Document Title:	Document Number:	ORAUT-TKBS-0007-6
Idaho National Engineering and Environmental Laboratory – Occupational External Dosimetry	Revision:	01
	Effective Date:	11/07/2006
	Type of Document:	TBD
	Supersedes:	Revision 00

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Approval:	Brant A. Ulsh Signature on File for James W. Neton, Associate Director for Science	Approval Date:	11/07/2006

☐ New ☒ Total Rewrite ☐ Revision ☐ Page Change

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
04/06/2004	00	New technical basis document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational External Dosimetry. First approved issue. Initiated by Norman D. Rohrig.
11/07/2006	01	Document revised as a result of biennial review. Approved Revision 01. Incorporates formal internal review comments. Revision constitutes a total rewrite of document. Adds Section 6.5.1 to incorporate comments from the Worker Outreach meeting with the PACE Local 8-0652 on April 28, 2004. Incorporates formal NIOSH review comments. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig.

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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 Reactor Area/Auxiliary Reactor Area
CFA	Central Facilities Area
Ci	curie
cm	centimeter
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
EBR	Experimental Breeder Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EPRI	Electric Power Research Institute
ERDA	U.S. Energy Research and Development Administration
FNCF	facility neutron correction factor
g	gram
$H^*(10)$	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 (formerly CPP and now INTEC)
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
in.	inch
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center (formerly ICCP and CPP)
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
MRL	minimum reporting level
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
NRTS	National Reactor Testing Station
NTA	nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program
PBF	Power Burst Facility
POC	probability of causation
R	roentgen
RBE	relative biological effectiveness
RESL	Radiological and Environmental Services Laboratory
RWMC	Radioactive Waste Management Complex
SMC	Specific Manufacturing Capability
SPERT	Special Power Excursion Reactor Test
TAN	Test Area North
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor 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 Reactor, Zero Power 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 the NIOSH staff in the completion of the individual work required for each dose reconstruction.

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

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation[1]] guidelines established under subsection (c) …" [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation) 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:

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

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

6.1.1 Purpose

This document describes the processes and results for measuring external doses at the Idaho National Engineering and Environmental Laboratory (INEEL) and its predecessor organizations [the National Reactor Testing Station (NRTS) and the Idaho National Engineering Laboratory (INEL)]. In 2005, DOE shortened the name to the Idaho National Laboratory, but this document uses INEEL for consistency.

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 INEEL until 1989, when DOE transferred that responsibility to the prime operating contractor. Despite the fact that INEEL had several contractors at a time and that contractors changed often, the external dosimetry process has remained under technical management of a single organization with responsibilities for dosimetry development, operational dosimetry, and radiological records; thus providing a stable external dosimetry system.

6.2 BASIS OF COMPARISON

The Interactive RadioEpidemiological Program (IREP) calculates the probability of cancer induction in an organ from the external equivalent dose and internal dose received by that organ. Appendix B of the *External Dose Reconstruction Implementation Guideline* (NIOSH 2002) provides conversions from four photon dose quantities [deep dose equivalent $H_p(10)$, ambient dose equivalent $H^*(10)$, exposure X, and air kerma K_a] and three neutron quantities [fluence φ, ambient dose equivalent $H^*(10)$, and deep dose equivalent, $H_{p,slab}(10)$] to the organ doses. Over the years, 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 INEEL dosimetry system have received further definition. INEEL has reported doses as penetrating and nonpenetrating. The penetrating dose corresponds to the deep dose equivalent, and the sum of the nonpenetrating and penetrating doses corresponds to the shallow dose equivalent.

Horan and Braun (1993), Attix and Roesch (1968), and Meinhold (1975) discuss the history of radiation protection requirements from the 1930s. On September 29-30, 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-rays 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 roentgen equivalent physical (rep) units equivalent to 93 ergs absorbed per gm 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 or NCRP) 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. 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. 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 per year above age 18 (5 rem × [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-wk whole-body limit of 3 rem with "the provision that not more than one-fourth of the 15-rem maximum permissible yearly dose shall be taken in one-fourth of a year" (AEC 1958).

The quarterly limit of 3 rem or 12 rem/yr replaced the 15 R/yr 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 (AEC 1963) 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.

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 adding 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 contained in American National Standards Institute standards N13.5, 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 hr 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 reported in ICRP 21 (1973) do not present that problem.

In 1971, ICRU defined the quantity *dose equivalent index*, the maximum value in a 30-cm-diameter sphere, for describing ambient radiation fields for radiation protection purposes (ICRU 1971). ICRU extended this discussion in *Conceptual Basis for the Determination of Dose Equivalent* (ICRU 1976),

which defined the concept of deep and shallow dose equivalent indexes as those inside a 1-cm depth in the sphere and at a depth between 0.07 mm and 10 mm, respectively. A remaining issue was that the quantity was measured near the surface of the sphere but applied to the center of the sphere, a distance of 14 or 15 cm. In 1980, ICRU identified the deep and shallow dose equivalent indexes as restricted indexes (ICRU 1980). In 1985, ICRU Report 39, *Determination of Dose Equivalents Resulting from External Radiation Sources*, introduced the concepts of aligned and expanded fields to eliminate issues of field direction and nonuniform fields; it 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 (1977, 1979) introduced new dose limits and the associated quantity *effective dose equivalent* as averaged over the radiation-sensitive organs of the body.

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 1998), superseded DOE Order 5480.1A, Chapter XI. The Order 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 1983b) and the National Voluntary Laboratory Accreditation Program (NVLAP). DOE *Guidelines for the Calibration of Personnel Dosimeters* (Roberson and Holbrook 1984) revised the 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, invoked in NIOSH (2002), depend on neutron energy at the entrance to the body rather than on secondary particle LET where the dose is received. Dose conversion factors for organs and for ambient dose equivalent and personal dose equivalent were generated in ICRP Publication 74 (ICRP 1996) and are referenced in the external dose implementation guide (NIOSH 2002).

