<table>
<tr><td>ORAU Team
Dose Reconstruction Project for NIOSH

Technical Basis Document for the Argonne National Laboratory - West – Occupational External Dosimetry</td><td>Document Number:
ORAUT-TKBS-0026-6
Effective Date: 09/30/2004
Revision No.: 00
Controlled Copy No.: _______
Page 1 of 45</td></tr>
<tr><td>Subject Expert: Norman D. Rohrig

Document Owner
Approval: Signature on File Date: 09/27/2004
Norman D. Rohrig, TBD Team Leader

Approval: Signature on File Date: 09/22/2004
Judson L. Kenoyer, Task 3 Manager

Concurrence: Signature on File Date: 09/22/2004
Richard E. Toohey, Project Director

Approval: Signature on File Date: 09/30/2004
James W. Neton, Associate Director for Science</td><td>Supersedes:

None</td></tr>
</table>

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	06/30/2004	00-A	New technical basis document for the Argonne National Laboratory - West - Occupational External Dosimetry. Initiated by Norman D. Rohrig.
Draft	08/23/2004	00-B	Incorporates NIOSH and internal review comments. Initiated by Norman D. Rohrig.
09/30/2004	09/30/2004	00	First approved issue. Initiated by Norman D. Rohrig.

ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ANSI	American National Standards Institute
AP	anterior–posterior
ATLAS	Automatic Thermoluminescent Analyzer System
ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory - West
Ci	curie
cm	centimeter
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
ERDA	Energy Research and Development Administration
EBR	Experimental Breeder Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
FFTF	Fast Flux Test Facility
FNCF	Facility Neutron Correction Factor
g	gram
hr	hour
ICPP	Idaho Chemical Processing Plant (formerly CPP and now INTEC)
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
ID, IDO	Idaho Operations Office
in	inch
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center (formerly ICPP and CPP)
IREP	Interactive RadioEpidemiological Program
kerma	kinetic energy released to matter
keV	kilovolt-electron, 1,000 electron volts
LET	Linear Energy Transfer
MeV	megavolt-electron, 1 million electron volts
mg	milligram
mm	millimeter
mR	milliroentgen
mrad	millirad
mrem	millirem
MRL	minimum reporting level
MTR	Materials Test Reactor

NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurement
NIOSH	National Institute for Occupational Safety and Health
NRTS	National Reactor Testing Station
NTA	nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program
R	roentgen
RBE	relative biological effectiveness
rep	roentgen-equivalent-physical
RESL	Radiological and Environmental Services Laboratory
RWMC	Radioactive Waste Management Complex
TAN	Test Area North
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor Test
wk	week
yr	year
Z	atomic number
ZPPR	Zero Power Physics Reactor

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 mean 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)].

From the start of operations in 1951 until the present, Argonne National Laboratory – West (ANL-W) has been operated by the University of Chicago under supervision of the Chicago Operations Office of the U.S. Atomic Energy Commission (AEC), the Energy Research and Development Agency (ERDA), and the U.S. Department of Energy (DOE). A branch of the Idaho Operations Office (ID, previously IDO) provided external dosimetry resources and services at the Idaho National Engineering and Environmental Laboratory (INEEL) including ANL-W from the start of operations in 1951 [when it was called the National Reactor Testing Station (NRTS)] until 1989, when DOE transferred that responsibility to the prime operating contractor. Despite the fact that INEEL had several contractors at a time and that contractors changed often, the external dosimetry process has remained under technical management of a single organization with responsibilities for dosimetry development, operational dosimetry, and radiological records, which has provided a stable external dosimetry system.

6.2 BASIS OF COMPARISON

The Interactive RadioEpidemiological Program (IREP) calculates the probability of cancer induction in an organ from the external equivalent dose and internal dose received by that organ. Appendix B of the *External Dose Reconstruction Implementation Guidelines* (NIOSH 2002) provides conversions from four photon dose quantities [deep dose equivalent, $H_p(10)$; ambient dose equivalent, $H^*(10)$; exposure, X; and air kerma, K_a] and three neutron quantities [fluence, φ; ambient dose equivalent, $H^*(10)$; and deep dose equivalent, $H_{p,slab}(10)$] to the organ doses. Over the years, as the National Council on Radiation Protection and Measurements (NCRP), International Commission on Radiological Protection (ICRP), and their predecessor organizations have developed the definitions of dosimetry parameters, dose parameters measured by the INEEL dosimetry system have received further definition. INEEL has reported doses as penetrating and nonpenetrating. The penetrating dose corresponds to the deep dose equivalent, and the nonpenetrating dose plus the penetrating dose corresponds to the shallow dose equivalent.

Horan and Braun (1993), Attix and Roesch (1968), and Meinhold (1975) discuss the history of radiation protection requirements from the 1930s. In 1949, the newly formed National Committee (now Council) on Radiation Protection (NCRP) issued NCRP Report 7 [as National Bureau of Standards (NBS 1949, p 6) Handbook 42], which recommended a permissible dose of 0.3 R wk^{-1} (15 R yr^{-1}) for occupational workers. The term dose was undefined. A *roentgen* (R) was defined as the quantity or dose of X-rays such that the associated ionization per 0.001293 gram of air (1 cm^3 at standard temperature and pressure) produces 1 electrostatic unit of charge of either sign. A site manual in April 1952 stated the limit as “0.3 rep wk^{-1} at an effective depth in soft tissue of 5 cm, assumed to be the depth of the blood forming organs” (ACC 1952, p. IV: 1-1). It does not mention a quarterly or an annual limit. The unit roentgen only applies to x-rays or gamma rays, whereas the roentgen equivalent physical (rep) was used for all ionizing radiation and is the energy absorbed by tissue from the radiation (93 ergs/gm, the same as for one roentgen of gamma radiation).

In 1953, the International Commission on Radiation Units and Measurements (ICRU 1954) established a new unit, *absorbed dose*, which is the energy deposited in material per unit mass by radiation, using the *rad* (from radiation absorbed dose), a unit equal to 100 erg g^{-1}. ICRU specified the term *exposure dose*, later to become *exposure*, for the ionization capability in air for X- and gamma rays. In 1956, ICRU defined the term *relative biological effectiveness* (RBE) dose and the unit *rem* (from roentgen equivalent in man), and introduced the concept of adding all types of external doses together (ICRU 1956).

In 1957, the NCRP introduced an age prorating formula for the Maximum Allowable Dose of 5 rem × [age (yr) -18] (NBS 1958). This introduced 5 rem as an average annual dose, but deemphasized it as a limit. The AEC issued AEC Manual Chapter 0524 “Permissible Levels of Radiation Exposure” on January 9, 1958, which adopted the prorating formula. It retained 15 rem as the maximum annual dose and superseded the 13-wk whole-body limit of 3 rem with “the provision that not more than one-fourth of the 15 rem maximum permissible yearly dose shall be taken in one-fourth of a year” (AEC 1958).

The quarterly limit of 3 rem or 12 rem yr^{-1} replaced the 15 R yr^{-1} associated with the weekly limit (NBS 1958, p. 3, footnote 2). President Eisenhower approved these values in 1960 for Federal agencies. AEC Manual Chapter 0524 was reissued in 1963 and 1968 (AEC 1963, 1968) and later still as ERDA Manual Chapter 0524 (ERDA 1975, 1977), which provided requirements for radiation safety.

In 1957, NBS Handbook 63 (NBS 1957) specified a dependence of the RBE on the linear energy transfer (LET) of the charged particles that actually deliver the dose. NBS used this in the Snyder calculations of maximum permissible neutron flux (NBS 1961), and it is still used in the radiation control regulations for DOE.

At an April 1962 ICRU meeting, the use of the terms RBE and RBE dose in radiation protection was criticized, and the terms quality factor (QF, now Q) and dose equivalent (DE, now H) were introduced. The ICRU recommended the unit kerma (kinetic energy released in material) in 1962 (ICRU 1962).

In 1971, NCRP Report 39, *Basic Radiation Protection Criteria* (NCRP 1971a), recommended an annual dose limit of 5 rem, eliminating the quarterly limit. In April 1975, ERDA reissued Manual Chapter 0524 (ERDA 1975), which invoked the 5-rem annual dose limits in NCRP Report 39 and required adding internal and external dose equivalents if both are known. Monitoring was required “where the potential exists for the individual to receive a dose or dose commitment ... in excess of 10% of the quarterly standard” of 3 rem. Personnel monitoring equipment for each individual was required for external radiation: “To achieve optimum accuracy, personnel dosimeters should comply with the performance parameters contained in American National Standards Institute (ANSI) standards N13.5 (ANSI 1972), N13.7 (ANSI 1983a), and N13/42 WG1 Final draft 1974” (ERDA 1975, Appendix, p. 10). Quality factors from NCRP Report 38 are specified along with the neutron flux density for 100 mrem in 40 hr as a function of neutron energy (NCRP 1971b). The NCRP 38 guidance for interpolating in energy cannot be accomplished with an instrument. The dose equivalent conversion factors and the associated interpolation with energy reported in ICRP Publication 21 (ICRP 1973) do not present that problem.

In 1971, ICRU defined the quantity *dose equivalent index*, the maximum value in a 30-cm-diameter sphere, for describing ambient radiation fields for radiation protection purposes (ICRU 1971). ICRU extended this discussion in *Conceptual Basis for the Determination of Dose Equivalent* (ICRU 1976), which defined the concept of deep and shallow dose equivalent indexes as those inside a 1-cm depth in the sphere and at a depth between 0.07 mm and 10 mm, respectively. A remaining issue was that the quantity was measured near the surface of the sphere but applied to the center of the sphere, a distance of 14 or 15 cm. In 1980, ICRU identified the deep and shallow dose equivalent indexes as restricted indexes (ICRU 1980). In 1985, ICRU Report 39, *Determination of Dose Equivalents Resulting from External Radiation Sources*, introduced the concepts of aligned and expanded fields to eliminate issues of field direction and nonuniform fields; the document also introduced several dose equivalents: ambient dose equivalent, directional dose equivalent, individual dose equivalent penetrating, and individual dose equivalent superficial (ICRU 1985).

ICRP Publications 26 and 30 introduced new dose limits and the associated quantity *effective dose equivalent* as a weighted averaged over the radiation-sensitive organs of the body (ICRP 1977, 1979).

In 1981, DOE Order 5480.1A, Chapter XI, "Requirements for Radiation Protection" (DOE 1981), superseded ERDA Manual Chapter 0524 (ERDA 1977). In 1988, DOE Order 5480.11, "Radiation Protection for Occupational Workers" (DOE 1988) superseded DOE Order 5480.1A, Chapter XI. This order adopted much of the language of ICRP Publications 26 and 30 (1977, 1979), and the monitoring threshold became 100 mrem effective dose equivalent. The order imposed slight changes in quality factor value for neutrons in one table, but did not capture those changes in the table of permitted neutron flux density.

Because of questions of quality control for dosimetry, the Conference of Radiation Control Program Directors encouraged development of a dosimetry accreditation process, leading to the development of ANSI N13.11 (ANSI 1983b) and the National Voluntary Laboratory Accreditation Program (NVLAP). DOE *Guidelines for the Calibration of Personnel Dosimeters* (Roberson and Holbrook 1984) revised the ANSI (1983b) NVLAP processes. Calibration was to the quantities shallow and deep dose equivalent (H_s and H_d) and shallow absorbed dose (D_s), which are similar to the individual dose equivalent superficial and individual dose equivalent penetrating dose defined in ICRU (1985). These quantities were renamed to the personal dose equivalent $H_p(d)$ (ICRU 1993) where d is the depth in millimeters (0.07 mm for surface and 10 mm for deep) from the surface for which the dose is measured. In 1987, DOE Order 5480.15, "Department of Energy Laboratory Accreditation Program for Personnel Dosimetry," (DOE 1987) established the DOE Laboratory Accreditation Program (DOELAP) system for dosimetry accreditation. *Standard for the Performance Testing of Personnel Dosimetry Systems* (DOE 1986a) specified the measurement of deep and shallow dose equivalents at depths of 10 mm and 0.07 mm, respectively.

In 1990, the ICRP redefined the concept of dose equivalent to equivalent dose, redefined quality factor to radiation weighting factor, and generated new factors (ICRP 1990). These factors, invoked in NIOSH (2002), depend on neutron energy at the entrance to the body rather than on secondary particle LET where the dose is received. Dose conversion factors for organs and for ambient dose equivalent and personal dose equivalent were generated in ICRP Publication 74 (ICRP 1996) and are referenced in the external dose implementation guide (NIOSH 2002).

Thus, the quantities to be measured and reported by the dosimetry systems have evolved over the last 50 years. Although the standards organizations were changing definitions, this had little impact on dosimetry measurements because, for gamma radiation, the numerical differences are small.

6.3 DOSE RECONSTRUCTION PARAMETERS

6.3.1 Site Administrative Practices

It was ANL-W policy that personnel expected to receive any radiation dose or personnel whose work was centered at the site were assigned a radiation monitoring badge. These badges were usually stored at the respective operational area entrance security gates for ANL-W facilities. Control badges, which are used to subtract background radiation, have also been and are currently located where the badges are stored. This practice may lead to subtracting environmental radiation from site activities, which would reduce the reported doses. Environmental radiation levels have been monitored for most of the life of the ANL-W, originally with film badges and later with TLDs. Table 6-1 presents results of this monitoring at facility fence line locations near the security gates. A fraction (2000/8766) of these values can be added to an individual dose history or used for unmonitored workers working at the site.

Table 6-1. Facility fence direct gamma values (TLD – Background) (mR).

