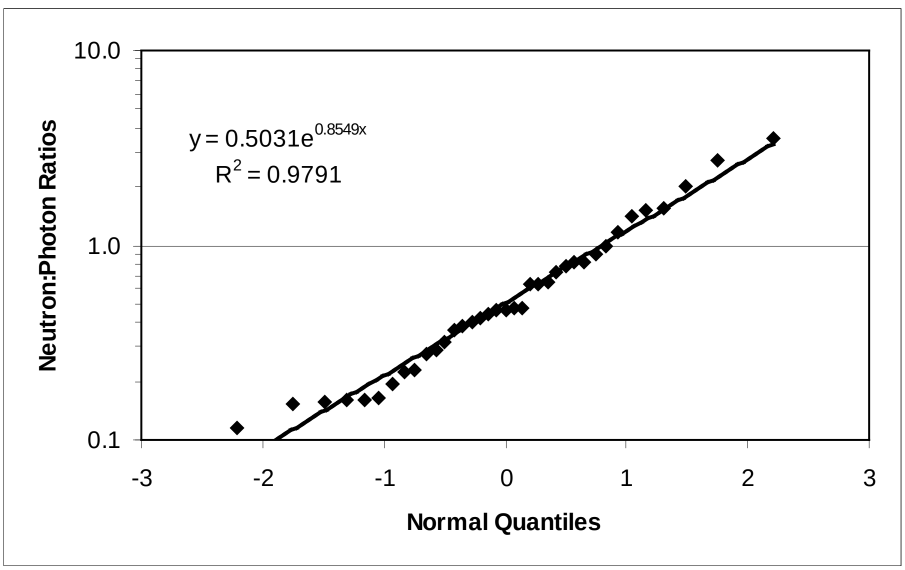

Figure 6-32. Normal probability plot for Group 13 neutron-to-photon dose ratios.

Figure 6-33 is a normal probability plot of the 44 positive neutron dose values for Group 13 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 18 mrem and a GSD of 1.6. They range from 10 to 53 mrem.

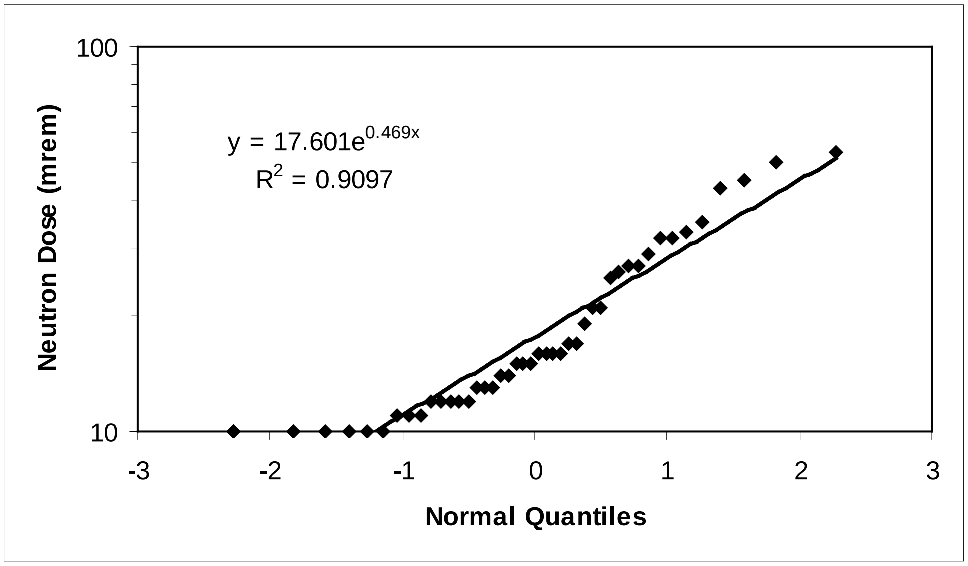

Figure 6-33. Normal probability plot for Group 13 neutron doses.

6.3.4.2.3.7 Neutron Source Make-up Workers: Group 14

Group 14 is used to evaluate neutron dose for personnel fabricating and handling isotopic neutron sources in ORNL's isotope production (Isotope Circle) facilities, Building 3038. The TLD response-to-neutron-dose conversion factors for Group 14 reflect a weighted average of shielded Pu:Be spectra.

Group 14 workers accounted for 222 (1%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 222 records, 72 (32%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 45 records for which both the neutron and photon dose were positive. Figure 6-34 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 45 results. The ratios are approximately lognormally distributed with a geometric mean of 1.1 and a GSD of 2.2. They range from 0.2 to 9.1.

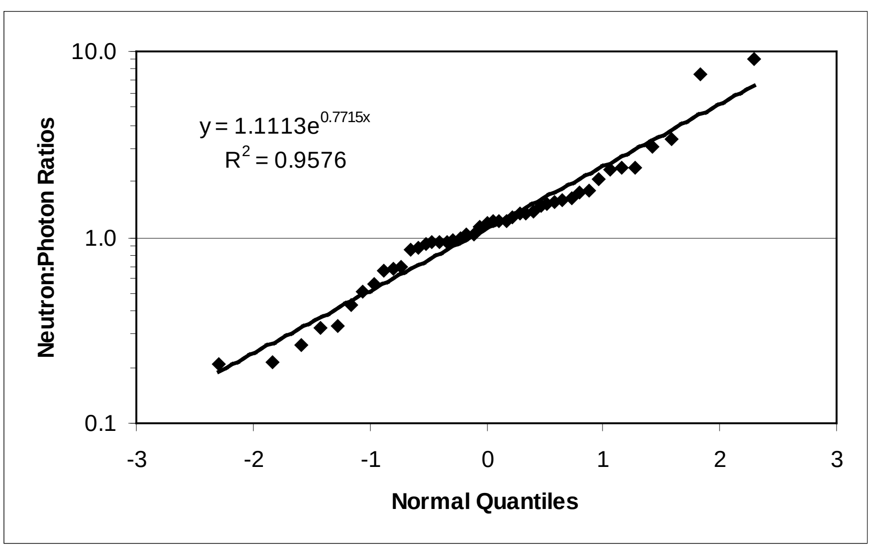

Figure 6-34. Normal probability plot for Group 14 neutron-to-photon dose ratios.

Figure 6-35 is a normal probability plot of the 72 positive neutron dose values for Group 14 workers in the period from 1990 through the second quarter of 2004. These data are approximately lognormally distributed with a geometric mean of 25 mrem and a GSD of 1.8. They range from 10 to 182 mrem.

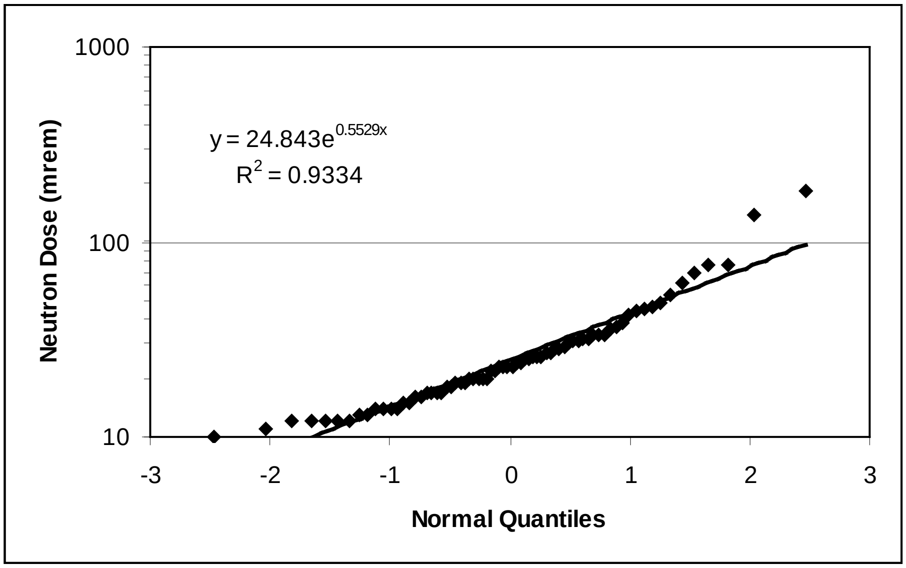

Figure 6-35. Normal probability plot for Group 14 neutron doses.

The neutron-to-photon dose ratio determined for Group 14 from the dosimetry records is significantly lower than that from the PNL characterization measurements.

6.3.4.2.3.8 Other Workers: Groups 1, 5, 10, and 22

There were not enough positive results recorded in any of these groups between 1990 and the second quarter of 2004 to allow distributions of either neutron-to-photon dose ratios or neutron doses to be established.

Group 1 is used to evaluate dose for workers at RADCAL. These individuals accounted for 406 (2%) of the 20,037 neutron dosimetry records obtained from ORNL. Of the 406, 5 (1%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were four records for which both the neutron and photon dose were positive. The neutron-to-photon dose ratios associated with these four records are 0.6, 0.9, 1.3, and 1.6. The measured neutron doses associated with these four ratios were 14, 26, 14, and 28 mrem, respectively.