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

6.3 DOSE RECONSTRUCTION PARAMETERS

6.3.1 Site Administrative Practices

It was INEEL policy that personnel 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 respective operational area entrance security gate for INEEL facilities. Control badges, which are used to subtract background radiation, have also been located there. This practice might have led to subtracting environmental radiation from site activities, which would have reduced the reported doses. Environmental radiation levels have been monitored for most of the life of the INEEL 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), which can be added to individual's dose history or used for unbadged workers on the site.

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

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

DOE provided the dosimetry information for former INEEL 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 as to the exchange period for the person for that time period. The following five figures are a partial example set of redacted dose reporting forms.

From 1951 to 1958, the INEEL 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. On this sample, on October 28, November 16, and December 9, 1954, the badges were pulled and read in response to high pencil chamber readings. The personnel monitoring badges have always been considered more reliable than pencil dosimeters, so after the film badge results became available the daily pencil readings were no longer considered doses of record. However, these values can be recovered from the earliest forms for a worst-case estimate of dose. In Figure 6-1, the pencil readings totaled 820 mR and the badges reported 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 (O. W.) result was zero, so the shielded film was not read. On the other two, the open window and shielded values were at the minimum recorded density of 0.02, which corresponded to a 30 mR penetrating dose.

After the pencil ionization chambers were replaced with self-reading pencil dosimeters (which were also ionization chambers), the INEEL operational health physics staff would rezero the dosimeters. The film reading was automated and results were stored in a computer. The form shown in Figure 6-3 reports badge reading for May 1959 when badges were exchanged every 2 wk. The column under

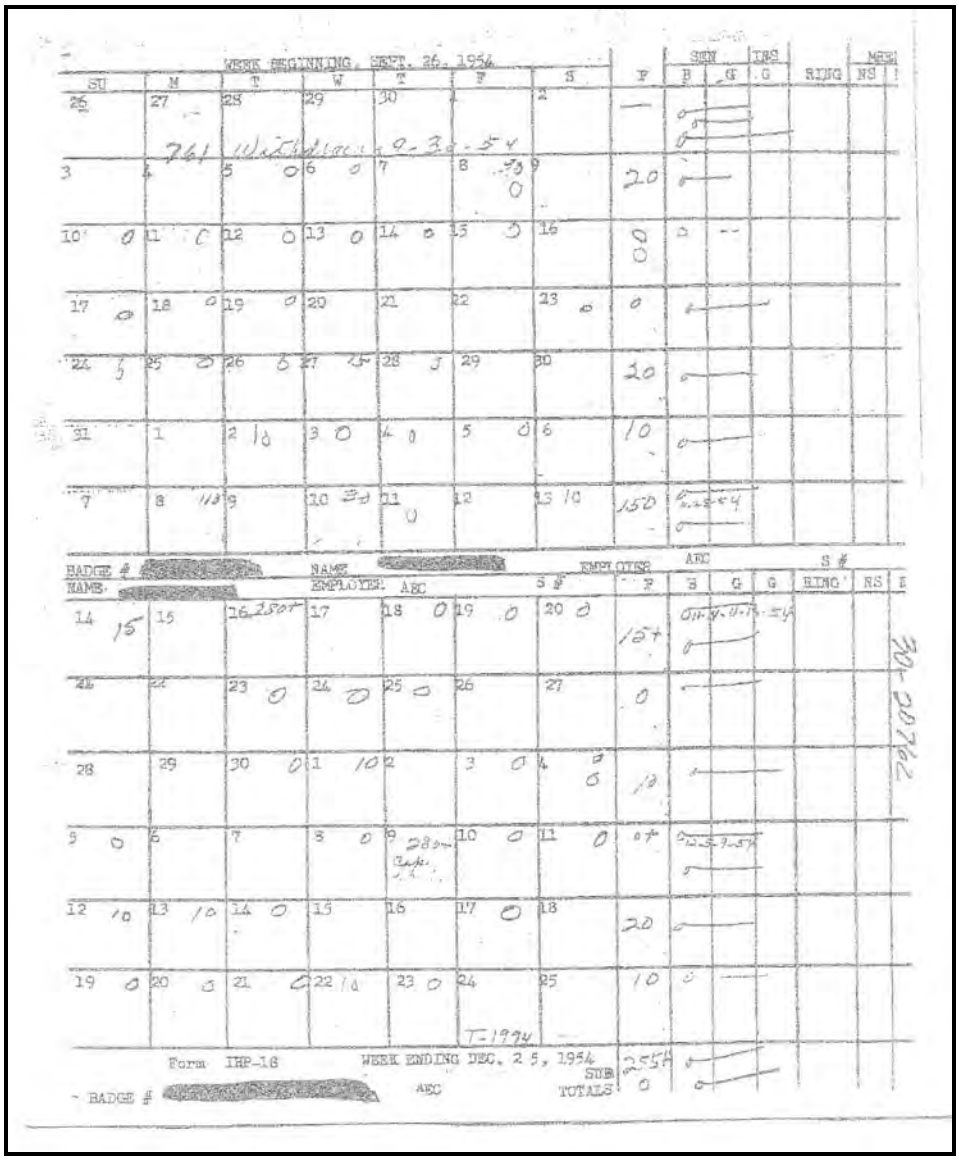

Figure 6-1. Individual dose reporting form in use until 1958.

