



ORAU TEAM Dose Reconstruction Project for NIOSH

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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.

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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
CFA	Central Facilities Area
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
CPP	Chemical Processing Plant
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
FNCF	facility neutron correction factor
g	gram
GSD	geometric standard deviation
$H^*(10)$	ambient dose equivalent
$H_p(10)$	personal dose equivalent at 10 mm depth in tissue
$H_p(d)$	personal dose equivalent
$H_{p,slab}(d)$	personal dose equivalent (slab phantom)
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
INEL	Idaho National Engineering Laboratory
INTEC	Idaho Nuclear Technology and Engineering 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

MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
mm	millimeter
mR	milliroentgen
mrad	millirad
mrem	millirem
mrep	millirep
MRL	minimum reporting level
mSv	millisievert
MTR	Materials Test Reactor
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
PBF	Power Burst Facility
PEQ	Personnel Exposure Questionnaire
PER	Program Evaluation Report
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
TAN	Test Area North
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor and Experiment Test
TSF	Tower Shielding Facility
U.S.C.	United States Code
WERF	Waste Experimental Reduction Facility
wk	week

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) define “performance of duty” for DOE employees with a covered cancer or restrict the “duty” to nuclear weapons work.

As noted above, the statute 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 contains an exclusion with respect to the Naval Nuclear Propulsion Program, 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 dosimetry monitoring results are considered valid for use in 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:

- Radiation from naturally occurring radon present in conventional structures
- Radiation from diagnostic X-rays received in the treatment of work-related injuries

¹ The U.S. Department of Labor 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 (INEL) and the Idaho National Engineering and Environmental Laboratory (INEEL). For consistency, this document uses INL except in special cases.

6.1.2 Scope

From the start of operations in 1951 (when it was NRTS), a branch of the U.S. Atomic Energy Commission (AEC) Idaho Operations Office provided external dosimetry resources and services at INL 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 DOE changed contractors often, the external dosimetry process has remained under technical management of a single organization with responsibilities for dosimetry development, operational dosimetry, and radiological records; thereby providing 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

The Interactive RadioEpidemiological Program (IREP) calculates the probability of cancer induction in an organ from the external equivalent dose and internal dose received by that organ. Appendix B of the *External Dose Reconstruction Implementation Guideline* (NIOSH 2006) provides conversions from four photon dose quantities [deep dose equivalent $H_p(10)$, ambient dose equivalent $H^*(10)$, exposure X , and air kerma K_a] and three neutron quantities [fluence ϕ , ambient dose equivalent $H^*(10)$, and deep dose equivalent, $H_{p,slab}(10)$] to the organ doses. Over the years, the National Council on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), and their predecessor organizations have developed the definitions of dosimetry parameters. Dose parameters measured by the INL dosimetry system have received further definition. INL has reported doses as penetrating and nonpenetrating. The penetrating dose corresponds to the deep dose equivalent, and the sum of the nonpenetrating and penetrating doses corresponds to the shallow dose equivalent.

Horan and Braun (1993), Attix and Roesch (1968, p. 6-18), and Meinhold (1975) discuss the history of radiation protection requirements from the 1930s. On September 29 and 30, 1949, representatives of the United Kingdom, Canada, and the United States met in Chalk River, Canada, at a Permissible Doses Conference (Warren et al. 1949) and picked the blood-forming organs as critical for hard X- or gamma rays at a depth of 5 cm. Skin is the critical organ for soft X-rays or beta radiation. Dosage was to be reported in units of rep (*rep* is from *roentgen-equivalent-physical*; 1 rep is equivalent to 93 ergs absorbed per gram of tissue). The maximum permissible dose to the bone marrow was 0.3 rep/wk.

In 1949, the newly formed National Committee on Radiation Protection (now the National Council on Radiation Protection and Measurement) issued NCRP Report 7 through the National Bureau of Standards as Handbook 42 (NBS 1949, p. 6), which recommended a permissible dose of 0.3 R/wk (15 R/yr) for occupational workers (NBS is now the National Institute of Standards and Technology). The term *dose* was undefined. A roentgen was defined as the quantity or dose of X-rays such that the associated ionization per 0.001293 g of air (1 cm^3 at standard temperature and pressure) produces 1 electrostatic unit of charge of either sign. The unit roentgen only applies to X-rays or

gamma rays, whereas the rep was used for all ionizing radiation and is the energy absorbed by tissue from the radiation (93 ergs/g, the same as for 1 R of gamma radiation).

An April 1952 site manual stated the limit as “0.3 rep/wk at an effective depth in soft tissue of 5 cm, assumed to be the depth of the blood forming organs” (ACC 1952, p. IV: 1-1). The manual does not mention a quarterly or an annual limit.

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

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

The quarterly limit of 3 rem or 12 rem/yr replaced the 15 R/yr that was associated with the weekly limit (NBS 1958, p. 3, footnote 2). President Eisenhower approved these values in 1960 for Federal agencies. AEC Manual Chapter 0524 was reissued in 1963 and 1968 (AEC 1963, 1968) and twice later as U.S. Energy Research and Development Administration (ERDA) Manual Chapter 0524 (ERDA 1975, 1977), which provided requirements for radiation safety.

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

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

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

conversion factors in ICRP Publication 21 and the associated interpolation with energy do not present that problem (ICRP 1973).

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

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

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

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

In 1990, the ICRP redefined the concept of dose equivalent to *equivalent dose*, redefined quality factor to *radiation weighting factor*, and generated new factors (ICRP 1991). These factors, which are invoked by NIOSH (2006), depend on neutron energy at the entrance to the body rather than on secondary particle LET where the dose is received. Dose conversion factors for organs and for ambient dose equivalent and personal dose equivalent were generated in ICRP Publication 74 (ICRP 1996) and are referenced in the external dose implementation guide (NIOSH 2006).

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

INL policy was 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 for INL facilities. Control badges, which are used to subtract background radiation, have also been located at the gates (Ruhter 2003). This practice might have led to the subtraction of environmental radiation from site activities, which would have reduced the reported doses (ORAUT 2003). Environmental radiation levels have been monitored for most of the life of INL using film badges first and thermoluminescent dosimeters (TLDs) later. Table 4-13 in the INL environmental dose TBD (ORAUT 2004a) lists the results of this monitoring at facility fence-line 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 workers working at the site.

Some individuals who might have 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. During the early years at INL, 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, and the frequency can be determined from the worker'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 the dosimetry information for former INL workers, which should include a dose summary for the employment period and a copy of each weekly, monthly, quarterly, etc. form with the work location, so the individual file could be several inches thick in hard copy. Each sheet is redacted so only the person of interest's name and applicable information are visible. This file provides the recorded information about the exchange period for the person for that period. The following five figures are a partial example set of redacted dose reporting forms.

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). 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 6-2 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.