Group 5 is used to evaluate doses for worker and researchers at HFIR. This group represents the largest number of individuals monitored for neutron exposure at ORNL, accounting for 7992 (40%) of the 20,037 neutron dosimetry records obtained. However, only 5 (<0.1%) of these 7992 records showed a positive neutron dose from 1990 through the second quarter of 2004, and all 5 values were at or near the 10-mrem reporting limit. The largest of the five values was 14 mrem. In addition, there were five records for which both the neutron and photon dose were positive. Four of these records gave a neutron-to-photon dose ratio of 0.1, and the other gave a value of 0.2.

Group 10 is used to evaluate neutron dose for individuals whose duties include work in the Building 3100 storage vault. These individuals accounted for only 53 (0.3%) of the 20,037 records obtained from ORNL. Only 1 of the 53 results exceeded the 10-mrem reporting limit in the period from 1990 through the second quarter of 2004 (the result was 11 mrem). There were no individuals for whom both the neutron and photon doses were positive.

Group 22 was created primarily to support dose evaluations for Instrumentation and Control technicians exposed to isotopic neutron sources in both moderated and unmoderated configurations. The TLD response-to-neutron-dose conversion factors for Group 22 reflect a weighted average of bare Pu:Be and D_2O-moderated ^{252}Cf fission spectra in a 20:80 ratio (ORNL 1994). Workers assigned to Group 22 represented only 48 (0.2%) of the 20,037 records obtained from ORNL. Of the 48 records, 11 (2%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were only two records for which both the neutron and photon doses were positive. The neutron-to-photon dose ratios associated with these two records are 1.1 and 1.4, which correspond to neutron dose values of 25 and 17 mrem, respectively.

6.3.4.2.3.9 Summary for all groups

Of the 20,037 neutron dosimetry records obtained from ORNL, 3,252 (16%) showed a neutron dose greater than or equal to the 10-mrem reporting limit. There were 2,684 records for which both the neutron and photon dose were positive. Figure 6-36 is a normal probability plot showing the distribution of the neutron-to-photon dose ratios determined from these 2684 results. The ratios are approximately lognormally distributed with a geometric mean of 1.2 and a GSD of 2.2. They range from 0.1 to 14.1.

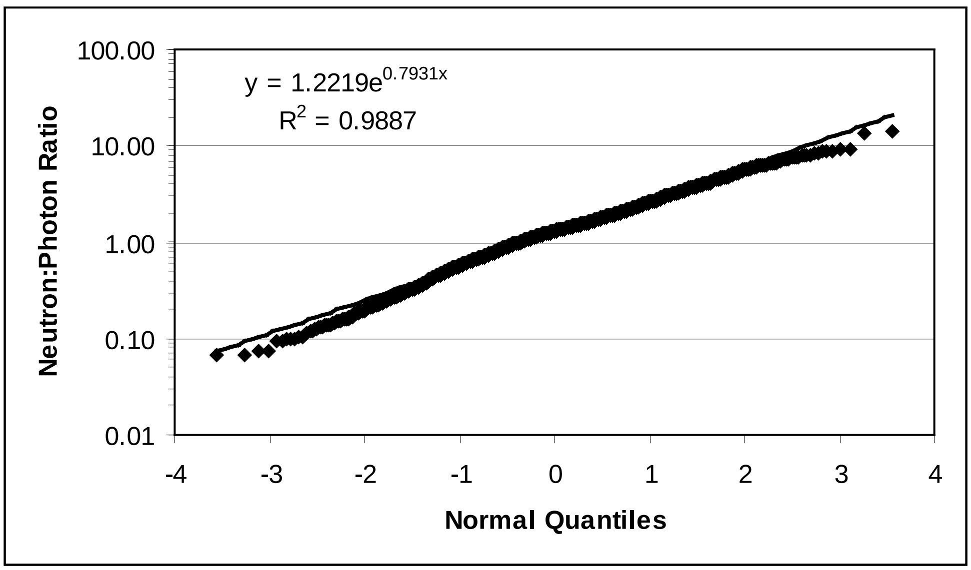

Figure 6-36. Normal probability plot for neutron-to-photon dose ratios derived from ORNL personnel neutron monitoring records for 1990 – 2004.

Figure 6-37 is a normal probability plot of the 3,252 positive neutron dose values from the neutron dosimetry records obtained from ORNL for the period from 1990 through the second quarter of 2004. The data are approximately lognormally distributed with a geometric mean of 31 mrem and a GSD of 2.1. They range from 10 to 584 mrem.

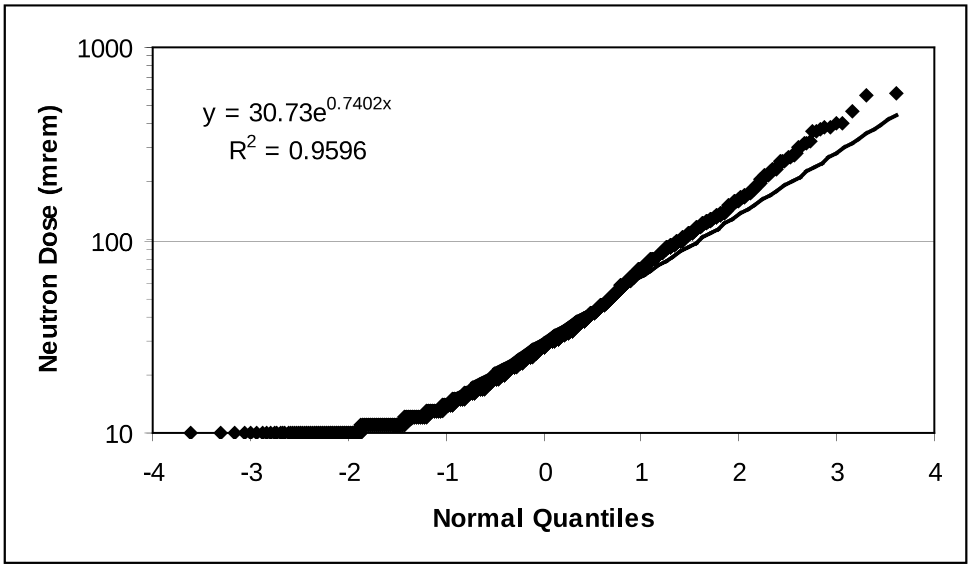

Figure 6-37. Normal probability plot for neutron doses ≥ 10 mrem for ORNL workers in the period from 1990 – 2004.

6.4 REPORTED QUANTITIES AND ADJUSTMENTS TO REPORTED DOSE

6.4.1 Reported Quantities and Associated Algorithms: Beta-Gamma

One version of the Individual Meter Record cards used to record film badge and pocket meter readings in the 1944 to 1946 period included columns labeled G and B. The G (gamma) value was the response of the film badge under the cadmium shield as determined from ^{226}Ra calibration curves. This value is analogous to the deep dose. The B (beta) value was determined from ^{226}Ra calibration curves; however, the meaning of this value is uncertain. Deal and Hart (1949) assert that the B value was simply the OW reading from the dosimeter, the same as the W or OW readings recorded in the years that followed. In addition, Deal and Hart state that it was an early practice not to record the B value unless the film density was at least twice that of the G value (recorded as S on other forms). However, Hart (1966) asserts that the B value was the computed difference between the dosimeter response from under the OW and that from under the cadmium shield (the G value). Review of claim files supports the case of the B value being the difference between the OW and S readings; therefore, dose reconstructors should interpret it in this fashion. Thus, G and B should be summed if the OW result from the reading of a particular dosimeter is desired. NOTE: (1) Treating the B value as the difference between the window and shield results is the claimant-favorable choice; (2) other versions of the Individual Meter Record cards from that period (1944 to 1946) had data columns for film badge results labeled W and S (corresponding to the OW and cadmium-filter responses, respectively) rather than G and B.

After 1946, ORNL practice was apparently to record results from the two-element film badges as OW and S, referring to readings behind the OW and the cadmium filter, respectively. The OW readings were sometimes called W. Data columns on the cards were used to record results for the sensitive and insensitive films in the beta-gamma badges. These results were noted as FILM [results from neutron dosimeters (NTA film) were called NEUTRON]. Values of S and OW (or W) were recorded for the sensitive beta-gamma films and for the neutron dosimeters. Apparently, only S values were recorded for the insensitive film, which was used only if the sensitive film *blacked out*. In 1949, threshold (maximum dose) values were administratively set to 5,000 mR(ep) and 20,000 mR(ep) for sensitive and insensitive films, respectively. These values, considered the upper range of measurement for the respective films, were entered on the data cards in the event a blacked-out film was encountered. Other notations indicated anomalies such as fogging of the film, evidence of X-ray (low-energy photon) exposure, contamination, etc. NOTE: By 1952, the threshold value for a blacked-out sensitive film appears to have been increased to 10,000 mR(ep) (Craft, Ledbetter, and Hart 1952).

Beginning in the second half of 1951, ORNL introduced the terms *probable total reading* (PTR) and *probable maximum exposure* (PME) into its dosimetry practices (Hart 1966). The intent of the PTR was to account for over-response of the OW reading by subtracting the gamma response from behind the cadmium shield. Thus, PTR was defined as

$$PTR \equiv OW - S$$

where OW is the response in millirep determined from the film density behind the OW element (based on calibration to uranium) and S is the response in milliroentgen based on the film density behind the cadmium shield (based on calibration to radium). However, if the S (shield) reading was higher than the PTR as computed above, the PTR was set to the value of S rather than OW minus S (Craft, Ledbetter, and Hart 1952).

