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☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

FOR DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE, REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
04/06/2004	00	New technical basis document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational External Dosimetry. First approved issue. Initiated by Norman D. Rohrig.
11/07/2006	01	Document revised as a result of biennial review. Approved Revision 01. Incorporates formal internal review comments. Revision constitutes a total rewrite of document. Adds Section 6.5.1 to incorporate comments from the Worker Outreach meeting with the PACE Local 8-0652 on April 28, 2004. Incorporates formal NIOSH review comments. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig.
06/18/2007	02	Approved Revision 02 initiated to incorporate Attribution and Annotation section. A minor change was made to Sections 6.3.4.5 and 6.5.4.2, which will reduce the dose for a very few individuals. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Jo Ann M. Jenkins.
04/19/2011	03	Initiated to incorporate Argonne National Laboratory – West – Occupational External Dose technical basis document (ORAUT-TKBS-0026-6 Rev 01) into the Idaho National Laboratory Occupational External Dosimetry technical basis document. Incorporates formal internal and NIOSH review comments. Removed the Site Expert line from page 1 since the former Site Expert served no role in this revision. Updated Section 6.8 to reflect that in all cases where the former site expert's information or prior studies or writings are included or relied on by the new Document Owner, those materials are fully attributed to the source in accordance with the "NIOSH Policy Statement: Management of Conflict or Bias in the Radiation Dose Reconstruction Program." Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by JoAnn M. Jenkins.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ANL-W	Argonne National Laboratory–West
ANSI	American National Standards Institute
AP	anterior-posterior
ARA	Army (later Auxiliary) Reactor Area
ATLAS	Automatic Thermoluminescent Analyzer System
AX	TAN Construction
CF	Central Facilities
CFA	Central Facilities Area
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
CPP	Chemical Processing Plant
CX	CPP construction
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
EBR	Experimental Breeder Reactor
ECF	Expended Core Facility (at NRF)
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EPRI	Electric Power Research Institute
ERDA	U.S. Energy Research and Development Administration
EX	EBR Construction
FNCF	facility neutron correction factor
g	gram
GSD	geometric standard deviation
HP	Health Physics
hr	hour
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
ILTSF	Intermediate-Level Transuranic Storage Facility
in.	inch
INEEL	Idaho National Engineering and Environmental Laboratory
IRC	INEL Research Center
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
LET	linear energy transfer
LOFT	Loss-of-Fluid Test
LPTF	Low Power Test Facility
LX	Construction at LOFT
m	meter

MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
ML-1	Mobile Low Power Reactor
mm	millimeter
mR	milliroentgen
mrad	millirad
mrem	millirem
mrep	millirep
mSv	millisievert
MTR	Materials Test Reactor
MX	TRA Construction
n	neutron
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurement
NIOSH	National Institute for Occupational Safety and Health
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
NTA	nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program
OMRE	Organic-Moderated Reactor Experiment
ORAU	Oak Ridge Associated Universities
OW	open window
OX	Construction at OMRE
PBF	Power Burst Facility
PEQ	Personnel Exposure Questionnaire
PER	Program Evaluation Report
PINS	Portable Isotopic Neutron Source
POC	probability of causation
R	roentgen
RBE	relative biological effectiveness
RESL	Radiological and Environmental Sciences Laboratory
RWMC	Radioactive Waste Management Complex
SL-1	Stationary Low-Power Reactor No. 1
SMC	Specific Manufacturing Capability
SPERT	Special Power Excursion Reactor Test
SRDB Ref ID	Site Research Database Reference Identification (number)
SWEPP	Stored Waste Examination Pilot Plant
SX	SPERT Construction
TAN	Test Area North
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor and Experiment Test
U.S.C.	United States Code
WERF	Waste Experimental Reduction Facility

wk	week
WP	Waste Processing Building
yr	year
Z	atomic number
ZPPR	Zero Power Plutonium (later Physics) Reactor
α	alpha particle
§	section

6.1 INTRODUCTION

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

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

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation1] guidelines established under subsection (c) …" [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation, 42 C.F.R. Pt. 82) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work (NIOSH 2010).

The statute also includes a definition of a DOE facility that excludes "buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program" [42 U.S.C. § 7384l(12)]. While this definition excludes Naval Nuclear Propulsion Facilities from being covered under the Act, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled "Exposure in the Performance of Duty"] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally-derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external occupational radiation exposures are considered valid for inclusion in a dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived (NIOSH 2010):

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

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

6.1.1 Purpose

This document describes the processes and results for measuring external doses at the Idaho National Laboratory (INL), formerly known as the National Reactor Testing Station (NRTS), the Idaho National Engineering Laboratory, and the Idaho National Engineering and Environmental Laboratory (INEEL), and at the Argonne National Laboratory–West (ANL-W). For consistency, this document uses INL or ANL-W, except in special cases.

6.1.2 Scope

From the start of operations in 1951 until early 2005, ANL-W was 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 Administration (ERDA), and DOE. In 2005, the functions of ANL-W were combined with research activities at INEEL to form INL, and ANL-W was renamed the Materials Fuel Complex (MFC). A branch of the Idaho Operations Office provided external dosimetry resources and services at INEEL, including ANL-W from the start of operations in 1951 (when it was called the NRTS) until 1989, when DOE transferred that responsibility to the prime operating contractor. Despite the fact that INL had several contractors at a time and that contractors changed often, the external dosimetry process has remained under the technical management of a single organization with responsibilities for dosimetry development, operational dosimetry, and radiological records, which has provided a stable external dosimetry system.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 6.8.

6.2 BASIS OF COMPARISON

Since the initiation of the Manhattan Engineer District project in the early 1940s, various radiation dose concepts and quantities have been used to measure and record occupational dose. The basis of comparison for reconstruction of dose is the *personal dose equivalent*, $H_p(d)$, where *d* identifies the depth (in millimeters) and represents the point of reference for dose in tissue. For weakly penetrating radiation of significance to skin dose, *d* is 0.07 mm and the dose equivalent is noted as *Hp(0.07)*. For penetrating radiation of significance to *whole-body dose*, *d* is 10 mm and the dose equivalent is noted as *Hp(10)*. Both *Hp(0.07)* and *Hp(10)* are the radiation quantities recommended by the International Commission on Radiation Units and Measurements (ICRU) for use as the operational quantities to be recorded for radiological protection purposes (ICRU 1993). In addition, *Hp(0.07)* and *Hp(10)* are the radiation quantities used since the 1980s in the U.S. Department of Energy (DOE) Laboratory Accreditation Program (DOELAP) to accredit personnel dosimetry systems (DOE 1986a). The International Agency for Research on Cancer selected *Hp(10)* as the quantity to assess error in historical recorded whole-body dose for workers in Agency nuclear worker epidemiologic studies (Thierry-Chef et al. 2002). The basis for comparison for neutron radiation is more complicated because the calibration of dosimeters to measure neutron dose was based historically on different dose quantities such as first collision dose, multiple collision dose, dose equivalent index, and so forth. The numerical difference in using these dose quantities compared to the *Hp(10)* dose used in current DOELAP performance testing could be evaluated by using the relative values of the dose conversion factors for the respective dose quantities in conjunction with characteristics of the respective INL neutron dosimeter response characteristics and workplace radiation fields.

The quantities to be measured and reported by the dosimetry systems have evolved over the last 50 years, but the changing definitions have had little effect on dosimetry measurements because, for gamma radiation, the differences are small [1].

6.3 DOSE RECONSTRUCTION PARAMETERS

6.3.1 Site Administrative Practices

It was INL and ANL-W policy that personnel who were expected to receive any radiation dose or whose work was centered at the site were assigned a radiation monitoring badge. These badges were usually stored at the operational area entrance security gates. Control badges, which are used to subtract background radiation, have also been located at the gates (Ruhter 2003). Environmental radiation levels have been monitored using film badges and then thermoluminescent dosimeters (TLDs). Table 4-13 in the environmental dose TBD (ORAUT 2010a) lists the results of this monitoring at facility fenceline locations near the security gates interpreted as net annual dose (i.e., in addition to natural background). A fraction (2000/8766) of these values can be added to an individual dose history or used for unmonitored site workers.

To provide consistency of radiation safety programs at the INL among a large variety of facilities and constantly changing contractors, the AEC established a Health and Safety (H&S) Laboratory at the INL to provide technical support for internal and external dosimetry programs. The name of this organization changed to Health Services Laboratory (HSL), then to the Health and Safety Division (HSD), then to the Idaho Center for Radiological and Environmental Sciences, and most recently to the Radiological and Environmental Sciences Laboratory (RESL).

Some individuals who occasionally visited site facilities but did little work with radiation had badges at several different facilities [2]. It is not appropriate to base missed doses on the multiple badges issued. Early on, the badge exchange frequency was not the same for all workers. Workers with low probability of exposure were placed on a longer exchange cycle than those with more chance of exposure (Cusimano 1972). Therefore, missed doses should be based on the actual exchange frequency for a person, when it can be determined from the individual's data package.

The INL 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 (Cipperley 1958).

DOE provided redacted dosimetry information for former workers, which included a dose summary for the employment period and a copy of each weekly, monthly, quarterly, etc., form with the work location. See Figures 6-1 to 6-5 for an example.

From 1951 to 1958, the INL dosimetry staff recorded dose daily on a dose card (Figure 6-1), rezeroed the pencil ionization chambers worn by workers, and entered the weekly badge result on the same card (Cipperley 1958). These dose cards were color coded to indicate at which facility the exposures were received; however, the copies being received from the DOE are only in black and white so work locations prior to1959 can only be determined from the "AEC-IDO-Locater File Cards" when provided or from the telephone interview information. On this sample, on October 28, November 16, and December 9, 1954, the badges were pulled and read in response to high pencil ionization 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 815 mR [255 + 280 (Oct 16) + 280 (Nov 9)] and the badges reported zero for 18 badges.

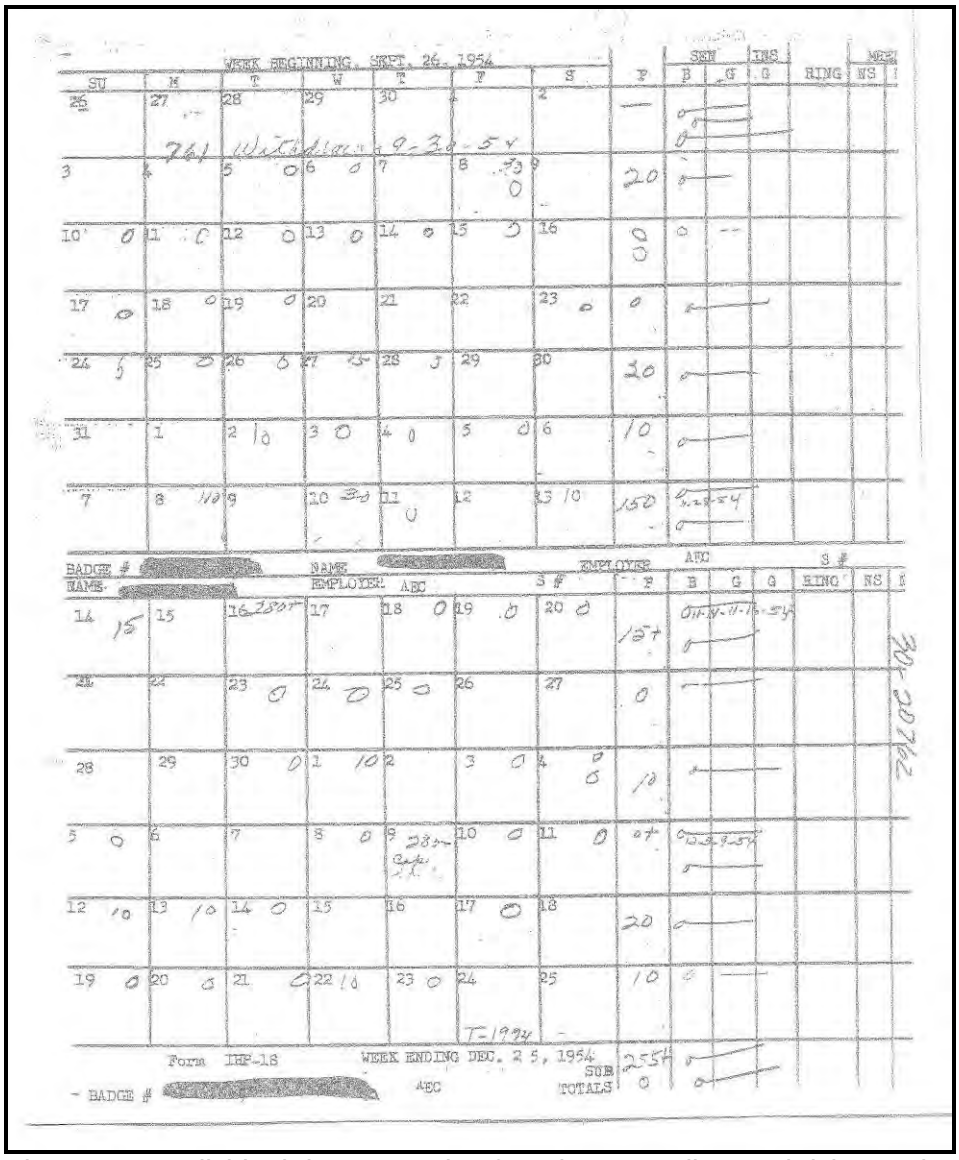
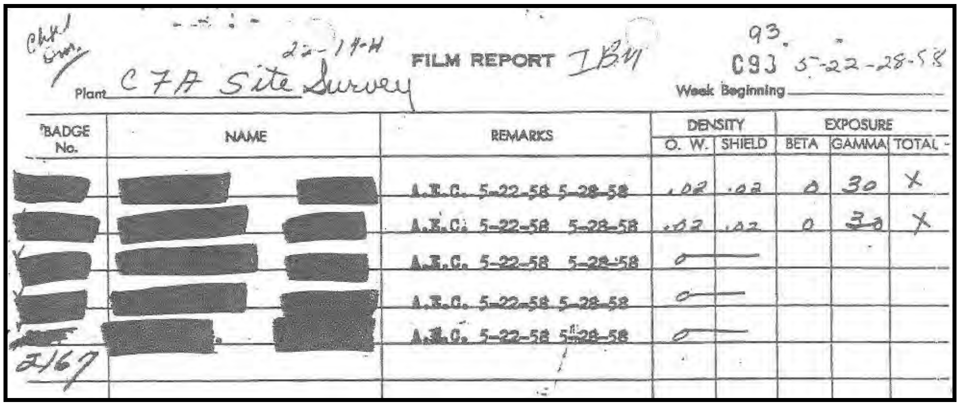


Figure 6-1. Individual dose reporting form in use until 1958 (Vivian and Rockhold 2003, p. 17).

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 (OW, "O. W." in the figure) result was zero, so the shielded film was not read. On the other two, the OW and shielded values were at the minimum recorded density of 0.02, which corresponded to a 30-mR penetrating dose.