Year	EBR-1 RWMC[a]	EBR-II	TREAT	Backgrnd
1952-72	32	36	18	
1973	32	37	19	121
1974	370	35	17	123
1975	265	32	8	118
1976	155	56	50	113
1977	189	22	0	132
1978	106	56	2	129
1979	65	59	5	113
1980	57	51	12	119
1981	42	28	9	118
1982	42	20	12	117
1983	50	24	10	115
1984	48	31	13	124
1985	48	31	13	124
1986	48	31	13	124
1987	48	31	13	124
1988	48	31	13	124
1989	48	31	13	124
1990	27	19	13	124
1991	27	19	13	124
1992	27	19	13	124
1993	24	28	16	111
1994	25	15	3	130
1995	42	17	7	116
1996	40	22	21	129
1997	17	16	16	128
1998	20	0	11	131
1999	22	13	13	122
2000	61	25	26	129
2001	25	0	3	140
2002	33	18	39	120

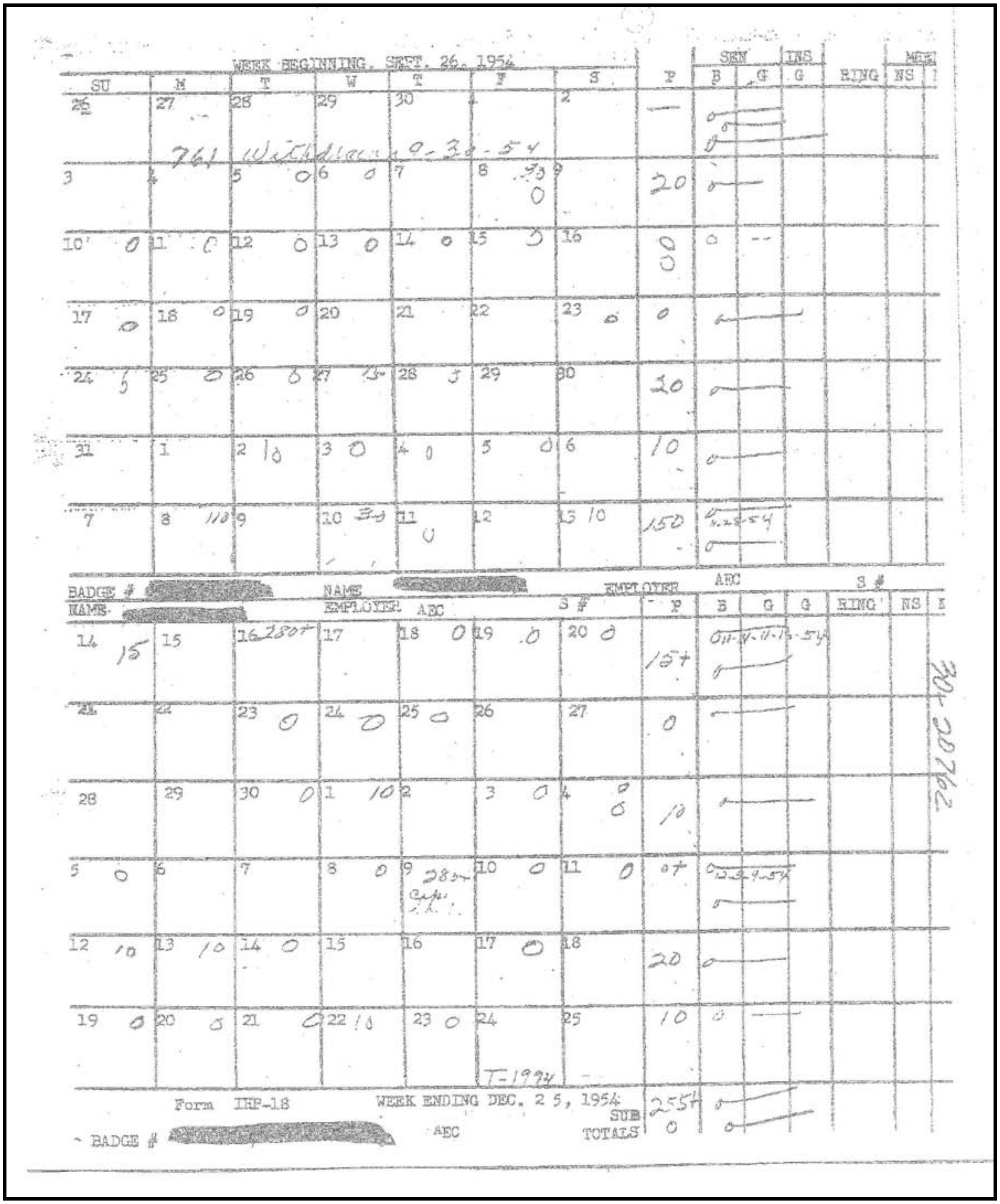
a. RWMC = Radioactive Waste Management Complex.

Some individuals who could occasionally have visited site facilities but did little work with radiation, had badges at several different facilities. It is not appropriate to base missed doses on the multiple badges issued. Early on, the badge change frequency was not the same for everyone. Workers with low probability of exposure were placed on a longer change cycle than those with more chance of exposure. Therefore, missed doses should be based on the actual change frequency for a person, and the frequency can be determined from the individual's data package.

The INEEL dosimetry organization developed a set of basic administrative practices in 1951, which have changed somewhat as the technologies of ionizing radiation dosimetry and recordkeeping have changed.

DOE provided dosimetry information for a former ANL-W worker will include copies of available dosimetry forms. These forms include a dose summary for each monitored employment period and a copy of each weekly, monthly, quarterly, etc., dosimeter result, which will also show the worker's work location. The information can easily be several inches in thickness. Each sheet is redacted so only the person of interest's name and applicable information are visible. This file provides the recorded information as to the exchange period for the person for that period. Figures 6-1 through 6-5 show a partial example set of redacted dose reporting forms.

From 1951 to 1958, the INEEL dosimetry staff recorded each worker's dose each day on a dose card (Figure 6-1), rezeroed pencil ionization chambers worn by workers, and entered the weekly badge result on the same card. On this sample, on October 28, November 16, and December 9, 1954, the badges were pulled and read in response to high pencil chamber readings. The personnel monitoring badges have always been considered more reliable than pencil dosimeters; so after the film badge results became available, the daily pencil readings were no longer considered doses of record. However, these values can be recovered from the earliest forms for a worst-case estimate of dose. In Figure 6-1, the pencil readings totaled 820 mR and the badges reported 0 mR for 18 badges.



WEEK BEGINNING SEPT. 26, 1954

SU	M	T	W	T	F	S	P	SEN B	SEN G	INS G	RING	NS
26	27	28	29	30	1	2	—	0				
	761 Withdrawn 9-30-54											
3	4	5 0	6 0	7	8 .20 0	9	20	0				
10 0	11 0	12 0	13 0	14 0	15 0	16	0 0	0	--			
17 0	18 0	19 0	20	21	22	23 0	0	0				
24 5	25 0	26 0	27 15	28 5	29	30	20	0				
31	1	2 10	3 0	4 0	5 0	6	10	0				
7	8 110	9	10 30	11 0	12	13 10	150	0 10-28-54 0				

BADGE # [redacted] NAME [redacted] EMPLOYER AEC S #
NAME [redacted] EMPLOYER AEC S #

SU	M	T	W	T	F	S	P	B	G	G	RING	NS
14 15	15	16 280+	17	18 0	19 0	20 0	15+	0 11-16-54 0				
21	22	23 0	24 0	25 0	26	27	0	—				
28	29	30 0	1 10	2	3 0	4 0 0	10	—				
5 0	6	7	8 0	9 280	10 0	11 0	0+	0 12-9-54 0				
12 10	13 10	14 0	15	16	17 0	18	20	0				
19 0	20 0	21 0	22 10	23 0	24	25	10	0	—			

T-1994

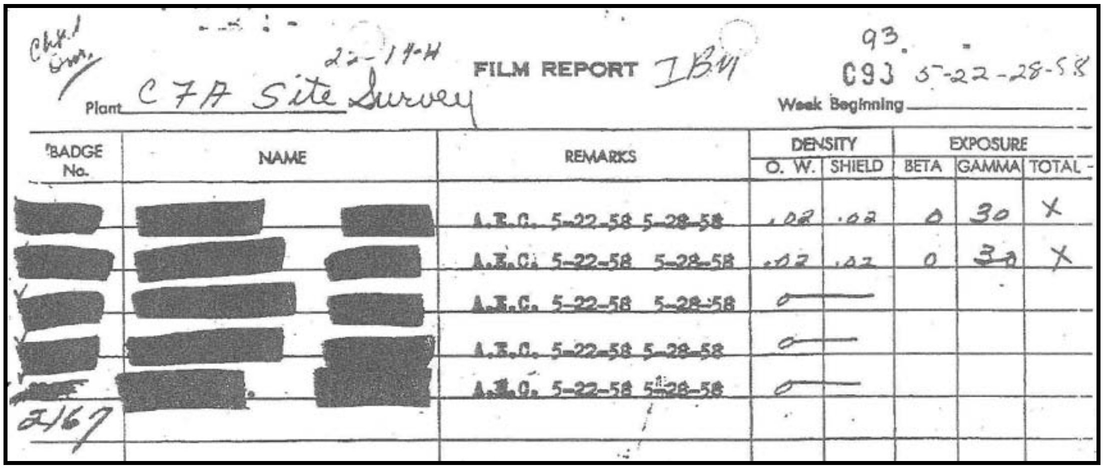
Form IHP-18 WEEK ENDING DEC. 25, 1954 SUB TOTALS 255+ 0 0

BADGE # [redacted] AEC

Figure 6-1. Individual dose reporting form in use until 1958.

Figure 6-2 is a report from reading the films in the same period. On three of the five badges, the more sensitive open window result was zero, so the shielded film was not read. On the other two, the open window and shielded values were at the minimum recorded density of 0.02, which corresponded to a 30 mR penetrating dose.

After the pencil ionization chambers were replaced with self-reading pencil dosimeters (also ionization chambers), the ANL-W operational health physics staff would rezero the dosimeters. The film reading was automated and results were stored in a computer. The form shown in Figure 6-3 reports badge

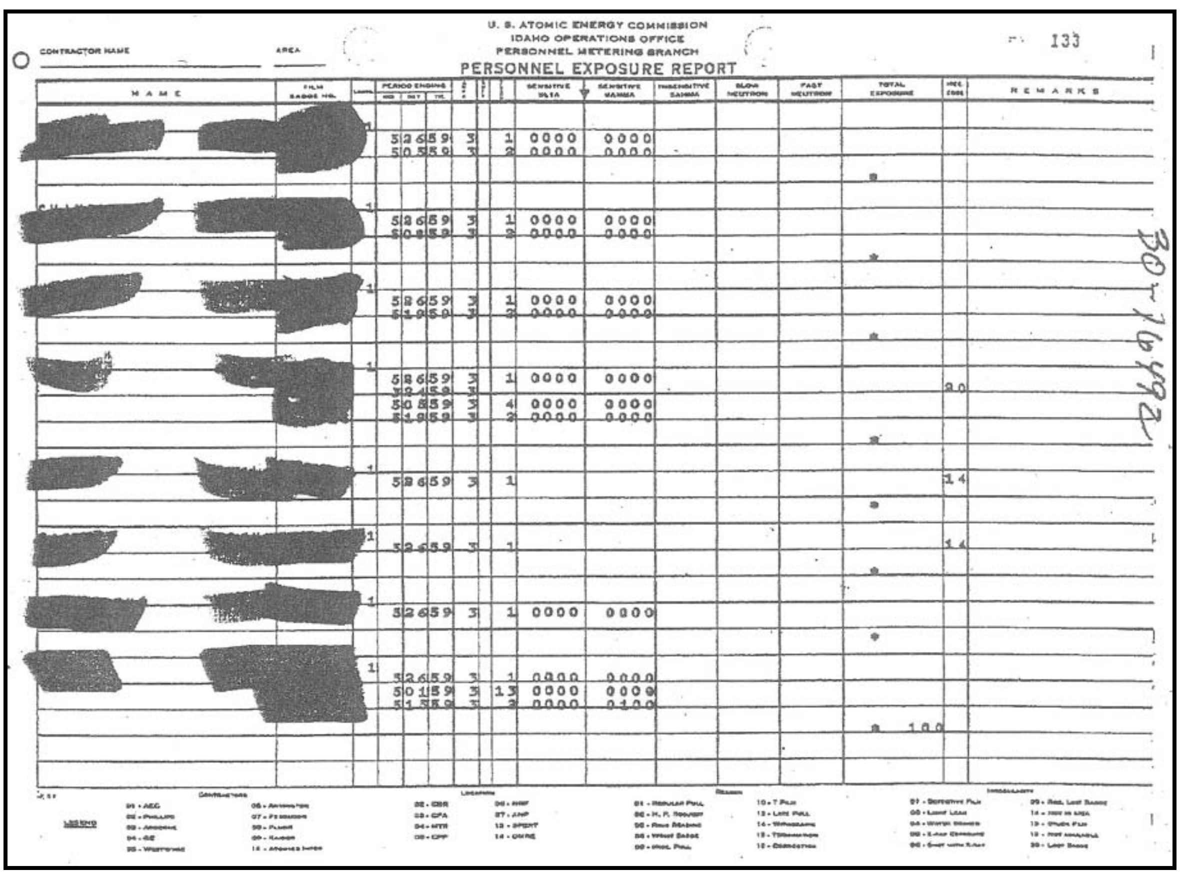


Chk'd Orig. 22-14-H FILM REPORT IBM 93 C93 5-22-28-58

Plant C 7 A Site Survey Week Beginning

BADGE No.	NAME	REMARKS	DENSITY O. W.	DENSITY SHIELD	EXPOSURE BETA	EXPOSURE GAMMA	EXPOSURE TOTAL
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	.02	.02	0	30	X
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	.02	.02	0	30	X
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	0				
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	0				
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	0				
2167							

Figure 6-2. Film report form used in 1958.



U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
PERSONNEL METERING BRANCH
PERSONNEL EXPOSURE REPORT

133

30-76492

NAME	FILM BADGE NO.	PERIOD ENDING			SENSITIVE BETA	SENSITIVE GAMMA	TOTAL EXPOSURE	REC. CODE
[redacted]	[redacted]	52659	3	1	0000	0000		
		50559	3	2	0000	0000		
[redacted]	[redacted]	52659	3	1	0000	0000		
[redacted]	[redacted]	52659	3	1	0000	0000		
[redacted]	[redacted]	52659	3	1	0000	0000		20
		50859	3	4	0000	0000		
[redacted]	[redacted]	52659	3	1				14
[redacted]	[redacted]	52659	3	1				14
[redacted]	[redacted]	52659	3	1	0000	0000		
[redacted]	[redacted]	52659	3	1	0000	0000		
		50159	3	13	0000	0009		
		51359	3	2	0000	0100	100	

Figure 6-3. Monthly badge reporting form from May 1959.

readings for May 1959 when badges were exchanged every two weeks. The column under the P of Personnel is an area designator with the code listed under Location at the bottom of the page. The next column was unused and dropped somewhat later. The next column is a reason code. Attachment 6A lists the codes and their meanings. Figure 6-4 is a listing of some doses received during recovery from the SL-1 accident. Workers from several areas were pulled into the accident recovery process, and it is notable that one result exceeds the dose limits and that there are few zeros. Figure 6-5 is a follow-up badge report for one result on Figure 6-4.

OFFICIAL USE ONL

42

NAME	CONTR.	AREA	BADGE NUMBER	BETA	GAMMA
[redacted]	71	22	[redacted]		50
	07	17			365
	01	03			
	02	05			140
	73	00			1175
	02	16			345
	30	00		550	740
	01	03			35
	04	07		120	250
	71	22		210	205
	02	03			60
	67	16		120000	23195
	07	16			570
	07	04			
	01	03			65
	01	03		1475	2370
	01	01			30
	02	05			
	01	03		190	145
	01	03			15
	02	00			
	01	03			245
	01	03		135	210
	01	03		530	35
	01	00			
	04	16		120	130
	01	03			180
	02	04			20
	04	07			

OFFICIAL USE ONLY

Figure 6-4. Badge pull results from January 1961 for work in recovery from the SL-1 accident.

10-105
(3-58)

BADGE REPORT AEC 07A

Copies: 1. H. P. Representative
2. File

Special Report master 032

TO ____________

AREA SL1 [redacted] Bldg. ____________ Date ____________

RE: [redacted] Badge No. [redacted]

The badge on the above named employee recorded:

SEN Beta	SEN Gamma	INS Beta	INS Gamma
530	35		

The period extending from ____________ through 1-31-61

Badge pulled for reason listed below:

☐ High Pencil Readings of Taken from C[illegible] [illegible] pull

☐ Damaged Pencil DO Special Report [illegible]

☐

Week 5

Signed ____________

Figure 6-5. Special badge report associated with a high beta reading listed in Figure 6-4.

When there has been a question about a dose value being assigned to an INEEL worker, a Personnel Exposure Questionnaire was normally initiated as shown in Figure 6-6 (shows a hypothetical case). Based on this form a beta dose of 500 and a gamma dose of 350 for a total dose of 850 mrem would override the pocket meter dose of 290 total.

PERSONNEL EXPOSURE QUESTIONNAIRE

Date 1-5-58

Name of employee Doe, Jim S# 12345 Badge Number 1003

Area CPP Exposure Date 12-29-57--1-4-58

Reason fo Investigation.

( ) A reportable weekly daily pocket meter reading total of

X Weekly film total of 300 mr or more.