The PME was the sum of the PTR values for a given monitoring period unless there were artifacts in any of the dosimeter readings that necessitated a field investigation (Craft, Ledbetter, and Hart 1952). If an investigation was required, the PME was assigned as an estimate of the probable maximum exposure (Hart 1966).

PTR and PME, along with OW (or W) and S, were apparently reported until the second half of 1956, when ORNL adopted the depth dose quantities of NBS Handbook 59 (NBS 1954). These quantities included a skin dose evaluated at a density thickness of 7 mg/cm^2, which at the time could not be determined from film badge readings. The skin (superficial) dose was designated as DS. Other quantities adopted at this time were the *moderately penetrating dose* (DM or MOP), a dose to the lens of the eye (DL or LEN), and a penetrating dose (DP or PEN). DP was determined from the reading under the cadmium filter, and thus was physically the same as S. NOTE: There were fields for values of DS on the data cards used to record personnel exposure results, even though the quantity was not actually computed until the second half of 1961.

In the second half of 1961, after ORNL began using the Model II film badge, it began reporting only the quantities DS and DC on the data cards used to record personnel exposure results. DC referred to *critical organ dose*, and was physically the same as the previously used penetrating dose. Both of these quantities are analogous to deep dose. ORNL began computing values for the skin (superficial) dose at this time, using the expression:

$$DS = 2.5(W - P) + C$$

where W was the response under the OW, P was the response under the plastic element, and C was the response under the cadmium filter (Thornton, Davis, and Gupton 1961). NOTE: ORNL used the quantities DS and DC to refer to superficial and deep dose, respectively, through the late 1980s and the adoption of the DOELAP-accredited CEDS.

The purpose of the term $2.5(W - P)$ in the expression for DS above was to represent the beta portion of the total dose, with the factor of 2.5 being a correction for the attenuation by the OW and plastic filters relative to a dose depth of 7 mg/cm^2. The factor of 2.5 was based on the beta spectrum from natural uranium, but different factors could be used if there was evidence of exposure to a different beta energy (Thornton, Davis, and Gupton 1961).

The above expression was used to compute values of DS from the second half of 1961 until the first quarter of 1968. However, it was not always applied because other administrative practices were in place that governed how values of DS (and DC for that matter) were computed. For example, it can be seen in the claim files that if the value of P was blank, the expression was not used and apparently DS was assigned as the value of W (i.e., the OW reading). However, if the value of C was greater than W (and P was blank), DS was assigned equal to DC (where DC = C, the reading from under the cadmium filter). Review of the data implies that blank entries for P on the data cards mean the value was less than the LOD of the dosimeter. NOTE: Computed values were apparently rounded to the nearest 10 mrem.

The above examples of cases where the DS equation was not used is by no means a complete treatment of how values of DS were computed. Indeed, there are many examples in the claim files where the means used to assign a value for DS cannot be determined from the information on the dosimetry record. Gupton (1968) reports that dose evaluation at ORNL could have involved:

- Calculations using data obtained from personnel dosimeters
- Knowledge of workplace radiation fields and exposure conditions
- Radiation survey data
- Investigatory actions
- Cohort dosimetry
- Reconstruction of exposure conditions, etc.

Thus, one should not assume that values of DS were computed using the above expression in cases where only summary-level information is available. However, it appears that in most cases dose reconstructors will have dosimetry record cards from which to work, which include dosimeter readings in addition to assigned doses and remarks.

Beginning with the first quarter of 1968, the expression used for computing the superficial dose DS from individual dosimeter element readings was revised to:

$$DS = 0.8\frac{(W-C)}{(P-C)}(W-P)+C$$

where *W*, *P*, and *C* are defined as above. As with the previous expression, this expression was not always applied to compute values of DS. Gupton (1968) states that this expression was used for routine exposure conditions, meaning:

- No individual dosimeter element readings exceeded 500 units.
- No ratio of adjacent elements exceeded a factor of 2.
- There was no unusual appearance of the film.

Apparently, other administrative practices were used as well, such as one involving the first term of the DS expression (i.e., 80% of the ratio of the net response under the window and plastic elements). From review of claim data, this term apparently was constrained such that it was never less than unity; that is, if a value less than unity were obtained, the calculation of DS reduced to:

$$DS = (W-P)+C$$

As with the initial expression for DS above, other administrative practices were associated with the assignment of values of DS, and there are instances in the claim files where the bases for assigned values are not immediately obvious from the dosimeter data. Gupton (1968) states that nonroutine cases were forwarded to external dosimetry supervisors, who could then assign a dose based on the dosimeter reading or initiate an investigation by the ORNL radiation survey section.

TLDs appear to have first been employed at ORNL to measure deep dose (DC) for workers in the first quarter of 1974. The advent of the use of TLDs for assigning values of DC meant a slight modification of the expression used to compute DS. Specifically, the *C* term representing the penetrating contribution to the superficial dose was replaced with DC, making the expression:

$$DS = 0.8\frac{(W-C)}{(P-C)}(W-P)+DC$$

NOTE: The values of *C* from the film badge were still used to compute the net response of *W* and *P* in the first term.

After 1975, when ORNL switched to TLDs exclusively for routine monitoring, the methods used to compute the quantities DS and DC become less clear, complicated by the fact that several different TLD designs were in use. Thus, the methods used to compute deep and shallow dose varied somewhat depending on the type of dosimeter assigned to a worker. NOTE: The dosimeter model a worker was wearing for a given period can be ascertained from the Beta-Gamma Dosimetry Record cards by looking at the TLD elements employed (see Table 6-1).

For TLDs issued to nonradiation workers, shallow (skin) dose was determined from the OW, which was covered by an effective density thickness of 60 mg/cm^2 (Parrish 1979). The basic assumption behind this practice was that individuals assigned this dosimeter would not be exposed to radiation fields where there would be an appreciable difference between dose at the effective depth of the OW and that at 7 mg/cm^2 (Gupton 1978).

According to Gupton (1978), the skin dose from the Class 2 (yellow or yellow dot) and Class 3 (red or red dot) dosimeters (both of which were issued to radiation workers) was determined for a dose depth of 7 mg/cm^2 with the expression

$$DS = A + \frac{(OW-A)^2}{(P-A)}$$

where *OW* is the response from the element under the OW, *A* is the response from the element under the aluminum filter, and *P* is the response from the element under the plastic filter. However, a criterion was applied against skin dose evaluations made in this manner to identify cases where the calculated dose could be inappropriate due to artifacts such as the energy of the incident radiation, the angle of incidence, read errors, and so forth. Specifically, if the ratio of the calculated skin dose to the response of the OW TLD element was greater than 4, this was an indication that further action was required in either an evaluation of the Type 2 film packet in each dosimeter or an exposure investigation (Gupton 1978). Claims data have not been reviewed to confirm if this expression for DS was used, or under what conditions it might have been applied.

After 1988, ORNL changed to the DOELAP-accredited CEDS dosimeter and associated dose algorithms, for which assigned shallow doses can be assumed equivalent to *Hp(0.07)*.

6.4.2 Reported Quantities and Associated Algorithms: Neutrons

In the period when NTA film was used exclusively for neutron personnel monitoring, the film was processed for a given individual only if the field health physicist recommended it. Similarly, when neutron personnel monitoring was performed using neutron-sensitive TLDs, the TLDs were issued only to individuals for whom neutron exposure was anticipated. The other TLDs in use were capable of indicating neutron exposure, however, and cases have occurred where neutron dose was added to a worker's deep and shallow dose totals on the beta-gamma dosimetry record cards.

Data fields (i.e., entry blanks) for neutron exposure data first appeared on ORNL dosimetry data cards in early 1948. Fields were added for OW and S readings under the heading NEUTRON. Thus, at that time, beta-gamma and neutron data were recorded on the same card. In 1952, ORNL changed to the use of separate data cards for beta-gamma and neutron dosimetry data. Photographic Film Dosimetry Record cards (or similar) were used for beta-gamma data, and Nuclear Track Film Dosimetry Record cards (or similar) were used for neutron data.

The original ORNL method for computing what it called *thermal* neutron dose was to subtract the NTA film reading under the cadmium shield from that under the OW, assuming the tracks observed under the window were the result of fast and thermal neutrons, and those under the shield were from fast neutrons alone. However, it is unclear if neutron dose was reported in terms of different energy components. The dosimetry data cards examined either had fields for recording the *window* and *shield* readings separately or had single values for neutron dose (representing the contribution from all energies).

Between the late 1940s and early 1950s, neutron *dose* data were reported in terms of a fraction of tolerance, where the neutron exposure and tolerance were both stated in terms of tracks per field or, more typically, tracks for a number of fields (Hart and Walters 1950). For example, for the first half of 1950 tolerance values asserted on the Neutron Film Record cards were in terms of a number of tracks per 12 fields. The tolerance values differed for the OW and S readings, with that for the OW being 22 tracks per 12 fields and that for the S reading being 18 tracks per 12 fields. To convert such values to dose equivalent, dose reconstructors should apply a factor of 10 tracks/cm^2/mrem (Thornton, Davis, and Gupton 1961). This factor is for fast neutrons and inherently includes a quality factor of 10. The value 2×10^{-4} cm^2/field (Thornton, Davis, and Gupton 1961) should be used to convert a number of fields to area. Thus, for a reported value of "3/22" for the first half of 1950, which implies 3 tracks per 12 fields observed relative to a tolerance level of 22 tracks per 12 fields, the dose conversion is as follows:

$$\left(\frac{3\text{ tracks}}{12\text{ fields}}\right)\left(\frac{1\text{ field}}{2\times 10^{-4}\text{ cm}^2}\right)\left(\frac{1\text{ mrem}\cdot\text{cm}^2}{10\text{ tracks}}\right) = 125\text{ mrem.}$$

Dose reconstructors will need to consider other factors before assigning neutron dose in this manner, such as missed dose and the adequacy of NTA film for the neutron field in question. The dose equivalent indicated by the values reported on the Neutron Film Record cards should also be considered versus the LOD for NTA film dosimeters. For instance, in the example above, a value less than 3 tracks per 12 fields would equate to less than 100 mrem, suggesting that the tracks could be from photon events or effects other than a small neutron dose.