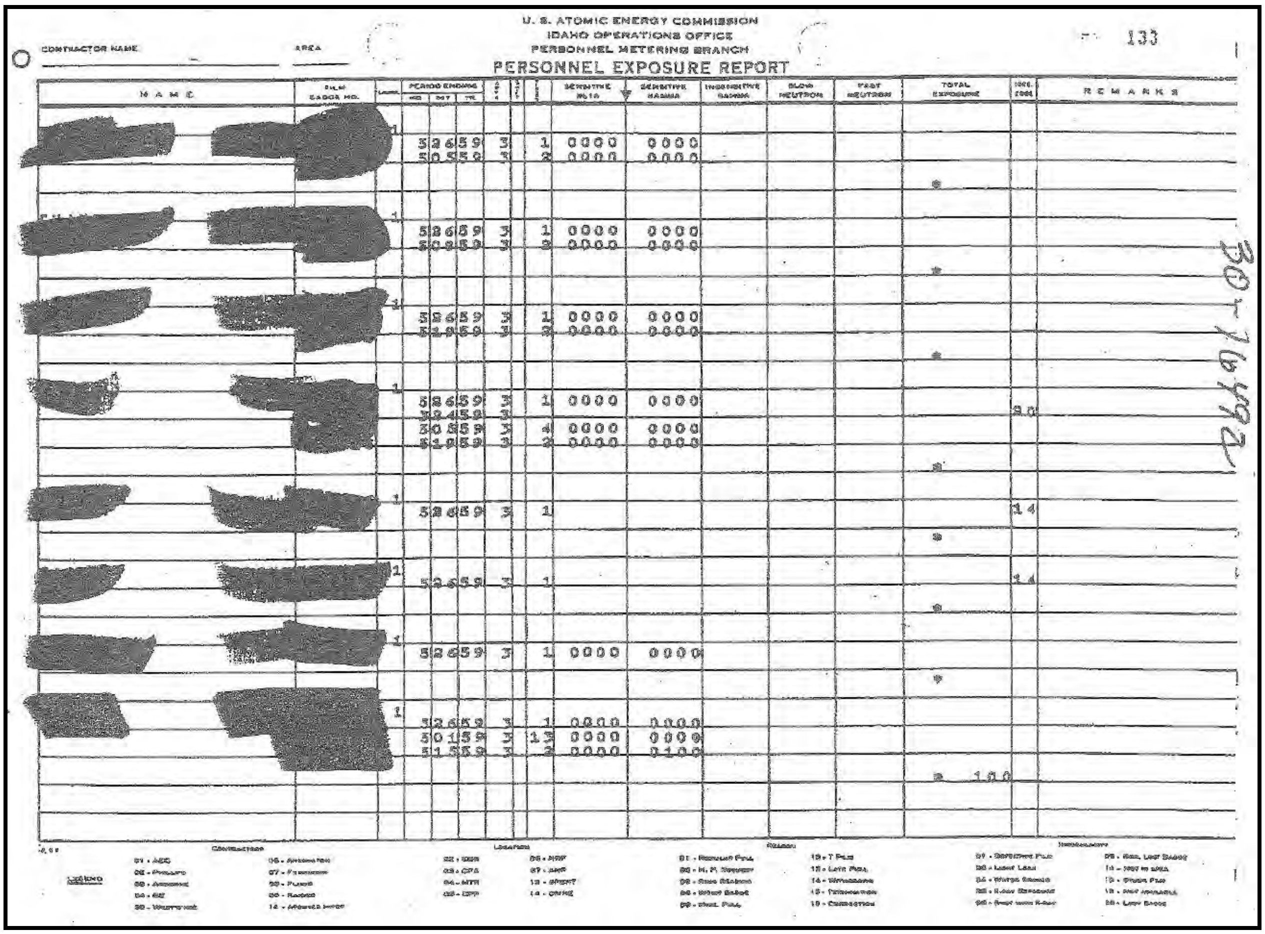
After the pencil ionization chambers were replaced with self-reading pencil dosimeters (also called ionization chambers), the INL operational health physics staff would rezero the dosimeters (Horan 1959). The film reading was automated, and results were stored in a computer. The form shown in Figure 6-3 shows a representative example of the records format used from March 1958 through November 1966. During that timeframe there were some periods with slightly different formats, but the general layout of the records was the same. The column under the P of Personnel is an area designator with the code listed under Location at the bottom of the page. Table 6-1 lists the codes for the areas. The next column was unused and dropped later. The next column was for the reason codes listed in Table 6-2. Computer input card codes are listed in Tables 6-3 and 6-4, respectively. Figures 6-3 through 6-6 provide further examples of records formats used at INL.



Plant C7A Site Survey 22-14-H FILM REPORT IBM 93 C93 5-22-28-58 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	—				
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	—				
[redacted]	[redacted]	A.E.C. 5-22-58 5-28-58	—				
2167							

Figure 6-2. Film report form used in 1958 (Vivian and Rockhold 2003, p. 43).



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

CONTRACTOR NAME AREA 133

NAME	FILM BADGE NO.		PERIOD ENDING			SENSITIVE BETA	SENSITIVE GAMMA	INSENSITIVE GAMMA	SLOW NEUTRON	FAST NEUTRON	TOTAL EXPOSURE		REMARKS
[redacted]	[redacted]	1	52659	3	1	0000	0000						
			50559	3	2	0000	0000						
[redacted]	[redacted]	1	52659	3	1	0000	0000						
			50959	3	2	0000	0000						
[redacted]	[redacted]	1	52659	3	1	0000	0000						
			51959	3	2	0000	0000						
[redacted]	[redacted]	1	52659	3	1	0000	0000					30	
			52459	3									
			50859	3	4	0000	0000						
			51959	3	2	0000	0000						
[redacted]	[redacted]	1	52659	3	1							14	
[redacted]	[redacted]	1	52659	3	1							14	
[redacted]	[redacted]	1	52659	3	1	0000	0000						
[redacted]	[redacted]	1	52659	3	1	0000	0000						
			50159	3	13	0000	0000						
			51559	3	2	0000	0100				100		

LEGEND

30r76492

Figure 6-3. Example of dosimeter reporting format used from March 1958 through November 1966 (Vivian and Rockhold 2003, p. 46).

Table 6-1. Area codes that could be in worker dose files (Hill-Ellis 2003).

Area code	Description	Area code	Description
1	AEC Headquarters Building	20, 261, 264	TREAT
2	EBR-I	21	LX
3, 034, 035	CFA	22	Gas-cooled reactor experiment
4, 042,045	MTR, TRA	23	OX at OMRE
5, 053, 055	ICPP	24	ARHG
6	NRF	25	No information available
7	TAN (GE)	26, 263,265	EBR-II
8	Services	27	ML-1
9	NX (X is construction) at NRF	28	Onsite survey
10	AX at TAN	29	Offsite survey
11, 113	CX at CPP	30	Advanced Nuclear Propulsion Program at SL-1
12	EX at EBR	31	Shield test pool facility
13, 133,135	SPERT, PBF	65	ECF
14	OMRE	66	Nonsecurity
15	SX at SPERT	67	Division of Compliance
16	SL-1	68	STEP
17, 333	MX at MTR	69	LPTF (Phillips and AEC)
18, 814,815	WP, RWMC	71	CADRE (guard force)
19, 772, 775	TAN (Phillips and AEC)	774,776	SMC

Table 6-2. Reasons codes (Columns 68 and 69) (Hill-Ellis 2003).

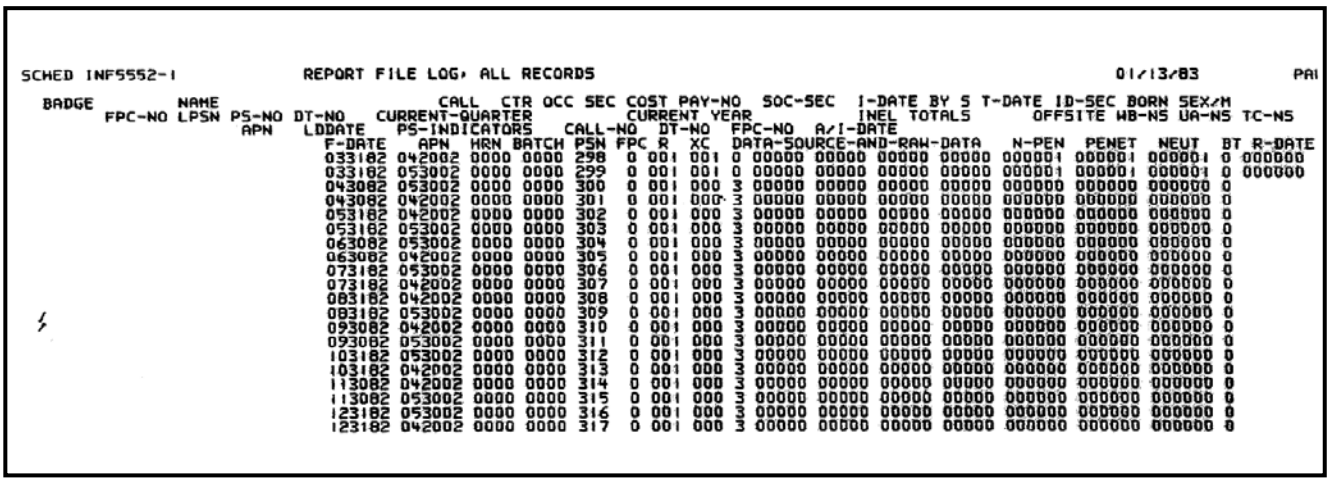
	Old	Later years		Old	Later years
01	Regular pull		11	Lost pencil (or damaged)	Visitor HP request
02	HP request	Miscellaneous pull	12	HP check	
03	High dosimeter reading	Withdrawn	13	Late pull	
04	Recover lost badge	Termination	14	Withdrawn badge	
05	Ring reading		15	Termination	
06	Wrist badge reading	HP request	16	Correction	
07	Recovered lost badge and withdrawn		17	Records withdrawn	
08		Late pull or 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 6-3. Irregularity codes (columns 70-71) (Hill-Ellis 2003).

01	Defective film	12	Dropped in canal or reactor
02	Impossible to read	13	
03	Light leak	14	Not in area
04	Water soaked	15	
05		16	
06	OW 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		

Table 6-4. Column 20 codes (Hill-Ellis 2003).

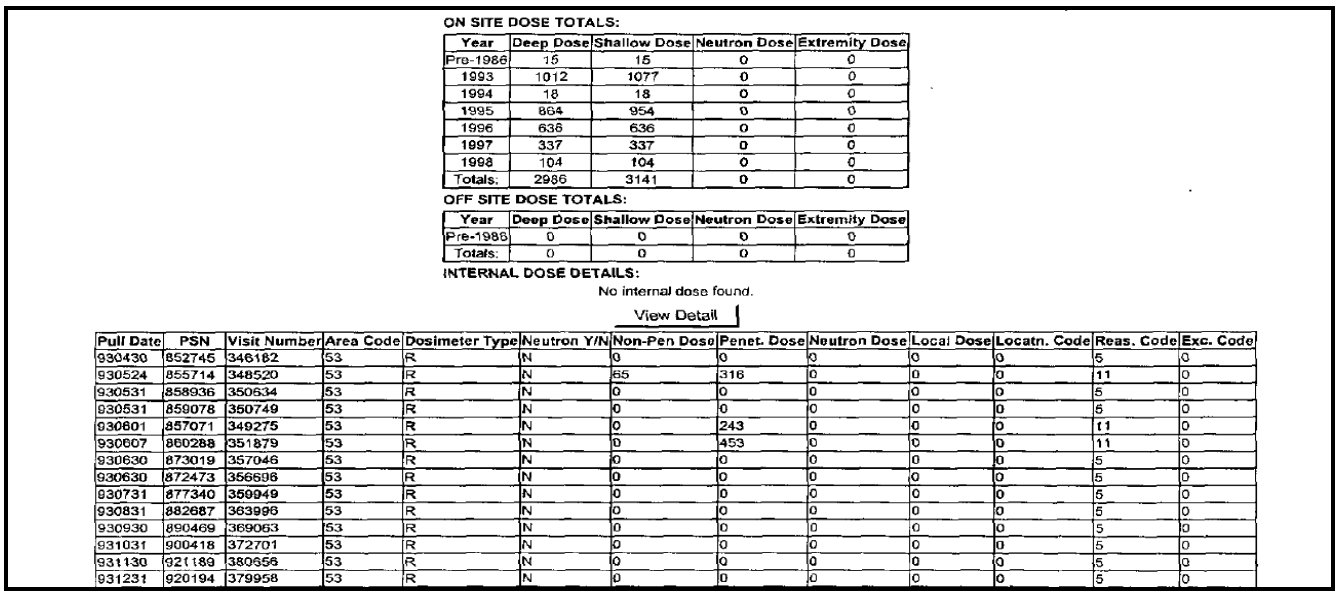
“X”	Master card	6	Fast neutron
1	Summary card	7	Urinalysis
3	Sensitive beta-gamma	8	Summary card
4	Insensitive gamma	9	Summary card
5	Slow neutron	0	Total body results card



SCHED INF5552-1 REPORT FILE LOG, ALL RECORDS 01/13/83 PAI

BADGE FPC-NO	NAME LPSN	PS-NO APN	DT-NO LDDATE	CALL CTR OCC SEC COST PAY-NO SOC-SEC I-DATE BY 5 T-DATE ID-SEC BORN SEX/M CURRENT-QUARTER CURRENT YEAR INEL TOTALS OFFSITE WB-NS UA-NS TC-NS PS-INDICATORS CALL-NO DT-NO FPC-NO A/I-DATE F-DATE APN HRN BATCH PSN FPC R XC DATA-SOURCE-AND-RAW-DATA N-PEN PENET NEUT BT R-DATE
				033182 042002 0000 0000 298 0 001 001 0 00000 00000 00000 00000 000001 000001 000001 0 000000
				033182 053002 0000 0000 299 0 001 001 0 00000 00000 00000 00000 000001 000001 000001 0 000000
				043082 053002 0000 0000 300 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				043082 042002 0000 0000 301 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				053182 042002 0000 0000 302 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				053182 053002 0000 0000 303 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				063082 053002 0000 0000 304 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				063082 042002 0000 0000 305 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				073182 053002 0000 0000 306 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				073182 042002 0000 0000 307 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				083182 042002 0000 0000 308 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				083182 053002 0000 0000 309 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				093082 042002 0000 0000 310 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				093082 053002 0000 0000 311 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				103182 053002 0000 0000 312 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				103182 042002 0000 0000 313 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				113082 042002 0000 0000 314 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				113082 053002 0000 0000 315 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				123182 053002 0000 0000 316 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0
				123182 042002 0000 0000 317 0 001 000 3 00000 00000 00000 00000 000000 000000 000000 0

Figure 6-4. Example of dosimeter reporting format used from December 1966–1985.



ON SITE DOSE TOTALS:

Year	Deep Dose	Shallow Dose	Neutron Dose	Extremity Dose
Pre-1986	15	15	0	0
1993	1012	1077	0	0
1994	18	18	0	0
1995	884	954	0	0
1996	636	636	0	0
1997	337	337	0	0
1998	104	104	0	0
Totals:	2986	3141	0	0

OFF SITE DOSE TOTALS:

Year	Deep Dose	Shallow Dose	Neutron Dose	Extremity Dose
Pre-1986	0	0	0	0
Totals:	0	0	0	0

INTERNAL DOSE DETAILS:

No internal dose found.

View Detail

Pull Date	PSN	Visit Number	Area Code	Dosimeter Type	Neutron Y/N	Non-Pen Dose	Penet. Dose	Neutron Dose	Local Dose	Locatn. Code	Reas. Code	Exc. Code
930430	852745	346182	53	R	N	0	0	0	0	0	5	0
930524	855714	348520	53	R	N	65	316	0	0	0	11	0
930531	858936	350534	53	R	N	0	0	0	0	0	5	0
930531	859078	350749	53	R	N	0	0	0	0	0	5	0
930601	857071	349275	53	R	N	0	243	0	0	0	11	0
930607	860288	351879	53	R	N	0	453	0	0	0	11	0
930630	873019	357046	53	R	N	0	0	0	0	0	5	0
930630	872473	356696	53	R	N	0	0	0	0	0	5	0
930731	877340	359949	53	R	N	0	0	0	0	0	5	0
930831	882687	363996	53	R	N	0	0	0	0	0	5	0
930930	890469	369063	53	R	N	0	0	0	0	0	5	0
931031	900418	372701	53	R	N	0	0	0	0	0	5	0
931130	921189	380656	53	R	N	0	0	0	0	0	5	0
931231	920194	379958	53	R	N	0	0	0	0	0	5	0

Figure 6-5. Example of dosimeter reporting format used from 1986–1998.