( )

Film total covers period extending from 12-29-57 through 1-4-58

FILM RESULTS

BETA	GAMMA
500	350

EXPOSURE RESUME

Week Ending	Meters	SUN.	MON.	TUES.	WED.	THURS.	FRI.	SAT.
1-4-58	Pocket Meters	-	20	40	60	90	80	-
	Badge Meters	-	-	-	-	-	-	B-500 G-350

Remarks Total 850 mrem

Investigation

a. Findings of Health Physics Representative and or Supervisor.

b. Recommendations.

Investigated by ______ Date ______ Noted ______

Health Physics Supervisor

Figure 6-6. Personnel exposure questionnaire partially completed for a hypothetical case.

6.3.2 Personnel Monitoring Systems Used at INEEL/ANL-W

6.3.2.1 Initial Film Badge

The badging system in place when operations began at the NRTS was the Self-Service System (Cipperley 1958). This film system, in use from August 1951 to March 1958, used the Oak Ridge National Laboratory stainless-steel holder design, which was 1.875-in. long, 1.375-in. wide, and 0.25-in. thick. Badges were processed weekly. The upper portion of the badge was shielded with 1 mm of cadmium and the lower portion was an open window. Sensitive and insensitive DuPont type 552 film was used for beta-gamma dosimetry for most locations; DuPont type 558 film (type 508 sensitive and type 1290 insensitive) was used at two reactor areas.

Gamma calibration was to a radium source, and beta calibration was to a metallic uranium plate. To determine doses, the film densities were read to ±0.02 density unit. A calibration curve was used to convert the cadmium-shielded portion to penetrating gamma exposure in roentgen. The open window density corresponding to the gamma exposure was subtracted from the measured open window density and the remainder was converted to beta dose in rep.

Type 552 and 508 films have Minimum Reporting Levels of approximately 30 and 10 mR, respectively, to radium gamma radiation. The open window responds to beta radiation as well as X-rays and low-energy gamma rays. Because of the high atomic number (Z) of film in relation to air or tissue, the open window overresponds per unit exposure to low-energy photon radiation, as shown in Figure 6-7, by about a factor of 30 at 40 keV. Using a cadmium filter with its high Z severely attenuates the photons that get to the film, so the overresponse is reduced to about a factor of 2 at 125 keV and becomes less-than 1 at about 50 keV. The beta particle range is independent of Z; the range depends only on the density, so the 1-mm cadmium filter (~900 mg cm^{-2}) acts like a tissue depth of 9 mm for beta radiation.

Wrist badges used the same package attached to a wrist band. A finger ring used a small piece of film with a silver or cadmium filter. Pencil ionization chambers were used to monitor daily doses and control operational activities. The dosimetry group read and recorded these pencil readings on cards. Film badge readings were written on the same cards to indicate the dose of record. In 1958, the Victoreen 352 pencil ionization chambers being read by the dosimetry group were replaced with self-reading dosimeters that were read and rezeroed by the field health physics technicians (AEC 1959, p. 11). Film readings remained the dose of record.

6.3.2.2 Multiple-Filter NRTS Film Badge

In March 1958, the security badge and film badge were combined in a film badge containing filters of 1 mm cadmium, 0.013 mm silver, and 0.5 mm aluminum with thicknesses of 950 mg cm^{-2}, 203 mg cm^{-2}, and 175 mg cm^{-2}, respectively, including the plastic in which they were mounted (Cipperley 1968). This NRTS badge was also a security badge, resulting in an absorber thickness of 100 mg cm^{-2} in the open window, which filtered out beta radiation below 360 keV.

With the four absorbers, it was possible to separate beta radiation from low-energy photon radiation and to determine photon energy to a degree. Figure 6-7 shows photon energy dependence of the darkening behind the four filters for a combination of uranium beta and X-ray irradiation provided by NBS (Cipperley and Gammill 1959). With Dupont type 508 film, mixed exposures of radium gamma and uranium beta of 10, 20, and 30 mR or mrep were measurable within ±12 mR with 95% confidence. A minimum reporting level of 10 mrem was used for both beta and gamma radiation (AEC 1962).

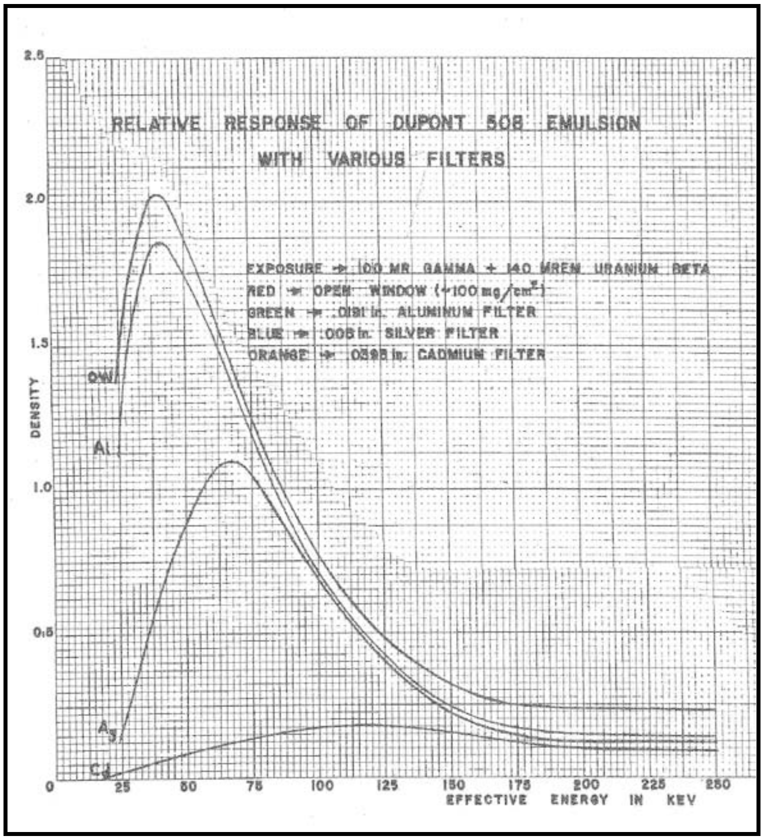


Figure 6-7. Response of DuPont 508 film with various filters to 140 mrem uranium beta and 100 mR of different energy photon irradiation provided by NBS. The original badge used the open window and cadmium shielded films. The multiple filter badge used all three filters plus the open window (Cipperley and Gammill 1959).

The ID Instrument and Development Branch developed an automatic film reader and densitometer (Purcell and McGary 1963). An algorithm based on probit corrected densities was developed to determine the high-energy photon, beta, and low-energy photon contributions separately (Cipperley 1968, p. 94). The cadmium filter provided the hard gamma component. The uranium beta responses under the open window, aluminum, and silver filters were 1, 0.2, and 0.1, respectively. By assuming a beta signal and subtracting it, the remaining signal could be attributed to low-energy photons and the energy could be estimated. For beta sources other than uranium, the analysis had greater uncertainty.

Because about 95% of the weekly badge films had doses less than 30 mrem, in 1958 the badging interval was increased to biweekly or monthly with the exception of the high-dose areas where the weekly schedule continued (AEC 1959). The introduction of punch cards increased the efficiency of report and record generation. A 12-point calibration curve was generated for radium and for ^{137}Cs gamma and uranium beta. Calibration did not use a phantom.

Experience following the SL-1 accident showed a wide variation of beta-to-gamma ratios and necessitated controlling both radiations rather than just the gamma. A set of as many as 18 badges could and in many cases was fastened on a belt around the worker to determine a beta:gamma ratio for each particular entry for entries into SL-1 during recovery operations (Cipperley, Henry, and Cusimano 1965).

6.3.2.3 Original Lithium Fluoride Teflon TLD System

Beginning in November and December 1966, individuals projected to receive doses of less than 0.5 rem yr^{-1} were placed on a lithium fluoride (LiF) disk thermoluminescent dosimeter (TLD) badge, which was exchanged quarterly (Cusimano and Cipperley 1968). Two 13-mm-diameter Teflon disks, 0.4 mm thick [100 mg (75 mg cm^{-2}) impregnated with 28 mg LiF], were mounted in a badge behind an open window and a 1-mm cadmium filter (Watkins, date unknown). The disks, manufactured by Teledyne Isotopes, were read with the Teledyne Model 7300 TLD reader. LiF was chosen because the average Z is close to that of air and tissue, resulting in little energy correction for beta or gamma radiation. The badge could read 30 mR on a quarterly basis, so more small doses were reported. The angular dependence of the gamma response (within 10% to 70º) is superior to film because the material acts like an ionization chamber. For normal monitoring, only the open window TLD was read and considered penetrating dose unless it read more than 125 mrem, in which case the shielded TLD was also read.

The pilot tests were successful, and the LiF Teflon TLD system was phased into use in 1966, particularly for individuals who would receive low doses, with longer exchange cycles, typically 3 or 6 months. In July 1968, the monitoring period was increased from 3 to 6 months (AEC 1969). In December 1972, annual processing was used for 1,190 low-dose individual TLDs and 960 were processed quarterly (Cusimano 1972). Employees on a monthly badge change were moved to this system as late as September 1973.

The system had an automatic badge calibrator that did not involve a phantom to provide backscatter (Cipperley 1966; AEC 1970, p. 8). A later discussion introduced the use of a ^{137}Cs source, so these earlier calibrations probably used radium.

6.3.2.4 INEEL ATLAS TLD System

Development began in 1969 on a patented Automatic Thermoluminescent Analyzer System (ATLAS). It used LiF in a homogeneous mixture with Teflon and replaced the film in the multi-element badge using the same filters. ATLAS became operational for monthly badge changes in February 1974. In June 1974, questions about this system were formalized (Black 1974; Walker 1974) and the system was soon replaced.

6.3.2.5 Harshaw Two-Chip TLD System

Several unstable characteristics with ATLAS led to rapid implementation of a two-chip TLD system in May 1975 for Argonne. This commercial Harshaw system used two LiF TLDs 240 mg cm^{-2} thick. In 1976, holes were punched in the security badges to restore the open window. One chip was covered by 540 mg cm^{-2} of aluminum and the other was under 4 mg cm^{-2} of Mylar. The aluminum-covered chip provided penetrating dose at a nominal tissue depth of 1 cm. The beta dose was calculated from the difference between the two chips. Because of the thickness of the Mylar-covered chip, the beta dose was accurate only for the beta energy used in calibration. Field calibrations were used to reduce the problem with beta energy dependence. The thin aluminum filter (density thickness 350 mg cm^{-2}) allowed higher-energy beta radiation to expose the chip used for measuring the penetrating (1,000 mg cm^{-2}) dose.

The practice was to read only the open window chip to determine if the nonpenetrating dose was above 15 mrem and thus reportable. If the threshold dose was exceeded, both chips were read and the penetrating and nonpenetrating doses were computed (Kalbeitzer 1983).

6.3.2.6 Panasonic Four-Chip System

In 1986, with the advent of DOELAP, INEEL went to a four-element system, the Panasonic 814 AS4 (Gesell, Hall, and Andersen 1992; INEEL 2001). Lithium borate ($Li_2B_4O_7$) TLD elements with plastic and aluminum filtration provide an improved measurement of deep dose equivalent and, with a thinner filter, an improved measurement of the shallow dose equivalent. A calcium sulfate ($CaSO_4$) TLD provides a strong low-energy photon response. The elements are 15 mg cm^{-2} thick. Element one has filtration of 16 mg cm^{-2}, element two has filtration of 58 mg cm^{-2} plastic, and elements three ($CaSO_4$) and four have filtration of 550 mg cm^{-2} of plastic and 50 mg cm^{-2} of aluminum. Although none of the elements are at a depth of 7 mg cm^{-2}, the specified depth for the shallow dose equivalent, an acceptable response can be obtained by using elements at 16 and 58 mg cm^{-2}. This system is qualified in DOELAP beta, photon, and mixture performance categories.

The minimum reporting level was 15 mrem beta and gamma from January to July 1986 (Gesell 1986), 10 mrem gamma and 30 mrem beta from July 1986 to about September 1989, and 15 mrem for gamma and 30 mrem for beta until 1993 (Perry, Andersen, and Ruhter 1993), when it returned to 10 mrem gamma.

6.3.2.7 Nuclear Track Emulsion, Type A for Neutrons

Kodak nuclear track emulsion, type A (NTA) was used for neutron monitoring when the field radiation protection staff requested it. NTA responds to neutrons with energies above 500 to 800 keV, for which the proton recoil tracks leave enough energy to expose at least three (four in some references) grains of emulsion.

Before 1958, if a proton recoil track was counted in 40 microscope fields, it was read twice more for a total of 120 fields (Cipperley 1958). The reported MRL was 14 mrem. This is confirmed based on inspection of a March 1958 processing data sheet for several workers with positive neutron doses of 14, 14, 14, and 42 mrem, and also on examining doses for one worker during January through March 1958 with positive recorded doses of 14, 28, 42 and 14 mrem. On the March data sheet, six people had 0 in the neutron column and 17 had blanks. The zeros had NTA with nothing observed and the blanks probably were not monitored for neutrons or the film was not read.

After 1959, if more than three proton recoil tracks were counted in 40 microscope fields, a different location on the film was counted by two other technicians, providing three independent results (Cipperley 1968). Two tracks or fewer were attributed to background. This resulted in a somewhat higher MRL. In November 1959, a data sheet shows 10 and 20 mrem neutron dose equivalents. In January 1962 a data sheet shows a 20 mrem dose. A data sheet from April 1959 shows neutron dose equivalents of 20, 20, and 40 mrem. These values suggest an MRL of 20 mrem.

Calibration was with a polonium-beryllium (PoBe) source (approximately 30 Ci), resulting in 5.87×10^{-4} tracks/neutron (Cusimano 1963). Uncertainties were assigned at the 90% confidence level. Cipperley (1968, pp. 102-115) discusses this process.

The field health physics staff was aware of the energy limitations of the NTA badge (Sommers 1967, 1969) and compensated with neutron-detecting pencil dosimeters and field measurements. A request to read NTA film occurred if the hard spectra neutron exposure was likely to exceed 10 mrem.

6.3.2.8 Neutron Albedo Dosimetry

Because of the missed dose from neutrons below the NTA energy threshold of 0.5 to 0.8 MeV, particularly at plutonium facilities, and because of the advent of TLD techniques, several development efforts in neutron dosimetry occurred in the early 1970s. The result was several designs using the albedo technique in which scattered neutrons from the wearer's body were absorbed by ^{6}Li in a TLD.

In the Hankins dosimeter used at the INEEL and ANL-W (Hankins 1973), TLDs (^{6}Li to capture thermal neutrons) are inside a polyethylene case (to lower the neutron energy) inside a cadmium shell (to eliminate thermal neutrons from outside). Lithium-7 TLDs are used to subtract the gamma dose. Because the ^{6}Li(n,α)^{3}He reaction has a strong energy dependence, the response does not follow the flux-to-dose-equivalent conversion, so the neutron signal is divided by a facility neutron correction factor (FNCF). An FNCF that converts the TLD neutron response signal to neutron dose equivalent can be generated from the ratio of the dose equivalent measured with a 9-in.-diameter Eberline PNR-4 and the corresponding signal (reads in mrem, but not dose equivalent) with the detector in the 3-in.-diameter PNR-4 insert (Hankins 1976). The FNCF values shown in Table 6-2 (Cusimano 1981) were measured for different fields at INEEL, were tabulated for assigning the dose equivalent from the badge results, and were routinely updated.

Table 6-2. INEEL Facility Neutron Correction Factors from 1981.