In later years neutron exposure data were expressed in terms of tracks per scan rather than tracks per field, where a *scan* was defined as a unit of volume equal to the depth of the emulsion, the width of the field of view, and a length of 2 mm (Gupton 1968). NTA films were read using 900- to 950-

power magnification under dark field illumination. Gupton (1968) states that as many as five scans could have been observed for a dosimeter depending on the track density.

Neutron dose was assumed to be penetrating and thus was added to both superficial and penetrating dose values from other radiations as necessary when compiling summary information. As of 1969, ORNL began doubling neutron dose values to obtain the recorded dose for unknown spectra (Gupton 1969).

Following the adoption of TLDs for routine monitoring, personnel neutron dosimetry at ORNL was effected through use of the ORNL neutron dosimeter, known as the Class 3, red, or red dot badge. This dosimeter used combinations of neutron-sensitive and -insensitive TLD chips in conjunction with NTA film. Empirical algorithms were developed based on knowledge of workplace radiation fields and the neutron energy response characteristics of the dosimeter. These algorithms enabled assignment of a total neutron dose that reflected contributions from four energy groups: Thermal, intermediate, 0.05 MeV to 0.5 MeV, and greater than 0.5 MeV. Dose from the thermal group was measured by the TLDs (pairs of TLD-600 and TLD-700 chips) alone. Similarly, dose from the greater-than-0.5-MeV energy group was determined from the NTA film alone. Doses from the other groups were computed by empirical algorithms based on the response of the TLDs and the NTA film. Thus, neutron dose assigned for ORNL personnel from 1975 to the present represent the dose contribution from all energies based on the known energy response characteristics of the dosimeters and characterization of workplace spectra.

The algorithms employed to compute neutron dose for the thermal, intermediate, and fast energy groups using the ORNL neutron dosimeter between 1975 and approximately 1985 were as follows.

Determine the parameters T_1 and T_2, where T_1 is the albedo response from under the 1-mm cadmium filter, and T_2 is the incident plus albedo response from under the aluminum filter. T_1 is computed as the response of the TLD-600 chip minus that of the TLD-700 chip (from the pair under the cadmium filter). T_2 was the same difference computed for the pair of chips under the aluminum filter. The thermal neutron dose was then computed as:

$$NT = 0.02(T_2 - T_1)$$

where

NT = thermal neutron dose equivalent
0.02 = thermal neutron constant

The fast neutron dose equivalent was computed as:

$$NF = 0.1\left(\frac{T_1}{T_2}\right)(T_1 - NTA)$$

where

NF = fast neutron dose equivalent
0.1 = fast neutron constant
NTA = NTA film reading

NOTE: The ratio of T_1 to T_2 was a function of the hardness of the neutron energy spectrum (Gupton 1978).

The dose from the intermediate neutron energy group was computed as:

$$NI = 0.035\left[T_1 - \left(NF + NTA\right)\right]$$

where

NI = intermediate energy neutron dose equivalent;
0.1 = intermediate energy neutron constant.

The total neutron dose equivalent was then computed as the sum of the four components – *NT*, *NI*, *NF*, and *NTA*.

Between 1981 and 1985, the NTA film packet was eliminated from routine use. The algorithm used to determine neutron dose equivalent from the ORNL red dot TLD was then modified as follows (Berger and Lane 1985):

Compute the parameter *A*, which indicated the response from incident, thermal neutrons as:

$$A = \frac{\left(T_2 - T_1\right)}{T_2}$$

and the parameter *R*, which indicated the quality of the neutron spectrum where:

$$R = \frac{T_1}{T_2}.$$

An empirical calibration factor *CAL* was then computed as:

$$CAL = 1.46e^{-7.64(AR)},$$

and finally, the neutron dose equivalent was computed as the product of T_1 and *CAL*.

Sometime around the mid-1980s, ORNL began using four-element Panasonic TLDs for neutron monitoring (MMES 1992). In 1989, ORNL began implementing the CEDS, which used two dosimeters for whole-body monitoring: A four-element beta-gamma dosimeter (blue holder) and a four-element neutron-sensitive dosimeter (red holder). The CEDS beta-gamma dosimeter became the dosimeter of record as of January 1989. However, the use of the Panasonic neutron dosimeter continued as the neutron dosimeter of record until January 1990. The transition to CEDS at ORNL included a comprehensive effort to further characterize the workplace neutron spectra. As before, this information was used in combination with the known neutron energy response characteristics of the CEDS neutron dosimeter to develop methods for estimating neutron dose for individual workers based on evaluations of time spent in different workplace neutron fields.

6.4.3 Adjustments to Recorded Dose: Beta-Gamma

6.4.3.1 Deep Dose

All personnel dosimetry employed at ORNL over its history included suitably filtered elements for the determination of deep dose. The ORNL film dosimeters, which were calibrated in terms of exposure from ^{226}Ra photons, show either good agreement with or an over-response to the quantity *Hp(10)* (see Figure 6-1). However, because dosimeters were calibrated in terms of exposure over most of the Laboratory's history, dose reconstructors should apply exposure to organ dose conversion factors to deep dose results recorded for ORNL workers for the period before 1989 (i.e., for the period before DOELAP accreditation). After 1989, reported deep dose values are considered equivalent to *Hp(10)*.

6.4.3.2 Shallow Dose

Shallow dose was not explicitly reported at ORNL until 1961 when the Laboratory began reporting the quantity DS, referred to as the superficial, nonpenetrating, or skin dose. DS was defined at a dose depth of 7 mg/cm^2 and thus was intended to represent a skin dose. ORNL adopted this quantity (from NBS Handbook 59) in the second half of 1956, but had no algorithm by which to assign it at that time (NBS 1954). Before 1961, the Laboratory reported film badge results for the individual dosimeter elements with the exception of the period from 1944 through 1946 when the quantities B and G were used. Dose reconstructors should sum the B and G values to obtain the corresponding OW reading, and should use the OW results in cases where shallow dose data are required before 1961 when ORNL began reporting the quantity DS. However, some adjustments to the OW readings are required for certain periods so they are consistent with one another.

The window dose assigned in the period from late June 1944 through the eighth week of February 1947 was based on calibration curves determined with the use of a radium photon source. After that time, the conversion of film density under the OW to dose was based on calibration to a uranium slab. The uranium slab came into use in the ninth week of February 1947; however, the surface dose rate of 270 mrep/hr originally assumed for the slab at that time was revised to 240 mrep/hr on March 1, 1951 (Hart 1966). Therefore, corrections are necessary for OW results reported between 1944 and March 1, 1951, as follows.

- Multiply OW results recorded from 1944 through the eighth week of February 1947 by a factor of 2.2. This factor represents the ratio of the window reading based on calibration to uranium to that based on ^{226}Ra, with the surface dose rate for the uranium source taken to be 240 mrep/hr (Hart 1966).

- Multiply OW results recorded from the ninth week of February 1947 through February 1951 by 0.9 to adjust from the original dose rate assumed for the uranium source (270 mrep/hr) to the 240 mrep/hr value adopted later (Hart 1966).

These corrections should enable comparison of all OW (or W) readings recorded by ORNL on a consistent basis.

6.4.4 Adjustments to Recorded Dose: Neutron

Adjustments to neutron dose data for ORNL workers are required to separate doses into appropriate energy groups for input into IREP and to account for differences between the neutron quality factors in use at the time dose assignments were made and the radiation weighting factors were promulgated in ICRP Publication 60 (ICRP 1990). Neutron dose data for ORNL workers should be assigned to IREP neutron energy groups using the information in Attachment 6B. Adjustments required to state the values in terms of the neutron radiation weighting factors of ICRP 60 are described below.

Neutron quality factors historically used at ORNL were

- 5 for fast neutrons and 2 for thermal for the period from 1943 until 1950
- 10 for fast neutrons and 2.5 for thermal for the period from 1950 until 1971
- Values equivalent to those reported in Table 2 of NCRP Report 38 (NCRP 1971) from 1971 to the present

Fast neutrons at ORNL consisted primarily of those associated with nuclear fission, and thus should correspond to the 0.10-to-2.00-MeV energy group defined in ICRP Publication 60 (ICRP 1990) and used in IREP. The ICRP 60 radiation weighting factor for the 0.10-to-2.00-MeV energy group is 20. Thus, if any reported neutron dose data are used, dose reconstructors should multiply fast neutron doses reported for ORNL workers between 1943 and 1950 by a factor of 4 to account for the difference in historical and current quality factors. Similarly, dose reconstructors should multiply fast neutron doses reported for the period from 1950 to 1968 by a factor of 2. After 1968, ORNL began applying a factor of 2 to neutron dose data, apparently to account for uncertainty associated with unknown neutron spectra (Gupton 1969). From review of claim files it should be apparent when this factor of 2 was applied, thus dose reconstructors should know if a given result needs to be doubled to account for differences in quality factor or if a factor of 2 has been applied. NOTE: The factor of 2 appears to have been applied for conservatism. It does not appear to have been a quality factor. This practice appears to have continued until the adoption of TLDs for personnel neutron monitoring in 1975.