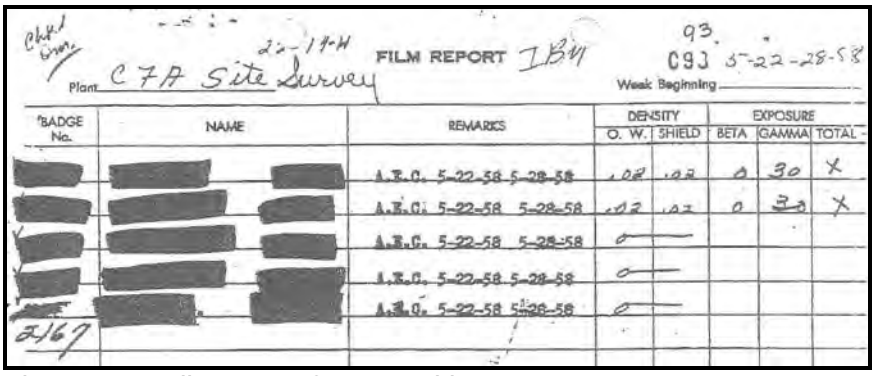

Figure 6-2. Film report form used in 1958.

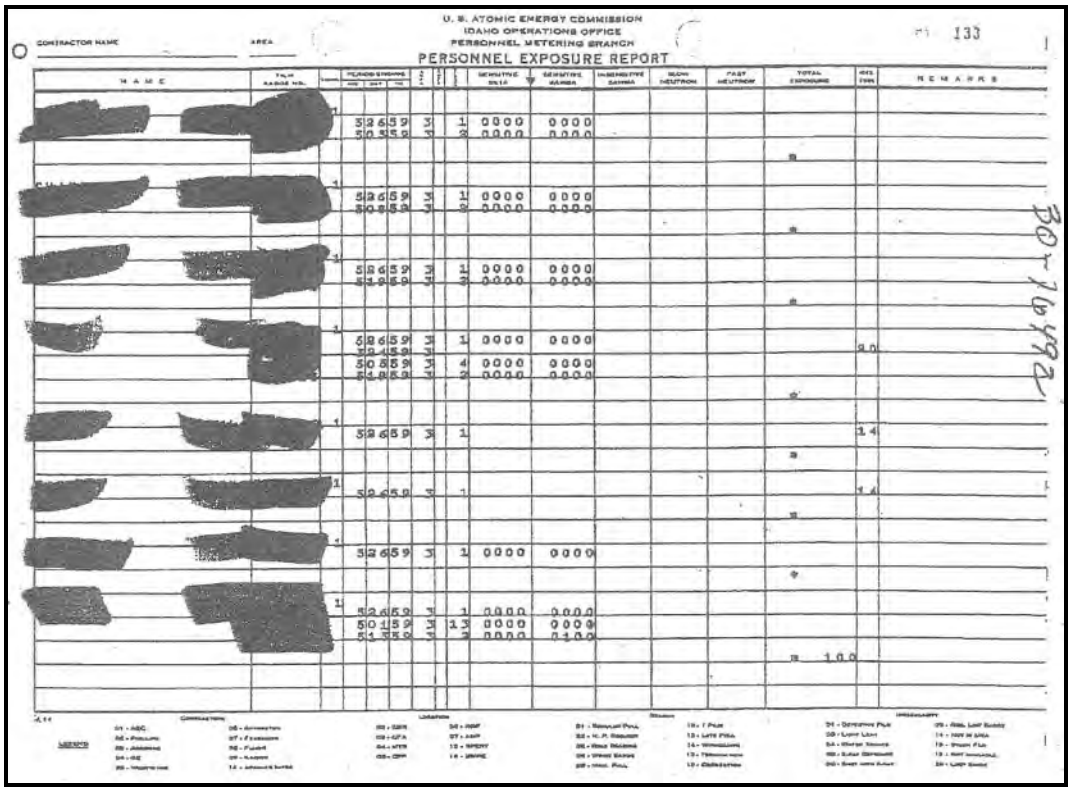

Figure 6-3. Monthly badge reporting form from May 1959.

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 somewhat later. The next column was for the reason codes listed in Table 6-2. Computer input card codes are listed in Table 6-3, "Irregularity Codes," and in Table 6-4, "Column 20 codes," respectively.

Table 6-1. Area codes that could be in claimant dose files.

Area code	Description	Area code	Description
1	AEC Headquarters Bldg	20, 261, 264	TREAT
2	EBR – I	21	LX
3, 034, 035	CFA	22	GCRE
4, 042,045	MTR, TRA	23	OX at OMRE
5, 053, 055	ICPP	24	ARHG
6	NRF	25	No information available
7	TAN (GE)	26, 263,265	EBR -II
8	Services	27	ML-1
9	NX (X is construction) at NRF	28	On-Site Site Survey
10	AX at TAN	29	Off-Site Site Survey
11, 113	CX at CPP	30	ANP Program at SL-1
12	EX at EBR	31	STPF
13, 133,135	SPERT, PBF	65	ECF
14	OMRE	66	Non-Security
15	SX at SPERT	67	Division of Compliance
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

With the advent of computers, the reports are all computer generated with the effect that even though many workers were not exposed to neutrons and did not receive neutron dosimeters, zeros are 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, (~1975-1984) the penetrating and nonpenetrating labels are reversed from the data that is printed. The nonpenetrating result is reported first, then the penetrating, and then the neutron result.

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

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

Table 6-3. Irregularity codes (columns 70-71).

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

Table 6-4. Column 20 codes.