WEEK BEGINNING SEPT. 26, 1954

SU	M	T	W	TH	F	S	S	F	B	G	G	RING	NS	I
25	27	28	29	30		1			0	0	0			
3	4	5	6	7	8	9	10	11	0	0	0			
10	11	12	13	14	15	16	17	18	0	0	0			
17	18	19	20	21	22	23	24	25	0	0	0			
24	25	26	27	28	29	30	31	1	0	0	0			
31	1	2	3	4	5	6	7	8	0	0	0			
7	8	9	10	11	12	13	14	15	150	0	0			

741 Withhold 9-30-54

BADGE #	NAME	EMPLOYEE	ABC	S #	B	G	G	RING	NS	I
14	15	16	17	18	19	20	21	22	23	24
24	25	26	27	28	29	30	31	1	2	3
28	29	30	1	2	3	4	5	6	7	8
5	6	7	8	9	10	11	12	13	14	15
12	13	14	15	16	17	18	19	20	21	22
19	20	21	22	23	24	25	26	27	28	29

Form IHP-18 WEEK ENDING DEC. 25, 1954 SUB TOTALS: 254

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

22-14-H 93. C93 5-22-28-58

Plant: CFA Site Survey Week Beginning

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

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

Figure 6-3. Monthly badge reporting form from May 1959 (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 Bldg	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	Naval Reactors Facility (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 Stationary Low-Power Reactor No. 1 (SL-1)
12	EX at EBR	31	Shield Test Pool Facility
13, 133, 135	SPERT, PBF	65	ECF
14	Organic-Moderated Reactor Experiment (OMRE)	66	Nonsecurity
15	SX at SPERT	67	Division of Compliance

Area code	Description	Area code	Description
16	SL-1	68	STEP
17, 333	MX at MTR	69	LPTF (Phillips & AEC)
18, 814,815	WP, RWMC	71	CADRE (guard force)
19, 772, 775	TAN (Phillips & AEC)	774,776	SMC

After the pencil ionization chambers were replaced with self-reading pencil dosimeters (also 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 reports badge reading for May 1959 when badges were exchanged every 2 weeks. 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.

Table 6-2. Reasons Codes (Column 68-69) (Hill-Ellis 2003).

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

Table 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

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 by the computer 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 (about 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 6-4 is a listing of some doses received during recovery from the SL-1 accident. Workers from several areas were pulled into the accident recovery process, and it is notable that one result exceeds the dose limits and that there are few zeros. Figure 6-5 is a follow-up badge report for one result on

				OFFICIAL USE ONLY	
NAME	CONTR.	AREA	BADGE NUMBER	BETA	GAMMA
	71	22			50
	07	17			365
	01	03			
	02	05			140
	73	00			1175
	02	16			345
	30	00		550	740
	01	03			35
	04	07		120	250
	71	22		210	205
	02	03			60
	67	16		120000	23195
	07	16			570
	07	04			
	01	03			65
	01	03		1475	2370
	01	01			30
	02	05			
	01	03		190	143
	01	03			15
	02	00			
	01	03			245
	01	03		135	230
	01	03		530	35
	01	00			
	04	16		120	130
	01	03			180
	02	04			20
	04	07			

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Figure 6-4. Badge results from January 1961 for work in recovery from the SL-1 accident (Vivian and Rockhold 2003, p. 50).

10-105 (2-55)

BADGE REPORT AEC
072

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

TO _____

AREA 261 Bldg. _____ Date _____

RE: _____ Bldg. No. _____

The badge on the above named employee recorded:

SEN		INS	
Beta	Gamma	Beta	Gamma
530	35		

The period extending from _____ through 1-31-61

Badge pulled for reason listed below:

High Pencil Readings of Taken from CIP bag pull

Damaged Pencil DO Special Report by

week 5

Signed _____

Figure 6-5. Special badge report associated with a high beta reading in Figure 6-4 (Vivian and Rockhold 2003, p. 57).

Figure 6-4. When there was a question about an assigned dose value, a *Personnel Exposure Questionnaire* (PEQ) was normally initiated as shown in Figure 6-6 (shows 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.

PERSONNEL EXPOSURE QUESTIONNAIRE

Date 1-5-58

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

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

Reason for Investigation:

A reportable weekly daily pocket meter reading total of _____

Weekly film total of 300 mr or more.

() _____

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

FILM RESULTS		EXPOSURE RESUME								
BETA	GAMMA	Week Ending	Meters	SUN.	MON.	TUES.	WED.	THURS.	FRI.	SAT.
500	350	<u>1-4-58</u>	Pocket Meters	-	20	40	60	90	80	-
			Badge Meters	-	-	-	-	-	-	B-500 G-350

Remarks Total 850 mrem

Investigation

a. Findings of Health Physics Representative and or Supervisor.

b. Recommendations.

Investigated by _____ Date _____ Noted _____

Health Physics Supervisor

Figure 6-6. Personnel exposure questionnaire partially completed for a hypothetical case (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 OW. 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 two reactor areas (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 about 30 mR, and DuPont 558 film has a threshold level of about 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 6-7, by about 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

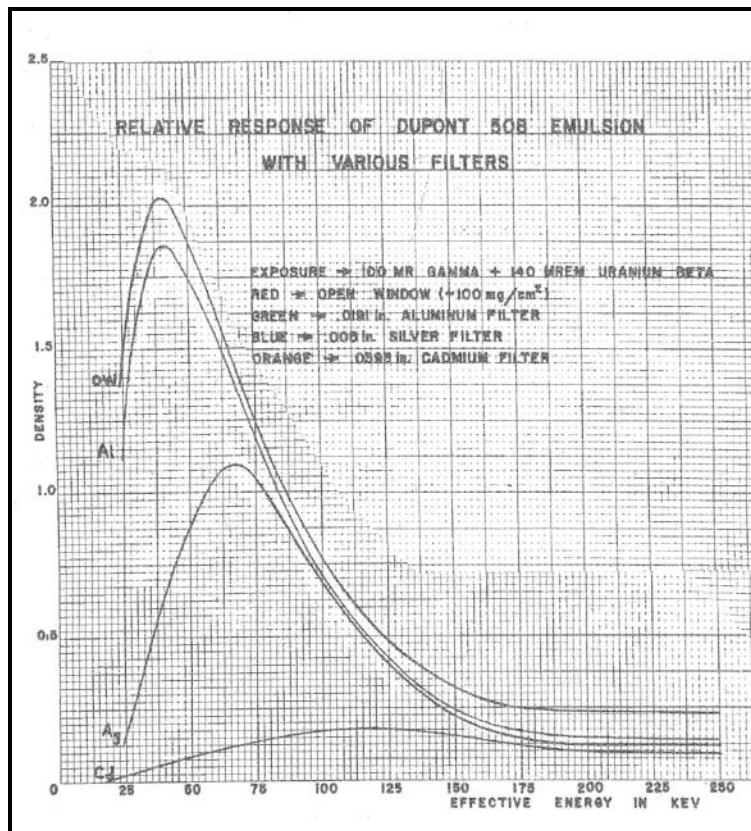


Figure 6-7. Response of DuPont 508 film with various filters to 140 mrem from uranium beta and 100 mR of different energy photon irradiation. The original badge used the OW and cadmium-shielded films. The multiple-filter badge used all three filters plus the OW (Cipperley and Gammill 1959).

high Z severely attenuates the photons that get to the film, so the over-response is reduced to about a factor of 2 at 125 keV and becomes less than 1 at about 50 keV. The beta particle range is independent of the atomic number Z (it depends only on the density) so the 1-mm cadmium filter ($\sim 900 \text{ mg/cm}^2$) 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.

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/cm², 203 mg/cm², and 175 mg/cm², 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/cm² 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 6-7 shows photon energy dependence of the darkening behind the four filters for a combination of uranium beta and X-ray irradiation provided by NBS (Cipperley and Gammill 1959). With DuPont 508 film, mixed exposures of radium gamma and uranium beta of 10, 20, and 30 mR or mrep were measurable within ± 12 mR with 95% confidence. A minimum reporting level (MRL) of 10 mrem was used for both beta and gamma radiation (Horan 1963).