Organization	FNCF	Organization	FNCF
DOE-CFA	0.092	EG&G-TRA (Bare PuBe)	0.06
EG&G-CFA	0.092	EG&G-TRA (PuBe in poly)	0.23
ANL-TREAT	1.05	EG&G-LOFT	3.5
ANL-ZPPR	0.94	EG&G-ARA III	2.0
EG&G-TRA (L & S)	1.2	EG&G-RWMC	0.33
EG&G-TRA (SA)	2.7		

The date of changeover from NTA to albedo neutron monitoring is somewhat in dispute. Typically, different organizations would transition to new monitoring systems at different times. The present record suggests it occurred near the end of 1974 or early 1975 (Ruhter and Perry 2002; Gesell et al. 1996), but an informal list of "Dosimetry Branch Changes" from 1978 (INEL c. 1978) states "initial testing of albedo neutron dosimeter and replacement of NTA neutron monitoring film with same" in October 1976. Aoki (1979) says the Albedo system replaced the NTA badge in 1977. Dose reconstructions should make the claimant-favorable assumption that this transition occurred in October 1976.

6.3.3 Calibration

6.3.3.1 Beta-Gamma Radiation

Gamma calibration initially used a radium source. Victoreen R meters standardized by NBS were used to measure radiation fields (AEC 1959, p. 132). Uranium metal bars 5 mm thick were used for beta calibrations. Cesium-137 was considered for a calibration source in 1959 (AEC 1960, p. 83) and was installed in the instrument calibration facility in 1961 (AEC 1962). An automatic badge irradiator developed in the 1960s (Cipperley 1966) did not use a phantom to provide backscatter.

As reported in 1981, an extrapolation chamber was built for the measurement of beta doses (Gupta 1981). The chamber window was polycarbonate, the gas was air, and the thick collecting electrode was Shonka tissue-equivalent plastic. The chamber was used to calibrate a 2.5 Ci ^{90}Sr/Y source to tissue rad. The source with area, 2.5 cm^{2}, was constructed by the Amersham Searle Corporation in

February 1975. This source was used to measure beta correction factors for several instruments following the Three Mile Island TMI-2 reactor accident in 1978. TLD badges were calibrated to 500 mrad tissue using a 1.78-cm-thick phantom 50 cm (300 rad hr^{-1}) from the source.

In January 1983, the natural uranium slab again became the primary calibration source for nonpenetrating radiation to better approximate field beta spectra (Gesell 1982a).

Use of a phantom in calibration apparently started about 1981, with the NVLAP certification process developed for non-DOE dosimetry processors. About this time, calibration techniques developed in terms of absorbed dose to tissue rather than exposure. Beginning in January 1981, in response to a draft National Voluntary Laboratory Accreditation Program (NVLAP) standard, dosimeters for calibration were irradiated with ^{137}Cs using a phantom backing. To convert from exposure in roentgen to dose equivalent index in rem, the Radiological and Environmental Services Laboratory (RESL) used a conversion factor (C_x) of 1.08 was used (RESL 1981). The current recommended C_x value of 1.03 for ^{137}Cs (DOE 1986b, Table 2) was used beginning in June 1981 (Gesell 1982b; Kalbeitzer 1984).

In 1989, the INEEL dosimetry service transferred from RESL to EG&G, Inc., the prime contractor. Calibrations continued to use DOE RESL sources, and no changes were made to the dosimetry system. The 1991 Tiger Team Review of the INEEL site indicated that the joint use by the INEEL contractor and the Idaho Operations Office of the same sources for calibration led to a conflict of interest or an advantage in DOELAP tests. As a result, EG&G purchased a Shepherd panoramic irradiator with a ^{137}Cs source for badge irradiations. This irradiator does not use a phantom, but was cross-referenced using many TLD irradiations to the DOE source using a phantom (Andersen 1994). In addition, the contractor developed and characterized a uranium slab it owned for beta irradiations (Bean 1995).

Table 6-3 lists common sources of laboratory bias for personnel beta/photon dosimeter calibration based on comparison of the recorded dose with $H_p(10)$.

6.3.3.2 Neutron Calibration

The initial NTA neutron badges were calibrated using a PoBe neutron source (30 Ci in 1958) (AEC 1959). In 1982, an AmBe source was used (Cusimano 1982). Alpha particles from the americium or polonium interact in the $^{9}Be(\alpha,n)^{12}C$ reaction and generate a broad spectrum of neutrons up to about 11 MeV (mean energy about 5 MeV) as shown in Figure 6-8 (Kluge and Weise 1982). The yield of the AmBe source should be only about 3% larger than that for the PoBe source (Anderson 1971). Kluge and Weise (1982) calculate conversion factors of 3.51 to 3.76×10^{-8} rem $cm^2\,n^{-1}$ depending on the particular measure of dose equivalent chosen. IAEA (1988) provides a dose conversion factor for AmBe of 3.8×10^{-8} rem $cm^2\,n^{-1}$ for the maximum average dose equivalent. A dose equivalent of 1.5 rem required 3.6×10^{7} n cm^{-2} (Cusimano 1963), corresponding to a dose conversion factor of 4.17×10^{-8} rem $cm^2\,n^{-1}$, so the recorded dose will be about 11% high. Monte Carlo calculations for 5-MeV neutrons show a dose equivalent of about 4.2×10^{-8} rem $cm^2\,n^{-1}$ averaged over the 0- to 2-cm shell on a 30-cm-diameter cylindrical phantom (NCRP 1971b). Use of the 50-Ci AmBe source continued until 1993.

Table 6-3. Laboratory sources of uncertainty for beta/photon dosimeter calibration parameters.

Parameter	Historical description	Uncertainty[a]	Comment
In-air calibration	In 1981, INEEL began exposing calibration dosimeters on phantoms (used to simulate worker body). Previous calibrations do not include response from radiation backscatter response.	+10%	Recorded dose of record **too high**. Backscatter radiation from worker body is highly dependent on dosimeter design.
Radiation quantity	Before 1981, INEEL dosimeter systems were typically calibrated to a photon beam measured as *exposure*.	-5%	For higher energy ^{226}Ra and ^{137}Cs gamma radiation used to calibrate dosimeters, this caused a slight (about 3%) **underresponse** in recorded dose.
Tissue depth of dose	Historically, INEEL used an unspecified depth to estimate the deep dose.	±5%	The numerical effect of this for photon radiation is comparatively low. INEEL dosimeter designs had filtration density thickness of about 1,000 mg cm^{-2} that would relate closely to the 1-cm depth in tissue.
Angular response	INEEL dosimeter system is calibrated using anterior-posterior (AP) laboratory irradiations.	>300 keV, ~20%	Recorded dose of record likely **too low** because the dosimeter response is usually lower at non-AP angles. Effect is highly dependent on dosimeter type, radiation type, energy, and angle.
Environmental stability	INEEL film dosimeter and TLD systems are subject to signal fade with time, heat, humidity, light, etc.	±10%	Recorded dose of record depends strongly on dosimetry parameters such as when calibration dosimeters were irradiated and processed. Mid-cycle calibration minimizes effects.

a. Uncertainty estimate in recorded dose compared to $H_p(10)$ based on judgment.

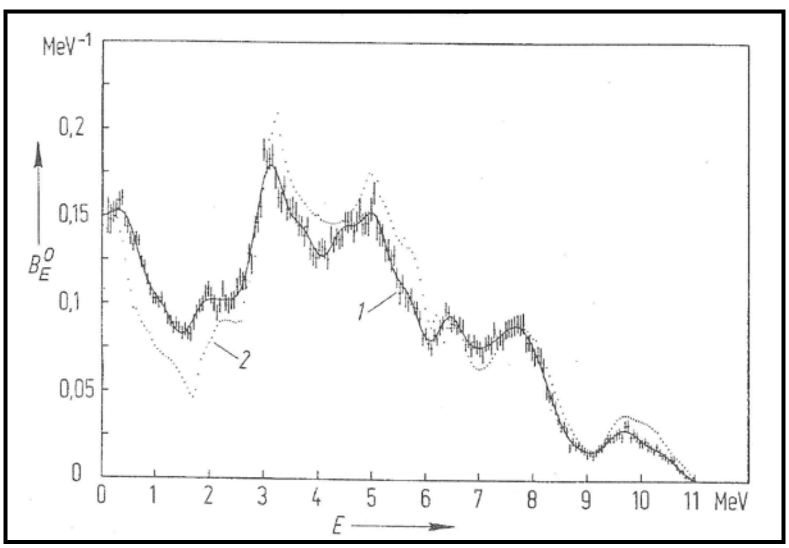


Figure 6-8. Probability density of neutron spectrum from a $^{241}AmBe$ (α,n) source (Kluge and Weise 1982).

In 1993, a 40- by 40- by 15-cm polymethyl methacrolate phantom was placed near the unmoderated ^{252}Cf source used for instrument calibration, and the system was characterized for TLD calibration (Gesell et al. 1996, Appendix A). This system has since been used for neutron dosimeter quality assurance measurements. Calibration factors from the DOELAP manual are used (DOE 1986b).

Table 6-4 outlines several common sources of expected laboratory bias for personnel neutron dosimeters based on comparison of the recorded dose with $H_p(10)$.

Table 6-4. Common sources of laboratory bias in the calibration parameters for neutron dosimeters.[a]

Parameter	Historical description	Anticipated laboratory bias[b]
Source energy spectrum	In 1976, INEEL began using dosimeters calibrated on a phantom to simulate a worker's body. The previous calibrations did not include response from backscattered radiation.	NTA film tends to be insensitive to albedo neutrons, so probably had minimal effect.
Radiation quantity	Neutron dose quantities used to calibrate INEEL neutron dosimeters have varied historically. The *first collision dose* for fast neutrons and a *quality factor* of 10 were used for many years.	As noted above, NTA calibration would result in the reported dose being about 11% high. The effects of the respective neutron dose quantities used to calibrate INEEL dosimeters is uncertain and could be evaluated in comparison to the Hp(10) dose used in DOELAP performance testing.
Angular response	INEEL dosimeters calibrated using AP laboratory irradiation.	Recorded dose of record is likely too low because the dosimeter response is lower at non-AP angles. The effect is highly dependent on neutron energy.
Environmental stability	INEEL NTA film and TLD dosimeters are subject to signal fade with time, heat, humidity, light, etc.	Recorded dose of record is likely too low; however, this depends strongly upon when the calibration dosimeters were irradiated during the dosimeter exchange cycle. Mid-cycle calibration minimizes the effects.

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

6.3.3.3 Gamma Radiation

In response to a Tiger Team finding, radiation fields at INEEL have been characterized by making field measurements with a NaI(Tl) gamma spectrometer and TLDs mounted on a phantom, then comparing the results (Reilly 1998). Figure 6-9 shows the percentage bias for the beta and gamma measurements. Most results lie between +27% to -43 %.

The high gamma bias results were for locations at the Radioactive Waste Management Center (RWMC) looking at skyshine (back scattered and thus low-energy photons) from low-level waste in the Subsurface Disposal Area. The doses measured with NaI(Tl) were low (6 and 11 mrem) and the threshold energy on the NaI(Tl) detector was about 100 keV, so some low energy photons are likely to have been missed.

6.3.4 Workplace Radiation Fields

The radiation fields at ANL-W, with a few exceptions, have been generated primarily by mixed fission and activation products. Therefore, most of the photon doses have come from photons with energy greater than 250 keV. The INEEL dosimeters are judged to measure these fields well.

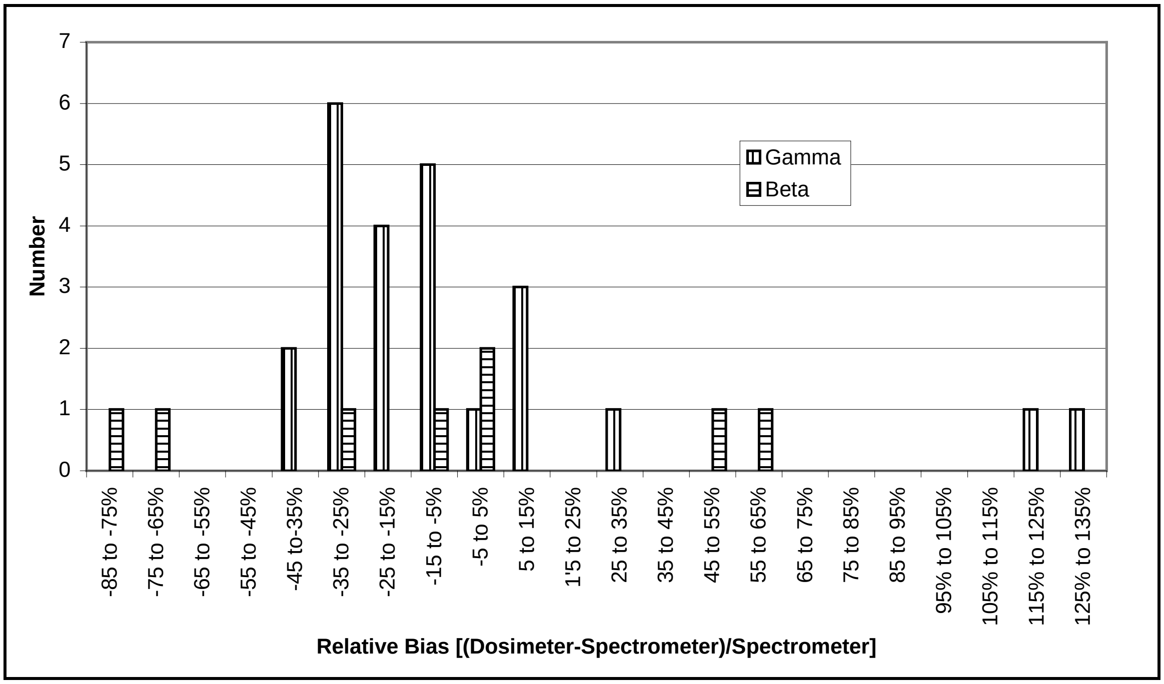


Figure 6-9. Gamma and beta radiation field characterization.

The few exceptions have usually been characterized by low dose rates. Analytical X-ray generators operating below 100 keV have been used in several laboratories. These are easily shielded so the fields have usually been low.

There have been a few 250-keV X-ray generators used for radiography or radiography development studies. Wall shielding has generally been adequate and any transmitted photons would have had energy near the operating voltage because of the hardening caused by the shielding.

Essentially all ANL-W radiological work areas involved beta/photon radiation covering a wide range of energies. These fields can be generally classified according to the IREP codes in Table 6-5.

Table 6-5. Selection of IREP beta and photon energies for ANL-W facilities.

Process/ buildings	Description	Operations Begin	Operations End	Radiation type	Energy group (keV)	Percentage
Reactors	Highly dispersed fields of higher energy photon radiation fields from fission process, activation and fission product nuclides. Potential for significant airborne nuclides, and there could be significant higher energy beta radiation			Beta Photon	>15 30–250 >250	100 25 75
	EBR, ANL					
		1952	2003			
Processing plants	Highly dispersed fields of higher energy photon radiation fields from activation and fission product nuclides dominant to most exposure profiles. Potential for higher-energy beta radiation during sampling and maintenance work resulting from fission products.			Beta Photon	>15 30–250 >250	100 25 75
	ICPP, ANL	1952	2003			
Calibrations	Calibration of instruments and dosimeters			Beta Photon	>15 30 – 250 >250	100 25 75
		1952	2003			

6.3.4.1 Beta Radiation

Beta radiation fields are usually associated with activation or fission product radioactivity that is lightly shielded, handled by workers such as in hot cells, or outside of a container such as a spill. The high bias results in Figure 6-9 from comparison of TLD to a phoswich beta spectrometer are for beta sources at contact or at 1 cm, which results in a geometry that is difficult to reproduce. The low-bias results are for large area beta sources for which even the spectrometry results have large variations. The beta occupational radiation fields in Figure 6-9 (only 3) have a bias less than 15%.