ORNL eventually began reporting neutron dose in terms of the contribution from all energies based on characterizations of workplace spectra and calibration of dosimeters to well-characterized calibration spectra. Because assigned dose equivalents from this era were based on the neutron quality factors of NCRP Report 38 (NCRP 1971), a factor is needed to convert from these factors to those of ICRP Publication 60 (ICRP 1990). This conversion is complicated by the fact that the neutron quality factors in NCRP 38 are point values, while those in ICRP 60 are in terms of neutron energy groups. Therefore, establishing a factor to adjust from NCRP 38 values to those from ICRP 60 requires application of a group-averaging technique to the NCRP 38 data. This process has been detailed in other site profiles developed for the NIOSH Dose Reconstruction Project [for the Y-12 plant for instance – see ORAU (2003)] and is not repeated here. However, Table 6-23 summarizes the results of these calculations for neutron energy groups of interest.

Table 6-23. Group-weighted NCRP 38 neutron quality factors and ICRP 60 radiation weighting factors.[a]

ICRP 60 (IREP) neutron energy group (MeV)[b]	NCRP 38 group-averaged quality factor	ICRP 60 radiation weighting factor	Ratio (ICRP 60-to-NCRP 38)
Thermal (< 0.01)	2.35	5	2.1
0.01 – 0.10	5.38	10	1.9
0.10 – 2.00	10.49	20	1.9
2.0 – 14.0[a]	7.56	10	1.3

a. ICRP 60 = ICRP (1991); NCRP 38 = NCRP (1971).
b. The ICRP energy group is actually from 2.0 to 20.0 MeV; it was truncated to accommodate the group averaging process.

Thus, to adjust more recent reported neutron dose data for ORNL workers (i.e., results for which a neutron energy group was not specified), one would multiply by a factor of 1.9 to account for the difference between the previously applied neutron quality factors and the ICRP 60 weighting factors (assuming an intermediate or fast neutron energy group) (ICRP 1990). Thermal results would be scaled up by a factor of 2.1.

In light of the above discussion on historical neutron quality factors in relation to the neutron weighting factors of ICRP 60 (ICRP 1991), dose reconstructors should consider applying a single factor of 2 to all reported neutron dose data from 1950 to the present for simplicity, with the caveat that data reported from 1969 through 1974 could already reflect application of such a factor. However, the data cards should show both the corrected and uncorrected dose values for the 1969 to 1974 period if the factor of 2 was used.

6.5 LIMITS OF DETECTION AND MISSED DOSE

6.5.1 Deep Dose

Table 6-24 lists LODs for deep dose and dosimeter exchange frequencies for ORNL radiation workers. Dosimeters for other individuals (i.e., nonradiation workers) were sometimes exchanged less frequently, and dosimeters for radiation workers are sometimes exchanged more frequently under circumstances such as potentially high doses or pregnant workers.

Table 6-24. LODs for deep dose and dosimeter exchange frequencies for ORNL radiation workers.

Period of use	Dosimeter	LOD (rem)	Exchange frequency
June 1944 through 2nd qtr. 1956	Two-element and multielement film	0.03	Weekly
3rd qtr. 1956 through 1974	Multi-element film	0.03	Quarterly
1975 - 1988	ORNL TLDs	0.01	Quarterly
1989 - present	CEDS TLD	0.01	Quarterly

NOTE: Deviations from the LOD values in Table 6-24 can be observed in the claim files. However, the deviations noted have always been cases where the LOD was greater than the assigned dose. For example, instances of assigned DC (deep dose) values as low as 10 mR are seen as early as the mid-1960s, and cases of assigned deep doses of 20 mR are seen as early as the mid-1950s.

6.5.2 Shallow Dose

Errors associated with reading film badges for beta exposure are essentially the same as those for X-ray and gamma exposure, and it appears to have been ORNL practice to equate the LOD for OW with that for the cadmium element (Morgan 1961). This practice continued after the adoption of the quantity DS and its associated algorithms and administrative practices, because DS values were generally assigned the deep (DC) LOD value in the event of a dosimeter reading less than its sensitivity threshold. This practice appears to have continued until the adoption of the CEDS dosimetry system in January 1989.

Similar to the Morgan (1961) assertion for film badges, the technical basis for use of CEDS dosimeters (ORNL 1994) implies that the LODs for beta and photon radiations are essentially equivalent in terms of shallow dose from ^{90}Sr/Y beta particles and photons from ^{137}Cs. ORNL (1994) shows that the shallow dose, lower LOD for the CEDS dosimeter calculated on the basis of background variability is approximately 40% higher for a ^{204}Tl beta spectrum (average energy approximately 0.24 MeV) than for a filtered ^{90}Sr/Y spectrum (average energy approximately 0.8 MeV).

Therefore, for assigning missed shallow dose, dose reconstructors should apply the deep dose LOD values from Table 6-24 unless there is reason to suspect exposure to a field consisting primarily of low-energy electrons. In the latter case, assigning a missed shallow dose would be problematic except in cases where the worker was monitored with the CEDS TLD, for which the shallow LOD for (pure) low-energy beta exposure would be approximately 14 mrem/quarter.

6.5.3 Neutron Dose

There is considerable uncertainty about neutron doses that ORNL workers could have received before the advent of TLD technology in 1975. The available information indicates the use of NTA film on an experimental basis to monitor personnel at the Graphite Reactor during the period before its formal introduction. In addition, neutron radiation surveys were used to control personnel neutron dose and to define the need for shielding. The available information is vague on documentation of these surveys or of any potential neutron dose determined from them in employee exposure records. Some references indicate that neutron doses were relatively low until the onset of production of isotopic neutron sources, while others suggest potentially significant exposures for certain operations in the 1944 to 1945 period (Wirth, Morgan, and Curtis 1945). These potentially significant exposure conditions could have been isolated events, or they could have been corrected by radiation protection measures such as shielding or limits on occupancy times.

The LOD for personnel neutron dosimetry using NTA film was 100 mrem (based on a quality factor of 10) for neutron energies greater than 0.5 MeV. The sensitivity of NTA film to neutrons with energies less than 0.5 MeV is essentially zero; that is, neutrons at this energy or lower cannot produce a discernible track in the emulsion. Therefore, two components of missed dose must be considered for the period when NTA film was used exclusively for personnel neutron monitoring: Dose that was below the LOD of the dosimeter and dose associated with neutron energies below 0.5 MeV.

Because the leakage neutron spectrum that would have been present in general areas of the Graphite Reactor was highly thermalized, it is doubtful that NTA film dosimeters were capable of providing an accurate and reliable estimate of personnel neutron exposures for workers in these areas. This logic can be extended to other ORNL facilities where there was potential for neutron exposures from degraded spectra. The neutron spectral characterizations performed by PNL in 1989 through 1991 showed that NTA film should have provided reasonable results in harder spectra such as those associated with ^{238}Pu:Be sources or glovebox operations where there was relatively little shielding. The characterization data show that NTA film probably would have under-responded in softer spectra where there was significant neutron scatter with the magnitude of this under-response ranging from modest to substantial (see Attachment 6F). Therefore, unless it is known that an individual's neutron exposure for a given period occurred in a field where NTA film should have provided reliable monitoring, dose reconstructors should assign neutron dose when necessary by applying neutron-to-photon dose ratios (see Attachments 6C and 6D). Appropriate ratios should be applied to both monitored and missed photon dose for individuals and periods for which neutron dose could be anticipated and NTA film was used exclusively for personnel neutron monitoring (i.e., before 1975). Missed photon dose is addressed in Section 6.5.1. However, because an individual's neutron dose could have been assigned based on survey data, a dose investigation, or other judgment, dose reconstructors should compare neutron dose values computed using neutron-to-gamma dose ratios to any neutron dose recorded for the same period (using consistent neutron quality factors) and select the higher value. As an alternative, the neutron dose distributions given in Section 6.3.4.2.3 and Attachment 6E could be considered as a means for estimating an individual's neutron dose.

Facilities at ORNL where neutron dose could be anticipated include reactors and those listed in Attachment 6B. The neutron-to-photon dose ratios given in Attachment 6D and discussed in Section

6.3.4.2.3 are derived from personnel neutron monitoring data provided by ORNL for the period from 1990 through the second quarter of 2004. These values reflect actual personnel neutron and photon exposures and should be used in favor of the values given in Attachment 6C as practicable. The values in Attachment 6C are derived from characterization of workplace neutron spectra performed in conservatively chosen locations with high neutron dose rates.