The 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 other than uranium, the analysis had greater uncertainty.

Because about 95% of the weekly badge films had doses less than 30 mrem, in 1958 the badging interval was increased to biweekly or monthly with the exception of the high-dose areas at which 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 ¹³⁷Cs 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 (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/cm²) 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 TLA 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, date unknown), so more small doses were reported (Cusimano and Cipperley 1968). The angular dependence of the radium-gamma response (within 10° to 70°) 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, date unknown).

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 MRL of 30 mR was used for gamma and beta (Walker 1974).

6.3.2.5 Harshaw Two-Chip Thermoluminescent Dosimeter System

Several unstable characteristics 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 TAN, TRA, etc., and in May 1975 for Argonne National Laboratory–West (ANL-W). This commercial Harshaw system used two LiF TLDs that were 240 mg/cm² thick. In 1976, holes were punched in the security badges to restore the OW. One chip was covered by 540 mg/cm² (initially 350 mg/cm²) of aluminum and the other was under 4 mg/cm² 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/cm²) allowed higher energy beta radiation to expose the chip that was used for measuring the penetrating dose (at 1,000 mg/cm²), 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 (Kalbeitzner 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 (Li₂B₄O₇) 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 (CaSO₄) TLD provided a strong low-energy photon response. The elements were 15 mg/cm² thick. Element 1

had filtration of 16 mg/cm², element 2 had filtration of 58 mg/cm² of plastic, and elements 3 (CaSO₄) and 4 had filtration of 550 mg/cm² of plastic and 50 mg/cm² of aluminum, respectively. Although none of the elements was at a depth of 7 mg/cm² (the specified depth for the shallow dose equivalent), an acceptable response can be obtained by using elements at 16 and 58 mg/cm². This system is qualified in DOELAP for the beta, photon, and mixture performance categories. The angular dependence of this system has been measured for ¹³⁷Cs 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 MRL was 15 mrem beta and gamma from January to July 1986 (Gesell 1986), 10 mrem gamma and 30 mrem beta from July 1986 to about September 1989, and 15 mrem for gamma and 30 mrem for beta until 1993 (Perry, Andersen, and Ruhter 1993), when it returned to 10 mrem gamma.

6.3.2.7 Nuclear Track Emulsion, Type A for Neutrons

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

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 MRL 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 MRL 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 MRL. 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 MRL 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 (BF₃) 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 ^6Li in a TLD (Gesell et al. 1996).

In the Hankins dosimeter used at INL (Hankins 1973), TLDs with ^6Li 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 $^6\text{Li}(n,\alpha)^3\text{He}$ 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). An 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 MRL 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 FNCf's (Cusimano 1981).^a

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

a. ZPPR = Zero Power Plutonium (later Physics) Reactor.

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 that was developed in the 1960s (Cipperley 1966) did not use a phantom to provide backscatter.

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 (about 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 about 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 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.

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 ⁹⁰Sr/Y source to tissue rad. The source, with an area of 2.5 cm², 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).

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 about 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 ¹³⁷Cs 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 1981b). 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. The 1991 Tiger Team Review (DOE 1991) of the INL site indicated that use by the INL contractor and the Idaho Operations Office of the same sources for calibration led to a conflict of interest or an advantage in DOELAP tests. As a result, EG&G purchased a Shepherd panoramic irradiator with a ^{137}Cs source for badge irradiations. This irradiator 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).

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 $H_p(10)$.

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 $^9\text{Be}(\alpha,n)^{12}\text{C}$ reaction and generate a broad spectrum of neutrons up to about 11 MeV (mean energy about 5 MeV) as shown in Figure 6-8 (Kluge and Weiss 1982). The yield of the AmBe source should be only about 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/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-cm²/n for the maximum average dose equivalent. A dose equivalent of 1.5 rem required 3.6×10^7 n/cm² (Cusimano 1963), which corresponds to a dose conversion factor of 4.17×10^{-8} rem-cm²/n, so the recorded dose is about 11% high. Monte Carlo calculations for 5-MeV neutrons show a dose equivalent of about 4.2×10^{-8} rem-cm²/n averaged over the 0- to 2-cm shell on a 30-cm-diameter cylindrical phantom (NCRP 1971b). Use of the 50-Ci AmBe source continued until 1993.

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).

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.	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 about 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 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 of signal fade with time, heat, humidity, light, etc.	Recorded dose of record is likely too low ; however, this depends strongly upon when the calibration dosimeters 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 $H_p(10)$.

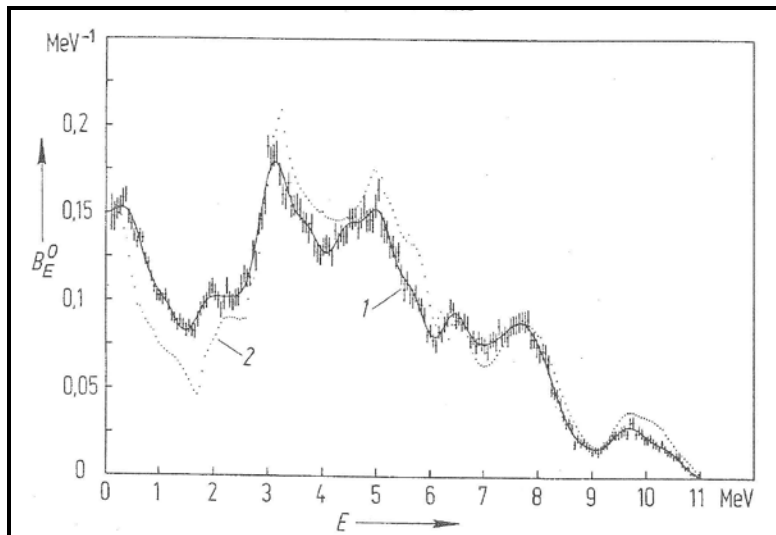


Figure 6-8. Probability density of neutron spectrum from a $^{241}\text{Am-Be}(\alpha,n)$ source (Kluge and Weiss 1982).

6.3.4 Workplace Radiation Fields

Radiation fields varied a great deal at INL as discussed in the Site Description TBD (ORAUT 2005a). 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

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
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

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 6-9 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.

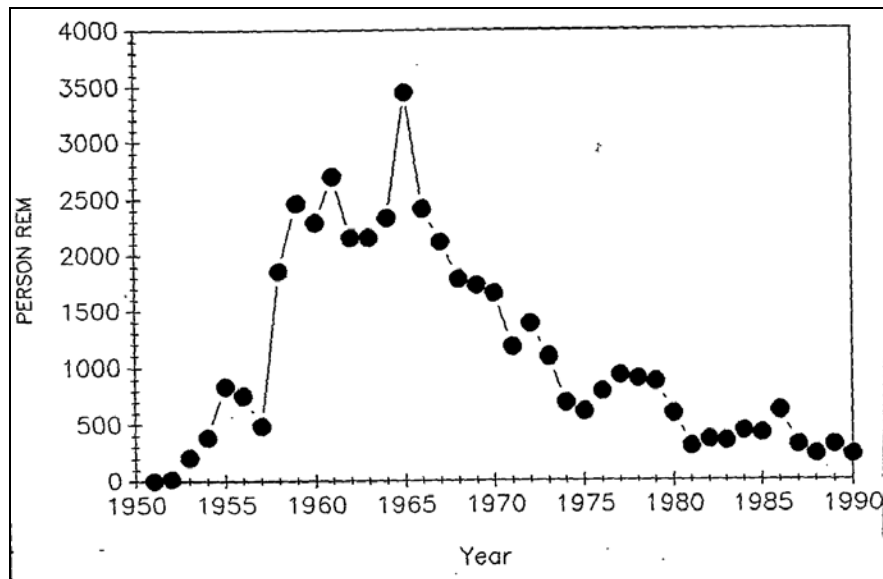


Figure 6-9. Collective dose for INL personnel, 1951 to 1990 (Horan and Braun 1993).