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

Figure 6-4 is a listing of some doses received during recovery from the SL-1 accident. Workers from several areas were pulled into the accident recovery process, and it is notable that one result exceeds the dose limits and that there are few zeros. Figure 6-5 is a follow-up badge report for one result on Figure 6-4. When there was a question about an assigned dose value, a Personnel Exposure Questionnaire was normally initiated as shown in Figure 6-6 (shows a hypothetical case).

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NAME	CONTR.	AREA	BADGE NUMBER	BETA	GAMMA
[redacted]	71	32	[redacted]		50
[redacted]	07	17	[redacted]		365
[redacted]	01	03	[redacted]		
[redacted]	02	05	[redacted]		140
[redacted]	73	00	[redacted]		1175
[redacted]	02	16	[redacted]		345
[redacted]	30	00	[redacted]	550	740
[redacted]	01	03	[redacted]		35
[redacted]	04	07	[redacted]	120	350
[redacted]	71	22	[redacted]	210	205
[redacted]	02	03	[redacted]		60
[redacted]	67	16	[redacted]	120000	23195
[redacted]	07	16	[redacted]		570
[redacted]	07	04	[redacted]		
[redacted]	01	03	[redacted]		65
[redacted]	01	03	[redacted]	1475	2370
[redacted]	01	01	[redacted]		30
[redacted]	02	05	[redacted]		
[redacted]	01	03	[redacted]	190	145
[redacted]	01	03	[redacted]		15
[redacted]	02	00	[redacted]		
[redacted]	01	03	[redacted]		245
[redacted]	01	03	[redacted]	135	210
[redacted]	01	03	[redacted]	530	35
[redacted]	01	00	[redacted]		
[redacted]	04	16	[redacted]	120	130
[redacted]	01	03	[redacted]		180
[redacted]	02	04	[redacted]		20
[redacted]	04	07	[redacted]		

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

10-105
(3-58)

BADGE REPORT AEC 07A

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

Special Report
master

032

TO

AREA SL-1 Bldg. Date

RE: [redacted] Badge No [redacted]

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 CP dog pull

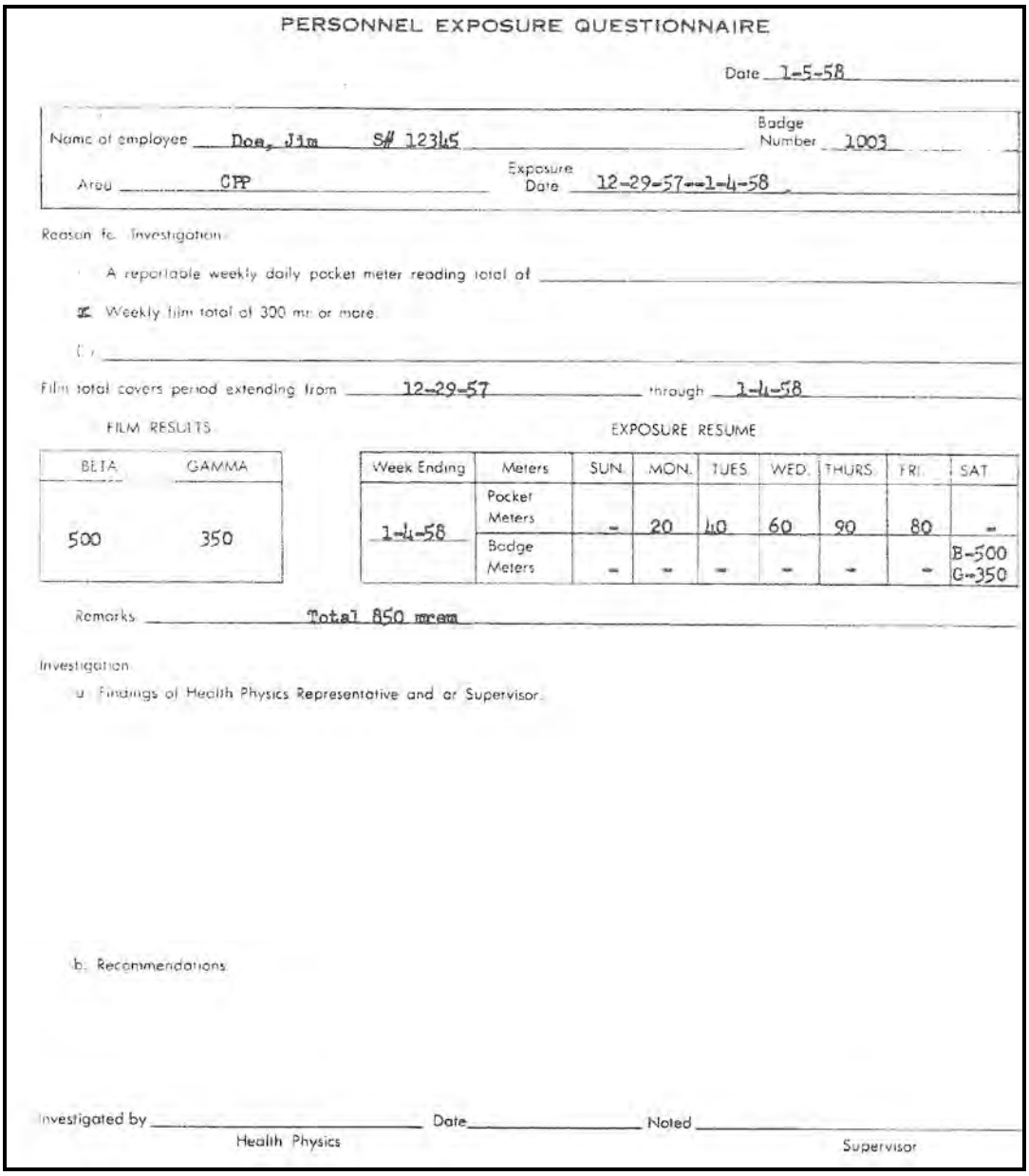
☐ Damaged Pencil DO Special Report

☐

week 5

Signed

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



PERSONNEL EXPOSURE QUESTIONNAIRE

Date 1-5-58

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

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

Reason for Investigation:

A reportable weekly daily pocket meter reading total of

X Weekly film total of 300 mr or more.