FAC	TLD ID	TYPE	ISSUED	RETURN	MULTIPACK	STAT	GAMMA	NEUT	BETA	LOCATN	BETAEXT	EYEDOSE
INL	041099	BG	04/01/2010			ISSUED	-1	-1	-1	WHOLE		0
INL	032672	BG	01/01/2010	03/08/2010		USE	0	-1	0	WHOLE		0
***	2010	*****	*****	*****	*****	*****				MAXQUAD	0	
INL	026978	BG	10/01/2009	12/31/2009		USE	0	-1	0	WHOLE		0
RWM	024570	BG	08/24/2009	08/31/2009		USE	0	-1	0	WHOLE		0
INL	043079	BG	07/01/2009	09/30/2009		USE	0	-1	0	WHOLE		0
INL	023437	BG	04/01/2009	06/30/2009		USE	0	-1	0	WHOLE		0
INL	026900	BG	01/01/2009	03/31/2009		USE	0	-1	0	WHOLE		0
***	2009	*****	*****	*****	*****	*****				MAXQUAD	0	

Figure 6-6. Example of dosimeter reporting format used from 1999–present.

With the advent of computers, the reports were all computer-generated with the effect that, even though many workers were not exposed to neutrons and did not receive neutron dosimeters, zeros were entered in the dose reports. A zero is not an indication that a dosimeter was assigned in a computer-generated report.

For the early computer records (1975 to 1984), the penetrating and nonpenetrating labels are reversed from the printed data. The nonpenetrating result is reported first, then the penetrating result, and then the neutron result.

Figure 6-7 is a listing of some doses received during recovery from the Stationary Low-Power Reactor No. 1 (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. NIOSH has a significant dosimetry history for first-responder and recovery workers at the SL-1 facility. The dosimetric records provide enough data to accurately reconstruct doses. In some instances SL-1 specific coworker dose is used. Figure 6-8 is a follow-up badge report for one result on Figure 6-7. When there was a question about an assigned dose value, a *Personnel Exposure Questionnaire* (PEQ) was normally initiated as shown in Figure 6-9 (a hypothetical case) (Cipperley 1958). Based on this form, a beta dose of 500 mrem and a gamma dose of 350 mrem for a total dose of 850 mrem would override the pocket meter dose of 290 mrem total.

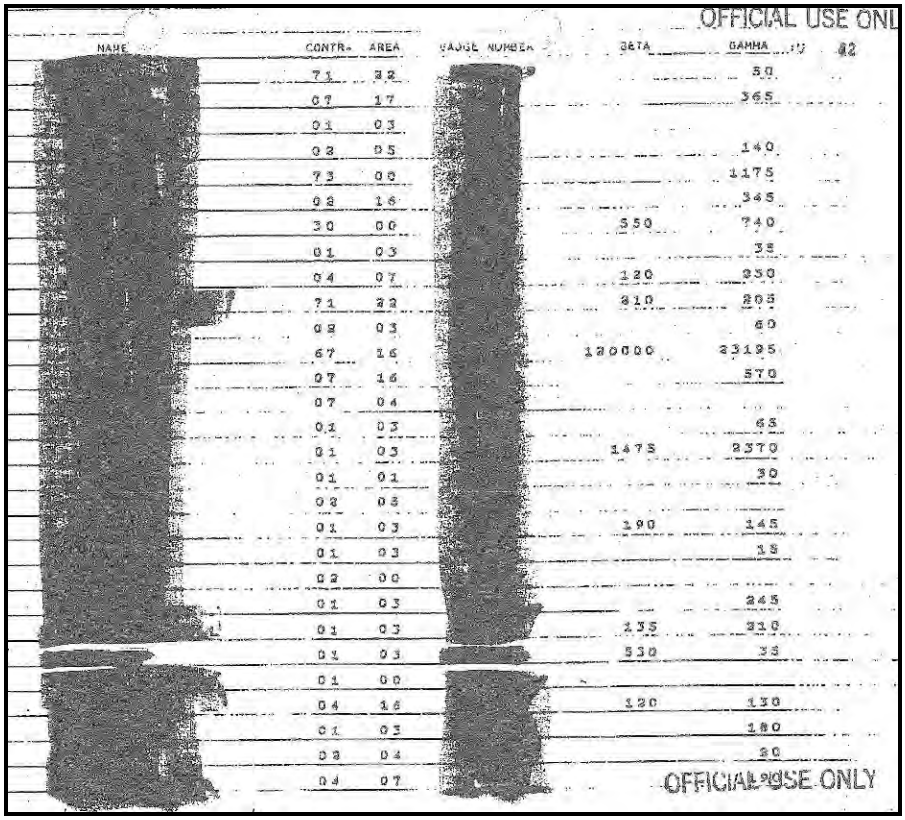


Figure 6-7. Badge results from January 1961 for work in recovery from the SL-1 accident (Vivian and Rockhold 2003, p. 50).

10-105 (2-58)

BADGE REPORT AEC [illegible]

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

Special Report master 032

TO ______

AREA [illegible] ______ 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 [illegible]

☐ Damaged Pencil [illegible]

☐ ______

Week 5

Signed ______

Figure 6-8. Special badge report associated with a high beta reading in Figure 6-7 (Vivian and Rockhold 2003, p. 57).

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 for 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 ______ Health Physics Date ______ Noted ______ Supervisor

Figure 6-9. Personnel exposure questionnaire partially completed for a hypothetical case (Cipperley 1958).

6.3.2 Personnel Monitoring Systems

6.3.2.1 Initial Film Badge

The badging system in place when operations began at the NRTS was called 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, 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 552 film was used for beta-gamma dosimetry for most locations; DuPont 558 film (a combination of types 508 sensitive and 1290 insensitive films) was used at the Materials Test Reactor and the Experimental Breeder Reactor (Cipperley 1958).

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 (Cipperley 1958). A calibration curve was used to convert the cadmium-shielded portion to penetrating gamma exposure in roentgen. The OW density that corresponded to the gamma exposure was subtracted from the measured OW density, and the remainder was converted to beta dose in rep (Cipperley 1958).

DuPont 552 film has a threshold level of 30 mR, and DuPont 558 film has a threshold level of 10 mR (Cipperley 1958). The OW 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 OW over-responds per unit exposure to low-energy photon radiation, as shown in Figure 6-10, by a factor of 30 at 40 keV (Cipperley and Gammill 1959). Because there was no isolated plutonium at the NRTS, the nonpenetrating radiation is considered to be beta radiation. Using a cadmium filter with its high *Z* severely attenuates the photons that get to the film, so the over-response is reduced to a factor of 2 at 125 keV and becomes less than 1 at 50 keV. The beta particle range is independent of the atomic number Z (it depends only on the density) so the 1-mm cadmium filter (~900 mg/cm^2) simulates a tissue depth of 9 mm for beta radiation.

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

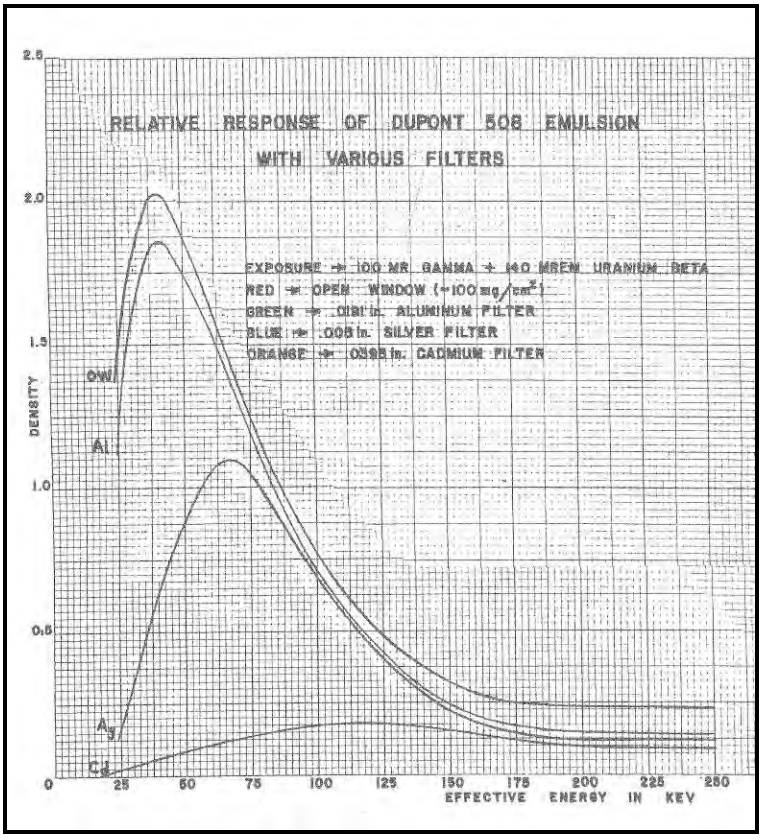


Figure 6-10. Response of DuPont 508 film with various filters to 140 mrem from uranium beta and 100 mR of different energy photon irradiation. The original badge used the OW and cadmium-shielded films. The multiple-filter badge used all three filters plus the OW (Cipperley and Gammill 1959).

6.3.2.2 Multiple-Filter NRTS Film Badge

In March 1958, the security badge and film badge were combined in a film badge that contained filters of 1 mm of cadmium, 0.013 mm of silver, and 0.5 mm of 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, which resulted in an absorber thickness of 100 mg/cm^2 in the OW that filtered out beta radiation below 360 keV.

With the four absorbers, it was possible to separate beta radiation from photon radiation and to somewhat determine photon energy. Figure 6-10 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 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 limit of detection (LOD) of 10 mrem was used for both beta and gamma radiation (Horan 1963).

The DOE Idaho Operations Office 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 OW, 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 approximately 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 (Horan 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 after 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 be, and in many cases was, fastened on a belt around the worker to determine a beta:gamma ratio for each particular entry into SL-1 during recovery operations (Cipperley, Henry, and Cusimano 1965).

6.3.2.3 Original Lithium Fluoride Teflon Thermoluminescent Dosimeter System

Beginning in November and December 1966, individuals who were projected to receive doses of less than 0.5 rem/yr were given a lithium fluoride (LiF) disk 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) and impregnated with 28 mg LiF], were mounted in a badge behind an OW 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, which results in little energy correction for beta or gamma radiation. The badge could read 30 mR on a quarterly basis (Watkins undated), so more small doses were reported (Cusimano and Cipperley 1968). The angular dependence of the radium-gamma response (within 10% to 70º) is superior to film because the material acts like an ionization chamber. For normal monitoring, only the OW TLD was read, and that was considered the penetrating dose unless it read more than 125 mrem, in which case the shielded TLD was also read (Watkins undated).

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 (i.e., with longer exchange cycles, typically 3 or 6 months). In July 1968, the monitoring period was increased from 3 to 6 months (Voelz 1969). In December 1972, annual processing was used for 1,190 low-dose individual TLDs and quarterly processing was used for 960 TLDs (Cusimano 1972). Employees on a monthly badge exchange 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; Voelz 1970, p. 8). A later discussion introduced the use of a ^{137}Cs source, so these earlier calibrations probably used radium, which was used for testing (Cusimano and Cipperley 1968).

6.3.2.4 Automatic Thermoluminescent Analyzer 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 multielement badge using the same filters. This system 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. An LOD of 30 mR was used for gamma and beta (Walker 1974).

6.3.2.5 Harshaw Two-Chip Thermoluminescent Dosimeter System

Several unstable characteristics including over response, warping of the TLD material and reproducibility of results, of the ATLAS led to rapid implementation of a two-chip TLD system beginning in December 1974 for the Idaho Chemical Processing Plant (ICPP), in February 1975 for

the prime contractor at Test Area North (TAN), Test Reactor Area (TRA), etc., and in May 1975 for ANL-W. These issues were evaluated by the HSL at the time and no modifications to worker doses were deemed necessary (Black 1974; Walker 1974). This commercial Harshaw system used two LiF TLDs that were 240 mg/cm^2 thick. In 1976, holes were punched in the security badges to restore the OW. One chip was covered by 540 mg/cm^2 (initially 350 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 of beta energy dependence. The initial thin aluminum filter (density thickness of 350 mg/cm^2) allowed higher energy beta radiation to expose the chip that was used for measuring the penetrating dose (at 1,000 mg/cm^2), so it was changed to 2 mm in July 1977 (INEL ca. 1978). The practice was to read only the OW chip to determine if the nonpenetrating dose was above 15 mrem and thus reportable. If the threshold dose was exceeded, both chips would be read and the penetrating and nonpenetrating doses would be computed (Kalbeitzer 1983).

6.3.2.6 Panasonic Four-Chip Thermoluminescent Dosimeter System

In 1986, with the advent of DOELAP, INL changed to the four-element Panasonic 814 AS4 System (Gesell, Hall, and Andersen 1992; INEEL 2001). Lithium borate ($Li_2B_4O_7$) TLD elements with plastic and aluminum filtration provided 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 provided a strong low-energy photon response. The elements were 15 mg/cm^2 thick. Element 1 had filtration of 16 mg/cm^2, element 2 had filtration of 58 mg/cm^2 of plastic, and elements 3 ($CaSO_4$) and 4 had filtration of 550 mg/cm^2 of plastic and 50 mg/cm^2 of aluminum, respectively. Although none of the elements was 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 for the beta, photon, and mixture performance categories. The angular dependence of this system has been measured for ^{137}Cs gamma rays and four X-ray energies from 16 to 70 keV (INEEL 2001). For the lower energies, attenuation in the absorbers reduces the measured dose. (The attenuation also reduces the delivered deep dose, but this effect is not incorporated in the results.)

The LOD was 15 mrem beta and gamma from January to July 1986 (Gesell 1986), 10 mrem gamma and 30 mrem beta from July 1986 to 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 Film for Neutrons

Kodak nuclear track emulsion, type A (NTA) film was used for neutron monitoring when the field radiation protection staff requested it. NTA film 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.

The minimum assigned dose was 14 mrem (Cipperley 1958, p. 6). 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). On one data sheet from March 1958 with 10 neutron readings, three persons received 14 mrem and one received 42 mrem. Two of the four had gamma readings (the LOD was 30 mrem). Blanks are recorded for 17 people on the data sheet, probably because they were not monitored for neutrons or the film was not read. Only the two individuals received measurable gamma doses. A person on weekly exchange from January to March 1958 received neutron dose equivalents of 14, 28, 42, and 14 mrem, and gamma doses of 130, 70, 30, 30, 50, 30, and 20 mrem with both neutron and

gamma doses received twice (Vivian and Rockhold 2004, p. 5). These values indicate that the data sheets support the LOD of 14 mrem.

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, which provided three independent results (Cipperley 1968). Two tracks or fewer were attributed to background. This resulted in a somewhat higher LOD. In November 1959, a data sheet shows a 10- and 20-mrem neutron dose equivalent (Vivian and Rockhold 2004, p. 15). In January 1962 a data sheet shows a 20-mrem dose (Vivian and Rockhold 2004, p. 6). A data sheet from April 1959 shows neutron dose equivalents of 20, 20, and 40 mrem (Vivian and Rockhold 2004, p. 3). These values suggest an LOD of 20 mrem.

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

The field health physics personnel were aware of the energy limitations of the NTA badge (Sommers 1967, 1969), and they 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. Procedures were established using boron trifluoride (BF_3) pencils to monitor neutrons in the Radioactive Waste Management Complex (RWMC) transuranic waste areas where NTA would not respond to low-energy neutrons (Sommers 1975).

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 results were several designs using the albedo technique in which scattered neutrons from the wearer's body were absorbed by ^{6}Li in a TLD (Gesell et al. 1996).

In the Hankins dosimeter used at INL (Hankins 1973), TLDs with ^{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 (Gesell et al. 1996). Because the $^{6}Li(n,\alpha)^{3}He$ 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) (Gesell et al. 1996). A FNCF that converts the TLD gamma-equivalent 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 (in millirem but not dose equivalent) with the detector in the 3-in.-diameter PNR-4 insert. A plot of FNCF versus 9- to 3-in. ratio is used to determine the FNCF from the measured ratio (Hankins 1976). Values of the FNCF as shown in Table 6-5 (Cusimano 1981) were measured for different fields at INL, were tabulated for assigning the dose equivalent from the badge results, and were routinely updated. This correction was applied to generate the reported neutron dose. An LOD of 15 mrem was used (Gesell et al. 1996). The angular dependence of this system has been measured for moderated ^{252}Cf neutrons (INEEL 2001).