6.3.4.1 Gamma Radiation

In response to a Tiger Team finding (DOE 1991), radiation fields at INL have been characterized by comparing field measurements with a NaI(Tl) gamma spectrometer and TLDs mounted on a phantom (Reilly 1998). Figure 6-10 shows the percentage bias for the beta and gamma measurements. Most results lie between +27% and -43%. The high gamma-bias results are for locations at RWMC looking at skyshine (back-scattered, low-energy photons) from low-level waste in the Subsurface Disposal Area. The doses measured with NaI(Tl) were low (6 and 11 mrem), and the threshold energy on the NaI(Tl) detector was about 100 keV, so some low-energy photons were likely to have been missed. The radiation fields at INL, 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].

There are a few 250-keV X-ray generators 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 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 ¹⁹²Ir or 1.25-MeV ⁶⁰Co [10].

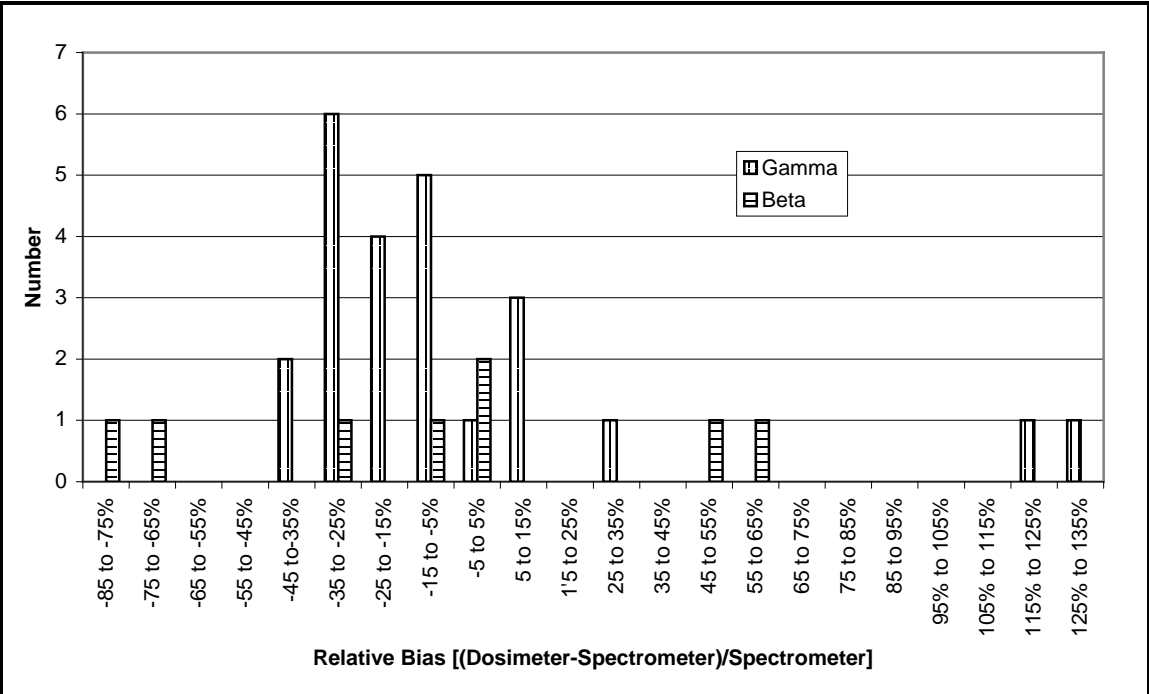


Figure 6-10. Gamma and beta radiation field characterization (Reilly 1998).

All INL 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 Beta Radiation

Beta 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 beta fields were not unusual at the Idaho Nuclear Technology and Engineering Center (INTEC; previously ICPP) where there are large quantities of fission products. Pure high-energy beta 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 beta fields would therefore be measured as gamma fields (Black 1974; Jenson 1976; INEL ca. 1978).

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

Beta field dosimetry became reasonably accurate with the definition of DOELAP requirements in the early 1980s. Before then, beta monitoring systems had various flaws, primarily in a detector too thick

to give a good surface result or one that was covered with extra material. Calibration was to high-energy betas from either uranium or strontium (see § 6.3.3.1). The dose from low-energy betas can be missed altogether if the beta energy is not sufficient to penetrate the detector cover, and the dose can be under-reported if the beta energy is not sufficient to penetrate the entire detector. The mean beta energy for the spectrum from a particular nuclide is about one-third of the maximum beta energy for that nuclide.

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

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 beta radiation PBF, TRA, ARA, TAN, EBR, ANL-W, SPERT			Beta	>15	100
				Photon	30-250	25
					>250	75
Processing plants	Highly dispersed fields of higher energy photon radiation fields from activation and fission product nuclides dominant to most exposure profiles. Potential for higher energy beta radiation during sampling and maintenance work resulting from fission products. ICPP			Beta	>15	100
				Photon	30-250	25
					>250	75
Calibrations	Calibration of instruments and dosimeters. CF 633, 636			Beta	>15	100
				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. RWMC, WERF			Beta	>15	100
				Photon	30-250	25
					>250	75
Uranium handling	Produced special armor from depleted uranium. Primarily beta radiation from U-238 daughters. Some gamma from contaminants and Cs-137 sources used in process. SMC			Beta	>15	100
				Photon	30-250	90
					>250	10

a. PBF = Power Burst Facility; SMC = Special Manufacturing Capability.

Based on the range vs. energy curve for beta particles and the beta energy distribution of beta emitters (BRH 1970, pp. 90, 91, 123), the fraction of beta radionuclides with ranges greater than the abscissa is plotted on the ordinate in Figure 6-11. Beta-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 beta 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 beta particles enter the detector at an angle, a curve is provided for 45° 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 beta 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 beta dose correctly at a depth of 7 mg/cm², which was chosen in the early 1980s. These curves demonstrate why the beta 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/cm², and for most persons at depths greater than 1 cm.

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/cm². The appropriate curve should be the curve of the mean range for the beta

spectrum and the angular distribution of the radiation exposure. To estimate this, this analysis added the mean energy curve for perpendicular incidence and 1.4 (relative path length) times that for 45° incidence for the mean energy and added one-half of that value for 45° 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.