At present, measured values of neutron-to-photon dose ratios for ORNL reactor facilities are available only for the Graphite Reactor, the HPRR, and the HFIR. The neutron-to-photon dose ratio estimated for the Graphite Reactor compares well with values for the early Hanford production reactors in ORAU (2004). The ratios for the HFIR from both the PNL characterization data and the ORNL neutron dosimetry records are essentially the same as that for the Graphite Reactor. This is consistent because both facilities are characterized by highly thermalized spectra. Similarly, the neutron-to-photon dose ratio information reported by Kerr and Johnson (1968) for the HPRR is comparable with the values observed for the spectra encountered at the REDC. The highest ratio derived from the ORNL personnel neutron monitoring records for any of the neutron worker groups was a geometric mean of 2.5 having a GSD of 1.6. This ratio, from the dosimetry data for Group 3 (see Section 6.3.4.2.3.1), corresponds to individuals working in a fission neutron spectrum in a low-scatter environment. Lower values are seen for other facilities and job duties. Looking at personnel neutron monitoring results for ORNL workers collectively, the records for the period from 1990 through the second quarter of 2004 yield a geometric mean neutron-to-photon dose ratio of 1.2 with a GSD of 2.2 (see Section 6.3.4.2.3.9). This value is derived from 2,684 individual monitoring records where both the measured neutron and photon dose were greater than or equal to 10 mrem. These records are a subset of a total of 20,037 personnel neutron monitoring records obtained from ORNL for this period, representing monitoring in all workplace neutron spectra. Therefore, for reactors or other facilities not addressed in Attachments 6C or 6D, dose reconstructors can evaluate the applicability of the distribution of neutron-to-photon dose ratios derived from the ORNL personnel monitoring records for similar facilities or job duties, the distribution for all facilities (a geometric mean of 1.2 with a GSD of 2.2), or the distribution yielding the maximum ratios (a geometric mean of 2.5 with a GSD of 1.6 for Building 7930 workers). Dose reconstructors should refer to the neutron-to-photon dose ratios in Attachments 6C and 6D to assign missed and unmonitored dose for individuals who received neutron exposures in comparable facilities or job duties.

Distributions of neutron-to-photon dose ratios or dose values should be used to estimate missed and unmonitored neutron dose for ORNL workers with potential neutron exposure for the period before 1975. Dose reconstructors should use these methods in cases where individuals were monitored via NTA film and it is known or suspected there was significant dose associated with neutrons having energies less than 500 keV. Photon missed dose values for this period should be in accordance with Table 6-24.

For 1975 to the present, neutron monitoring for ORNL workers should have been adequate in terms of sensitivity of the methods used to the various workplace neutron energy spectra. Thus, missed neutron dose can be estimated on the basis of LODs and dosimeter exchange frequencies in the same manner used for deep dose. Dose reconstructors should note that they could encounter instances where an individual was issued more than one dosimeter for a given monitoring period. If so, missed dose should be addressed for each dosimeter worn during the period. Some workers are issued multiple dosimeters because they work in different neutron groups, and dosimeters are sometimes processed more frequently than quarterly at the request of field health physics staff. The latter case typically applies to REDC workers for whom higher doses are anticipated. In any event, applying the quarterly LOD values from the table below for more frequent exchanges should be conservative.

Table 6-25 lists LODs for neutron dose and dosimeter exchange frequencies for ORNL neutron radiation workers. An LOD for the brief period in which the Panasonic neutron dosimeter was used is not given because the value for the ORNL neutron TLD should be sufficiently bounding.

Table 6-25. Limits of detection for neutron dose and dosimeter exchange frequencies for radiation workers from 1975 to the present.

Period of use	Dosimeter	LOD (rem)	Exchange frequency
1975 through 1989	ORNL neutron TLDs (red dot)	0.02	Quarterly
1990 - present	CEDS neutron TLD	0.01	Quarterly

6.6 UNCERTAINTY IN PHOTON AND NEUTRON DOSE

For film badges, the LODs that are quoted in the literature range from about 30 to 50 mrem for beta-gamma irradiation (Morgan 1961; Parrish 1979; West 1992) and from 50 to 100 mrem for neutrons (Morgan 1961; Parrish 1979). These are not the expected uncertainties at larger photon and neutron dose readings. For example, it was possible to read a photon dose of 100 mrem to with ± 15 mrem if the exposure involved photons with energies between several hundred keV and several MeV (Morgan 1961). If the exposure involved photons with energies less than several hundred keV, the uncertainty was at least twice that for the more energetic photons. Thus, the standard error in the recorded film badge doses from photons of any energy is estimated here to be ± 30%. The standard error for the recorded dose from beta irradiation was essentially the same as that for photon irradiation, but when an unknown mixture of beta and photon irradiation was involved the standard error for the dose from beta irradiation was somewhat larger than 30% (Morgan 1961). The situation for neutrons was not as favorable as that for photons. With NTA films, the estimated standard error was much larger and varied significantly with the energy of the neutrons. Thus, the standard error for a neutron dose reading of approximately 100 mrem is estimated here to be ± 50%. For the TLD beta-gamma and neutron dosimeters used at ORNL after the mid-1970’s, the standard errors for a recorded dose reading of 100 mrem or more are estimated here to be approximately ± 15% for photons, beta particles, and neutrons. The standard errors for TLD dose measurements less than 100 mrem and those in mixed radiation fields would be expected to be somewhat larger.

REFERENCES

Beck, J., P. Stansbury, and J. Watson, 1979, *Interview with Ed Gupton on Oak Ridge National Laboratory's Personnel Monitoring Program*, August 6, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Berger, C. D., and B. H. Lane, 1985, *Neutron Dosimetry with the ORNL Badge*, ORNL/TM-9450, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Cameron, J. R., N. Sunthanalingham, and G. N. Kennedy, 1968, *Thermoluminescent Dosimetry*, University of Wisconsin Press, Madison, Wisconsin.

Craft, H. R., J. C. Ledbetter, and J. C. Hart, 1952, *Personnel Monitoring Operating Techniques*, ORNL-1411, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Davis, D. M., E. D. Gupton, and J. C. Hart, 1954, *Applied Health Physics Radiation Survey Instrumentation*, ORNL-332, Revision 1, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Deal, L. J., and J. C. Hart, 1949, *An Explanation Concerning Personnel Monitoring Records and Methods of Filing Prior to January 1, 1947*, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

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

Gupton, E. D., 1968, *Methods and Procedures for External Radiation Dosimetry at ORNL January 1, 1968*, Central Files Number 68-1-73, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Gupton, E. D., 1969, *Methods and Procedures for External Radiation Dosimetry at ORNL January 1969*, Central Files Number 69-1-59, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Gupton, E. D., 1974, "Badge Dosimetry Components for 1975," Intra-Laboratory Correspondence, December 6, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Gupton, E. D., 1975, "Photobadges and Dosimetry Components", Intra-Laboratory Correspondence, August 13, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Gupton, E. D., 1978, *Dosimetry with the ORNL Badge, 1978*, ORNL/TM-6357, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Gupton, E. D., 1981, *Methods and Procedures for External Radiation Dosimetry at ORNL*, ORNL/TM-7984, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September.

Hart, J. C., 1966, *A Progress Report Dealing with the Derivation of Dose Data From ORNL Personnel Exposure Records Applicable to the Mancuso Study*, Central Files Number 66-1-84, January 14, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Hart, J. C., and P. O. Martin, 1949, *Personnel Monitoring Records and Filing at Oak Ridge National Laboratory*, Health Physics Division Report Number SUM-535-49, May 27, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Hart, J. C., and B. T. Walters, 1950, *Personnel Monitoring Recording and Filing Methods of the Health Physics Division of the Oak Ridge National Laboratory as Effected July 2, 1950*, Health Physics Division Report Number SM-188-50, November 7, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Horowitz, Y. S., 1984, *Thermoluminescence and Thermoluminescent Dosimetry*, CRC Press, Boca Raton, Florida.

IAEA (International Atomic Energy Agency), 1990, *Compendium of Neutron Spectra and Detector Responses for Radiation Protection Purposes*, Technical Report Series No. 318, Vienna, Austria.

ICRP (International Commission on Radiological Protection), 1971, *Data for Protection Against Ionizing Radiation from External Sources: Supplement to ICRP Publication 15*, ICRP Publication 21, Pergamon Press, Oxford, England.

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

ICRU (International Commission on Radiation Units and Measurements), 1993, *Quantities and Units in Radiation Protection Dosimetry*, Report No. 51, Bethesda, Maryland.

Kerr, G. D., 2003, "Neutron Dosimetry and Neutron-to-Photon Ratios for Various ORNL Facilities", letter to R. Burns, November 29, Kerr Consulting Company, Knoxville, Tennessee.

Kerr, G. D., and D. R. Johnson, 1968, *Radiation Survey and Dosimeter Intercomparison Study at the Health Physics Research Reactor*, ORNL-TM-2334, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

McLendon, J. D., 1980, *External Radiation Monitoring of Union Carbide Nuclear Division Personnel*, Y/DD-268, Y-12 Plant, Oak Ridge, Tennessee.

Mitchell, T. J., G. Ostrouchov, E. L. Frome, and G. D. Kerr, 1993, *A Method for Estimating Occupational Radiation Dose to Individuals, Using Weekly Dosimetry Data*, ORNL-6778, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

MMES (Martin Marietta Energy Systems, Inc.), 1992, *Dosimeter Response Characterization for Personnel Neutron Dosimetry*, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Morgan, K. Z., 1961, "Dosimetry Requirements for Protection from Ionizing Radiations," in *Selected Topics in Radiation Dosimetry*, proceedings of the symposium on selected topics in radiation dosimetry, International Atomic Energy Agency, Vienna, Austria.