Table 6-5. 1981 FNCFs (Cusimano 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

Note: The facility neutron correction factors in this table were used by the site to generate the reported neutron doses.

The date of the change from NTA to albedo neutron monitoring is somewhat in dispute. Different organizations would typically have changed to new monitoring systems at different times. The present record suggests the switch 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 ca. 1978) states, "initial testing of albedo neutron dosimeter and replacement of NTA neutron monitoring film with same," in October 1976. Aoki (1979) said the albedo system replaced the NTA badge in 1977. Dose reconstructions should make the assumption that this transition occurred on October 1, 1976.

6.3.3 Calibration

6.3.3.1 Beta-Gamma Radiation

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

Gamma calibration initially used a radium source. Victoreen R meters standardized by NBS were used to measure radiation fields (Horan 1959, p. 132). Uranium metal bars that were 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 (Horan 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 an area of 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 after the Three Mile Island-2 reactor accident in 1978. TLD badges were calibrated to 500 mrad tissue using a 1.78-cm-thick phantom 50 cm from the source (300 rad/hr).

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

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

Parameter	Historical description	Uncertainty[a]	Comment
In-air calibration	In 1981, INL 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, INL dosimeter systems were typically calibrated to a photon beam measured as exposure.	-5%	For higher energy Ra-226 and Cs-137 gamma radiation used to calibrate dosimeters, this caused a slight (3%) **under-response** in recorded dose.
Tissue depth of dose	Historically, INL used an unspecified depth to estimate the deep dose.	±5%	The numerical effect of this for photon radiation is comparatively low. INL dosimeter designs had filtration density thickness of 1,000 mg/cm^2 that would relate closely to the 1-cm depth in tissue.
Angular response	INL dosimeter system is calibrated using anterior-posterior (AP) laboratory irradiations.	>300 keV, ~20%	Recorded dose of record **likely to be 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	INL 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. Midcycle calibration minimizes effects.

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

Separation of penetrating from nonpenetrating dose was an issue in 1957 (Bennett 1957) and 1976 (Jenson 1976), particularly for ICPP where strong high-energy beta fields were not unusual.

Use of a phantom in calibration apparently started in 1981 with the NVLAP certification process developed for non-DOE dosimetry processors. About this time, calibration developed in terms of absorbed dose to tissue rather than exposure. Beginning in January 1981, in response to a draft NVLAP (a precursor for DOELAP) 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, a conversion factor C_x value of 1.08 was used (DOE 1981). The current recommended C_x value of 1.03 for ^{137}Cs (DOE 1986a, Table 2) was used beginning in June 1981 (Gesell 1982b; Kalbeitzer 1984).

In 1989, the INL dosimetry service transferred from the DOE Radiological and Environmental Sciences Laboratory (RESL) to EG&G Idaho, the prime contractor. Calibrations continued to use DOE RESL sources and no changes were made to the dosimetry system. EG&G purchased a Shepherd panoramic irradiator with a ^{137}Cs source for badge irradiations. This irradiator did not use a phantom; it was cross-referenced using many TLD irradiations to the DOE source using a phantom (Andersen 1995). In addition, the contractor developed and characterized a uranium slab for beta irradiations (Bean 1995).

Radiation fields have been characterized by comparing field measurements with a NaI(Tl) gamma spectrometer and TLDs mounted on a phantom (Reilly 1998). Most results lie between +27% and -43%. The high gamma-bias results are for locations at RWMC looking at skyshine (backscattered 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 100 keV, so some low-energy photons were likely to have been missed.

6.3.3.2 Neutron Calibration

Table 6-7 describes several common sources of expected laboratory bias for personnel neutron dosimeters based on comparison of the recorded dose with $H_p(10)$.

Table 6-7. Common sources of laboratory bias in the calibration parameters for neutron dosimeters [4].[a]

Parameter	Historical description	Anticipated laboratory bias[b]
Source energy spectrum	In 1976, INL began using dosimeters that were calibrated on a phantom to simulate a worker's body. The previous calibrations did not include response from backscattered radiation.	Depends on workplace neutron spectra. NTA recorded dose of record **likely to be too low** because of high 500-keV threshold for detection of neutrons. NTA film tends to be insensitive to albedo neutrons, so this probably had minimal effect.
Radiation quantity	Neutron dose quantities that were used to calibrate INL 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 **11% high**. The effects of the respective neutron dose quantities used to calibrate INL dosimeters is uncertain and could be evaluated in comparison to the $H_p(10)$ dose used in DOELAP performance testing.

Angular response	INL dosimeters are calibrated using AP laboratory irradiation.	Recorded dose of record is **likely to be too low** because the dosimeter response is lower at non-AP angles. The effect is highly dependent on neutron energy.
Environmental stability	INL NTA film dosimeters and TLDs are subject to signal fade with time, heat, humidity, light, etc.	Recorded dose of record is **likely to be too low**; however, this depends strongly on when the calibration dosimeters are irradiated during the dosimeter exchange cycle. Midcycle calibration minimizes the effects.

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

The initial NTA neutron badges were calibrated using a PoBe neutron source (30 Ci in 1958) (Horan 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 11 MeV (mean energy about 5 MeV) as shown in Figure 6-11 (Kluge and Weiss 1982). The yield of the AmBe source should be only 3% larger than that for the PoBe source (Anderson and Hertz 1971). Kluge and Weiss (1982) calculate conversion factors of 3.51 to 3.76 × 10^{-8} rem-cm^2/n depending on the particular measure of dose equivalent chosen. IAEA (1988) provided a dose conversion factor for AmBe of 3.8 × 10^{-8} rem-cm^2/n for the maximum average dose equivalent. A dose equivalent of 1.5 rem required 3.6 × 10^{7} n/cm^2 (Cusimano 1963), which corresponds to a dose conversion factor of 4.17 × 10^{-8} rem-cm^2/n, so the recorded dose is 11% high. Monte Carlo calculations for 5-MeV neutrons show a dose equivalent of 4.2 × 10^{-8} rem-cm^2/n averaged over the 0- to 2-cm shell on a 30-cm-diameter cylindrical phantom (NCRP 1971). Use of the 50-Ci AmBe source continued until 1993.

In 1993, a 40- by 40- by 15-cm polymethyl methacrylate 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).

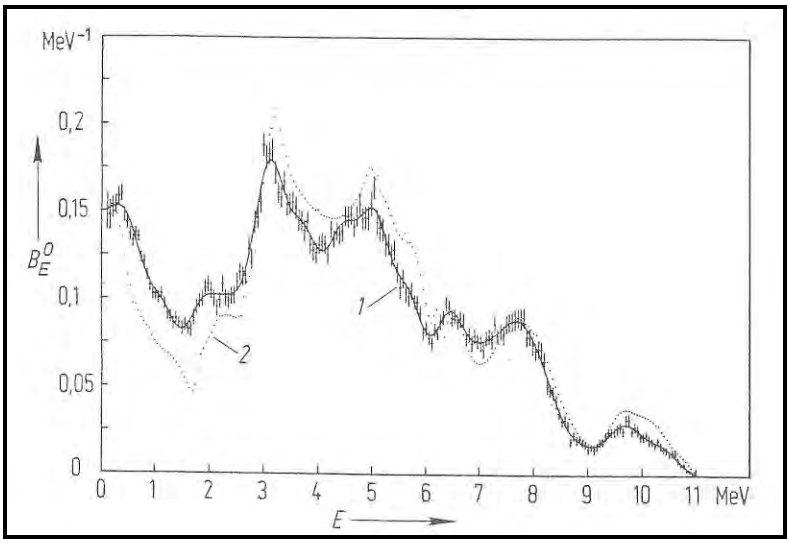


Figure 6-11. Probability density of neutron spectrum from a ^{241}Am-Be(α,n) source (Kluge and Weiss 1982).

6.3.4 Workplace Radiation Fields

Radiation fields varied a great deal at INL and ANL-W as discussed in the Site Description TBD (ORAUT 2010b). At times, very high radiation fields existed, particularly at ICPP, and work was sometimes done in those fields under tight controls.

The epidemiological study of INL (Schubauer-Berigan et al. 2005) provides the excellent summary of radiation exposure history at INL in Table 6-8. The table shows the number of people monitored in the last column, the mean dose received (1 mSv = 100 mrem), and doses at several percentages on the distribution of doses received. For example, in 1952 90% of the 913 monitored workers had doses less than 30 mrem and 99.5% of the workers had doses less than 727 mrem. The mean dose was 2.2 mrem, and the standard deviation of dose was 100.2 mrem.

Figure 6-12 shows the collective dose received by all workers through 1990 from Horan and Braun (1993). The highest collective dose was 3,448 rem in 1965.

6.3.4.1 Penetrating Radiation

Penetrating gamma rays and X-rays are considered to be ≥30 keV photons. The radiation fields at the site, with a few exceptions, have been generated primarily by mixed fission and activation products. Therefore, most of the photon dose has been from photons with energy greater than 250 keV [5]. The INL dosimeters are judged to measure these fields well [6].

The few exceptions are usually characterized by low dose rates. Much of the waste at the RWMC is transuranic waste from the Rocky Flats Plant. This contains predominantly plutonium and americium with 17- and 59-keV photons, which are highly adsorbed by the waste and the shipping container. The dominant fields at the RWMC come from mixed fission and activation products dominated by ^{137}Cs and ^{60}Co [7].

Analytical X-ray generators operating below 100 keV are used in several laboratories. These are easily shielded so the fields are usually low [8].

Table 6-8. Photon dose (rem) percentiles for monitored workers (Schubauer-Berigan et al. 2005).

Year	0%	25%	50%	75%	90%	97.5%	99.5%	100%	Mean	St. dev.	Number
1951	0	0	0	0	0	0	0.4	0.4	0.0022	0.0292	188
1952	0	0	0	0	0.03	0.222	0.727	1.275	0.0216	0.1002	913
1953	0	0	0.06	0.25	0.56	1.09	2.097	7.57	0.2002	0.4034	1,408
1954	0	0	0	0.21	0.9	2.498	4.077	5.76	0.2838	0.6567	2,449
1955	0	0	0.1	0.59	1.6565	3.237	5.162	8.47	0.5157	0.9314	2,946
1956	0	0	0	0.2	1.015	2.555	4.267	22.06	0.3142	0.8423	3,209
1957	0	0	0	0.085	0.54	1.513	3.045	5.12	0.1706	0.4517	4,695
1958	0	0	0.04	0.325	1.26	2.89	4.378	10.51	0.3766	0.8015	5,079
1959	0	0	0.04	0.26	0.96	2.347	4.127	21.85	0.3099	0.7402	5,344
1960	0	0	0.05	0.255	1.075	2.647	3.734	5.01	0.3268	0.6755	5,827
1961	0	0	0.04	0.42	1.704	3.566	4.972	27.26	0.5063	1.2806	5,192
1962	0	0	0.03	0.175	1.115	3.225	5.048	9.885	0.3571	0.8942	5,339
1963	0	0	0.025	0.18	1.0795	2.954	4.022	5.1	0.318	0.7292	5,520
1964	0	0	0.01	0.215	1.2765	3.111	3.98	4.815	0.3538	0.7886	5,446
1965	0	0	0	0.43	2.2045	4.39	6.018	9.815	0.579	1.2107	5,520
1966	0	0	0.02	0.375	1.5595	3.467	4.466	6.045	0.4383	0.8926	5,180
1967	0	0	0	0.17	1.0725	3.084	4.377	4.805	0.3194	0.7702	6,304
1968	0	0	0	0.215	1.14	3.104	4.194	5.295	0.3364	0.7781	4,922
1969	0	0	0	0.24	1.195	2.73	3.982	4.45	0.3279	0.7151	4,758
1970	0	0	0	0.175	1.028	2.624	4.207	4.68	0.2952	0.702	5,051
1971	0	0	0	0.17	0.82	1.899	3.252	4.71	0.2357	0.5427	4,764
1972	0	0	0	0.161	0.855	2.375	3.835	4.665	0.2606	0.634	4,762
1973	0	0	0	0.115	0.6525	2.126	3.909	5.2	0.2185	0.5878	4,494
1974	0	0	0	0.1	0.515	1.696	2.993	4.065	0.1734	0.4574	4,878
1975	0	0	0.006	0.09	0.405	1.412	2.789	3.945	0.1531	0.4071	5,025
1976	0	0	0	0.11	0.506	1.646	2.643	4.145	0.1712	0.4294	5,489

Year	0%	25%	50%	75%	90%	97.5%	99.5%	100%	Mean	St. dev.	Number
1977	0	0	0	0.094	0.485	1.947	3.385	10.77	0.1869	0.526	5,677
1978	0	0	0.009	0.087	0.394	1.535	2.872	4.386	0.1563	0.4295	6,551
1979	0	0	0	0.074	0.369	1.46	2.732	4.18	0.1419	0.4064	6,863
1980	0	0	0	0.047	0.277	1.062	2.112	16.93	0.1073	0.3579	7,380
1981	0	0	0	0.046	0.252	0.824	1.674	3.289	0.0876	0.2358	6,722
1982	0	0	0	0.038	0.195	0.638	1.5	2.904	0.0715	0.204	6,556
1983	0	0	0	0.035	0.187	0.516	0.976	1.577	0.0582	0.1473	6,610
1984	0	0	0	0.034	0.179	0.577	1.198	2.285	0.0619	0.1766	7,476
1985	0	0	0	0.039	0.219	0.872	1.24	2.415	0.082	0.226	7,917
1986	0	0	0	0.024	0.192	0.773	1.749	9.338	0.077	0.2623	8,568
1987	0	0	0	0.025	0.155	0.733	1.417	3.158	0.0659	0.2098	8,575
1988	0	0	0	0.022	0.145	0.537	1.086	3.086	0.0545	0.1641	8,667
1989	0	0	0	0.016	0.101	0.559	1.315	7.811	0.0516	0.2169	8,848
1990	0	0	0	0.012	0.0954	0.566	1.266	2.728	0.049	0.1852	10,165
1991	0	0	0	0	0.037	0.249	0.694	4.577	0.0232	0.1094	10,742
1992	0	0	0	0	0.037	0.169	0.376	1.276	0.0157	0.0563	9,571
1993	0	0	0	0	0.065	0.342	1.06	1.535	0.0311	0.1233	9,048
1994	0	0	0	0.01	0.0786	0.335	0.729	1.394	0.0322	0.1121	8,473
1995	0	0	0	0.011	0.1031	0.449	1.153	1.844	0.0428	0.1533	7,818
1996	0	0	0	0.016	0.108	0.421	0.79	1.368	0.0393	0.1215	6,459
1997	0	0	0	0.01	0.081	0.259	0.574	1.108	0.0261	0.082	6,280
1998	0	0	0	0.002	0.062	0.208	0.462	0.844	0.0205	0.065	5,875

There are a few 250-keV X-ray generators and a 320-keV and a 160-keV X-ray generator that were used for radiography or radiography development studies. Wall shielding is generally adequate, and any transmitted photons have energy near the operating voltage because of the hardening caused by the shielding [9]. The radiography facility at TRA has no roof shielding, so nearby dose rates of less

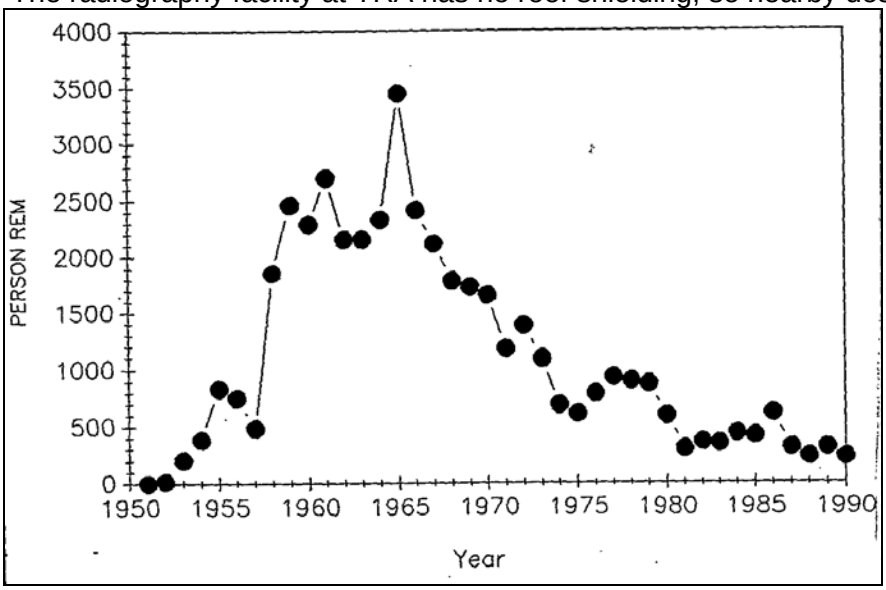


Figure 6-12. Collective dose for INL personnel, 1951 to 1990 (Horan and Braun 1993).

than 10 mrem/hr are possible, usually for short periods. The scattered photons have lower energy than the primary beam. The people likely to be exposed are radiographers who receive fairly high exposures, mostly from the ambient radiation fields at their worksites and some from 300-keV ^{192}Ir or 1.25-MeV ^{60}Co [10].