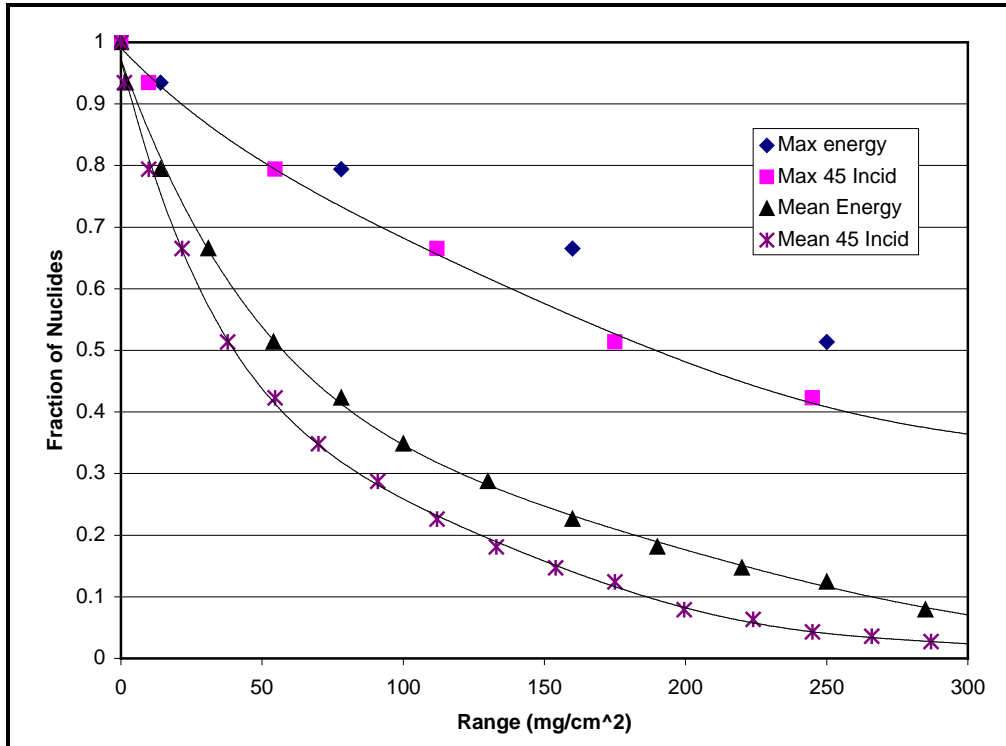


Figure 6-11. Distribution of beta ranges [14].

Table 6-10 provides the cover and detector thicknesses for the beta badges used at INL. Thicknesses with a “~” are estimated. The fraction of measured beta dose shown in Table 6-10 is the average as described above. To determine the corrected beta 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. Beta dosimeter thicknesses and associated underreporting [17].

Dosimeter system	Period	Covers (mg/cm ²)	Detector thickness (mg/cm ²)	Beta Correction Factor
Two-filter film	1951–1958	~50	~50	2.0
Multifilter film	1958–1974	100	~50	2.8
Low-dose TLD	1969–1974	100	75	3.0
ATLAS	1974–1975	100	~100	3.3
Harshaw TLD	1975–1976	104	240	4.8
Harshaw TLD	1976–1985	4	240	2.4
Panasonic TLD	1986–2006	16	15	1.0

6.3.4.3 Neutron Radiation

Most INL workers have not been exposed to neutrons and so have not been badged to measure neutrons. Neutron fields at INL have been specific to a few facilities. The high-dose locations where most of the gamma and beta 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].^a

Location	Period	Comments
Programmatic		
TREAT	1958–1994	
ZPPR	1969–1992	Between reactor halves
MTR	1952–1970	Research floor
TAN Warm Shop	1986–1988	See EPRI reports
TAN	1986–2006	Spent fuel storage pad
ANL 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

a. ILTSF = Intermediate-Level Transuranic Storage Facility; SWEPP = Stored Waste Examination Pilot Plant.

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 6-12 (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 is used 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 $[(\text{Dosimeter} - \text{Spectrometer})/\text{Spectrometer}]$ for the neutron fields are shown in Figure 6-13. 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 ²⁵²Cf 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 ²⁵²Cf instrument calibration source at a 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).

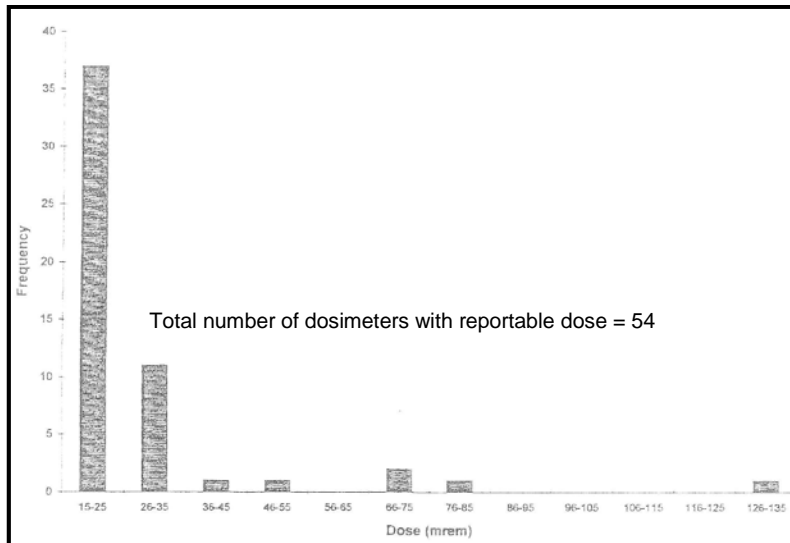


Figure 6-12. Distribution of reportable neutron dose for the first 9 months of 1995. Of 1,461 dosimeters, 1,407 were below the 15-mrem reporting level (Gesell et al. 1996).

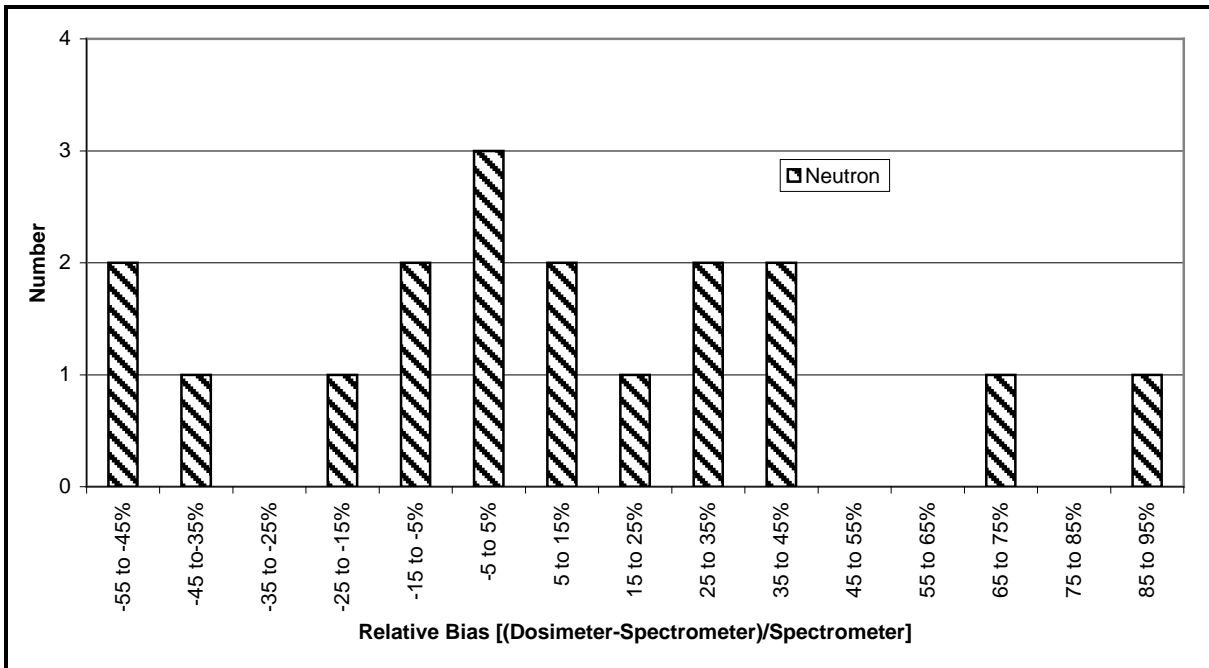


Figure 6-13. Neutron radiation field characterization (Reilly 1998).