C.

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

FILM RESULTS

BETA	GAMMA
500	350

EXPOSURE RESUME

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

Remarks: Total 850 mrem

Investigation:

a. Findings of Health Physics Representative and or Supervisor.

b. Recommendations

Investigated by _____ Date _____ Noted _____

Health Physics Supervisor

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

6.3.2 Personnel Monitoring Systems

6.3.2.1 Initial Film Badge

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

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

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 open window responds to beta radiation as well as X-rays and low-energy gamma rays. Because of the high atomic number (*Z*) of film in relation to air or tissue, the open window 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. Because there was no isolated plutonium at the NRTS, the non-penetrating radiation is considered to be beta radiation. Using a cadmium filter with its high *Z* severely attenuates the photons that get to the film, so the over-response is reduced to about a factor of 2 at 125 keV. The beta particle range is independent of the atomic number Z depending only on the density, so the 1-mm cadmium filter (~900 mg/cm^2) simulates a tissue depth of 9 mm for beta radiation.

Wrist badges used the same package attached to a wrist band. A finger ring used a small piece of film with a silver or cadmium filter. Pencil ionization chambers were used to monitor daily doses and control operational activities. The dosimetry group read these chambers and recorded the results on cards. Film badge readings were written on the same cards to indicate the dose of record. In 1958, the Victoreen 352 pencil ionization chambers 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 containing filters of 1 mm cadmium, 0.013 mm silver, and 0.5 mm aluminum with thicknesses of 950 mg/cm^2, 203 mg/cm^2, and 175 mg/cm^2, respectively, including the plastic in which they were mounted (Cipperley 1968). This NRTS badge was also a security badge, which resulted in an absorber thickness of 100 mg/cm^2 in the open window 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 of 10 mrem was used for both beta and gamma radiation (Horan 1962).

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 contributions of the betas and the high- and low-energy photons separately (Cipperley 1968, p. 94). The cadmium filter provided the hard gamma component. The uranium beta responses under the open window, aluminum, and silver filters were 1, 0.2, and 0.1, respectively. By assuming a beta signal and subtracting it, the remaining signal could be attributed to low-energy photons and the energy could be estimated. For beta 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

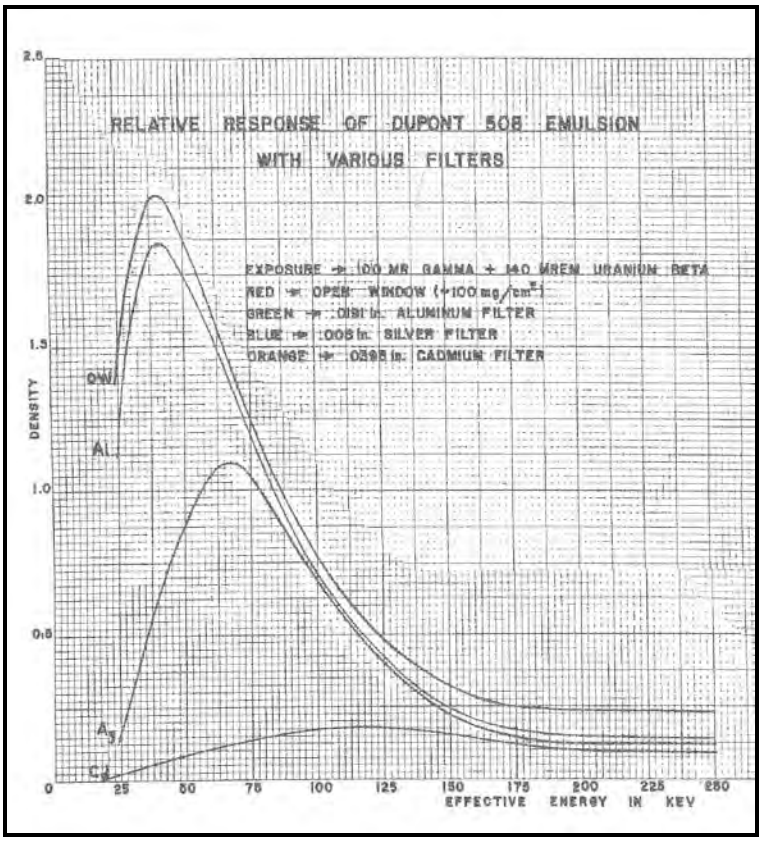


Figure 6-7. Response of DuPont 508 film with various filters to 140 mrem uranium beta and 100 mR of different energy photon irradiation. The original badge used the open window and cadmium shielded films. The multiple filter badge used all three filters plus the open window (Cipperley and Gammill 1959).

weekly schedule continued (Horan 1959). The introduction of punch cards increased the efficiency of report and record generation. A 12-point calibration curve was generated for radium and for ^{137}Cs gamma and uranium beta. Calibration did not use a phantom.

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

6.3.2.3 Original Lithium Fluoride Teflon TLD System

Beginning in November and December 1966, individuals projected to receive doses of less than 0.5 rem/yr were given a lithium fluoride (LiF) disk TLD badge, which was exchanged quarterly (Cusimano and Cipperley 1968). Two 13-mm-diameter Teflon disks, 0.4 mm thick [100 mg (75 mg/cm^2) impregnated with 28 mg LiF], were mounted in a badge behind an open window and a 1-mm cadmium filter (Watkins no date). 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, resulting in little energy correction for beta or gamma radiation. The badge could read 30 mR on a quarterly basis, so more small doses were reported (Cusimano and Cipperley 1968). The

angular dependence of the radium-gamma response (within 10% to 70º) is superior to film because the material acts like an ionization chamber. For normal monitoring, only the open window 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.