All radiological work areas essentially involved beta and photon radiation covering a wide range of energies. These fields can be generally classified according to the IREP codes in Table 6-9.

6.3.4.2 Non-Penetrating Radiation

Non-penetrating dose is important for certain cancers (e.g. skin, breast, testes, etc…). Non-penetrating radiations are considered to be >15 keV electrons and <30 keV photons and X-rays. Because the site had essentially no separated plutonium, all non-penetrating doses should be assigned as >15 keV electron doses. Electron radiation fields are usually associated with activation or fission products outside a container such as in spills or hot cells or when only lightly shielded [12]. High electron fields were not unusual at the Idaho Nuclear Technology and Engineering Center (previously the ICPP) where there are large quantities of fission products. Pure high-energy electron fields in some locations, particularly in the exhaust stream, have caused dosimetry problems because the badge shielding or instrument packages did not provide a full 10-mm tissue-equivalent coverage and the electron fields would therefore be measured as gamma fields (Black 1974; Jenson 1976; INEL ca. 1978).

The high-bias electron results in Figure 6-10 from comparison of TLDs to a Phoswich electron spectrometer are for electron sources at contact or at 1 cm, which results in hard-to-reproduce geometry. The low-bias electron results are for large area sources for which even the spectrometry results have large variations. The electron occupational radiation fields (only three) have a bias of more than 15% (Reilly 1998).

Electron field dosimetry became reasonably accurate with the definition of DOELAP requirements in the early 1980s. Before then, electron 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

Table 6-9. Selection of IREP electron and photon energies for INL facilities [11].

Process/ buildings	Description			Radiation type	Energy group (keV)	Percentage
	Operations	Begin	End			
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 might be significant higher energy electron radiation			Electron	>15	100
				Photon	30–250	25
					>250	75
	PBF, TRA, ARA, TAN, EBR, ANL-W, SPERT	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 electron radiation during sampling and maintenance work resulting from fission products.			Electron	>15	100
				Photon	30–250	25
					>250	75
	ICPP, ANL-W	1952	2003			
Calibrations	Calibration of instruments and dosimeters.			Electron	>15	100
	CF 633, 636	1952	2003	Photon	30–250	25
					>250	75
Waste handling	Radiation characteristics are highly dependent on source of waste, but typically fission product nuclides (Sr/Y-90, Cs-137) are dominant. Transuranic waste from Rocky Flats Plant contains Am-241 with 59-keV photon.			Electron	>15	100
				Photon	30–250	25
	RWMC, WERF	1953	2003		>250	75
Uranium handling	Produced special armor from depleted uranium. Primarily electron radiation from U-238 daughters. Some			Electron	>15	100
				Photon	30–250	90

	gamma from contaminants and Cs-137 sources used in process.			>250	10
	SMC	1985	2003		

high-energy electrons from either uranium or strontium (see Section 6.3.3.1). The dose from low-energy electrons can be missed altogether if the electron energy is not sufficient to penetrate the detector cover, and the dose can be under-reported if the electron energy is not sufficient to penetrate the entire detector. The mean electron energy for the spectrum from a particular nuclide is about one-third of the maximum electron energy for that nuclide.

Based on the range vs. energy curve for electron particles and the electron energy distribution of electron emitters (BRH 1970, pp. 90, 91,123), the fraction of electron radionuclides with ranges greater than the abscissa is plotted on the ordinate in Figure 6-13. Electron-emitting nuclides varied from location to location and time to time at INL, so a correction factor common for all facilities was estimated [13]. This analysis used the entire mixture of radionuclides to obviate whether the choice is correct and to reflect the wide variety of radionuclides used at INL. To reflect that the electron spectrum is not monoenergetic because of the energy carried off by the neutrino, a curve is presented for the mean energy or one-third of the maximum energy. To reflect that some electron 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 with a larger range essentially show the depth dependence of electron dose, because the energy loss of electrons does not have much energy dependence. These curves also demonstrate why early dosimeters with thicker, more sensitive elements failed to report the electron dose correctly at a depth of 7 mg/cm^2, which was chosen in the early 1980s. These curves demonstrate why the electron dose that was assigned for skin is inappropriate to use for the breast and testes, where much of the organ is at a depth greater than 1 mm or 100 mg/cm^2, and for most persons at depths greater than 1 cm.

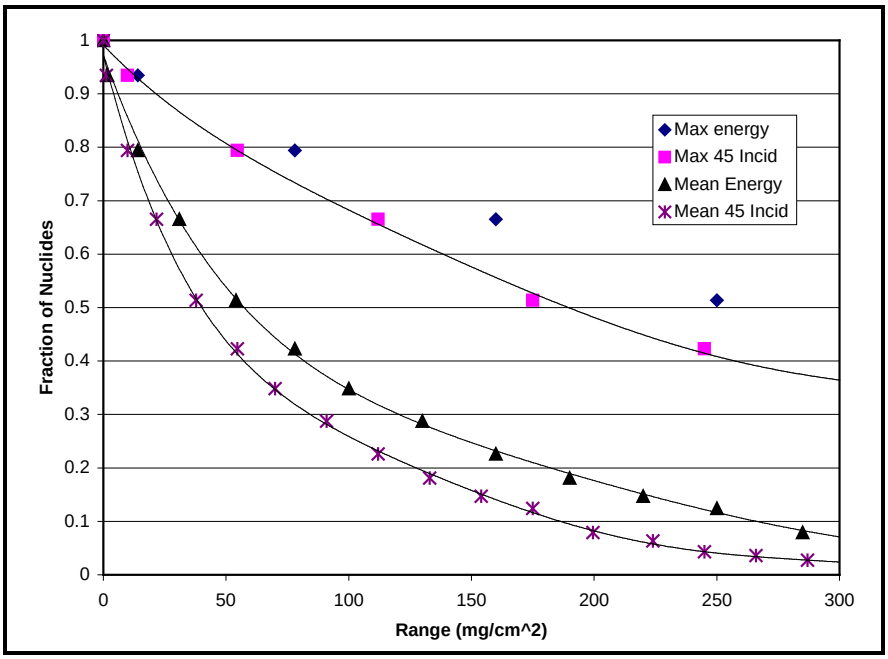


Figure 6-13. Distribution of electron ranges [14].

To calculate the fraction of dose missed by a dosimeter, the dose reconstructor needs only to 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 electron spectrum and the angular distribution of the radiation exposure. To estimate this, 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 of that value for 45º incidence for the maximum energy [15]. The curves are the result of a polynomial trend line to the data, so averaging the fraction of radionuclides is relatively easy.

Table 6-10 provides the cover and detector thicknesses for the electron badges used at INL. Thicknesses with a "~" are estimated. The fraction of measured electron dose shown in Table 6-10 is the average as described above. To determine the corrected electron dose, the measured and missed nonpenetrating results from the dosimetry system should be multiplied by the values in the last column of Table 6-10. 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 [16]. For the Panasonic system, such a correction has already been made (Gesell 1986), so the recommended correction is 1.

Table 6-10. Electron dosimeter thicknesses and associated under-reporting [17].

Dosimeter system	Period	Covers (mg/cm^2)	Detector thickness (mg/cm^2)	Electron correction factor
Self-service system	Aug 1951–Feb1958	~50	~50	2.0
Multifilter film	Mar 1958–Sept 1973	100	~50	2.8
LiF Teflon TLD	Nov 1966–1974	100	75	3.0
ATLAS	Feb 1974–May1975	100	~100	3.3
Harshaw 2 chip TLD	Dec 1974–1975	104	240	4.8
Modified Harshaw TLD	1976–1985	4	240	2.4
Panasonic TLD	1986–2006	16	15	1.0

a. Note: The site began phasing out the use of the multifilter film dosimeter in November 1966 but didn't completely phase out its use until September 1973.

6.3.4.3 Neutron Radiation

Most workers have not been exposed to neutrons and so have not been badged to measure neutrons. Neutron fields are specific to a few facilities. The high-dose locations where most of the gamma and electron dose has been received, such as the ICPP and SL-1, have not had associated neutron dose. Table 6-11 lists the facilities and periods for which neutron exposure is considered likely.

Table 6-11. Facilities and periods for neutron exposure [18].

Location	Period	Comments
Programmatic		
ANL-W, TREAT	1958–1994	Experiment floor
ANL-W, ZPPR	1969–1992	Between reactor halves during maintenance
MTR	1952–1970	Experiment floor
TAN Warm Shop	1986–1988	See EPRI 1986, 1987
TAN	1986–2006	Spent fuel storage pad
ANL-W Neutron Radiography Facility	1977–2006	
TRA Hot Cell Cave		Cf-252 on filters
RWMC ILTSF		Cargo container
RWMC WM 632		Transuranic waste drum
14 MeV		
RWMC SWEPP	1990–2004	14 MeV for waste characterization
TAN Warm Shop	1991–1994	Refurbish 14 MeV for waste characterization
TRA 635	1990–2006	PINS Cf-252 6 and 14 MeV
Sealed sources		
CPP-1649	1985–2006	PuBe Calibration Facility
CF-633	~1970–2002	Cf-252 Calibration Facility
CF-636	1952–1994	AmBe Calibration Facility
IRC		AmBe Cf-252
ANL-W, ZPPR	1969–1992	For instrument calibration

In 1969, 150 workers were involved in radiation work that required their NTA neutron dosimeters to be evaluated. At the time, there were 2,900 film-badged employees and more than 3,000 TLD-monitored personnel (Vallario, Hankins, and Unruh 1969).

In 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, only 1,461 neutron dosimeters were issued (both monthly and quarterly badges) in comparison with about 50,000 beta/gamma badges. Only 54 badges had reportable doses (≥15 mrem) as shown in Figure 6-14 (Gesell et al. 1996). Only six were above 35 mrem. The Hankins albedo dosimeter badges in use since 1975 are sensitive to all neutron fields. An FNCF determined from the 9- to 3-in. ratio in the worker location has been used by the site to adjust the measurement result to dose equivalent.

In 1997, several workplace neutron fields were measured with TLDs that were mounted on a phantom and, at nearly the same time, with a ROSPEC neutron spectrometer (Reilly 1998). The relative biases [(Dosimeter – Spectrometer)/Spectrometer] for the neutron fields are shown in Figure 6-15. These results show a greater dispersion than the gamma results. The two lowest values (-52% and -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 of the radiation field. The next lowest value (-38%) is for the ^{252}Cf instrument calibration source at a

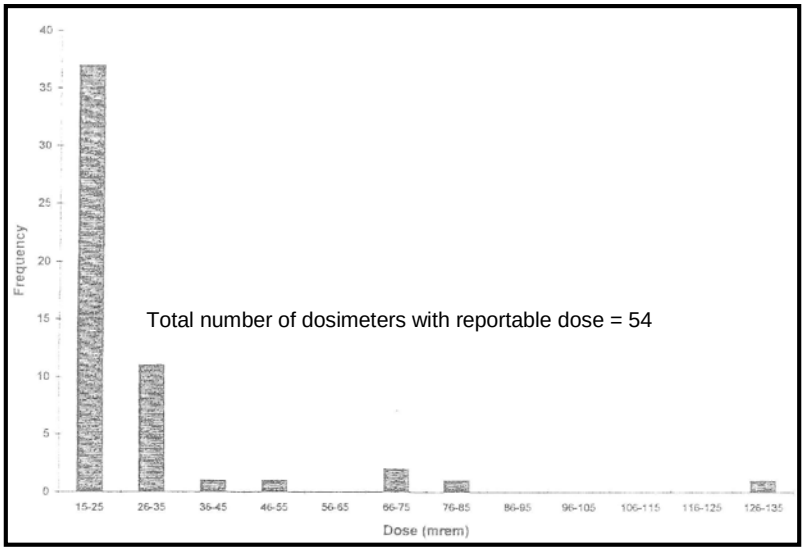


Figure 6-14. Distribution of reportable neutron dose for the first 9 months of 1995. Of 1,461 dosimeters, 1,407 were below the 15-mrem reporting level (Gesell et al. 1996).

distance of 3.5 m, where the operator stands. The two highest values (94% and 71%) are for a waste drum that was reanalyzed and a new 9- to 3-in. ratio that was determined because of the unsatisfactory initial result. The report suggests that other waste barrels might have had neutron sources that caused interference. The remaining bias values lie between -16% and 44% (Reilly 1998).

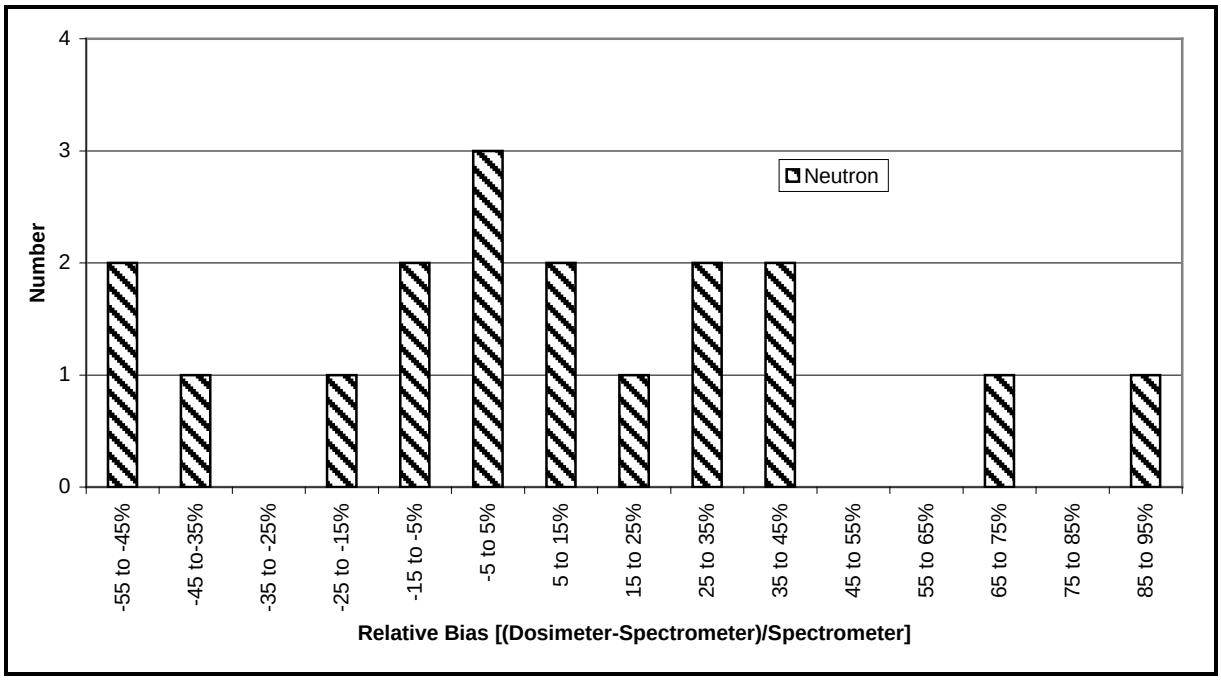


Figure 6-15. Neutron radiation field characterization (Reilly 1998).