Figure 6-14 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.

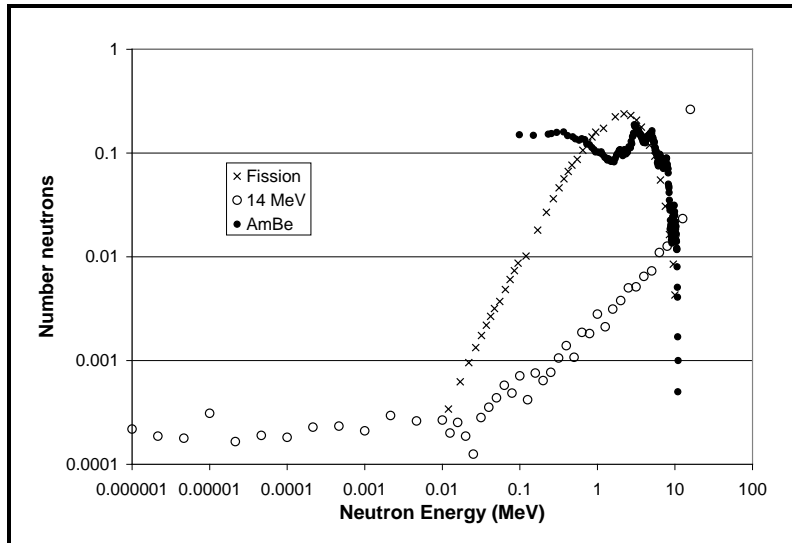


Figure 6-14. Neutron spectra that simulate INL facilities [19].

Most of the reactors built at INL had no beam ports. The neutrons were thus 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.

6.3.4.4 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 a research floor. ZPPR and TREAT, both at ANL-W, are also in this category. Some neutron surveys of the MTR experimental floor have been recovered (Sommers 1959, 1962; Hankins 1961), but these do not individually provide all components of the radiation field. Hankins (1961) used 2-, 3-, and 8-in. polyethylene Bonner balls in a cadmium shield to characterize the intermediate and fast neutrons at 21 locations around the MTR floor 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 6-15 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 6-16 shows the correlations of the thermal and intermediate neutron dose equivalents to the fast neutron dose equivalent for the ORAUT (2006a) 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, the fraction of neutron dose equivalent above 0.8 MeV has an average value of 0.52 ± 0.08 and varies

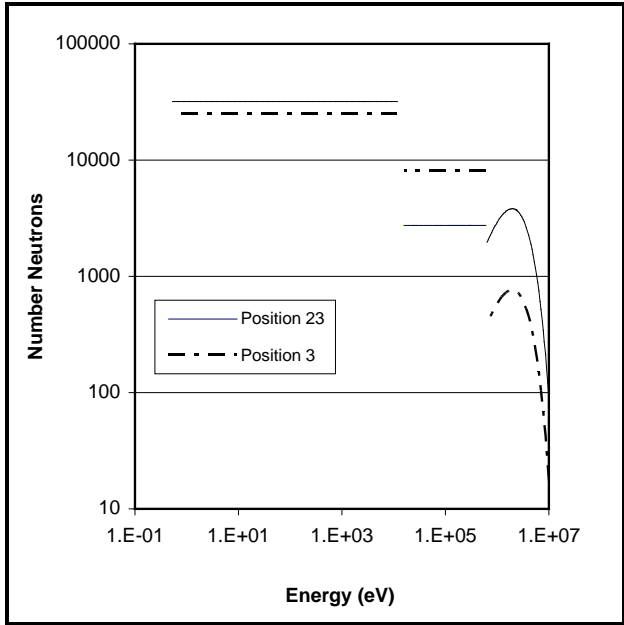


Figure 6-15. Sample MTR spectra from Hankins Bonner measurements (ORAUT 2006a).

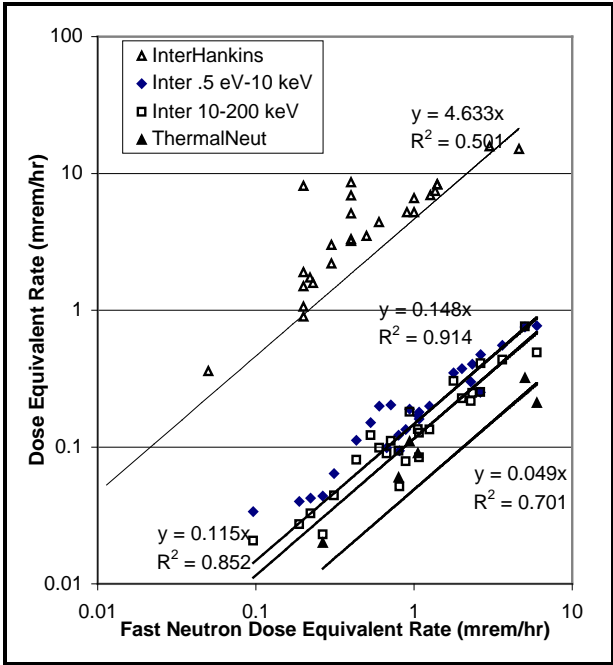


Figure 6-16. MTR neutron field components (ORAUT 2006a).

from 35% to 66% depending on the location (ORAUT 2006a). The dosimetry record location code for the TRA was 4 (later 40 to 45). To correct for missed dose on the MTR experiment floor, the NTA results from MTR should be multiplied by 2 ± 0.3 ($1/0.52$, $0.08/0.52^2$) for a Monte Carlo dose reconstruction (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 6-17 shows the correlation of fast neutron dose equivalent to the gamma dose for these measurements.

The fast neutron component was insignificant for several of the measurements. The values in Figure 6-17 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 experimental floor [20]:

$$\frac{TotalNeutronDose}{GammaDose} = \frac{FastN}{GammaDose} \left(1 + \frac{Thermal}{FastN} + \frac{LoInter}{FastN} + \frac{HiInter}{FastN}\right) \quad (6-1)$$

$$= (0.42 \pm 0.35)(1 + 0.071 + 0.177 + 0.149) = 0.58 \pm 0.48$$

The variations within 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 experimental floor while the reactor was operating. For example, health physics technicians would

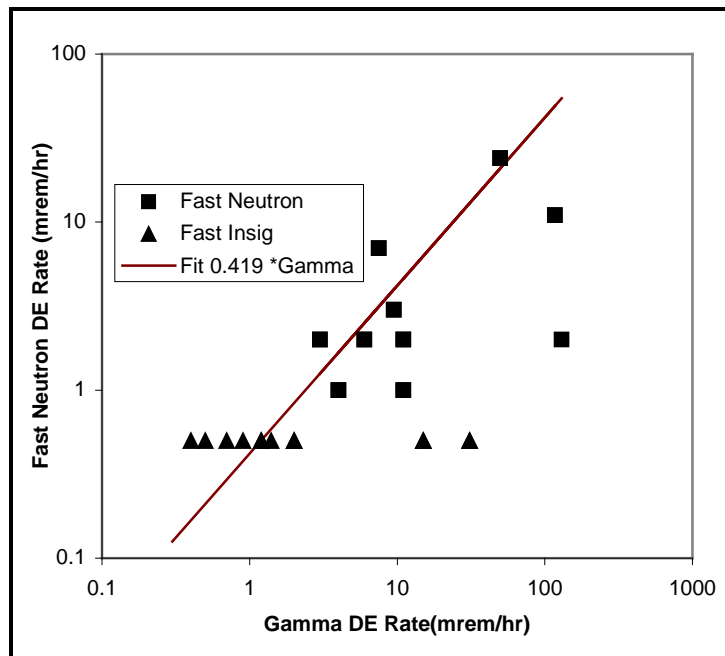


Figure 6-17. Correlation of fast neutron dose equivalent to gamma dose at MTR (ORAUT 2006a).