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 960 were processed quarterly (Cusimano 1972). Employees on a monthly badge change were moved to this system as late as September 1973.

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

6.3.2.4 Automatic Thermo Luminescent Analyzer System

Development began in 1969 on a patented Automatic Thermo Luminescent 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). A minimum reporting level (MRL) of 30 mR was used for gamma and beta (Walker 1974).

6.3.2.5 Harshaw Two-Chip TLD 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 240 mg/cm^2 thick. In 1976, holes were punched in the security badges to restore the open window. One chip was covered by 540 mg/cm^2 of aluminum and the other was under 4 mg/cm^2 of Mylar. The aluminum-covered chip provided penetrating dose at a nominal tissue depth of 1 cm. The beta dose was calculated from the difference between the two chips. Because of the thickness of the Mylar-covered chip, the beta dose was accurate only for the beta energy used in calibration. Field calibrations were used to reduce the problem with beta energy dependence. The initial thin aluminum filter (density thickness 350 mg/cm^2) allowed higher-energy beta radiation to expose the chip used for measuring the penetrating (at 1,000 mg/cm^2) dose so it was changed to 2 mm in July 1977 (INEL 1978).

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

6.3.2.6 Panasonic Four-Chip System

In 1986, with the advent of DOELAP, INEEL changed to the four-element Panasonic 814 AS4 System (Gesell, Hall, and Andersen 1992; INEEL 2001). Lithium borate ($Li_2B_4O_7$) TLD elements with plastic and aluminum filtration provide an improved measurement of deep dose equivalent and, with a thinner filter, an improved measurement of the shallow dose equivalent. A calcium sulfate ($CaSO_4$) TLD provides a strong low-energy photon response. The elements are 15 mg/cm^2 thick. Element 1 has filtration of 16 mg/cm^2, element 2 has filtration of 58 mg/cm^2 plastic, and elements 3 ($CaSO_4$) and 4

have filtration of 550 mg/cm^2 of plastic and 50 mg/cm^2 of aluminum. Although none of the elements is at a depth of 7 mg/cm^2, the specified depth for the shallow dose equivalent, an acceptable response can be obtained by using elements at 16 and 58 mg/cm^2. This system is qualified in DOELAP for performance categories beta, photon, and mixture. The angular dependence of this system has been measured for ^{137}Cs gamma rays and four X-ray energies from 16 to 70 keV (INEEL 2001). For the lower energies, attenuation in the absorbers reduces the measured dose. (The attenuation also reduces the delivered deep dose, but this effect is not incorporated in the results.)

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

6.3.2.7 Nuclear Track Emulsion, Type A for Neutrons

Kodak nuclear track emulsion, type A (NTA) 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 dose assigned was 14 mrem. 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). A blank is 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. 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, providing three independent results (Cipperley 1968). Two tracks or fewer were attributed to background. This resulted in a somewhat higher MRL. In November 1959, a data sheet shows a 10- and 20-mrem neutron dose equivalent. In January 1962 a data sheet shows a 20-mrem dose. A data sheet from April 1959 shows neutron dose equivalents of 20, 20, and 40 mrem. These values suggest an MRL of 20 mrem.

Calibration was with a polonium-beryllium (Po-Be) 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 staff was aware of the energy limitations of the NTA badge (Sommers 1967, 1969) and compensated with neutron-detecting pencil dosimeters and field measurements. A request to read NTA film occurred if the hard spectra neutron exposure was likely to exceed 10 mrem. Procedures were established using boron trifluoride (BF_3) pencils to monitor neutrons in the Radioactive Waste Management Complex (RWMC) transuranic waste areas where NTA would not respond to low-energy neutrons (Sommers 1975).

6.3.2.8 Neutron Albedo Dosimetry

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

In the Hankins dosimeter used at the INEEL (Hankins 1973), TLDs with ^{6}Li to capture thermal neutrons are inside a polyethylene case (to lower the neutron energy) inside a cadmium shell (to eliminate thermal neutrons from outside) and ^{7}Li TLDs are used to subtract the gamma dose. Because the $^{6}Li(n,\alpha)^{3}He$ reaction has a strong energy dependence, the response does not follow the flux-to-dose-equivalent conversion, so the neutron signal is divided by a facility neutron correction factor (FNCF). A FNCF that converts the TLD gamma equivalent signal to neutron dose equivalent can be generated from the ratio of the dose equivalent measured with a 9-in.-diameter Eberline PNR-4 and the corresponding signal (in millirem but not dose equivalent) with the detector in the 3-in.-diameter PNR-4 insert. A plot of FNCF versus 9" to 3" 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 INEEL, 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 FNCFs.[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 Physics Reactor.

The date of the change from NTA to albedo neutron monitoring is somewhat in dispute. Different organizations would typically have transitioned 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 c. 1978) states, "initial testing of albedo neutron dosimeter and replacement of NTA neutron monitoring film with same," in October 1976. Aoki (1979) says the albedo system replaced the NTA badge in 1977. Dose reconstructions should make the assumption that this transition occurred on October 1, 1976.