Figure 6-16 provides spectra for the 14-MeV neutron generator as seen through 10 cm of polyethylene shielding typical of the INL facilities using the ^{252}Cf (fission) neutron source (Ing and Makra 1978) and the AmBe neutron source (Kluge and Weiss 1982). Sources of neutron exposure include neutron sources at the instrument calibration laboratories and 14-MeV neutron generators used to characterize waste. For these spectra, the NTA film works reasonably well. Use of small ^{252}Cf sources for research began after albedo badge use.

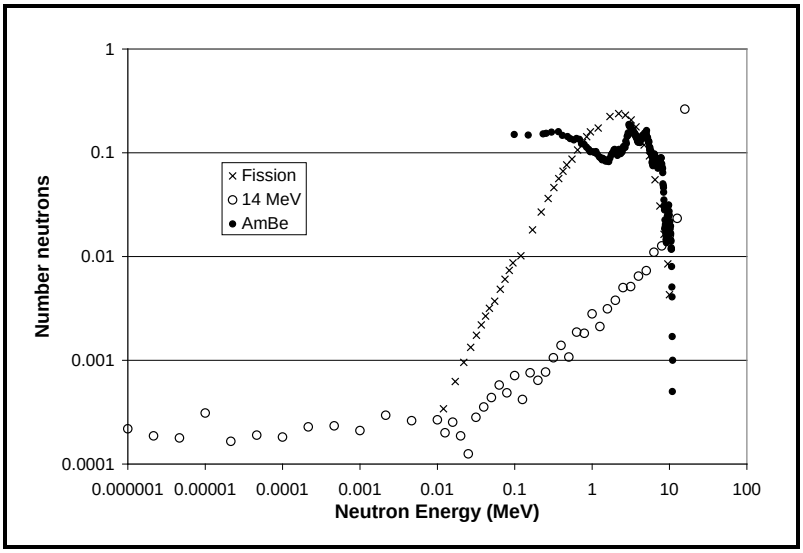


Figure 6-16. Neutron spectra that simulate INL facilities [19].

Most of the reactors built at INL and ANL-W had no beam ports. The neutrons were generally well contained away from the workplace. The 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 energy) are attenuated much more quickly in concrete or water than the gamma fields. This is

not true for lead or iron, but these are usually not used as gamma shields where neutrons also exist. Neutron fields are thus generally not a problem at an enclosed reactor.

6.3.4.3.1 Materials Test Reactor Neutron Radiation

The exception to the above discussion is the Materials Test Reactor (MTR), which operated from 1952 to 1970 and had beam ports and neutron beams extending onto an experiment floor. The Zero Power Plutonium (later Physics) Reactor (ZPPR) and the Transient Reactor and Experiment Test (TREAT), both at ANL-W, are also in this category. Some neutron surveys of the MTR experiment floor have been recovered (Sommers 1959, 1962; Hankins 1961), but these do not individually provide all components of the radiation field. Hankins 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 and measured the thermal neutron component at six other locations. The Hankins data have been reanalyzed (ORAUT 2006a) using more recent Bonner response curves (Hertel and Davidson 1985). Figure 6-17 shows the resultant neutron spectra for locations 3 and 23, which have higher doses and nearly the maximum low-energy intermediate and fission components, respectively. Figure 6-18 shows the correlations of the thermal and intermediate neutron dose equivalents to the fast neutron dose equivalent for the ORAUT reanalysis of the Hankins data.

The trend line for the reanalyzed intermediate-energy neutron dose equivalent has R^2 values of 0.86 and 0.92 in comparison with an R^2 value of 0.5 for the original analysis, which demonstrates a better fit to the data. The average ratio of thermal to fast neutrons is 0.071 ± 0.025, that for low-energy intermediate to fast neutrons is 0.177 ± 0.057, and that for the higher energy intermediate to fast neutrons is 0.149 ± 0.046, where fast neutrons are those above 0.2 MeV.

The MTR personnel who were likely to receive neutron dose were assigned NTA film in their dosimetry packets, but it would have missed the dose below 0.5 to 0.8 MeV. For the MTR spectra,

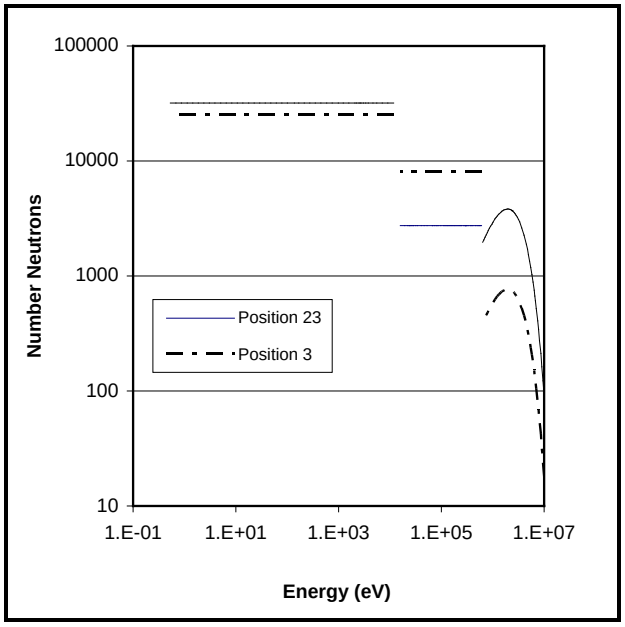


Figure 6-17. Sample MTR spectra from Hankins Bonner measurements (ORAUT 2006a).

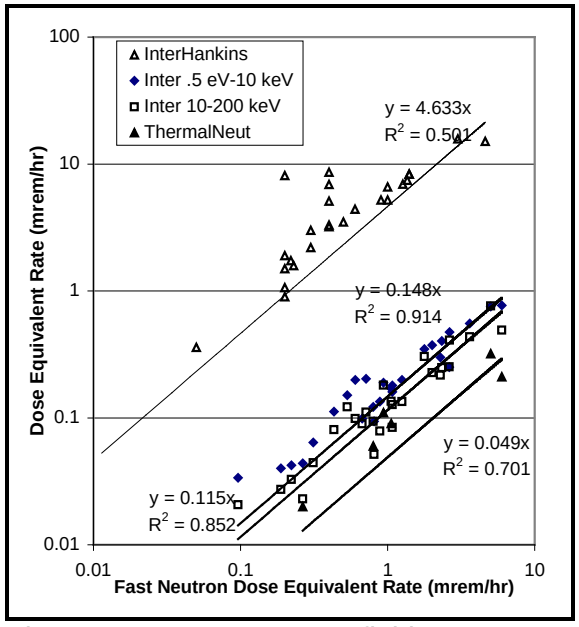


Figure 6-18. MTR neutron field components (ORAUT 2006a).

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 (ORAUT 2006a). To correct for the NTA film dosimeter's energy response limitations, neutron doses for the MTR workers measured with NTA film should be adjusted by a factor of 2 ± 0.3 ($1/0.52 \pm 0.08/0.52^2$) (ORAUT 2006a).

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 be not representative of the general workplace. Figure 6-19 shows the correlation of fast neutron dose equivalent to the gamma dose for these measurements.

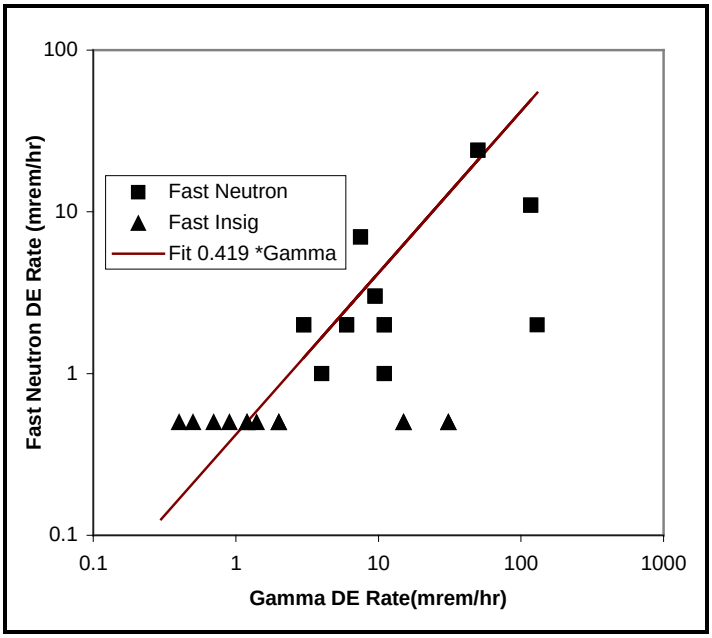


Figure 6-19. Correlation of fast neutron dose equivalent to gamma dose at MTR (ORAUT 2006a).

The fast neutron component was insignificant for several of the measurements. The values in Figure 6-19 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 a normal distribution is a slightly better description of the data than a lognormal distribution. The fast neutron dose equivalent is 0.42 ±0.35 of the gamma dose rate for this data set. Combining these results as shown in Equation 6-1 yields a total neutron dose equivalent of 0.58 ±0.48 of the gamma dose equivalent on the MTR experiment floor [20]:

$$\frac{TotalNeutronDose}{GammaDose} = \frac{FastN}{GammaDose}\left(1 + \frac{Thermal}{FastN} + \frac{LoInter}{FastN} + \frac{HiInter}{FastN}\right) \quad (6\text{-}1)$$
$$= (0.42 \pm 0.35)(1 + 0.071 + 0.177 + 0.149) = 0.58 \pm 0.48$$

The variations in the different components of neutron dose rate are so small in comparison with the variation between the fast neutron and gamma dose equivalent rate that they are unimportant [21].

Many of the people wearing NTA film would receive gamma dose at locations other than on the MTR experiment floor while the reactor was operating. For example, health physics technicians would often have been covering jobs with only beta/gamma fields. A craftsperson might have serviced pumps that carried radioactive water and not received any neutron dose. Therefore, simply multiplying the gamma dose by 1.6 (i.e., average = 1 + 0.58) or 2.1 (i.e., 84% confidence = 1 + 0.58 + 0.48), although favorable to claimants, is probably inappropriate. Further extrapolating experience at the MTR at TRA to TREAT or ZPPR at ANL-W is not recommended [22]. While similarities of the neutron spectra and associated corrections are likely to be valid, the relative amount of gamma and neutron radiation might not be. In addition, 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 [23].

A fairly large neutron field existed between the halves of the ZPPR when it was off due to ^{240}Pu spontaneous fission decay and subsequent multiplication in the subcritical reactor. Rather than trust the NTA film in use at the time, the practice was to conduct neutron and gamma surveys using a 9-in. ball detector. Simons, Young, and Thalgott (1972) report spectral measurements (shown in Figure 6-20) and neutron dose assignments based on time and neutron dose rate. Neutron doses were estimated based on the measured neutron:gamma ratio and gamma doses measured with pencil dosimeters. These results were entered in the dosimetry records (Rohrig 2006a). Shields were also used to minimize the reactor surface open to workers. Figure 6-21 shows the integral distribution of neutron dose with energy for this spectrum. The fraction of dose below the NTA cutoff is quite consistent with the MTR result.

6.3.4.3.2 Test Area North Fuel Storage Casks

As noted in the Site Description TBD (ORAUT 2010b, Section 2.2.1.3), fuel storage casks are on a storage pad at TAN. The dose rates are 25 to 30 mrem/hr gamma and 40 mrem/hr neutron. The metal cask attenuates the gamma radiation, but does not appreciably affect the neutron field. The loaded casks were temperature-tested in the TAN Warm Shop in 1985 and 1986. The shop has offices nearby at one end on the second floor. Neutron radiation levels were discovered in the offices, and the six people in these offices were not wearing albedo neutron badges but were wearing beta/gamma dosimetry. Each of the three casks was in the area about 2 weeks while temperature measurements were made before they were moved outside to the storage pad. The casks contain irradiated oxide fuel in which the $^{18}O(\alpha,n)$ reaction generates neutrons in the range of a few megaelectron-volts. This radiation was attenuated by distance and the building concrete between the cask and the offices. A TLD area-monitoring albedo system identified the radiation field. Based on Electric Power Research Institute (EPRI) documents (EPRI 1986, 1987; PNL, VPC, and EG&G

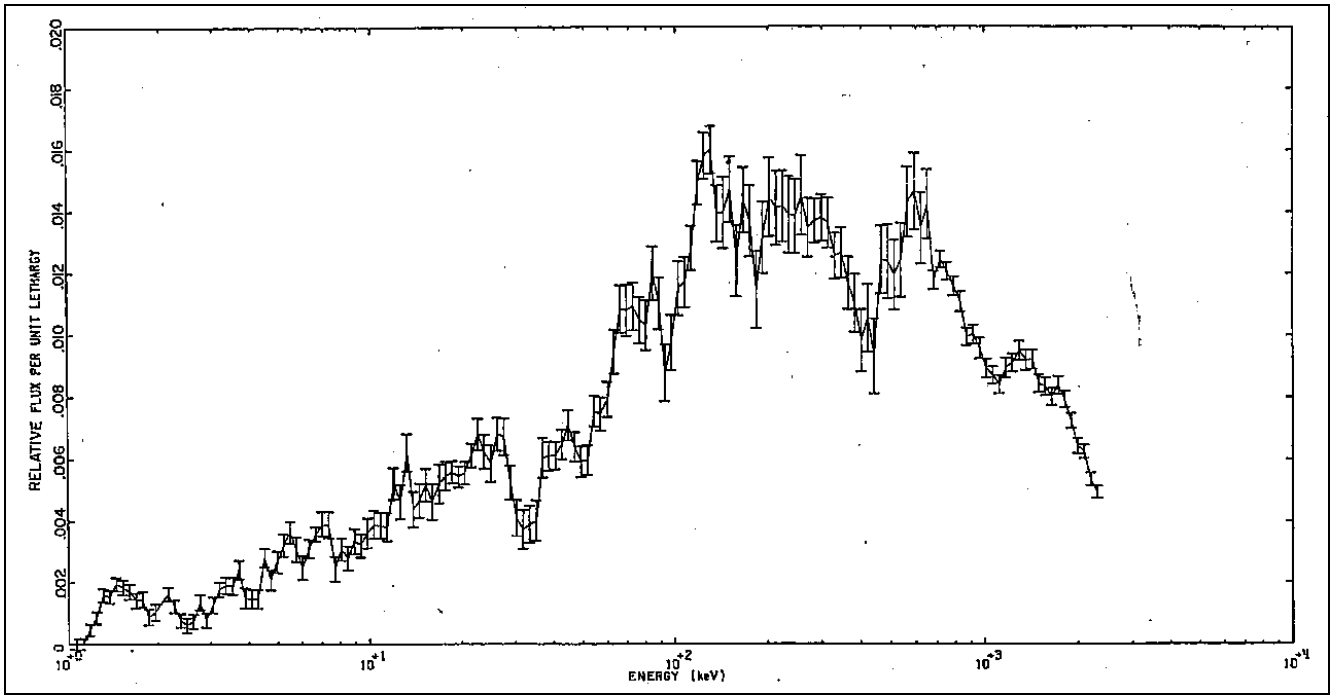


Figure 6-20. ZPPR Assembly-3 Phase-1A fast neutron spectrum measured midway between halves with both shields withdrawn (Simons, Young, and Thalgott 1972).