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 [22].

6.3.4.5 Test Area North Fuel Storage Casks

As noted in the INL Site Description TBD (ORAUT 2005a, 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 people (about six) 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 onto the storage pad. The casks contain irradiated oxide fuel in which the $^{18}\text{O}(\alpha, n)$ 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 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 [23].

6.3.4.6 Typical Workplace Neutron Dosimeter $H_p(10)$ Performance

Table 6-12 summarizes typical neutron personnel dosimeter parameters important to $H_p(10)$ performance in the workplace. The most important parameter in relation to $H_p(10)$ performance of the neutron dosimeters is the difference between calibration and workplace neutron energy spectra [24].

Table 6-12. Typical workplace neutron dosimeter $H_p(10)$ performance [25].^a

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

a. Judgment based on INL dosimeter response characteristics.

b. Recorded dose compared to $H_p(10)$.

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Neutron Weighting Factor

All measured and reported dose equivalents from INL in this document 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 [26]. 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 INL data require correction to change from dose equivalent (pre-ICRP Publication 60) to the newer dose quantity (ICRP 1991; NIOSH 2006). 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 (2006), 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:

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

Error! Bookmark not defined. These factors are incorporated into 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 1971b). 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 about 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 [27].

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 [28].

Table 6-11 summarizes the locations at INL where neutron dose is credible. Table 6-13 lists the calculated fractions of dose equivalent in the IREP energy groups and the conversion factors from

Table 6-13. Calculated and recommended dose equivalent fractions and quality factor corrections [29].

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 74 H^*_{10} / NCRP 38 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

dose equivalent to equivalent dose for INL 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 [30]. 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 6-14.

Table 6-14. Recommended IREP neutron energy fractions and correction factors [31].

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

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 2004b). The latest revision of *Use of Coworker Dosimetry Data for External Dose Assignment* (ORAUT 2005b) 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 Photon Missed Dose

Missed photon dose for INL workers would occur when a zero dose was recorded for the dosimeter systems for any response less than the site dose recording threshold (the MRL). The missed dose for dosimeter results less than the MRL is particularly important for earlier years when MRLs were higher and dosimeter exchange was more frequent [32]. The missed dose is calculated as described in NIOSH (2006) using MRL/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 MRLs 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.

6.5.3 Missed Beta Dose

Beta dose is important for certain cancers. Beta electrons are always above 15 keV if they are considered as external dose. Because INL had essentially no separated plutonium, the electron dose

is the difference between the shallow and deep doses. All nonpenetrating dose for INL should be considered to be electron or beta dose. An alternative method is to use the nonpenetrating dose. The number of positive readings for beta exposure is likely to be much smaller than for photons, so the missed beta dose is generally larger. The calculation of missed beta dose should (1) determine the measured beta dose and the number of nonzero readings, (2) determine the missed dose using MRL/2, the MRLs shown in Table 6-14, and the number of zero readings; then (3) correct the total dose by multiplying by the factor in the last column of Table 6-10 for the appropriate period.

Table 6-15. Beta/photon dosimeter period of use, type, MRL, exchange frequency, and potential annual missed dose [33].

Period of use ^a	Dosimeter	Exchange frequency	MRL ^b (mrem)		Annual missed dose (mrem) ^c	
			Photon	Beta	Photon	Beta
August 1951– March 1958	INL initial film, 552 DuPont film	Weekly (n=52)	30	30	780	780
		Monthly (n=12)			180	180
March 1958 – December 1966	Reactor areas, DuPont 558 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

- 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.
- MRLs 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).
- Maximum annual missed dose calculated using $n \times \text{MRL}/2$ from NIOSH (2006).
- ICPP began using the Harshaw TLD in December 1974, the prime contractor began in February 1975, and ANL-W began in May 1975.
- The MRL was 15 mrem from January 1, 1986, to July 7 1986; 10 mrem from July 7, 1986, to about September 1989; and 15 mrem until 1993, when it returned to 10 mrem.
- The MRL was 15 mrem from January 1, 1986, to July 7 1986, and 30 mrem after that.

6.5.4 Missed Neutron Dose

Neutron radiation was present at the INL reactors, the 14-MeV neutron generators at RWMC, for a short time at TRA and TAN, in small sources used for research at TRA and the INL Research Center, in calibration laboratories CF633 and CF636, and for calibration of criticality alarms at TRA and ICPP [34]. For other locations, there is likely limited missed neutron dose because of the very low potential for neutron exposure. To calculate the missed dose, the dose reconstructor must first determine if the

person worked near neutrons and the category of neutrons. This can best be accomplished by looking for the work location and at whether a coworker or others in the badge reporting group were assigned a neutron dose equivalent. The work location code for TRA where the MTR operated is 4 (later 40 to 45). If no neutron dose was assigned to the worker or coworkers for several months, the dose reconstructor should assume that the person was not exposed to neutrons [35].

If a worker was likely to have been exposed to neutrons, the dose reconstructor should assign missed neutron dose equivalent using Table 6-16 for the times when workers did not have reported neutron dose [36]. For the period when NTA film was used, the missed neutron dose should be multiplied by 1.25 for all facilities except the MTR experimental floor and by 2 for the MTR experimental floor when the MTR was operating between 1953 and 1970 [37]. The dose equivalent is then apportioned into the IREP groups using Table 6-16.

Table 6-16. Neutron dosimeter type, period of use, exchange frequency, laboratory minimum detectable limit, and maximum annual missed dose [38].

Dosimeter	Period	Exchange frequency	Laboratory MRL (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

For example, if in 1955 a person was an experimenter at the MTR, and seven of the weekly badges recorded a total of 185 mrem neutron dose equivalent, then the missed dose would be 315 mrem $[(52 - 7) \times 14 \div 2]$ so the total dose by the badges would be 500 mrem. Because the badge only sees about one-half of the MTR neutron dose equivalent (see Section 6.3.4.4), the total dose equivalent is 1 rem. To convert the 1 rem received from neutrons on the MTR experimental floor to equivalent dose, the total dose equivalent is multiplied by the last column of Table 6-14, which results in 200 mrem to the <10-keV group, 60 mrem to the 10- to 100-keV group, 700 mrem to the 0.1- to 2-MeV group, and 280 mrem to the above-2-MeV group for a total equivalent dose of 1.24 rem.

The neutron missed dose is divided into two historical periods in the following discussion. The first is before October 1976 when only NTA film dosimeters were used with supplemental recording of thermal neutron doses from B-10 pencil dosimeters [39]. The second period is from and including October 1976 to the present when only Hankins albedo dosimeters were used [40]. Table 6-16 summarizes the estimated MRLs for these neutron dosimeters. It is possible to estimate the missed neutron dose using the MRLs because the neutron dosimeters were calibrated with neutron sources of energies similar to those in the workplace and because most of the neutrons to which workers were normally exposed had energies greater than the 500- to 800-keV threshold of the NTA film dosimeters [41]. There was, of course, no threshold energy for the measurements using neutron albedo TLD badges.