Beta field dosimetry became fairly accurate with the definition of DOELAP requirements in the early 1980s. Before then, beta monitoring systems had various flaws, primarily in a detector too thick to give a good surface result or one that was covered with extra material. Calibration was to high-energy betas from either uranium or strontium. The dose from low-energy betas can be missed altogether if the beta energy is not sufficient to penetrate the detector cover, and the dose can be underreported if the beta energy is not sufficient to penetrate the entire detector. The mean beta energy for the spectrum from a particular nuclide is about one-third of the maximum beta energy for that nuclide.

Based on the range energy curve for beta particles and the beta energy distribution of beta emitters (HEW 1970, pp. 90, 91, and 123), the fraction of beta radionuclides with ranges greater than the abscissa is plotted on the ordinate in Figure 6-10. The beta nuclides varied by location and time so a correction factor common for all facilities was estimated. This analysis used the entire mixture of radionuclides to avoid questions of whether the choice is correct and to reflect the wide variety of radionuclides used. To reflect that the beta spectrum is not monoenergetic because of the energy carried off by the neutrino, a curve is presented for the mean energy (one-third of the maximum energy). To reflect that some beta particles enter the detector at an angle, a curve is provided for 45º incidence at the maximum energy and the mean energy. These curves of the fraction of nuclides

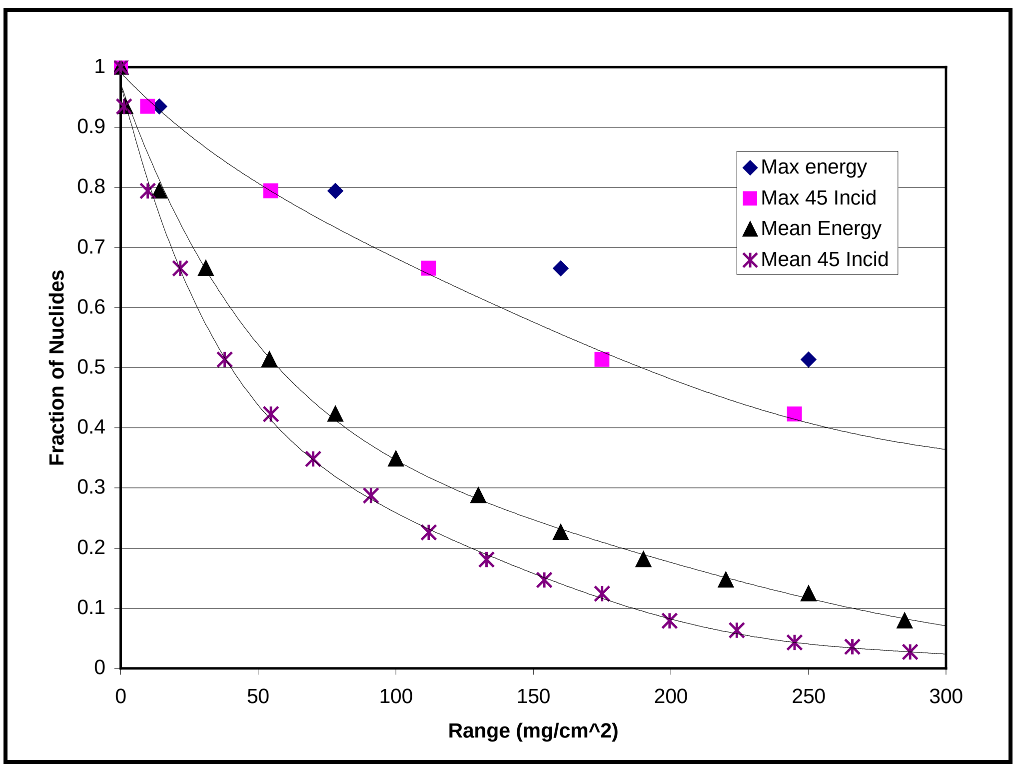


Figure 6-10. Distribution of beta ranges.

with a larger range essentially show the depth dependence of beta dose because the energy loss of electrons does not have much energy dependence. These curves also demonstrate why early dosimeters with thick, sensitive elements failed to correctly report the beta dose at a depth of 7 mg cm^{-2}, which was chosen in the early 1980s. In addition, these curves demonstrate why the beta dose assigned for skin is inappropriate to use for the breast and testes in which much of the organ is at a depth greater than 1 mm or 100 mg cm^{-2}, and for most of persons at depths greater than 1 cm.

To calculate the fraction of dose missed by a dosimeter, the dosimeter should average the appropriate curve of this nature over the depth of the active detector and compare it to the value at a depth of 7 mg cm^{-2}. The appropriate curve should be the curve of the mean range for the beta spectrum and the angular distribution of the radiation exposure. To estimate this value, this analysis added the mean energy curve for perpendicular incidence and 1.4 (relative path length) times that for 45º incidence for the mean energy and added one-half that value for 45º incidence for the maximum energy. The curves are the result of a polynomial trend line to the data, so averaging the fraction of radionuclides is relatively easy.

Table 6-6 provides the cover and detector thicknesses for the INEEL beta badges. The fraction of measured beta dose shown in Table 6-6 is the average as described above. To determine the corrected beta dose, the reconstructor should divide the nonpenetrating result from the dosimetry system by the values in the last column of Table 6-6. The reported dose will likely be somewhat higher than this because the calibration probably did not consider such a correction and reported the dose for the calibration exposure.

Table 6-6. Beta dosimeter thicknesses and associated underreporting.

Dosimeter system	Period	Covers (mg cm^{-2})	Detector thickness (mg cm^{-2})	Estimated dose recorded
Two filter film	1951–1958	50[a]	50[a]	0.49
Multi-filter film	1958–1974	100	20[a]	0.35
Low dose TLD	1969–1974	100	75	0.33
ATLAS	1974–1975	100	100[a]	0.30
Harshaw TLD	1975–1976	104	344	0.21
Harshaw TLD	1976–1985	4	240	0.41
Panasonic TLD	1986–present	16	15	0.86

a. Estimated.

6.3.4.2 Neutron Radiation

Most ANL-W workers have not been exposed to neutrons and so have not been badged to measure neutrons. Neutron fields are specific to a few facilities.

In 1969, 150 out of 2,900 film-badged and more than 3,000 TLD-badged employees at the INEEL, including ANL-W, were involved in radiation work that required their NTA neutron dosimeters to be evaluated (AEC 1969).

For calendar year 1979, 5 people received neutron doses between 0.5 and 1 rem, and 79 received measurable neutron doses below 0.5 rem (Jones 1980).

Individuals who have the potential to receive neutron dose currently wear albedo badges, and experience has shown that most do not receive significant doses. In the first 9 months of 1995 for INEEL, only 1,461 neutron dosimeters were issued (both monthly and quarterly badges) compared to about 50,000 beta/gamma badges. Only 54 badges had reportable doses (≥15 mrem) as shown in Figure 6-11 (Gesell et al. 1996). Only six were above 35 mrem. The Hankins albedo dosimeter badges in use since 1975 see all neutron fields. An FNCF determined from the 9- to 3-in. ratio in the worker location is used to adjust the measurement result to dose equivalent.

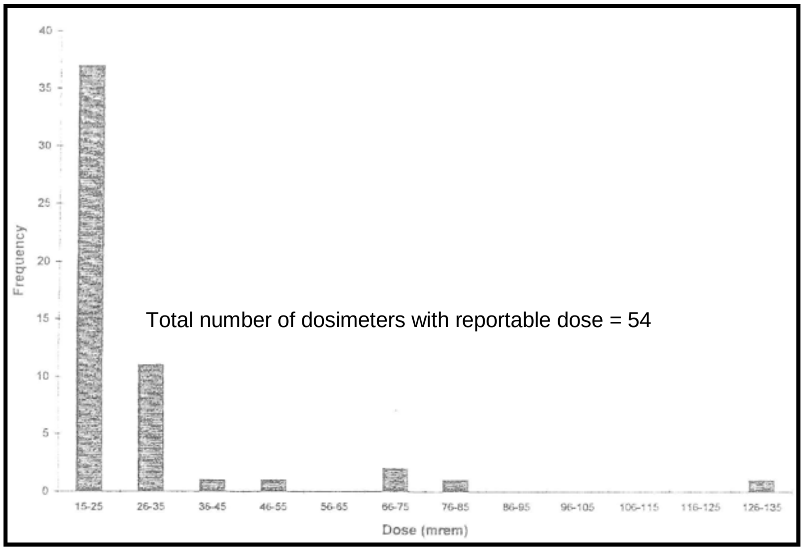


Figure 6-11. Distribution of reportable neutron dose at INEEL for the first 9 months of 1995.

In 1997, several workplace neutron fields were measured with TLDs mounted on a phantom and at nearly the same time a ROSPEC neutron spectrometer (Reilly 1998). Figure 6-12 shows the relative biases [(Dosimeter-Spectrometer)/Spectrometer] for the neutron fields. These results show a greater dispersion than the gamma results.

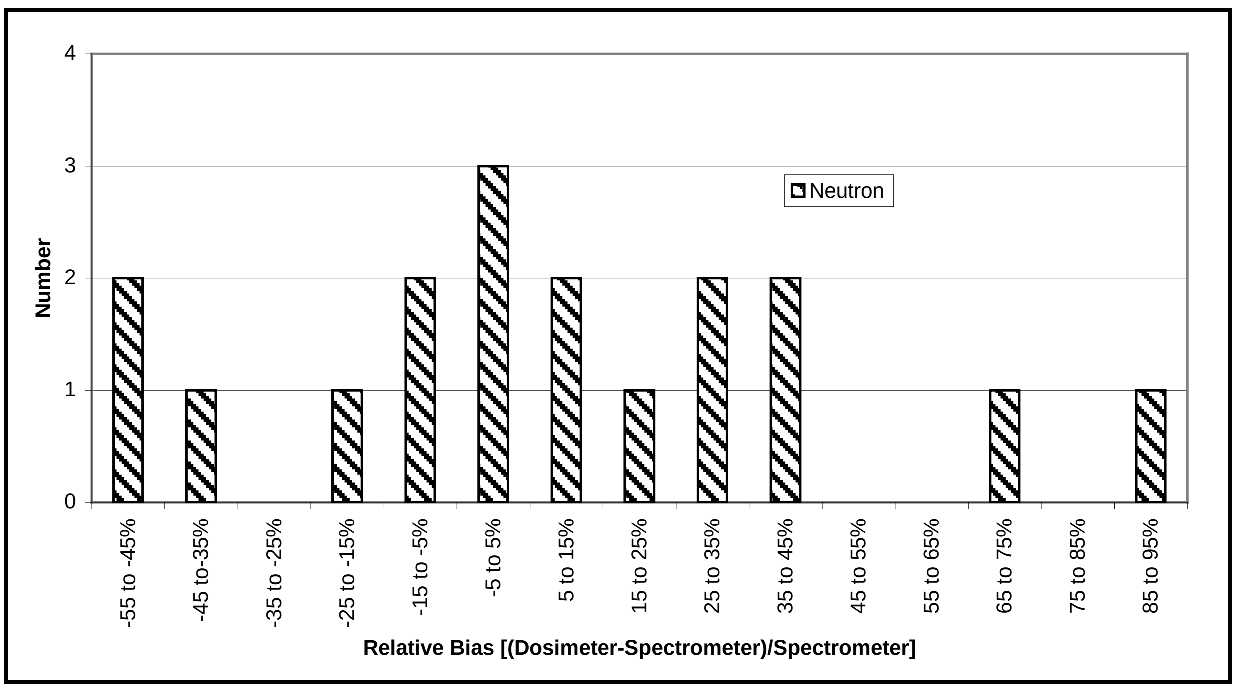


Figure 6-12. Neutron radiation field characterization.

The two lowest values (-0.52 and -0.51) are for TLD measurements on opposite sides of a phantom where the field is from ^{252}Cf on an overhead filter bank. The phantom attenuates the radiation from each side so the TLDs see only half the radiation field. The next lowest value (-0.38) is for the ^{252}Cf instrument calibration source at a distance of 3.5 m where the operator stands. The two highest values (0.94 and 0.71) are for a waste drum that was reanalyzed and a new 9-in. to 3-in. ratio determined because of the unsatisfactory initial result. Reilly (1998) suggests that other waste barrels may have had neutron sources causing interference. The remaining bias values lie between -0.16 and 0.44.

Sources of neutron exposure include neutron sources at the instrument calibration laboratories. For the spectra from these sources, the NTA works reasonably well. Use of small ^{252}Cf sources for research began after use of albedo badges. Figure 6-13 provides spectra for the AmBe (Kluge and Weise 1982) and ^{252}Cf (fission) neutron sources and the 14-MeV neutron generator as seen through 10 cm of polyethylene shielding (Ing and Makra 1978), which is typical of the INEEL/ANL-W facilities.

Most of the reactors at the INEEL/ANL-W have not had beam ports, and the neutrons have therefore been generally well contained away from the workplace. A reactor core environment is characterized not only by high neutron levels, but also by very high gamma levels. The gamma shielding is often water and concrete, which are also very good neutron shields. The neutron fields in the energy spectrum for reactors and lower are attenuated much more quickly in concrete or water than are gamma fields. This is not true for lead or iron, but these have not usually been used as gamma shields where neutrons exist. Therefore, neutron fields are generally not a large concern at an enclosed reactor.

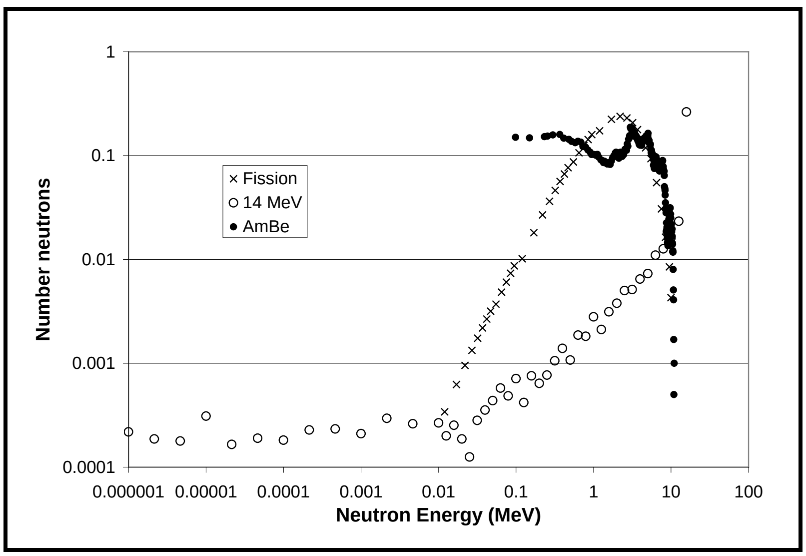


Figure 6-13. Neutron spectra simulating INEEL facilities.