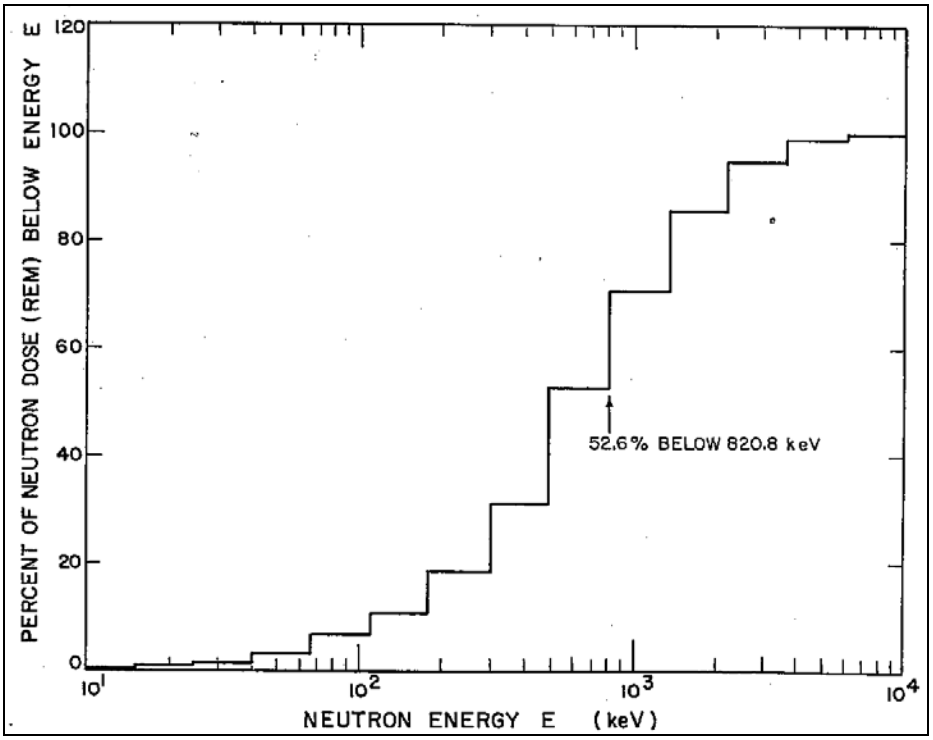


Figure 6-21. ZPPR Assembly-3 Phase-1A fast neutron dose spectrum midway between halves with both shields withdrawn (Simons, Young, and Thalgott 1972).

1987), the temperatures were measured from September 11 to 23, 1985; January 14 to February 6, 1986; and June 2 to 27, 1986. The person most affected was in operations. Safety and radiological engineers were also affected. The estimated dose equivalent for full-time occupancy is less than 50 mrem for each of the three cask evolutions. The records indicate that manual additions to the dosimetry records were made at the time for neutron dose based on area-monitoring TLDs [24].

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Photon Dose Adjustments

No adjustments to photon doses are necessary.

6.4.2 Electron Dose Adjustments

Electron field dosimetry became reasonably accurate with the definition of DOELAP requirements in the early 1980s. Before then, electron 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. The dose from low-energy electrons can be missed altogether if the electron energy is not sufficient to penetrate the detector cover, and the dose can be under-reported if the electron energy is not sufficient to penetrate the entire detector. To determine the corrected electron dose, the measured and missed electron results from the dosimetry system should be multiplied by the values in Table 6-12. 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. For the Panasonic system that was used after 1985, such a correction has already been made (Gesell 1986), so the recommended correction is 1.

Table 6-12. Electron dosimeter under-reporting correction factors.

Period	Electron correction factor
1951–1957	2.0
1958–1965	2.8
1966–1973	3.0
1974–1975	4.8
1976–1985	2.4
1986–2006	1.0

6.4.3 Neutron Dose Adjustments

6.4.3.1 Neutron Energy Response Adjustments

Because of the NTA film dosimeter's energy response limitations, the following adjustment factors should be applied to the measured and missed neutron doses reported prior to October 1976. In October 1976 the site began using TLDs to measure neutron doses, which did not have any significant energy response limitations.

As previously recommended in section 6.3.4.3, the neutron doses for the MTR workers measured with NTA film should be adjusted by a factor of 2 ± 0.3. Section 6.3.4.3 also indicated that the neutron spectra for the TREAT and ZPPR facilities were comparable to the MTR's neutron spectra. Therefore, the adjustment factor for the MTR neutron doses is applicable to the TREAT and ZPPR facilities. To simplify the dose reconstruction process, the measured and missed neutron doses from the MTR, TREAT, and ZPPR should be multiplied by 2.3, which is the upper-bound for this correction factor.

As indicated in Table 6-15, the fraction of lower energy neutrons at the other site facilities is ≤ 20%, which means the NTA film dosimeters were capable of measuring at least 80% of the neutron doses at those facilities. Therefore, the neutron doses for workers at facilities other than the MTR, TREAT, or ZPPR should be adjusted by a factor of 1.25 (1/0.80).

6.4.3.2 Radiation Weighting Factor Adjustments

All measured and reported dose equivalents from INL in this TBD used the quality factors based on the LET of the ionizing secondary particles that was established in the 1950s and used since by U.S. regulatory agencies [27]. 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 1991, Table 1).

The reported dosimetry data require correction to change from dose equivalent (pre-ICRP Publication 60) to the newer dose quantity (ICRP 1991; NIOSH 2007). ICRP Publication 74 tabulates the ambient dose equivalent (dose equivalent at 10-mm depth in a 30-cm-diameter sphere) for neutrons (ICRP 1996). The ratios of organ to ambient dose equivalents are tabulated in Appendix B of NIOSH (2007), so this quantity is 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 those in other TBDs.

The dose equivalent for a spectrum of particle energies is the result of an integral of the fluence spectrum $\Phi(E)$ times a dose equivalent conversion factor $DECF(E)$, which also depends on energy over the considered range of energies:

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

Error! Bookmark not defined. These factors are incorporated in statements of dose equivalent values and calibrations that follow 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 1971). The primary geometry is conventionally considered 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 10 due to attenuation in the human body. Dosimeters are designed to respond to radiation entering the body on the side where they are located, and they work best for an AP irradiation geometry with the dosimeter on the front of the body [28].

For ambient dose equivalent, the same equation applies 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 to calculate the correction factors [29].

Table 6-13 summarizes the locations where neutron dose is credible. Table 6-14 lists the calculated fractions of dose equivalent in the IREP energy groups and the conversion factors from dose equivalent to equivalent dose for site 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 [31]. 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 6-13. This correction should be applied to the measured and missed neutron doses.

Table 6-13. Calculated and recommended dose equivalent fractions and quality factor corrections [30].

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 polyethylene	2.4E-08	3.1E-06	1.5E-03	1.00
MTR experiment floor average	0.18	0.06	0.49	0.28
MTR experiment floor maximum	0.24	0.08	0.52	0.35
MTR experiment floor minimum	0.13	0.03	0.46	0.19
ICRP (1996) H^*_{10} /NCRP (1971b) *H*				
Bare fission		1.46	1.32	1.09
AmBe			1.41	1.05
14-MeV 10-cm polyethylene	0.69	1.47	1.36	0.93
MTR experiment floor average	0.86	1.08	1.33	1.12
MTR experiment floor maximum	0.80	1.08	1.37	1.12
MTR experiment floor minimum	0.92	1.08	1.30	1.12
Recommended defaults				
Dose equivalent fractions				
14-MeV 10-cm polyethylene			0.05	0.95
Source calibrations			0.20	0.80
MTR experiment floor	0.2	0.05	0.50	0.25
H*(10)/*H	1	1.1	1.4	1.1

Table 6-14. Recommended IREP neutron energy fractions and correction factors [32].

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
Waste characterization	RWMC SWEPP 14-MeV neutron generator	~1980	2003	0.1-2 MeV	5	1.4	0.07
				2-20 MeV	95	1.1	1.05
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	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

6.5 MISSED DOSE

Missed dose for site workers would occur when a zero dose was recorded for the dosimeter systems for any response less than the site dose recording threshold (the LOD). In the 1975–1985 dosimetry records, when the electron, penetrating, and neutron doses are all reported as "000001" mrem, the worker was not in the area and no missed doses should be assigned for those dosimeters.

6.5.1 Dosimeter Not Worn

Workers have reported that, on some occasions, they did not wear dosimeters while working in radiation areas (Wages et al. 1998; ORAUT 2004). The latest revision of *Use of Coworker Dosimetry Data for External Dose Assignment* (ORAUT 2008a) provides guidance to assist dose reconstructors in the evaluation of the support for a claimant's allegation that he or she did not wear a dosimetry badge at all times.

6.5.2 Missed Photon Dose

Missed photon dose for workers would occur when a zero dose was recorded for the dosimeter systems for any response less than the site dose recording threshold (the LOD). The missed dose for dosimeter results less than the LOD is particularly important for earlier years when LODs were higher and dosimeter exchange was more frequent [33]. The missed dose is calculated as described in NIOSH (2007) using LOD/2 multiplied by the number of zero dose results. Table 6-15 lists the potential missed photon doses by year, dosimeter type, and badge exchange frequency. The LODs shown are based on Cipperley (1958, 1968) and Cusimano (1963) for film; Kalbeitzer (1983), Gesell (1986), Gesell, Hall, and Anderson (1992), and Perry, Anderson, and Ruhter (1993) for TLDs; and Ruhter and Perry (2002) for film and TLD. The exchange frequency must be determined from the individual worker's dose submittal package for each year because it was shorter for highly exposed individuals and longer for those with lower doses.

Table 6-15. Electron/photon dosimeter period of use, type, LOD, exchange frequency, and potential annual missed dose [34].

Period of use[a]	Dosimeter	Exchange frequency	LOD[b] (mrem)		Annual missed dose (mrem)[c]	
			Photon	Electron	Photon	Electron
August 1951–March 1958	INL initial film, 552 DuPont film	Weekly (n=52)	30	30	780	780
		Monthly (n=12)			180	180
	Reactor areas, DuPont 558	Weekly (n=52)	10	30	260	780

	film					
March 1958–December 1966	INL multielement DuPont 508 film	Weekly (n=52)	10	30	260	780
		Biweekly (n=26)			130	390
		Monthly (n=12)			60	180
December 1966–February 1974	INL multielement DuPont 508 film	Weekly (n=52)	10	30	260	780
		Biweekly (n=26)			130	390
		Monthly (n=12)			60	180
	INL LiF TLD	Quarterly (n=4)	15	15	30	30
		Semi-ann (n=2)			15	15
		Annual (n=1)			7.5	7.5
February 1974–May 1975[d]	INL Atlas TLD LiF in Teflon	Monthly (n=12)	30	30	180	180
		Quarterly (n=4)			60	60
		Semi-ann(n=2)			30	30
		Annual (n=1)			15	15
December 1974-December 1985[d]	INL Harshaw two-chip TLD	Monthly (n=12)	15	15	90	90
		Quarterly (n=4)			30	30
		Annual (n=1)			7.5	7.5
January 1986–2006	INL Panasonic four-chip TLD	Monthly (n=12)	15[e]	15[f]	90	90
		Quarterly (n=4)			30	30
		Monthly (n=12)	10[e]	30[f]	60	180
		Quarterly (n=4)			20	60

a. For many years, INL workers had a dosimeter assigned to each operating area where they worked, or they were issued visitor dosimetry. All area dosimetry was issued beginning in January 2000.
b. LODs are based on Cipperley (1958, 1968), Cusimano (1963), Kalbeitzer (1983), Gesell (1986), Gesell, Hall and Anderson (1992), Perry, Anderson, and Ruhter (1993), and Ruhter and Perry (2002).
c. Maximum annual missed dose calculated using n × LOD/2 from NIOSH (2007).
d. ICPP began using the Harshaw TLD in December 1974, the prime contractor began in February 1975, and ANL-W began in May 1975.
e. The LOD was 15 mrem from January 1, 1986, to July 7 1986; 10 mrem from July 7, 1986, to September 1989; and 15 mrem until 1993, when it returned to 10 mrem.
f. The LOD was 15 mrem from January 1, 1986, to July 7 1986, and 30 mrem after that.

6.5.3 Missed Electron Dose

Non-penetrating dose is important for certain cancers (e.g. skin, breast, testes, etc…). Because the non-penetrating and penetrating doses were measured by the same dosimeter, because the dosimeters' LODs for electron doses are sometimes higher than the LODs for photon doses, and because the dosimeter correction factors that only get applied to electron doses are sometimes significant, special instructions are needed for assigning the missed doses for cancer locations affected by non-penetrating radiation. The following are the special instructions for each situation that may be encountered for the affected cancer locations.

1. When the reported non-penetrating result for a dosimeter is less than its electron LOD/2 value and the corresponding penetrating result is greater than its photon LOD/2 value, a missed dose is assigned as a >15 keV electron dose using the applicable parameters for electron doses.

2. For instances when the non-penetrating and penetrating doses for a dosimeter are both below their respective electron and photon LOD/2 values, the missed dose is calculated as an electron dose using the applicable parameters for electron doses, but is assigned as a more favorable to claimant 30-250 keV photon dose.

3. When the reported non-penetrating result for a dosimeter is greater than its electron LOD/2 value and the corresponding penetrating result is less than its photon LOD/2 value, a missed dose is assigned as a 30-250 keV photon dose using the applicable parameters for photon doses.

6.5.4 Missed Neutron Dose

As previously indicated, only workers that had a potential to receive more than an incidental neutron dose were the workers that were monitored for neutron dose and had their neutron dosimeters read. The only exception to this might have been the workers that worked around the fuel storage casks on a storage pad at TAN during the period of 1986-2006. However, those workers were unmonitored and should only have unmonitored neutron doses assessed. Therefore, missed neutron doses should only be assessed for the workers with reported neutron dosimeter results using the Table 6-16 information and any applicable dosimeter adjustment factors.

Table 6-16. Neutron dosimeter type, period of use, exchange frequency, laboratory minimum detectable limit, and maximum annual missed dose [41].

Dosimeter	Period	Exchange frequency	Laboratory LOD (mrem)	Maximum 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 1976–2006	Biweekly	15	195
		Monthly	15	90
		Quarterly	15	30

Because of the use of placeholder zeros in the 1975–1985 dosimetry records, determining a worker's neutron exposure potential can be difficult; the following guidance should be used for this period.

1. If a worker had a reported neutron dose greater than zero during the period of 1975–1985 for a given dosimeter location code, measured and missed neutron doses should be assessed for all dosimeter results during the 1975–1985 period with that location code.

2. If all of a worker's reported neutron doses for the 1975–1985 period were zero, but neutron monitoring can be confirmed for a given dosimeter location code during the years immediately prior to 1975 or immediately after 1985, missed neutron doses should be assessed for all dosimeter results during the 1975–1985 period with that location code.

3. For any situations outside of those described in 1) and 2), a worker's potential to receive neutron dose was low and no missed neutron doses should be assessed.

6.5.4.1 Before October 1976

The use of NTA films for neutron dosimetry before 1976 is documented in INL reports (Cusimano 1963; Cipperley 1958, 1968). As noted above, it is possible to estimate the missed dose using the LODs. There are many recorded zeros in the neutron dose data for INL workers for two reasons: (1) An NTA film was developed and not read in accordance with the standard criteria, or (2) an NTA film indicated a neutron dose equivalent that was less than the film's 14-mrem LOD [45]. When the LOD for NTA film is used to estimate the missed neutron dose, it should be multiplied by 1.25 for most workers and by 2 for workers on the MTR experiment floor and on the TREAT or ZPPR experiment floor [46].

An estimate of the missed neutron dose in some facilities might appear to be attainable through use of neutron-to-photon dose ratios (NIOSH 2007). However, for other site facilities there are several other sources of gamma exposure with no associated neutron exposure, so that approach would be erroneous.

6.5.4.2 October 1976 to Present

Since and including October 1976, the neutron dose has been measured using the Hankins albedo TLD. The characteristics of this dosimeter are well documented (Gesell et al. 1996), and the LOD to be used to estimate missed dose is 15 mrem. A location-specific FNCF has been applied to convert the reading to dose equivalent, so no additional adjustments should be required (Gesell et al. 1996).

6.6 ORGAN DOSE

Once the $H_p(10)$ adjusted doses have been calculated for each year, the values are used to calculate organ doses of interest using the external dose reconstruction guideline (NIOSH 2007). For photons before 1981, the conversion factor from exposure to organ dose should be used [47]. For 1981 and after, the conversion factor from deep dose equivalent to organ dose should be used [48].