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 MRLs. 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 MRL [42]. When the MRL 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 experimental floor [43].

An estimate of the missed neutron dose in some facilities might appear to be attainable through use of neutron-to-photon dose ratios (NIOSH 2006). However, for the INL 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 MRL 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 2006). It is recommended that the AP geometry should be assumed for the irradiation geometry and for conversion to organ dose [44]. The calculated neutron doses in each energy group should be multiplied by the conversion factors from ambient dose equivalent to organ dose for AP irradiation from Appendix B of NIOSH (2006). For photons before 1981, the conversion factor from exposure to organ dose should be used [45]. For 1981 and after, the conversion factor from deep dose equivalent to organ dose should be used [46].

6.7 UNCERTAINTY

Measurement uncertainties arise from many sources. A lognormal distribution should generally be assumed unless stated otherwise. 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 only causes a minor change for high-energy photon dosimetry [47]. 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 [48]. 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 beta radiation. A realistic estimate of total uncertainty for photon dosimetry is about 35% at 1 sigma [49]. This is roughly consistent with the results in Figure 6-10. For those measurements, the standardization instrument contributed some significant uncertainty.

For beta radiations in relation to skin cancer, etc., the reported nonpenetrating dose must be divided by the fraction of measured dose in Table 6-12. The uncertainty for beta radiation is somewhat larger at an estimated 50% at 1 sigma [50]. This is driven by uncertainties in the field geometry and because beta radiation is often stopped by air and thin materials such as clothing [51]. Algorithms are used to estimate the dose at a depth of 7 mg/cm² from dosimeters at depths of 15 to 250 mg/cm², and such depth differences can change the signal significantly [52]. The difference between point- and planar-source irradiation can confuse an algorithm [53]. 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 ${}^6\text{Li}(n,\alpha)$ reactions [54]. 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 a geometric standard deviation (GSD) of about 1.6 [55]. The uncertainty of the base curve for calculating the FNCF is a GSD of about 1.35 (Gesell et al. 1996) although this uncertainty does not contribute in the DOELAP process, which results in a total GSD of 1.7 for albedo neutron measurements [56].

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 about a factor of 1,000 higher than the dose on the back side of a 30-cm-diameter sphere of tissue-equivalent material [57]. 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) [58]. Note that in Figure 6-12 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. 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. This revision has been overseen by a new Document Owner, who is fully responsible for the content of this document, including all findings and conclusions. Mr. Rohrig continues to serve as a Site Expert for this document because he possesses or is aware of information relevant for reconstructing radiation doses experienced by claimants who worked at the site. In all cases where such information or prior studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source. Mr. Rohrig's Disclosure Statement is available at www.oraucoc.org.

- [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, "DOE Technical Basis Document

Development, Review, and Approval Process” and uses values based on experience and from the Hanford Site Profile (ORAUT 2006b).

- [4] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is adapted to INL from the Hanford Site Profile (ORAUT 2006b) and uses values based on experience at the 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 2005a).
- [8] Rohrig, Norman D. ORAU Team. Health Physicist. Date.
The contact radiation field on these devices is less than 0.5 mrem/hr (phone conversation Larry Burke, Radiological Engineer, ANL-W and INL, 11/06/2006, 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. Also 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 6-9.
- [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 2005a) and work experience at the facility.
- [19] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This plot was generated from previously 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 may not be directly related to each other, they often have different sources.
- [23] Rohrig, Norman D. ORAU Team. Health Physicist. February 20, 2007.
Based on conversation on October 12, 2006 with a former health physicist who was in charge at TAN in 1986 and was in charge of INL dosimetry from about 1992 to 2006 (Rohrig 2006b).
- [24] 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 upon the similarity between the calibration and workplace radiation field.
- [25] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This standard TBD table is based on information in the Hanford TBD (ORAUT 2005a) and, in the judgment of Mr. Rohrig, performance of INL dosimeters for similar parameters of performance.
- [26] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on guidance in OTIB-0055 (ORAUT 2006c).
- [27] 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 thus need to face the radiation source to read reliably.

- [28] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. This is based on guidance in OTIB-0055 (ORAUT 2006c).
- [29] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. This table was derived with spectral data in a spreadsheet as shown in Figures 6-13 and 6-14 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.
- [30] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. The values in Table 6-12 are different, namely 1.08 and 1.46 or 1.47.
- [31] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. This table is derived from the values in Table 6-12.
- [32] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. The calculated missed dose, based on OCAS-IG-001 guidance 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.
- [33] 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 report.
- [34] 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.
- [35] 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."
- [36] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. This judgment is made by Mr. Rohrig based on reviews of historical INL dosimetry information.
- [37] 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.
- [38] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. The values in this Table 6-15 are required to calculate the missed dose per ORAUT-PROC-0031 and are based on judgment of Mr. Rohrig.
- [39] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. This reflects the period discussed in Section 6.3.2.7.
- [40] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003. This reflects the period discussed in Section 6.3.2.8.
- [41] 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.

- [42] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
By definition of the MRL, values less than the MRL are reported as zero (practice at INL) or a blank.
- [43] 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.
- [44] 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.
- [45] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This recommendation is made because the measurement quantity of dose in that period was exposure.
- [46] 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.
- [47] 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).
- [48] 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.
- [49] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This estimate is partially based on Figure 6-9 and partially based on uncertainties of many error processes that contribute uncertainty.
- [50] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This estimate is partially based on Figure 6-9 and partially based on uncertainties of many error processes that contribute uncertainty.
- [51] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The range of beta particles is shown in Figure 6-10.
- [52] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
These estimated corrections are provided in Table 6-9, and they are not small.
- [53] Rohrig, Norman. ORAU Team. Health Physicist. June-October 2003.
For this reason, DOELAP has had different categories for point and slab beta geometries.
- [54] 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 ${}^6\text{Li}(n,\alpha)$ neutron absorption reactions of the neutron dosimeter.

- [55] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This reflects the spread in Figure 6-13 and the factors in Table 6-12.
- [56] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Most of this uncertainty is from the FNCF for the field of the neutron exposure.
- [57] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
See Figure 16 of NCRP Publication 38 (NCRP 1971b).
- [58] 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

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

beta particle

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

dose equivalent (H)

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

dose equivalent index

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

deep dose equivalent index

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

DOE Laboratory Accreditation Program (DOELAP)

This program accredits DOE site dosimetry programs based on performance testing and onsite reviews performed on a 2-year cycle.

dosimeter

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

effective dose equivalent (H_E)

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

exposure

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

exposure-to-dose-equivalent conversion factor for photons (C_x)

The ratio of exposure in air to the dose equivalent at a specified depth in a material of specified geometry and composition. The 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 ignoring trace elements, or soft tissue including trace elements).

linear energy transfer (LET)

The lineal rate of local energy deposition by a charged particle.

minimum reporting level (MRL)

Based on a policy decision, the minimum dose level that is routinely recorded.

nonpenetrating dose

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

pencil dosimeters

A type of ionization chamber used by personnel to measure radiation dose. Other names: pencil, pocket dosimeter, pocket pencil, 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, $H_p(d)$

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

polymethyl methacrylate

Scientific name for plastic commonly known as Lucite or Plexiglas.

range

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

redacted

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

shallow absorbed dose (D_s)

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

shallow dose equivalent (H_s)

Dose equivalent at a depth of 0.07 mm in tissue (sum of penetrating and nonpenetrating dose equivalent).

tissue rad

Absorbed dose in tissue.