6.3.3 Calibration

6.3.3.1 Beta-Gamma Radiation

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

As reported in 1981, an extrapolation chamber was built for the measurement of beta doses (Gupta 1981). The chamber window was polycarbonate, the gas was air, and the thick collecting electrode was Shonka tissue-equivalent plastic. The chamber was used to calibrate a 2.5 Ci ^{90}Sr/Y source to tissue rad. The source with area 2.5 cm^2 was constructed by the Amersham Searle Corporation in February 1975. This source was used to measure beta correction factors for several instruments following the Three Mile Island TMI-2 reactor accident in 1978. TLD badges were calibrated to 500 mrad tissue using a 1.78-cm-thick phantom 50 cm (300 rad/hr) from the source.

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

Separation of penetrating dose 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 ^{137}Cs using a phantom backing. To convert from exposure in roentgen to dose equivalent index in rem, a conversion factor C_x value of 1.08 was used (DOE 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 INEEL dosimetry service transferred from the DOE Radiological and Environmental Services 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 of the INEEL site indicated that the INEEL contractor and the Idaho Operations Office using 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, but was cross-referenced using many TLD irradiations to the DOE source using a phantom (Anderson 1995). In addition, the contractor developed and characterized its own uranium slab for beta irradiations (Bean 1995).

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

6.3.3.2 Neutron Calibration

The initial NTA neutron badges were calibrated using a PoBe neutron source (30 Ci in 1958) (Horan 1959). In 1982, an AmBe source was used (Cusimano 1982). Alpha particles from the americium or polonium interact in the ^{9}Be(α,n)^{12}C reaction and generate a broad spectrum of neutrons up to about 11 MeV (mean energy about 5 MeV) as shown in Figure 6-8 (Kluge and 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) provides a dose conversion factor for AmBe of 3.8 × 10^{-8} rem-cm^2/n for the maximum average dose equivalent. A dose equivalent of 1.5 rem required 3.6 × 10^7 n/cm^2 (Cusimano 1963), corresponding to a dose conversion factor of 4.17 × 10^{-8} rem-cm^2/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-cm^2/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.

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

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

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

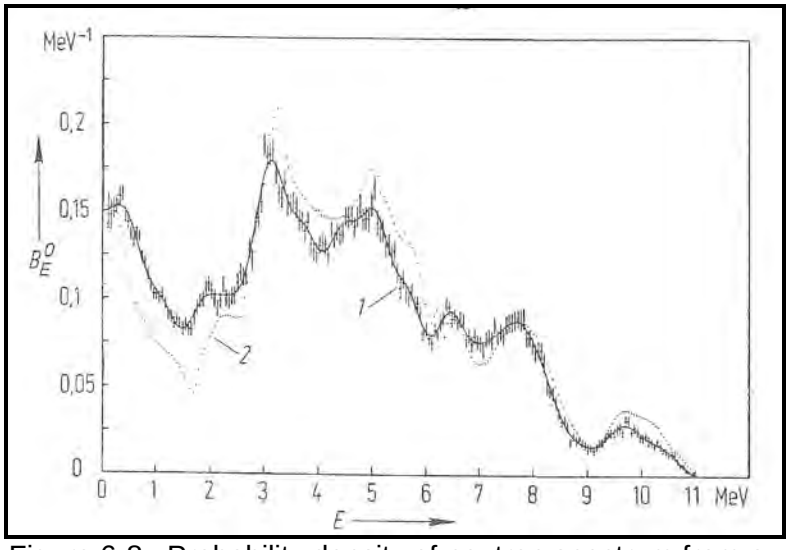


Figure 6-8. Probability density of neutron spectrum from a ^{241}Am-Be (α,n) source (Kluge and Weiss 1982).

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

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

Parameter	Historical description	Anticipated laboratory bias[b]
Source energy spectrum	In 1976, INEEL 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 probably had minimal effect.
Radiation quantity	Neutron dose quantities that were used to calibrate INEEL neutron dosimeters have varied historically. The first collision dose for fast neutrons and a quality factor of 10 was used for many years.	As noted above, NTA calibration would result in the reported dose being **about 11% high**. The effects of the respective neutron dose quantities used to calibrate INEEL dosimeters is uncertain and could be evaluated in comparison to the $H_p(10)$ dose used in DOELAP performance testing.
Angular response	INEEL 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	INEEL NTA film and TLD dosimeters 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. Mid-cycle calibration minimizes the effects.

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

6.3.4 Workplace Radiation Fields

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

The collective dose received by all workers through 1990 is shown in Figure 6-9 taken from Horan and Braun (1993). The highest collective dose was 3448 rem in 1965.

6.3.4.1 Gamma Radiation

In response to a Tiger Team finding, radiation fields at the INEEL 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