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. INL used ionization chambers standardized by NBS for its calibrations. Use of a phantom for dosimeter irradiation began in the early 1980s, but backscatter causes only a minor change for high-energy photon dosimetry [49]. The over-response of the multielement film badge to deep dose in tissue is due to calibration to exposure, which is somewhat greater at low energies than for the deep dose [50]. The INL environment did not have a significant low-energy photon field such as a plutonium finishing plant, so the nonpenetrating component was attributed to electron radiation. A realistic estimate of total uncertainty for photon dosimetry is ± 35% at 1 sigma [51]. This is roughly consistent with the results in a study that was performed on the dosimeters in 1998 (Reilly 1998). In that study, the standardization instrument contributed some significant uncertainty.

The uncertainty for beta radiation is somewhat larger at an estimated ± 50% at 1 sigma [52]. This is driven by uncertainties in the field geometry and because beta radiation is often stopped by air and thin materials such as clothing. 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 [53]. The difference between point- and planar-source irradiation can confuse an algorithm [54]. Earlier techniques did not provide a thin detector with minimal covering, which is important for simulating the skin for beta dosimetry.

For neutron radiations, the situation is more complex. The NTA films in use before 1975 did not react to low-energy neutrons below 0.5 to 0.8 MeV (Cusimano 1963; Cipperley 1968). Corrections are described for handling this issue. The TLD albedo system provides a very indirect way of measuring dose equivalent to a person. Dose to workers is primarily due to hydrogen recoils rather than the ^{6}Li(n,α) reactions [55]. The response of the 9-in. PNR-4 detector used to standardize the TLD measurements is also due to a different process than dose deposition in the human body. The total uncertainty for neutrons is probably larger at 60% at one sigma.

The cause of the greatest uncertainty for neutrons is the variation of dose caused by organ positions in the body. For 1-MeV neutrons, the dose facing the source is a factor of 1,000 higher than the dose on the back side of a 30-cm-diameter sphere of tissue-equivalent material [56]. In a work environment, the primary direction of the neutrons might 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, it is assumed in EEOICPA that the worker irradiation is in an AP geometry (from the front) [57]. In Figure 6-18 the discrepancy in which the dosimeters report about one-half of the spectrometer result is because the spectrometer does not simulate the attenuation of the body, so it reads high by a factor of two.

6.8 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in the preceding text, bracketed callouts have been inserted to indicate information, conclusions, and recommendations to assist in the process of worker dose reconstruction. These callouts are listed in this section with information that identifies the source and justification for each item. Conventional references are provided in the next section that link data, quotations, and other information to documents available for review on the Oak Ridge Associated Universities (ORAU) Team servers.

Norm Rohrig served as the initial Document Owner of this document that was published in 2004 and revised in 2006. Because Mr. Rohrig was previously employed at the INL site, where his work involved management, direction, or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site, the document was reviewed and revised by a new Document Owner in 2007. The 2007 revision was made in full accordance with the "NIOSH Policy Statement: Management of Conflict or Bias in the Radiation Dose Reconstruction Program." The new Document Owner was fully responsible for the content of the revised document, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied on by the new Document Owner, those materials are fully attributed to the source. Mr. Rohrig served no role in this latest revision.

[1] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Because gammas interact primarily with electrons and they are only weakly attenuated, different definitions of dose and different measurement techniques have little effect.

[2] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This statement generally applied to high-level supervisors such as DOE officials and company managers who occasionally came to the site for meetings.

[3] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is a requirement from ORAUT-PROC-0031, *Site Profile and TBD Development* (ORAUT 2007), and uses values based on experience and from the Hanford Site Profile (ORAUT 2010c).

[4] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is adapted to INL from the Hanford Site Profile (ORAUT 2010c) and uses values based on experience at INL.

[5] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The raw photon spectrum of the expected nuclide fission product mixes was considered, and the dose fraction in the 30- to 250-keV band was 1.7% to 5.1% of the total dose. Monte Carlo calculations were performed using a code provided by S. Cohen & Associates, and the dose contribution from 0- to 250-keV photons through a concrete wall was 0.4% to 7.4%. Combining these effects shows that the 25%-75% split is appropriate.

[6] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Photon dosimetry for higher energy photons is usually straightforward because of the minimal attenuation and because the primary interaction is by the Compton process with electrons.

[7] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
See the Site Description TBD (ORAUT 2010b).

[8] Rohrig, Norman D. ORAU Team. Health Physicist. Date.
The contact radiation field on these devices is less than 0.5 mrem/hr (Rohrig 2006a).

[9] Rohrig, Norman D. ORAU Team. Health Physicist. November 2006.
Because of the energy dependence of photon cross-sections, low-energy photons are absorbed more strongly than high-energy photons, so spectral hardening (reduction of low-energy photons) results.

[10] Rohrig, Norman. ORAU Team. Health Physicist. June-October 2003.
Radiographer exposure is generally to the radiography sources identified and to radiation surrounding the items being radiographed.

[11] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is an EEOICPA requirement and uses values based on experience and cited in the text. See item [5] above.

[12] Rohrig, Norman. ORAU Team. Health Physicist. June-October 2003.
This statement follows from the fact that beta particles are directly ionizing and thus have a limited range. For there to be any dose, they must be lightly shielded.

[13] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Retrospectively, there is no simple way of knowing the beta emitters that caused employee X's exposure in week Y, so this is a generic approach to the issue.

[14] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
As discussed in the paragraph above, the end-point energy distribution of beta-emitting nuclides was converted to a distribution of ranges and then used to generate this plot.

[15] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This choice of angles and energies was used to generate a simplified depth-to-dose curve.

[16] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This correction factor is unlikely to have been applied during calibration. The net correction is the ratio of corrections for the calibration spectrum and the field spectrum, so the net effect is smaller than the corrections in Table 6-10.

[17] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Dosimeter thicknesses have been stated in previous paragraphs. The correction factors are derived based on the described theoretical approach.

[18] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is based on information in the Site Description TBD (ORAUT 2010b) and work experience at the facility.

[19] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This plot was generated from cited spectra (Ing and Makra 1978; Kluge and Weiss 1982).

[20] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This equation is a direct result of using four energy groups for the neutrons and applying mean values as determined in ORAUT-OTIB-0009 for the different ratios (ORAUT 2006a).

[21] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The fractional error on gamma-to-fast-neutron ratio (83%) dominates that from the errors that are stated in the second paragraph of Section 6.3.4.4 of 1.8%, 4.1%, and 3.3%.

[22] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Based on the judgment of Mr. Rohrig, using the gamma dose to estimate the neutron dose is not recommended because the two fields might not be directly related to each other; they often have different sources. Because the gamma and neutron fields do not have a common source, there is no reason to believe the ratios from one facility would apply at another facility.

[23] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Using photon-to-neutron ratios to derive neutron doses assumes that valid photon data are considerably more likely to exist than valid neutron data. This does not seem to be the case on examining dose records.

[24] Rohrig, Norman D. ORAU Team. Health Physicist. February 20, 2007.
Based on conversation on October 12, 2006, with a former health physicist with knowledge of TAN and INL (Rohrig 2006b).

[25] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This subjective statement is based on the experience of Mr. Rohrig and knowledge of personnel neutron dosimetry; the performance of personnel neutron dosimeters is significantly dependent on the similarity between the calibration and workplace radiation field.

[26] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This standard TBD table is based on information in the Hanford TBD (ORAUT 2010b) and, in the judgment of Mr. Rohrig, performance of INL dosimeters for similar parameters of performance.

[27] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on guidance in ORAUT-OTIB-0055 (ORAUT 2006b).

[28] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
For neutrons, shielding by a human is significant (factor of 1,000 for 1-MeV neutrons), and dosimeters therefore need to face the radiation source to read reliably.

[29] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on guidance in ORAUT-OTIB-0055 (ORAUT 2006b).

[30] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table was derived with spectral data in a spreadsheet as shown in Figures 6-15 and 6-16 with the associated energy-dependent quality and dose effectiveness factors. The last four rows are averages of the associated calculated values higher in the table.

[31] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The values in Table 6-13 are different, namely 1.08 and 1.46 or 1.47.

[32] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is derived from the values in Table 6-13.

[33] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The calculated missed dose, based on OCAS-IG-001 guidance (NIOSH 2007) and described in the succeeding text, is based on the number of dosimeter exchanges multiplied by the minimum detection level divided by 2 for all reported results less than the minimum detection level divided by 2.

[34] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table uses values in columns 3 to 5 that are previously cited in this TBD.

[35] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This statement is made based on studying dose record sheets and conferring with INL dosimetry staff.

[36] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This judgment is based on reviews of historical INL dosimetry information and Watson et al. (1994).

[37] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This statement is based on an understanding of the type and variety of work at INL.

[38] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This statement is based on guidance provided in ORAUT-OTIB-0023, *Assignment of Missed Neutron Doses Based on Dosimeter Records* (ORAUT 2008b).

[39] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This judgment is based on reviews of historical INL dosimetry information.

[40] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
These values are based on weighting neutron spectra with dose conversion factors to determine the fraction of the dose below 0.8 MeV.

[41] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The values in Table 6-15 are required to calculate the missed dose per ORAUT-PROC-0031 (ORAUT 2007) and are based on the judgment of Mr. Rohrig.

[42] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This reflects the period discussed in Section 6.3.2.7.

[43] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This reflects the period discussed in Section 6.3.2.8.

[44] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
A perfect calibration source would have all its radiation detected by the detector being calibrated. If most of the radiation is detected, the adjustment would be small and would not contribute a major uncertainty.

[45] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
By definition of the LOD, values less than the LOD are reported as zero (practice at INL) or a blank.

[46] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
These values are based on weighting neutron spectra with dose conversion factors to determine the fraction of the dose below 0.8 MeV.

[47] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on the common practice of wearing the dosimeter on the front of the body and often facing the radiation source during work. Early program guidance recommended exclusive use of the AP geometry.

[48] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This recommendation is made because the measurement quantity of dose in that period was deep dose equivalent.

[49] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
For high energy, the angular distribution is primarily forward. See for example Figure 2-19 of Knoll (1989).

[50] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
At low energies, air that has very little hydrogen in comparison to tissue has a higher cross-section than tissue because of the Z dependence of the photoelectric effect.

[51] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This estimate is based partially on Figure 6-12 and partially on uncertainties of many error processes that contribute uncertainty.

[52] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This estimate is based partially on Figure 6-12 and partially on uncertainties of many error processes that contribute uncertainty.

[53] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
These estimated corrections are provided in Table 6-12, and they are not small.

[54] Rohrig, Norman. ORAU Team. Health Physicist. June-October 2003.
For this reason, DOELAP has had different categories for point and slab beta geometries.

[55] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Lithium is only a trace constituent of the human body. Dose from neutron radiation to people is primarily due to hydrogen atom recoils in tissue rather than the $^{6}Li(n,\alpha)$ neutron absorption reactions of the neutron dosimeter.

[56] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
See Figure 16 of NCRP Publication 38 (NCRP 1971).

[57] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on the common practice of wearing the dosimeter on the front of the body and often facing the radiation source during work. It is understood that early program guidance recommended exclusive use of the AP geometry.

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GLOSSARY

1/E spectrum
Typical neutron spectrum for neutrons emitted during prompt fissioning. Spectrum defined by the number of neutrons in an energy interval being proportional to the width of the energy interval divided by the average energy of the neutrons in that interval.

albedo dosimeter
Thermoluminescent dosimeter that measures the thermal, intermediate, and fast neutrons scattered and moderated by the body or a phantom from an incident fast neutron flux.

alpha particle (α)
See *alpha radiation*.

alpha radiation
Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2.

anterior–posterior (AP)
Physical orientation of the body relative to a penetrating directional radiation such that the radiation passes through the body from the front to the back.

backscatter
Reflection or refraction of radiation at angles over 90 degrees from its original direction.

beta particle (β)
See *beta radiation*.

beta radiation
Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

deep dose equivalent index
Dose equivalent in units of rem or sievert for a 1-centimeter depth in tissue (1,000 milligrams per square centimeter). See *dose*.

dose
In general, the specific amount of energy from ionizing radiation that is absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays.

dose equivalent (*H*)
In units of rem or sievert, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

dose equivalent index
Historical measure for neutron source calibration defined by the International Commission on Radiation Units and Measurements as the sum of the maximum dose equivalents delivered within a sphere at any depth for the respective neutron energies even though the maximum

dose occurred at different depths and discounting the outer 0.07-millimeter-thick shell. Also called unrestricted dose equivalent index.

DOE Laboratory Accreditation Program (DOELAP)
Program for accreditation by DOE of DOE site personnel dosimetry and radiobioassay programs based on performance testing and the evaluation of associated quality assurance, records, and calibration programs.

dosimeter
Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual. See *albedo dosimeter*, *film dosimeter*, *neutron film dosimeter*, *pocket ionization chamber*, and *thermoluminescent dosimeter*.

effective dose equivalent (H_E)
Average of the dose equivalents weighted for the susceptibility of harm to different tissues or organs in units of rem or sievert. See *dose*.

exposure
In general, the act of being exposed to ionizing radiation.

exposure-to-dose-equivalent conversion factor for photons (*Cx*)
Ratio of exposure in air to the dose equivalent at a specified depth in a material of specified geometry and composition. C_x 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 without trace elements, or soft tissue with trace elements).

film dosimeter
Package of film for measurement of ionizing radiation exposure for personnel monitoring purposes. A film dosimeter can contain two or three films of different sensitivities, and it can contain one or more filters that shield parts of the film from certain types of radiation. When developed, the film has an image caused by radiation measurable with an optical densitometer. Also called film badge.

gamma radiation
Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

gamma ray, particle, or photon (γ)
See *gamma radiation*.

linear energy transfer (LET)
Rate of energy absorption by media from particulate or electromagnetic radiation.

neutron (n)
Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

neutron film dosimeter
Film dosimeter with a nuclear track emulsion, type A, film packet.

neutron radiation

Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons. See *neutron*.

nonpenetrating dose

Dose from beta and lower energy photon (X-ray and gamma) radiation which does not penetrate the skin. It is often determined from the open window dose minus the shielded window dose. See *dose*.

nuclear track emulsion, type A (NTA)

Film sensitive to fast neutrons made by Eastman Kodak. The developed image has tracks caused by neutrons that become visible under oil immersion with about 1,000-power magnification.

occupational dose

Internal and external ionizing radiation dose from exposure during employment. Occupational dose does not include that from background radiation or medical diagnostics, research, or treatment, but does include dose from occupationally required radiographic examinations that were part of medical screening.

open window (OW)

Area of a film dosimeter that has little to no radiation shielding (e.g., only a holder and visible light protection). See *film dosimeter*.

pencil dosimeters

See *pocket ionization chamber*.

penetrating dose equivalent

Photon dose measured by shielded INL film or elements plus neutron dose equivalent. Essentially, personal dose equivalent $H_p(10)$.

personal dose equivalent, $H_p(d)$

Dose equivalent in units of rem or sievert in soft tissue below a specified point on the body at an appropriate depth *d*. The depths selected for personal dosimetry are 0.07 millimeters (7 milligrams per square centimeter) and 10 millimeters (1,000 milligrams per square centimeter), respectively, for the skin (shallow) and whole-body (deep) doses. These are noted as $H_p(0.07)$ and $H_p(10)$, respectively. The International Commission on Radiological Measurement and Units recommended $H_p(d)$ in 1993 as dose quantity for radiological protection.

photon radiation

Electromagnetic radiation that consists of quanta of energy (photons) from radiofrequency waves to gamma rays.

pocket ionization chamber (PIC)

Cylindrical monitoring device commonly clipped to the outer clothing of an individual to measure ionizing radiation. A PIC may be self-reading or require the use of a outside device to be able to read the dosimeter. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

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

range
Distance an energetic charged particle will go through a material before it stops. Range is an increasing function of energy and depends on the elemental makeup of the material and the density.

redact
To put into suitable form for publication by such processes as editing and revision.

shallow absorbed dose (D_s)
Absorbed dose at a depth of 0.07 millimeters (7 milligrams per square centimeter) in a material of specified geometry and composition.

shallow dose equivalent (H_s)
Dose equivalent in units of rem or sievert at a depth of 0.07 millimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

thermoluminescent dosimeter (TLD)
Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated by radiation, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

tissue rad
Absorbed dose in tissue.

X-ray
See *X-ray radiation*.

X-ray radiation
Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.