NBS (National Bureau of Standards), 1954, *Permissible Dose from External Sources of Ionizing Radiation*, Handbook 59, Gaithersburg, Maryland.

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

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

ORAU (Oak Ridge Associated Universities), 2003, *Technical Basis Document for the Y-12 Plant – Occupational External Dose*, ORAUT-TKBS-0014-6, Oak Ridge, Tennessee.

ORAU (Oak Ridge Associated Universities), 2004, *Technical Basis Document for the Hanford Site – Occupational External Dose*, ORAUT-TKBS-0006-6, Oak Ridge, Tennessee.

ORNL (Oak Ridge National Laboratory), 1958, *A Study of the Nuclear and Physical Properties of the ORNL Graphite Reactor Shield*, ORNL-2195, Oak Ridge, Tennessee.

ORNL (Oak Ridge National Laboratory), 1994, *Technical Basis for the Centralized External Dosimetry System*, Revision 2, Oak Ridge, Tennessee.

Pardue, L. A., N. Goldstein, and R. O. Wollan, 1944, *Photographic Film as a Pocket Radiation Dosimeter*, CH-1553, University of Chicago, Chicago, Illinois.

Parrish, B. R., 1979, *Analysis of External Personnel Monitoring Devices and Data for Oak Ridge National Laboratory Epidemiological Study*, University of North Carolina at Chapel Hill, Chapel Hill, North Carolina.

Roberson, J. H., 1947, excerpted material dated December 10, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Sims, C. S., and G. G. Killough, 1983, "Neutron Fluence-to-Dose Equivalent Conversion Factors: A Comparison of Data Sets and Interpolation Methods," *Radiation Protection Dosimetry*, volume 5, number 1, pp. 45–48.

Thierry-Chef, I., F. Pernicka, M. Marshall, E. Cardis, and P. Andreo, 2002, "Study of a Selection of 10 Historical Types of Dosimeters: Variation of the Response of Hp(10) with Photon Energy and Geometry of Exposure," *Radiation Protection Dosimetry*, volume 102, number 2, pp. 101–113.

Thornton, W. T., D. M. Davis, and E. D. Gupton, 1961, *The ORNL Badge Dosimeter and its Personnel Monitoring Applications*, ORNL-3126, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Vallario, E. J., D. E. Hankins, and C. M. Unruh, 1969, *AEC Workshop on Personnel Dosimetry*, BNWL-1340, Battelle Pacific Northwest Laboratory, Richland, Washington.

West, C. M., 1992, *Description of the ORNL External Monitoring Program 1943 – 1961*, Oak Ridge Associated Universities, Oak Ridge, Tennessee.

Wirth, J. E., K. Z. Morgan, and H. J. Curtis, 1945, *Health Division and Biological Section of the Research Division Report for Month Ending February 15,1945*, CH-2587, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

ATTACHMENT 6A
Selection of IREP Energy Groups for Beta-Gamma Exposures for ORNL Facilities

Process	Building/description		Approx. dates of operation		IREP energy group selection for deep dose
			Begin	End	
Accelerators	6000:	Holifield Ion Beam Facility	August, 1975	Present	Photons: > 250 keV (100%)
	6010:	ORELA	1969	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	6000:	ORIC	1962	Present	Photons: > 250 keV (100%)
Calibration facilities	2007:	RASCAL	Ca. 1951	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7735:	RADCAL (aka CalLab)	1989	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
Isotope/separation facilities	3019:	Hot Pilot Plant/Radiochemical Development Facility	1943	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3026-C:	Radiochem. laboratory/^{85}Kr facility	1944	Ca. 1988	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3026-D:	RaLa Bldg./segmentation facility	1945	Ca. 1988	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3028:	Radioisotope Processing Building A/Alpha Powder Facility	1950	1985	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3029:	Source Development Laboratory	1952	Late 1980s	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3030:	Radioisotope Processing Building C	Ca. 1950	Late 1980s	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3031:	Radioisotope Processing Building D	Ca. 1950	Late 1980s	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3032:	Radioisotope Processing Building E	Ca. 1950	Late 1980s	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3033:	Radioisotope Processing Building F/Radioactive Gas Processing Facility	Ca. 1950	1990	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3038:	Isotope Research Materials Laboratory/Alpha Handling Facility	1949	1990	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3047:	Radioisotopes Development Laboratory	1962	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3505:	Metals Recovery Facility	1952	1960	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3508:	Alpha Isolation Laboratory	1952	Ca. 1976	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3515:	Fission Product Pilot Plant (F3P)	1948	1958	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3517:	Fission Product Development Laboratory	1958	1975	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3525:	High Radiation Level Examination Laboratory	1963	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	4501:	High-Level Radiochemistry Laboratory	1951	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	4507:	High-Level Chemical Development Facility	1957	1980	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	4508:	Interim Plutonium Laboratory/Fuel Cycle Alpha Facility	1962	Present	Photons: 30 – 250 keV (100%)
	7025:	Tritium Target Preparation Facility	1967	Ca. 1989	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7920:	REDC	1966	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)

ATTACHMENT 6A (Continued)

Process	Building/description	Approx. dates of operation		IREP energy group selection for deep dose
		Begin	End	
	7930: REDC	1966	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
Reactors	3001: Graphite Reactor	Nov. 1943	Nov. 1963	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3005: Low-Intensity Test Reactor (LITR)	1949	Oct. 1968	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3010: Bulk Shielding Reactor/Pool Critical Assembly	1951	1987	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3042: ORNL Research Reactor (ORR)	March, 1958	Ca. 1987	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7500: Homogeneous Reactor Experiment (HRE)/ Homogeneous Reactor Test (HRT)	1952	1961	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7503: Molten Salt Reactor Experiment (MSRE)	1965	1969	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7702: Tower Shielding Reactor (TSR)	1954	1992	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7710: Health Physics Research Reactor (HPRR)	1963	1987	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7900: High Flux Isotope Reactor (HFIR)	August, 1965	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	9213: Critical Experiments Facility (at Y-12)	August, 1950	1987	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
Waste/storage facilities	3023: North Tank Farm	1943	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3039: Central Offgas Facility	1950	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3100: Source/SNM Storage Vault	1960	Present	Photons: 30 – 250 keV (100%)
	3507: South Tank Farm	1943	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	3513: Waste Holding Basin	1943	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)
	7830: SWSA 5 (Melton Valley storage area)	1959	Present	Photons: > 250 keV (75%) Photons: 30 – 250 keV (25%)

ATTACHMENT 6B
Selection of IREP Energy Groups for Neutron Exposures for ORNL Facilities

Process	Building/description	Approx. dates of operation		IREP energy group selection	
		Begin	End	Energy groups (MeV)	Dose fractions
Accelerators	6000: Holifield Ion Beam Facility	August 1975	Present	0.1 – 2.0	100% (default)
	6010: ORELA	1969	Present	< 0.01 0.1 – 2.0	14% 86%
	6000: ORIC	1962	Present	0.1 – 2.0	100% (default)
Calibration facilities	2007: RASCAL (238Pu:Be only)	Ca. 1951	Ca. 1990	0.1 – 2.0 2.0 – 20.0	29% 71%
	7735: RADCAL: beta room with moderated Cf-252 source exposed	1989	Present	< 0.01 0.1 – 2.0	16% 84%
	7735: RADCAL: beta room with bare Cf-252 source exposed	1989	Present	0.1 – 2.0 2.0 – 20.0	69% 31%
	7735: RADCAL: hallway with moderated Cf-252 source exposed	1989	Present	< 0.01 0.1 – 2.0 2.0 – 20.0	33% 53% 14%
	7735: RADCAL: equipment room with moderated Cf-252 source exposed	1989	Present	< 0.01 0.1 – 2.0 2.0 – 20.0	25% 57% 18%
Isotope/separation facilities	3038: Isotope Research Materials Laboratory (shielded 238Pu:Be and Cf-252 sources)	1949	1990	0.1 – 2.0 2.0 – 20.0	42% 58%
	7920: REDC: Room 111 gloveboxes	1966	Present	0.1 – 2.0 2.0 – 20.0	76% 24%
	7920: REDC: Room 211 gloveboxes	1966	Present	0.1 – 2.0 2.0 – 20.0	75% 25%
	7920: REDC: TDF	1966	Present	< 0.01 0.1 – 2.0	12% 88%
	7920: REDC: waste cask	1966	Present	< 0.01 0.1 – 2.0 2.0 – 20.0	14% 59% 27%
	7920: REDC: WTA tunnel exit	1966	Present	0.1 – 2.0	100%
Reactors	3001: Graphite Reactor (general area)	Nov. 1943	Nov. 1963	< 0.01	100%
	3005: Low-Intensity Test Reactor (LITR)	1949	Oct. 1968	0.1 – 2.0	100% (default)
	3010: Bulk Shielding Reactor/Pool Critical Assembly	1951	1987	0.1 – 2.0	100% (default)
	3042: ORNL Research Reactor (ORR)	March 1958	Ca. 1987	0.1 – 2.0	100% (default)
	7500: Homogeneous Reactor Experiment (HRE)/ Homogeneous Reactor Test (HRT)	1952	1961	0.1 – 2.0	100% (default)
	7503: Molten Salt Reactor Experiment (MSRE)	1965	1969	0.1 – 2.0	100% (default)
	7702: Tower Shielding Reactor (TSR)	1954	1992	0.1 – 2.0	100% (default)
	7710: Health Physics Research Reactor (HPRR)	1963	1987	0.1 – 2.0	100%
	7900: High Flux Isotope Reactor (HFIR) (beam room)	August 1965	Present	< 0.01	100%
	9213: Critical Experiments Facility (at Y-12)	August 1950	1987	0.1 – 2.0	100% (default)
Waste/storage facilities	3100: Source/SNM Storage Vault (241Am-oxide drums)	1960	Present	0.1 – 2.0 2.0 – 20.0	78% 22%
	SWSA 5 TRU storage bunker: waste casks	1959	Present	< 0.01 0.1 – 2.0 2.0 – 20.0	12% 70% 18%
	SWSA 5 TRU storage bunker: at wall shared with adjacent bunker	1959	Present	< 0.01 0.1 – 2.0 2.0 – 20.0	13% 51% 36%