6.3.4.2.1 MTR, ZPPR, and TREAT Neutron Radiation

The exceptions to the above discussion are:

- The Materials Test Reactor (MTR) at NRTS/INEL, which operated from 1952 to 1970 and had beam ports and neutron beams extending onto a research floor
- The Zero Power Physics Reactor (ZPPR) at ANL-W
- The Transient Reactor Test (TREAT) at ANL-W

Some neutron surveys of the MTR experimental floor have been recovered (Sommers 1959, 1962; Hankins 1961), but these individually do not provide all components of the radiation field. Hankins (1961) used 2-, 3-, and 8-in. polyethylene Bonner balls in a cadmium shield to characterize the intermediate and fast neutrons at 21 locations around the MTR floor. He made these measurements and also measured the thermal neutron component at six other locations. The Hankins data have been reanalyzed (Rohrig 2004) using more recent Bonner response curves (Hertel and Davidson 1985). Figure 6-14 shows the resultant neutron spectra for locations 3 and 23, which had higher doses and nearly the maximum low-energy intermediate and fission components, respectively. Figure 6-15 shows the correlations of the thermal and intermediate neutron dose equivalents to the

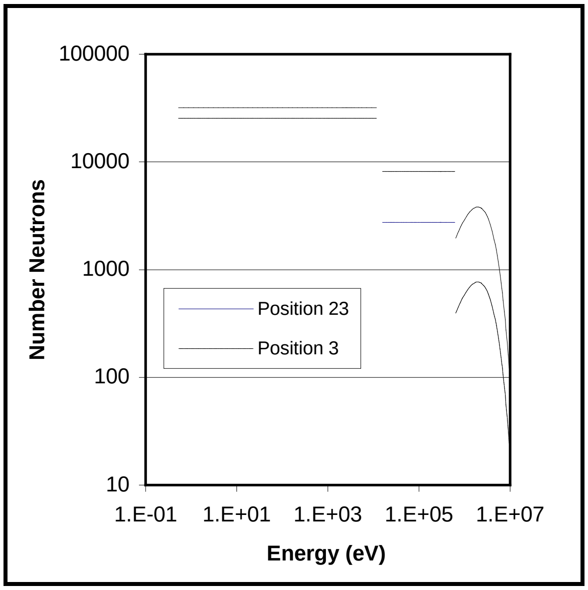


Figure 6-14. Sample MTR Spectra from Hankins Bonner Measurements.

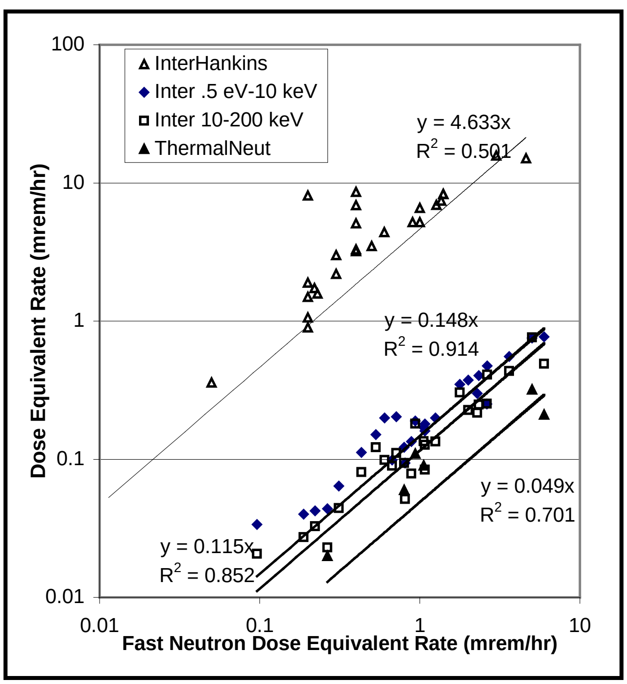


Figure 6-15. MTR neutron field components.

fast neutron dose equivalent for the Rohrig reanalysis of the Hankins data. The trend line for the reanalyzed intermediate energy neutron dose equivalent has a R^2 value of 0.85 and 0.91 as compared to a R^2 value of 0.50 for the original analysis, which fits the data better. The average ratio of thermal to fast neutron is 0.071 ±0.025, for the low-energy intermediate to the fast the ratio is 0.177 ±0.057, and for the higher energy intermediate to the fast the ratio is 0.149 ±0.046, where fast neutrons are taken as those above 0.2 MeV.

The MTR personnel likely to receive neutron dose were assigned NTA film in their dosimetry packets, but the packets would have missed doses below 0.5 to 0.8 MeV. For the MTR spectra, the fraction of neutron dose equivalent above 0.8 MeV has an average value of 0.52 ±0.08 and varies from 35% to 66% depending on the location. The dosimetry record location code for the Test Reactor Area (TRA) was 4 (later 40 to 45). To correct for missed dose on the MTR experiment floor, the NTA results from MTR should be multiplied by 2 ±0.2 (1/0.52, 0.08/0.52) for a Monte Carlo dose reconstruction or by 3 (1/0.35) for the less accurate worst-case reconstruction.

Sommers (1962) reported thermal and fast neutron dose equivalent rates and gamma dose rates around the MTR beam lines. The thermal measurements near beams are believed to not be representative of the general workplace. Figure 6-16 shows the correlation of fast neutron dose equivalent to the gamma dose for these measurements. The fast neutron component was less-than the measurement capability of 1 mrem/hr for several measurements. These values are shown with the triangles at one-half of the smallest measured value. Using the Shapiro-Wilks Normality Test (Gilbert 1987) and including the insignificant fast neutron values at one-half of the minimum reported value suggests that the normal distribution is a slightly better description of the data than a log-normal distribution. The fast neutron dose equivalent is 0.42 ±0.35 of the gamma dose rate for this data set. Combining these results, the total neutron dose equivalent is 0.58 ±0.48 of the gamma dose equivalent on the MTR Experimental Floor. Equation 6-1 elaborates this calculation. The variation in the different components of neutron dose rate are so much smaller than the variation between the fast neutron and gamma dose equivalent rate as to be unimportant.

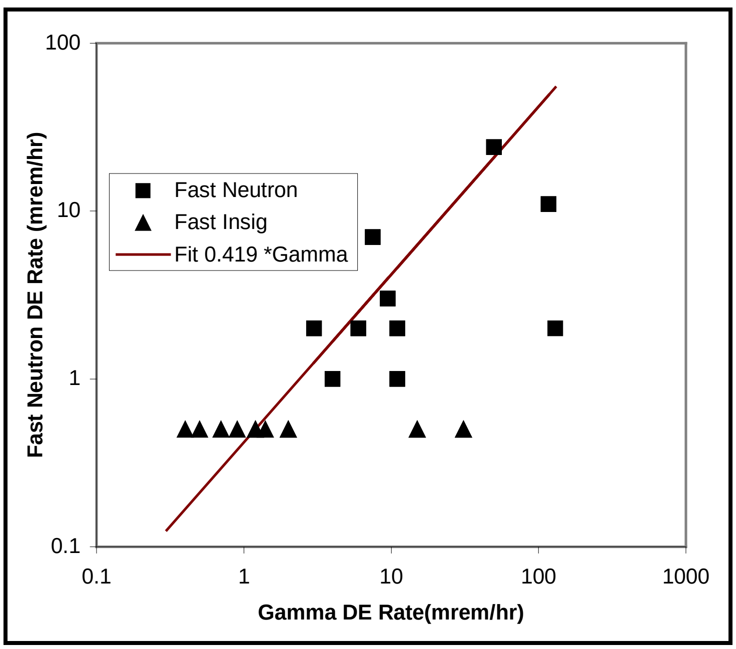


Figure 6-16. Correlation of fast neutron dose equivalent to gamma dose at MTR.

$$\frac{TotalNeutronDose}{GammaDose} = \frac{FastN}{Gamma}(1 + \frac{Thermal}{Fast} + \frac{LoInter}{Fast} + \frac{HiInter}{Fast}) \qquad \text{(Eq. 6-1)}$$
$$= (0.42 \pm 0.35)(1 + 0.071 + 0.177 + 0.149) = 0.58 \pm 0.48$$

Although this limited data for a small portion of the INEEL with weak correlation might suggest multiplying the gamma dose by 1.6 (i.e. 1+0.58) or 2.1 (i.e. 1+1.06), doing so is probably inappropriate because many of the people wearing NTA film would receive gamma dose at locations other than on the MTR experimental floor while the reactor was operating. For example, health physics technicians would often be covering jobs with only beta–gamma fields. A craftsperson could service pumps carrying radioactive water and not receive any neutron dose. Further extrapolating experience at the MTR at TRA to TREAT or ZPPR at ANL-W is a stretch. While similarities of the neutron spectra and associated corrections are likely valid, the relative amount of gamma and neutron radiation may not be. Also, dosimetry data sheets demonstrate that small neutron doses were as likely or more likely to be determined as small photon doses so this sort of correction is not necessary.

6.3.4.2.2 Typical Workplace Neutron Dosimeter Hp(10) Performance

Table 6-7 summarizes typical neutron personnel dosimeter parameters important to $H_p(10)$ performance in the workplace. The most important parameter related to $H_p(10)$ performance of the neutron dosimeters is the difference between calibration and workplace neutron energy spectra. Table 6B-2 in Attachment 6B summarizes the locations at ANL-W where neutron dose is credible.

Table 6-7. Typical workplace neutron dosimeter $H_p(10)$ performance.[a]

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

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

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Neutron Weighting Factor

All dose equivalents measured at ANL-W and reported in this document used the quality factors based on the LET of the ionizing secondary particles established in the 1950s and used since by U.S. regulatory agencies. In 1990 the ICRP developed new dose concepts that have been used by NIOSH. The quality factor, Q, as a function of LET was replaced with a radiation weighting factor, $w_{R,}$ which is a function of the neutron energy (ICRP 1990, Table 1).

A correction needs to be made to the ANL-W reported data to change from dose equivalent under NCRP 38 (NCRP 1971b) to a newer dose quantity (NIOSH 2002). ICRP Publication 74 (ICRP 1996) tabulates the ambient dose equivalent (dose equivalent at 10 mm depth in a 30 cm diameter sphere) for neutrons. NIOSH (2002) tabulates the ratios of organ dose equivalents to ambient dose equivalent, so this quantity is to be used for the conversion. Ambient dose equivalent is an ICRU quantity so it uses a revised Q(L) rather than a $w_{R,}$ so the correction factors are not as large as in other technical basis documents. The dose equivalent for a spectrum of particle energies is the result of an integral of the fluence spectrum, Φ(E), times a dose equivalent conversion factor, DECF(E), which also depends on energy over the range of energies considered:

$$H = \int_{E_1}^{E_2} DECF(E)\,\varphi(E)\,dE \qquad \text{(Eq. 6-2)}$$

These factors are incorporated into statements of dose equivalent values and calibrations following generally accepted principles. The conventional dose conversion factors are most clearly and correctly stated in ICRP Publication 21 (ICRP 1973). NCRP Report 38 tabulates a neutron flux density associated with the annual dose limit that is proportional to the reciprocal of the dose conversion factor (NCRP 1971b). Conventionally, the primary geometry considered is from one direction with the maximum dose in the body tabulated. More recent references (ICRU 1985; ICRP 1987, 1996) consider the dose to individual organs for different irradiation geometries, so the more recent tabulations give results lower by factors up to about 10 from attenuation in the human body. Dosimeters are designed to respond to radiation entering the body on the side where they are located and work best for an anterior-posterior (AP) irradiation geometry with the dosimeter on the front of the body.

Equation 6-2 also applies to ambient dose equivalent, except that a tabulation of the ambient dose equivalent dose conversion factor is used (ICRP 1996). The correction factor for an energy interval is then the ratio of the two integrals. Because IREP uses different radiation effectiveness factors for different radiation types and energies, it is appropriate to use the IREP energy intervals for calculating the correction factors.

Table 6-8 lists the calculated fractions of dose equivalent in the IREP energy groups and the conversion factors from dose equivalent to equivalent dose for ANL-W spectra. The ratios of average radiation weighting factor to average quality factor for the IREP energy groups have some variation, particularly for the 10- to 100-keV group where the energy dependence of the fluence is radically different for the fission and 14-MeV source than for the reactor spectrum. The lower part of the table lists the recommended default values for the dose equivalent fractions and quality factor corrections. These values are combined in Table 6B-4. This correction should be applied after the measured, the missed, and the unmonitored neutron doses are added together.

6.5 MISSED DOSE

6.5.1 Photon Unmonitored and Missed Dose

Unmonitored photon dose for ANL- W workers would occur where there is no recorded dose because workers were not monitored or the dose is otherwise unavailable. It was ANL-W policy to monitor everyone working at the site so unmonitored work is unlikely.

Table 6-8. Dose equivalent fractions and quality factor corrections, estimated and recommended.

IREP energy interval	<10 keV	10-100 keV	100 keV-2 MeV	2-20 MeV
Spectrum calculated values				
Dose equivalent fractions				
Bare fission		4.4E-05	0.20	0.80
AmBe			0.15	0.85
14 MeV 10 cm poly	2.4E-08	3.1E-06	1.5E-03	1.00
MTR exp floor ave	0.18	0.06	0.49	0.28
MTR exp floor max	0.24	0.08	0.52	0.35
MTR exp floor min	0.13	0.03	0.46	0.19
ICRP 74 H^*_{10} /NCRP 38 H				
Bare fission		1.46	1.32	1.09
AmBe			1.41	1.05
14 MeV 10 cm poly	0.69	1.47	1.36	0.93
MTR exp floor ave	0.86	1.08	1.33	1.12
MTR exp floor max	0.80	1.08	1.37	1.12
MTR exp floor min	0.92	1.08	1.30	1.12
Recommended defaults				
Dose equivalent fractions				
14 MeV 10 cm poly			0.05	0.95
Source calibrations			0.20	0.80
MTR exp floor	0.2	0.05	0.50	0.25
H*(10)/H	1	1.1	1.4	1.1

Missed photon dose for ANL-W workers would occur where a zero dose is recorded for the dosimeter systems for any response less than the site MRL. The missed dose for dosimeter results less than the MRL is particularly important for earlier years when dosimeter exchange was more frequent and MRLs were somewhat higher. One option to calculate the missed dose described in NIOSH (2002) is to estimate a claimant-favorable potential missed dose where MRL/2 is multiplied by the number of zero dose results. Table 6B-1 in Attachment 6B lists the potential missed photon dose according to year, dosimeter type, and exchange frequency. The MRLs shown are based on Cipperley (1958, 1968) and Cusimano (1963) for film; Kalbeitzer (1983), Gesell (1986), Gesell et al. (1992), and Perry et al. (1993) for TLDs; and Ruhter and Perry (2002) for film and TLD. The dose reconstructor should determine the exchange frequency from the dose submittal package for each individual and year because it was shorter for highly exposed individuals and longer for those with lower doses.

6.5.2 Neutron Unmonitored and Missed Dose

Neutron radiation was present only at the ANL-W reactors, and for calibration of criticality alarms, etc. Most ANL-W workers were not exposed to neutrons and were not monitored for them. For other locations, unmonitored neutron dose is very unlikely because of the very low potential for neutron exposure. The badge of a worker normally monitored for neutron dose may not have been read out and thus the dose would be unmonitored. This would be indicated by a blank under neutrons on the badge report at a time when other workers have zeros. A reasonable dose to assign for such a situation is the average dose for that worker for nearby monitoring periods.

To calculate the missed dose, the reconstructor must first determine if the person worked near neutrons and which category of neutrons. This is best accomplished by review of the work location(s) and whether a worker or others in the badge reporting group were assigned any neutron dose equivalent. For many years, individuals who were assigned NTA film which was read with minimal dose have zeros in the record. A blank means that that individual didn't have NTA film or it was not read because neutron exposure was believed to be unlikely. We are not certain that this practice was followed for all time. The work location code for TREAT is 20, but ZPPR did not have a unique code.

If no neutron dose was assigned to the individual in question or to coworkers for several months, the dose reconstructor should assume that the worker was not exposed to neutrons. Therefore, there would not be any unmonitored neutron dose.

If a worker was likely to have been exposed to neutrons, the reconstructor should assign missed neutron dose equivalent using Table 6B-2 in Attachment 6B for the times when workers did not have reported neutron dose. For the period when NTA film was used, the recorded neutron dose should be multiplied by 1.25 for all facilities except the TREAT or ZPPR and by 2 for the TREAT or ZPPR when they were operating. Then the dose equivalent is apportioned into the IREP groups using Table 6B-4.