ATTACHMENT 6C
Neutron-to-Photon Dose Ratios for ORNL Facilities for 100% Occupancy

Process	Building/description	Approx. dates of operation		Neutron-to-gamma dose ratios
		Begin	End	
Accelerators	6000: Holifield Ion Beam Facility	August, 1975	Present	19
	6010: ORELA	1969	Present	2.5
Calibration facilities	2007: RASCAL (^{238}Pu:Be only)	Ca. 1951	Ca. 1990	28.9
	7735: RADCAL: beta room with moderated Cf-252 source exposed	1989	Present	3.6
	7735: RADCAL: beta room with bare Cf-252 source exposed	1989	Present	15.6
	7735: RADCAL: hallway with moderated Cf-252 source exposed	1989	Present	2.1
	7735: RADCAL: edge of neutron source storage well	1989	Present	1.2
Isotope/separation facilities	3038: Isotope Research Materials Laboratory (shielded ^{238}Pu:Be and Cf-252 sources)	1949	1990	3.7
	3038: Isotope Research Materials Laboratory (^{238}Pu:Be source in Lucite glovebox)	1949	1990	32.2
	7920: REDC: Room 111 glovebox (face)	1966	Present	5.8
	7920: REDC: Room 211 glovebox (side)	1966	Present	11.7
	7920: REDC: Room 211 glovebox (face)	1966	Present	5.3
	7920: REDC: 4.6 m behind Room 211 gloveboxes	1966	Present	1.3
	7920: REDC: TDF	1966	Present	0.2
	7920: REDC: waste cask	1966	Present	4.4
	7920: REDC: WTA tunnel exit	1966	Present	6.7
	7920: REDC: WTA glovebox	1966	Present	1.2
	7930: REDC: low-scatter glovebox	1966	Present	16.7
	7920: REDC: Room 208 glovebox (face)	1966	Present	1.3
	7920: REDC: Room 208 low dose waiting area	1966	Present	1.3
Reactors	3001: Graphite Reactor (general area)	Nov. 1943	Nov. 1963	0.1
	7710: Health Physics Research Reactor (HPRR)	1963	1987	1.0
	7900: High Flux Isotope Reactor (HFIR) (beam room)	August, 1965	Present	0.2
Waste/storage facilities	3100: Source/SNM Storage Vault (^{241}Am-oxide drums)	1960	Present	1.2
	SWSA 5 TRU storage bunker: waste casks	1959	Present	2.8

NOTE: Geometric mean, minimum, and maximum should be doubled to account for ICRP 60 radiation weighting factors

ATTACHMENT 6D
Neutron-to-Photon Dose Ratios for ORNL Neutron Worker Groups

Neutron Group	Group Description	Neutron:Photon Dose Ratio				
		No. Records*	Geo. Mean	GSD	Min.	Max.
1	RADCAL workers	4	--	--	0.6	1.6
2	REDC Analytical Chemistry: Bldg. 7920 glovebox workers	1104	1.6	2.2	0.1	8.8
3	REDC TURF glove box: Bldg. 7930 workers	65	2.5	1.6	1.0	8.1
4	REDC General Area: control room, clerical, and support personnel.	982	1.1	1.7	0.1	3.7
5	HFIR Beam Room	5	--	--	0.1	0.2
6	Holifield workers (exclusively)	4	--	--	0.7	1.5
7	D_2O-moderated ^{252}Cf	24	0.3	3.2	0.1	2.2
9	Unshielded source handlers	163	1.0	2.8	0.1	14.1
10	Building 3100 storage vault	0	--	--	--	--
11	SWSA 5: storage bunker/WEAF workers	9	1.1	1.9	0.3	2.4
12	General TRU waste handlers	3	--	--	0.2	1.1
13	Shielded source users: well-loggers, I&C tasks	37	0.5	2.4	0.1	3.5
14	Neutron source make-up	45	1.1	2.2	0.2	9.1
17	REDC TRU decon. facility workers	86	2.1	2.3	0.4	8.0
18	ORELA workers (exclusively)	0	--	--	--	--
19	ORELA and Holifield: staff whom work at both facilities	3	--	--	0.7	1.9
21	SWSA 5: Bldg. 7879 drum storage handlers	148	0.5	1.9	0.1	2.8
22	R&D (primarily I&C techs.)	2	--	--	1.1	1.4
all	All groups combined	2684	1.2	2.2	0.1	14.1

*neutron and photon dose both ≥ 10 mrem

NOTE: Geometric mean, minimum, and maximum should be doubled to account for ICRP 60 radiation weighting factors

ATTACHMENT 6E
Neutron Dose Data for ORNL Neutron Worker Groups 1990 - 2004

Neutron Group	Group Description	Fraction of results ≥ 10 mrem	Neutron Dose (mrem)		
			Geo. Mean	GSD	Max.
1	RADCAL workers	1% (5 out of 406)	--	--	28
2	REDC Analytical Chemistry: Bldg. 7920 glovebox workers	78% (1197 out of 1535)	50	2.2	584
3	REDC TURF glove box: Bldg. 7930 workers	98% (79 out of 81)	42	1.8	137
4	REDC General Area: control room, clerical, and support personnel.	39% (1070 out of 2730)	23	1.6	91
5	HFIR Beam Room	0.1% (5 out of 7992)	--	--	14
6	Holifield workers (exclusively)	2% (8 out of 407)	--	--	28
7	D_2O-moderated ^{252}Cf	3% (37 out of 1329)	14	1.3	27
9	Unshielded source handlers	24% (405 out of 1687)	24	1.8	212
10	Building 3100 storage vault	2% (1 out of 53)	--	--	11
11	SWSA 5: storage bunker/WEAF workers	8% (38 out of 448)	14	1.3	26
12	General TRU waste handlers	1% (3 out of 501)	--	--	30
13	Shielded source users: well-loggers, I&C tasks	7% (44 out of 675)	18	1.6	53
14	Neutron source make-up	32% (72 out of 222)	25	1.8	182
17	REDC TRU decon. facility workers	42% (111 out of 264)	37	2.3	230
18	ORELA workers (exclusively)	0% (0 out of 191)	--	--	--
19	ORELA and Holifield: staff whom work at both facilities	1% (4 out of 290)	--	--	47
21	SWSA 5: Bldg. 7879 drum storage handlers	14% (160 out of 1148)	17	1.5	58
22	R&D (primarily I&C techs.)	23% (11 out of 48)	--	--	17
all	All groups combined	16% (3252 out of 20037)	31	2.1	584

NOTE: Geometric mean and maximum should be doubled to account for ICRP 60 radiation weighting factors

ATTACHMENT 6F
Fractions of Dose Equivalent above Energy Cutoff for NTA Film for ORNL Facilities

Process	Building/description	Approx. dates of operation Begin	End	Fraction of dose equivalent above the energy cutoff for NTA film (500 keV)
Accelerators	6010: ORELA	1969	Present	65%
Calibration facilities	2007: RASCAL (Pu-238:Be only)	Ca. 1951	Ca. 1990	94%
	7735: RADCAL: beta room with moderated Cf-252 source exposed	1989	Present	76%
	7735: RADCAL: beta room with bare Cf-252 source exposed	1989	Present	82%
	7735: RADCAL: hallway with moderated Cf-252 source exposed	1989	Present	29%
	7735: RADCAL: equipment room with moderated Cf-252 source exposed	1989	Present	61%
Isotope/separation facilities	3038: Isotope Research Materials Laboratory (shielded ^{238}Pu:Be and Cf-252 sources)	1949	1990	92%
	7920: REDC: Room 111 glovebox (face)	1966	Present	91%
	7920: REDC: Room 211 glovebox (side)	1966	Present	94%
	7920: REDC: TDF	1966	Present	0%
	7920: REDC: waste cask	1966	Present	72%
	7920: REDC: WTA tunnel exit	1966	Present	87%
Reactors	3001: Graphite Reactor (general area)	Nov. 1943	Nov. 1963	0% (estimated)
	7900: High Flux Isotope Reactor (HFIR) (beam room)	August, 1965	Present	0%
Waste/storage facilities	3100: Source/SNM Storage Vault (241Am-oxide drums)	1960	Present	80%
	SWSA 5 TRU storage bunker: waste casks	1959	Present	68%
	SWSA 5 TRU storage bunker: at wall shared with adjacent bunker	1959	Present	65%