For example, if in 1970 a person was an experimenter at the ZPPR, and 7 of the weekly badges recorded a total of 300 mrem neutron dose equivalent and the other sheets all showed zeros, the missed dose would be 450 mrem [(52-7)×20÷2] so the total dose by the badges would be 750 mrem. Note that if some reports had blanks rather than zeroes (and zeroes were entered for other individuals) this would be an unmonitored dose which must be estimated by the first paragraph of this section. Although both ZPPR and TREAT did not operate on full time schedules so neutrons would only exist occasionally, it is claimant friendly to assume that they did. Because the badge only responds to about one-half the ZPPR or TREAT neutron dose equivalent (from Section 6.3.4.2.1 and Table 6B-3), the total dose equivalent is 1.5 rem (this multiplication should also be done for the unmonitored dose). To convert the 1.5 rem received from neutrons at the ZPPR to equivalent dose, multiply the total dose equivalent by the last column of Table 6B-4 to get 300 mrem to the <10-keV group, 90 mrem to the 10- to 100-keV group, 1050 mrem to the 0.1- to 2-MeV group, and 420 mrem to the above-2-MeV group, for a total equivalent dose of 1.86 rem.

The neutron missed dose is divided into two historical periods in the following discussion. The first is before 1976 when only NTA film dosimeters were used with supplemental recording of thermal neutron doses from B-10 pencil dosimeters. The second period is after 1976 when only Hankins albedo dosimeters were used. The estimated MRLs for these neutron dosimeters are summarized in Table 6B-2 in Attachment 6B. It is possible to estimate the missed neutron dose using the MRLs because the neutron dosimeters were calibrated with neutron sources that had energies similar to those encountered in the workplace and because most of the neutrons to which workers were normally exposed had energies greater than the 500- to 800-keV threshold of the NTA dosimeters. There was, of course, no threshold energy for the measurements using neutron albedo TLD badges.

6.5.2.1 Before October 1976

The use of NTA films for neutron dosimetry before 1976 is documented in various INEEL reports (Cusimano 1963; Cipperley 1958, 1968). As noted above, it is possible to estimate the missed dose using the MRLs. There are many recorded zeros in the neutron dose data for INEEL workers because an NTA film indicated a neutron dose equivalent that was less than the 14-mrem MRL. When the MRL for NTA film is used in estimating the missed neutron dose, it should be multiplied by 1.25 for most workers and by 2 for workers on the TREAT or ZPPR Experimental Floor.

6.5.2.2 After October 1976

Since October 1976, the neutron dose has been measured using the Hankins albedo-type TLD. The characteristics of this dosimeter are well documented (Gesell et al. 1996), and the MRL to be used in estimating missed dose is 15 mrem. A location-specific FNCF has been applied to convert the reading to dose equivalent. The adjustment to equivalent dose provided in Table 6B-4 must then be made.

6.6 ORGAN DOSE

The dose values should be used to calculate organ doses of interest using NIOSH (2002). It is recommended that the AP (front to back) geometry should be assumed for the irradiation geometry and for conversion to organ dose. The calculated neutron doses in each energy group should be multiplied by the conversion factors from ambient dose equivalent to organ dose for AP irradiation from Appendix B of NIOSH (2002). For photons before 1981 the conversion factor from exposure to organ dose should be used. For 1981 and after, the conversion factor from deep dose equivalent to organ dose should be used.

6.7 UNCERTAINTY

Measurement uncertainties arise from many sources. For gamma rays the standards for exposure have existed with only minor changes since the 1930s as required for medical uses of radiation. The INEEL used ionization chambers standardized by NBS (now the National Institute of Standards and Technology) for their calibrations. Use of a phantom for dosimeter irradiation began in the early 1980s, but backscatter only causes a minor change for high-energy photon dosimetry. The overresponse of the multi-element film badge to deep dose in tissue arises from their calibration to exposure, which is somewhat more at low energies than the deep dose. The INEEL environment did not have a significant low-energy photon field like a plutonium finishing plant, so the nonpenetrating component was attributed to beta radiation. A realistic estimate of total uncertainty for photon dosimetry is about 35% at 1 sigma. This is roughly consistent with the results in Figure 6-9. For those measurements, the standardization instrument contributed some significant uncertainty.

To determine beta radiations in relation to skin cancer, the reconstructor should divide the reported nonpenetrating dose by the fraction of dose measured; Table 6-5 lists the measured doses. The uncertainty for beta radiation is somewhat larger than that for gamma rays at an estimated 50% at 1 sigma. This large uncertainty is driven by the uncertainties in field geometry and because beta radiation is often stopped by thin materials such as clothing and air. Algorithms are used to estimate the dose at a depth of 7 mg cm^{-2} from dosimeters at depths of 15 to 250 mg cm^{-2}, and such depth differences can change the signal significantly. The difference between a point source irradiation and a planar source can confuse an algorithm. Earlier techniques did not provide a thin detector with minimal covering, which is important for simulating the skin for beta dosimetry.

The determination of uncertainty for neutron radiations is more complex. The NTA films used before 1975 did not see low-energy neutrons below 0.5 to 0.8 MeV. Corrections are described in Section 6.5.2 for handling this issue. The TLD albedo system provides a very indirect way of measuring dose equivalent to a person. Dose to people is primarily due to hydrogen recoils rather than $^{6}Li(n,\alpha)$ reactions. The response of the 9-in. PNR-4 detector used to standardize the TLD measurements arises from a different process than dose deposition in the human body. The total uncertainty for neutrons is probably larger than that for beta radiation at about 60% at 1 sigma. The cause of the greatest uncertainty for neutrons is the variation of dose caused by the position of the organ within the body. For 1-MeV neutrons, the dose facing the source is about 1,000 higher than the dose on the back side of a 30-cm-diameter sphere of tissue-like material. In a work environment the direction of the neutrons may be unknown, but it is often from many directions, which reduces the impact of this uncertainty driver. For simplicity and because it often is true, this analysis assumed that the geometry of exposure is AP (from the front). The apparent discrepancy in Figure 6-11, in which the dosimeters showed about one-half of the spectrometer readings, occurred because the spectrometer does not simulate the attenuation of the body, so the reading was high by a factor of two.

REFERENCES

ACC (American Cyanamid Company), 1952, *CPP Health Physics Manual*, IDO-14055, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1958, *Permissible Levels of Radiation Exposure*, AEC Manual, Chapter 0524, Washington D.C.

AEC (U.S. Atomic Energy Commission), 1959, *Annual Report 1958 Health and Safety Division*, IDO-12012, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1960, *Annual Report of Health and Safety Division 1959*, IDO-12014, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1962, *1961 Health and Safety Division Annual Report*, IDO-12021, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1963, *Standards for Radiation Protection*, AEC Manual, Chapter 0524, Washington D.C. August 12.

AEC (U.S. Atomic Energy Commission), 1968, *Standards for Radiation Protection*, AEC Manual, Chapter 0524, Appendix 0524, Washington D.C.

AEC (U.S. Atomic Energy Commission), 1969, *1968 Annual Report of the Health Services Laboratory*, George L. Voelz, IDO-12071, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1969*, AEC Workshop on Personnel Neutron Dosimetry, September 23-24, 1969*, BNWL-1340, UC-48, Battelle Northwest, Richland, Washington.

AEC (U.S. Atomic Energy Commission), 1970, *1969 Annual Report of the Health Services Laboratory*, George L. Voelz, IDO-12073, Idaho Operations Office, Idaho Falls, Idaho.

Andersen, B. L. 1994, *Panoramic Irradiator to Acquire NIST Equivalent Irradiations for the INEL Dosimetry Programs*, EDF ODU-94-5.0.

Anderson, M. E., 1971, "Thick Target Yield for the ^{9}Be(α,n) Reaction", *Nucl. Sci. Engr.*, volume 44, pp. 437-439.

ANSI (American National Standards Institute), 1972, *Performance Specifications for Direct Reading and Indirect Reading Pocket Dosimeters for X- and Gamma radiation,* Standard No. ANSI N13.5, New York, New York.

ANSI (American National Standards Institute), 1983a (Reference was to previous version), *Photographic Film Dosimeters - Criteria for Performance,* Standard No. ANSI N13.7, New York, New York.

ANSI (American National Standards Institute), 1983b, *Criteria for Testing Personnel Dosimetry Performance,* Standard No. ANSI N13.11, New York, New York.

Aoki, I., 1979, "General Information for Evaluating Penetrating Exposure Summary at INEL for Years 1950 through 1979", memorandum to file, May 31, Dosimetry Branch, Idaho Operations Office, Idaho Falls, Idaho.

Attix, F. H., and W. C. Roesch, 1968, *Radiation Dosimetry*, Second Edition, Academic Press.

Bean, L. C., 1995, *Establishing a Beta Source for QA/QC Irradiations*, Engineering Design File RDRU 95-1.0.

Black, D. E., 1974, letter to D. I. Walker (Director, Health Services Laboratory, Idaho Operations Office, U.S. Atomic Energy Commission), June 21, Allied Chemical, Idaho Falls, Idaho.

Cipperley, F. V. 1966*, A General Summary of Personnel Monitoring Systems Utilized at the National Reactor Testing Station*, IDO-12056, Idaho Operations Office, Idaho Falls, Idaho.

Cipperley, F. V., 1958, *Personnel Metering Operating Techniques – Self-Service System – Effective August 20, 1951 through March 29, 1958,* IDO-12007, Idaho Operations Office, Idaho Falls, Idaho.

Cipperley, F. V., 1968, *Personnel Dosimetry Standard Operating Procedure Effective March 30, 1958*, IDO-12070, Idaho Operations Office, Idaho Falls, Idaho.

Cipperley, F. V., and W. P. Gammill, 1959, “Improvements in Personnel Metering Procedures at NRTS,” meeting of the Health Physics Society, June 19, 1959.

Cipperley, F. V., R. W. Henry, and J. P. Cusimano, 1965, “Personnel Monitoring Practices at the National Reactor Testing Station,” *Symposium on Personnel Dosimetry for Accidental High-Level Exposure to External and Internal Radiation,* International Atomic Energy Agency, March 8-12, 1965, Vienna, Austria.

Cusimano J. P., and F. V. Cipperley, 1968, “Personnel Dosimetry Using Thermoluminescent Lithium Fluoride-Teflon Dosimeters,” *Health Physics*, volume 14, pp. 339-344.

Cusimano, J. P., 1963, *Latent Image Fading in Personal Neutron Monitoring Film at the NRTS*, IDO-12031, Idaho Operations Office, Idaho Falls, Idaho.

Cusimano, J. P., 1972, *Suggestions and/or Alternatives concerning the ATLAS Dosimetry System*, Letter to F. V. Cipperley, Dosimetry Branch, Health Services Laboratory, Idaho Operations Office, Idaho Falls, Idaho.

Cusimano, J. P., 1981, *Update of Facility Neutron Correction Factor (FNCF)*, Idaho Operations Office, Idaho Falls, Idaho.

Cusimano, J. P., 1982, “Changeout and Storage of Hurst NADs until Disposition,” memorandum to Bob Burkhart, Idaho Operations Office, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1981, *Environmental Protection, Safety, and Health Protection Program for DOE Operations*, Order 5480.1, *Requirements for Radiation Protection*, Chapter XI, Washington, D.C. 4/29/81 as revised 8/13/81.

DOE (U.S. Department of Energy), 1986a, *Handbook for the Department of Energy Laboratory Accreditation Program for Personnel Dosimetry Systems*, DOE/EH-0026, Washington, D.C.

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

DOE (U.S. Department of Energy), 1987, *Department of Energy Laboratory Accreditation Program for Personnel Dosimetry*, Order 5480.15, Washington, D.C.

DOE (U.S. Department of Energy), 1988, *Radiation Protection for Occupational Workers*, Order 5480.11, Washington, D.C.

ERDA (Energy Research and Development Administration), 1975, Chapter 0524, *Standards for Radiation Protection*, April 8, Washington, D.C.

ERDA (Energy Research and Development Administration), 1977, Chapter 0524, *Standards for Radiation Protection*, March 30, Washington, D.C.

Gesell, T. F, 1986, "Minimum Reporting Levels for the Panasonic Dosimetry Systems," letter to Dosimetry Customers, Radiological and Environmental Sciences Laboratory, May 15, Dosimetry Branch, Radiological and Environmental Sciences Laboratory.

Gesell, T. F., 1982a, "Calibration of the INEL personnel dosimeter for non-penetrating radiation," letter to S. Marchetti (West Valley Nuclear Services), October 26, Dosimetry Branch, Radiological and Environmental Sciences Laboratory.

Gesell, T. F., 1982b, "Estimation of External Dose Equivalents from TL Dosimeters," memorandum to Dosimetry File, February 3, Dosimetry Branch, Radiological and Environmental Sciences Laboratory.

Gesell, T. F., M. A. Hall, and B. L. Andersen, 1992, *Technical Basis of the INEL Personal Photon-Beta Dosimeter*, Idaho State University, Department of Physics, Health Physics Program, Idaho Falls, Idaho.

Gesell, T. F., R. Boston, B. L. Andersen, and R. Perry, 1996, *Technical Basis of the INEL Personal Neutron Dosimeter*, Idaho State University Report 960112, Idaho Falls, Idaho.

Gupta, V. P., 1981, *Standardization and use of 2.5 Ci $^{90}Sr/^{90}Y$ Source for Calibration of Personnel Dosimeters and Portable Instruments*, IDO-12093, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

Hankins, D. E. 1961, *A Method of Determining the Intermediate Energy Neutron Dose*, IDO-16655, Phillips Petroleum Company, Idaho Falls, Idaho.

Hankins, D. E., 1973, *A small, inexpensive albedo-neutron dosimeter*, LA-5261, Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

Hankins, D. E., 1976, "The effect of energy dependence on the evaluation of albedo neutron dosimeters," *Proceedings of the Ninth Midyear Topical Symposium of the Health Physics Society*, pp. 861–868, Operational Health Physics Society, Denver, Colorado.

Hertel, N. E., and J. W. Davidson, 1985, "The Response of Bonner Spheres to Neutrons from Thermal Energies to 17.3 MeV," *Nucl. Instr. Meth.*, A238, pp. 509–516.

HEW (U.S. Department of Health, Education and Welfare), 1970, *Radiological Health Handbook,* Bureau of Radiological Health, Rockville, MD, January.

Horan, J. R., and J. B. Braun, 1993, *Occupational Radiation Exposure History of Idaho Field Office Operations at the INEL*, EGG-CS-11143, Idaho Falls, Idaho.

IAEA (International Atomic Energy Agency), 1988, *Guidelines on Calibration of Neutron Measuring Devices*, Technical Reports Series No. 285, Vienna, Austria.

ICRP (International Commission on Radiological Protection), 1973, *Data for Protection against Ionizing Radiation from External Sources: Supplement to ICRP Publication 15*, Publication 21, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1977, *Recommendations of the International Commission on Radiological Protection*, Publication 26, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1979, *Limits for Intakes of Radionuclides by Workers*, Publication 30, Pergamon Press, Oxford, England

ICRP (International Commission on Radiological Protection), 1987, *Data for Use in Protection Against External Radiation, ICRP Publication 51,* 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.

ICRP (International Commission on Radiological Protection), 1996, *Conversion Coefficients for Use in Radiological Protection Against External Radiation*, Publication 74, Pergamon Press, Oxford, England.

ICRU (International Commission on Radiation Units), 1954, *Recommendation of the International Commission on Radiation Units (ICRU)*, 7th International Congress of Radiology, Radiology 62, p. 106.

ICRU (International Commission on Radiation Units and Measurements), 1956, *RBE Dose, Report of the International Commission on Radiation Units and Measurements*, NBS Handbook 62, p. 7.

ICRU (International Commission on Radiation Units and Measurements), 1962, Radiation Quantities and Units, Report 10a. NBS Handbook 84, Washington, D. C.

ICRU (International Commission on Radiation Units and Measurements), 1971, *Radiation Quantities and Units*, Report 19, Washington, D.C.

ICRU (International Commission on Radiation Units and Measurements), 1976, *Conceptual Basis for the Determination of Dose Equivalent*, Report 25, Washington, D.C.

ICRU (International Commission on Radiation Units and Measurements), 1980, *Radiation Quantities and Units*, Report 33, Washington, D.C.

ICRU (International Commission on Radiation Units and Measurements), 1985, *Determination of Dose Equivalents Resulting from External Sources,* Report 39, Bethesda, Maryland.

ICRU (International Commission on Radiation Units and Measurements), 1993, *Quantities and Units in Radiation Protection Dosimetry*, Report 51, Bethesda, Maryland.

INEEL (Idaho National Engineering and Environmental Laboratory), 2001, *Technical Basis of the INEEL Personal Photon-Beta Dosimeter*, INEEL/EXT/01-00636, Idaho Falls, Idaho

INEL (Idaho National Engineering Laboratory), c. 1978, "Dosimetry Branch Changes from March 1978," memorandum to file, Dosimetry Branch, Health Service Laboratory, Idaho Operations Office, Idaho Falls, Idaho.

Ing, H., and S. Makra, 1978, *Compendium of Neutron Spectra in Criticality Accident Dosimetry*, Technical Report Series No. 180, International Atomic Energy Agency, Vienna, Austria.

Jones, D., 1980, "DOE Headquarters Request for Comments on Proposed Changes to Neutron Quality Factor," letter to J. Barry (Operational Safety Director, Idaho Operations Office), May 22, Radiological and Environmental Sciences Laboratory, Idaho Falls, Idaho.

Kalbeitzer, F. L., 1983, "Reportable Threshold Reading from the Unshielded Dosimeter of the Personnel Badge," letter to Dosimetry Branch, RSL/ID, March 25, Idaho Falls, Idaho.

Kalbeitzer, F. L., 1984, "Computerized Determination of Penetrating/Non-Penetrating Calibration Factors," memorandum to Dosimetry Branch Staff, September 17, Idaho Falls, Idaho.

Kluge, H., and K. Weise, 1982, "The Neutron Energy Spectrum of a ^{241}Am-Be(α,n) Source and Resulting Mean Fluence to Dose Equivalent Conversion Factors," *Radiat. Prot. Dosim.*, volume 2, pp. 85–93.

Meinhold, C. B., 1975, "The Unchanging Aspects of Radiation Protection Limits," *Population Dose Evaluation and Standards for Man and his Environment*, IAEA-SM-184/19, International Atomic Energy Agency, Vienna, Austria.

NBS (National Bureau of Standards), 1949, *Safe Handling of Radioactive Isotopes*, Handbook 42, NCRP Report No. 7, Washington, D.C.

NBS (National Bureau of Standards), 1957, *Protection Against Neutron Radiation up to 30 Million Electron Volts,* Handbook 63, Washington, D.C., Nov 22.

NBS (National Bureau of Standards), 1958, *Maximum Permissible Radiation Exposures to Man*, *Addendum to National Bureau of Standards Handbook 59, Permissible Dose from External Sources of Ionizing Radiations*, Washington, D.C. [in NBS Handbook 48, *Control and Removal of Radioactive Contamination in Laboratories* NCRP Report No. 8, Feb 15, 1951(subsequent printing)].

NBS (National Bureau of Standards), 1961, *Measurement of Absorbed Dose of Neutrons, and of Mixtures of Neutrons and Gamma Rays, Recommendations of the National Council on Radiation Protection and Measurements*, Handbook 75, NCRP Report No. 25, Washington, D.C.

NCRP (National Council on Radiation Protection and Measurements), 1971a, *Basic Radiation Protection Criteria*, Report 39, Bethesda, Maryland.

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

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

Perry, O. R., B. L. Andersen, and P. E. Ruhter, 1993, *Minimum Reporting Level for the Idaho National Engineering Laboratory (INEL) Thermoluminescent Dosimetry (TLD) Systems*, EGG-ESQ-10917, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

Purcell, R. B., and H. McGary, 1963, *Automatic Film Density Reading System*, IDO-12029, Idaho Falls, Idaho.

Reilly, P. S., 1998, *Characterization of Radiation Fields at the Idaho National Engineering and Environmental Laboratory*, EDF Report RDR-98005.

RESL (Radiological and Environmental Sciences Laboratory) 1981, *Personnel Dosimetry Performance*, Dosimetry Branch, Idaho Falls, Idaho.

Roberson, P. L. and K. L. Holbrook, 1984, *Guidelines for the Calibration of Personnel Dosimeters*, PNL-4515, Pacific Northwest Laboratory, Richland, Washington.

Rohrig, N. D., 2004, *Reanalysis of Hankins MTR Bonner Ball Surveys*, ORAUT-OTIB-0009, Rev. 0, February, Oak Ridge Associated Universities, Oak Ridge, Tennessee.

Ruhter, P. E., and O. R. Perry, 2002, *INEEL Dosimetry Systems History and Minimum Dose Reporting Level (MRL) Information (External Dose Summary Profile – 1951 to March 2002*).

Sommers, J. F., 1959, “Neutron Activity Levels at the MTR due to ANP-1,” letter to R. L. Doan (site manager), August 27.

Sommers, J. F., 1962, “Thermal Column Measurements,” letter to W. H. Burgus (engineer), December 18.

Sommers, J. F., 1967, “Change in TRA Neutron Monitoring Methods,” memorandum to Personnel, SOM-59-67, TRA Health Physics and the Nuclear Technology Branch.

Sommers, J. F., 1969, “Neutron Dosimetry,” interoffice correspondence to reactor HPs TRA, (Health Physicists at Test Reactor Area) SOM-30-69.

Walker, D. I., 1974, letter to D. E. Black (Allied Chemical, ICPP), July 23, Idaho Operations Office, Health Services Laboratory, Idaho Falls, Idaho.

Watkins, V., date unknown, *NRTS Routine Thermoluminescent Dosimetry System,* [likely from between 1966 and 1975].

GLOSSARY

1/E spectrum
For neutrons, the number of neutrons in an energy interval scales as the width of the energy interval divided by the energy of the neutrons in that interval.

beta particle
An electron or positron emitted spontaneously at high velocity from the nuclei of certain radioactive elements. Most of the direct fission products are (negative) beta emitters.

dose equivalent (H)
The product of the absorbed dose (D), the quality factor (Q), and any other modifying factors. The special unit is the rem, the International System (SI) unit is the sievert (1 sievert = 100 rem).

dose equivalent index
Maximum dose equivalent within the ICRU sphere centered at the point in space to which the quantity is assigned, H_I. The outer 0.07-mm-thick shell is ignored. It is also called the unrestricted dose equivalent index.

deep dose equivalent index
Maximum dose equivalent in the ICRU sphere within a core radius of 14 cm. The sphere is centered at the point in space to which the quantity is assigned. This quantity is one of the two restricted dose equivalent indices.

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

dosimeter
A device used to measure the quantity of radiation received. A holder with radiation-absorbing elements (filters) and an insert with radiation-sensitive elements packaged to provide a record of absorbed dose or dose equivalent received by an individual.

effective dose equivalent, H_E
The weighted average of the dose equivalents in certain organs or tissues of the body, H_T, each weighted by an organ weighting factor, W_T. The organ weighting factors were chosen by the ICRP to reflect the relative risk of death from cancer or occurrence of severe hereditary effects in the first generations after uniform whole body exposure.

exposure
In the technical sense, the ionization produced by gamma or X-rays in air (roentgen); in the generic sense, ionizing radiation applied to matter.

exposure-to-dose-equivalent conversion factor for photons (Cx)
The ratio of exposure in air to the dose equivalent at a specified depth in a material of specified geometry and composition. The Cx factors are a function of photon energy, material geometry (e.g., sphere, slab, or torso), and material composition (e.g., tissue-equivalent plastic, soft tissue ignoring trace elements, or soft tissue including trace elements).

linear energy transfer (LET)
The lineal rate of local energy deposition by a charged particle.

minimum reporting level (MRL)
Based on a policy decision, the minimum dose level that is routinely recorded.

nonpenetrating dose
Dose from beta and lower energy photon radiation. Determined from the open window minus the shielded.

pencil dosimeters
A type of ionization chamber used by personnel to measure radiation dose. Also pencil, pocket dosimeter, pocket pencil, and pocket ionization chamber.

penetrating dose equivalent
Photon dose measured by shielded INEEL film or elements plus neutron dose equivalent. Essentially, personal dose equivalent H_P (10).

personal dose equivalent, H_p(d)
Radiation quantity recommended for use as the operational quantity to be recorded for radiological protection purposes (ICRU 1993). The Personal Dose Equivalent is represented by H_p(d), where d identifies the depth (in mm) from the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose, d = 0.07 mm and is noted as H_p(0.07). For penetrating radiation of significance to "whole-body" dose, d = 10 mm and is noted as H_p(10).

polymethyl methacrolate
Scientific name for plastic commonly known as Lucite or Plexiglas.

redacted
To select item(s) to be visible for viewing or for publication by obscuring others.

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

shallow dose equivalent (Hs)
Dose equivalent at a depth of 0.07 mm in tissue (sum of penetrating and non-penetrating dose equivalent).

tissue rad
Absorbed dose in tissue.

ATTACHMENT 6A
INTERPRETATION OF INEEL DOSIMETRY CODES

Table 6A-1. Area Codes.

02	EBR #1/Argonne	26	EBR II
12	EX (EBR Construction)	263	EBR II (Monthly)
		265	EBRII (Quarterly)

Table 6A-2. Reasons Codes (Column 68-69).

	Old	Later Years		Old	Later Years
01	Regular Pull		11	Lost Pencil (or damaged)	Visitor HP Request
02	H.P. Request	Misc. Pull	12	H.P. Check	
03	High Dosimeter Reading	Withdrawn	13	Late Pull	
04	Recover Lost Badge	Termination	14	Withdrawn Badge	
05	Ring Reading		15	Termination	
06	Wrist Badge Reading	H.P. Request	16	Correction	
07	Recovered Lost Badge & Withdrawn		17	Records Withdrawn	
08		Late Pull for Not Available	18	Lost Film Reading	
09	Miscellaneous Pull		19	X-Ray Exposure	
10	Temporary Film	Late Pull resolved by PEQ	20	Experiment Exposure	

Table 6A-3: Irregularity Codes (Columns 70-71) .

01	Defective Film	12	Dropped in Canal or Reactor
02	Impossible to Read	13	
03	Light Leak	14	Not in Area[a]
04	Water Soaked	15	
05		16	
06	O.W. Shot with X-Ray	17	Old Lot Film
07	Lost in Processing	18	Stuck Film
08	Heat Exposure	19	Not Available
09	Recovered Lost Badge	20	Lost Badge
10	Contaminated Badge	21	No Film
11	Wore Two Badges at one Time		

a) Individual did not wear the badge in the area during the badge-pull period, that he did not enter the area during that period.

Table 6A-4. Column 20 Codes.

"X"	Master Card	6	Fast Neutron
1	Summary Card	7	Urinalysis
3	Sens. Beta-Gamma	8	Summary Card
4	Insen. Gamma	9	Summary Card
5	Slow Neutron	0	Total Body Results Card

ATTACHMENT 6B
OCCUPATIONAL EXTERNAL DOSE FOR MONITORED WORKERS

Table 6B-1. INEEL beta/photon dosimeter period of use, type, minimum reporting level, exchange frequency, and potential annual missed dose

Period of use[a]	Dosimeter	MRL[b] (mrem)	Exchange frequency	Potential annual missed dose (mrem)[c]
August 1951–March 1958	INEEL Initial Film 552 Dupont Film	30	Weekly (n=52)	780
			Monthly (n=12)	180
	Reactor Areas INEEL 558 Dupont Film	10	Weekly (n=52)	260
March 1958–December 1966	INEEL Multi-Element Dupont 508 Film	10	Weekly (n=52)	260
			Biweekly (n=26)	130
			Monthly (n=12)	60
December 1966–February 1974	INEEL Multi-Element Dupont 508 Film	10	Weekly (n=52)	260
			Biweekly (n=26)	130
			Monthly (n=12)	60
	INEEL LiF TLD	15	Quarterly (n=4)	30
			Semi-ann (n=2)	15
			Annual (n=1)	7.5
February 1974–May 1975[d]	INEEL Atlas TLD LiF in Teflon	30	Monthly (n=12)	180
			Quarterly (n=4)	60
			Semi-ann(n=2)	30
			Annual (n=1)	15
December 1974–December 1985[d]	INEEL Harshaw Two-chip TLD	15	Monthly (n=12)	90
			Quarterly (n=4)	30
			Annual (n=1)	7.5
January 1986–present	INEEL Panasonic Four-chip TLD	15[e]	Monthly (n=12)	90
			Quarterly (n=4)	30
		10[e]	Monthly (n=12)	60
			Quarterly (n=4)	20

a. For many years, INEEL workers had a dosimeter assigned to each operating area where they worked, or were issued visitor dosimetry. All Area dosimetry was issued beginning in January 2000.
b. Minimum reporting levels (MRL) are based on Cipperley (1958), Cipperley (1968), Cusimano (1963), Kalbeitzer (1983), Gesell (1986), Gesell (1992), Perry et al. (1993), and Ruhter (2002).
c. Maximum annual missed dose calculated using N × MRL/2 from NIOSH (2002).
d. Argonne began using the Harshaw in May 1975.
e. The MRL was 15 mrem from January 1, 1986, to July 7 1986, 10 mrem from July 7, 1986 to about September 1989, and 15 mrem until 1993 when it returned to 10 mrem.

Table 6B-2. Neutron dosimeter type, period of use, exchange frequency, laboratory minimum detectable limit, and potential annual missed dose.

Dosimeter	Period	Exchange frequency	Laboratory MRL (mrem)	Potential annual missed dose (mrem)
NTA film	1951-1958	Weekly	14	364
NTA film	1959–September 1976	Weekly	20	520
		Biweekly	20	260
		Monthly	20	120
TLD	October 1976–present	Biweekly	15	195
		Monthly	15	90
		Quarterly	15	30

Table 6B-3. Spectrum Correction for Measured and Missed Neutron Dose.

Period	Dosimeter	Work Location	Correction
1951-Sept 1976	NTA	TREAT or ZPPR	2
		All other	1.25
October 1976-present	Albedo TLD	All	1

Table 6B-4. Recommended IREP neutron energy fractions and correction factors for INEEL facilities.

Process	Description	Operations		Neutron energy	Default dose (%)	Ambient dose equiv/ dose equiv	Net correction factor
Instrument calibration	Alpha Be source calibrations	1951	1993	0.1-2 MeV	20%	1.4	0.28
	Cf-252 source calibrations	1993	2003	2-20 MeV	80%	1.1	0.88
Neutron source based research			2003	0.1-2 MeV	20%	1.4	0.28
				2-20 MeV	80%	1.1	0.88
MTR, ZPPR, and TREAT reactors	Experiment floor and adjacent rooms during operation	1953	1970	< 10 keV	20%	1	0.20
				10 -100 keV	5%	1.1	0.06
				0.1-2 MeV	50%	1.4	0.7
				2-20 MeV	25%	1.1	0.28