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

Year	0%	25%	50%	75%	90%	97.50%	99.50%	100%	MEAN	STD 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	1408
1954	0	0	0	0.21	0.9	2.498	4.077	5.76	0.2838	0.6567	2449
1955	0	0	0.1	0.59	1.6565	3.237	5.162	8.47	0.5157	0.9314	2946
1956	0	0	0	0.2	1.015	2.555	4.267	22.06	0.3142	0.8423	3209
1957	0	0	0	0.085	0.54	1.513	3.045	5.12	0.1706	0.4517	4695
1958	0	0	0.04	0.325	1.26	2.89	4.378	10.51	0.3766	0.8015	5079
1959	0	0	0.04	0.26	0.96	2.347	4.127	21.85	0.3099	0.7402	5344
1960	0	0	0.05	0.255	1.075	2.647	3.734	5.01	0.3268	0.6755	5827
1961	0	0	0.04	0.42	1.704	3.566	4.972	27.26	0.5063	1.2806	5192
1962	0	0	0.03	0.175	1.115	3.225	5.048	9.885	0.3571	0.8942	5339
1963	0	0	0.025	0.18	1.0795	2.954	4.022	5.1	0.318	0.7292	5520
1964	0	0	0.01	0.215	1.2765	3.111	3.98	4.815	0.3538	0.7886	5446
1965	0	0	0	0.43	2.2045	4.39	6.018	9.815	0.579	1.2107	5520
1966	0	0	0.02	0.375	1.5595	3.467	4.466	6.045	0.4383	0.8926	5180
1967	0	0	0	0.17	1.0725	3.084	4.377	4.805	0.3194	0.7702	6304
1968	0	0	0	0.215	1.14	3.104	4.194	5.295	0.3364	0.7781	4922
1969	0	0	0	0.24	1.195	2.73	3.982	4.45	0.3279	0.7151	4758
1970	0	0	0	0.175	1.028	2.624	4.207	4.68	0.2952	0.702	5051
1971	0	0	0	0.17	0.82	1.899	3.252	4.71	0.2357	0.5427	4764
1972	0	0	0	0.161	0.855	2.375	3.835	4.665	0.2606	0.634	4762
1973	0	0	0	0.115	0.6525	2.126	3.909	5.2	0.2185	0.5878	4494
1974	0	0	0	0.1	0.515	1.696	2.993	4.065	0.1734	0.4574	4878
1975	0	0	0.006	0.09	0.405	1.412	2.789	3.945	0.1531	0.4071	5025
1976	0	0	0	0.11	0.506	1.646	2.643	4.145	0.1712	0.4294	5489
1977	0	0	0	0.094	0.485	1.947	3.385	10.77	0.1869	0.526	5677
1978	0	0	0.009	0.087	0.394	1.535	2.872	4.386	0.1563	0.4295	6551
1979	0	0	0	0.074	0.369	1.46	2.732	4.18	0.1419	0.4064	6863
1980	0	0	0	0.047	0.277	1.062	2.112	16.93	0.1073	0.3579	7380
1981	0	0	0	0.046	0.252	0.824	1.674	3.289	0.0876	0.2358	6722
1982	0	0	0	0.038	0.195	0.638	1.5	2.904	0.0715	0.204	6556
1983	0	0	0	0.035	0.187	0.516	0.976	1.577	0.0582	0.1473	6610
1984	0	0	0	0.034	0.179	0.577	1.198	2.285	0.0619	0.1766	7476
1985	0	0	0	0.039	0.219	0.872	1.24	2.415	0.082	0.226	7917
1986	0	0	0	0.024	0.192	0.773	1.749	9.338	0.077	0.2623	8568
1987	0	0	0	0.025	0.155	0.733	1.417	3.158	0.0659	0.2098	8575
1988	0	0	0	0.022	0.145	0.537	1.086	3.086	0.0545	0.1641	8667
1989	0	0	0	0.016	0.101	0.559	1.315	7.811	0.0516	0.2169	8848
1990	0	0	0	0.012	0.0954	0.566	1.266	2.728	0.049	0.1852	10165
1991	0	0	0	0	0.037	0.249	0.694	4.577	0.0232	0.1094	10742
1992	0	0	0	0	0.037	0.169	0.376	1.276	0.0157	0.0563	9571
1993	0	0	0	0	0.065	0.342	1.06	1.535	0.0311	0.1233	9048
1994	0	0	0	0.01	0.0786	0.335	0.729	1.394	0.0322	0.1121	8473
1995	0	0	0	0.011	0.1031	0.449	1.153	1.844	0.0428	0.1533	7818
1996	0	0	0	0.016	0.108	0.421	0.79	1.368	0.0393	0.1215	6459
1997	0	0	0	0.01	0.081	0.259	0.574	1.108	0.0261	0.082	6280
1998	0	0	0	0.002	0.062	0.208	0.462	0.844	0.0205	0.065	5875

results lie within +27% to -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 INEEL, with a few exceptions, are generated primarily by mixed fission and

activation products. Therefore, most of the photon dose is from photons with energy greater than 250 keV. The INEEL dosimeters are judged to measure these fields well.

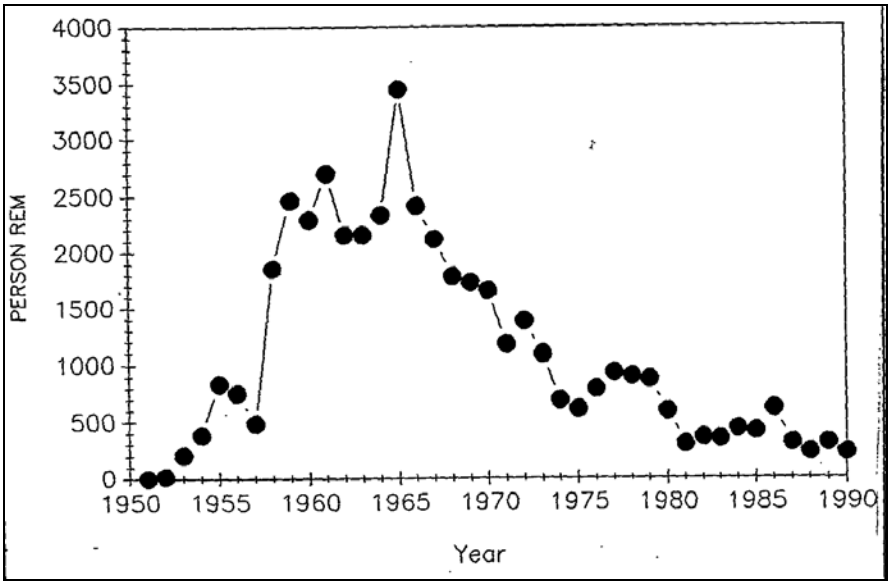


Figure 6-9. Collective dose for INEL personnel from 1951-1990 (Horan & Braun 1993).

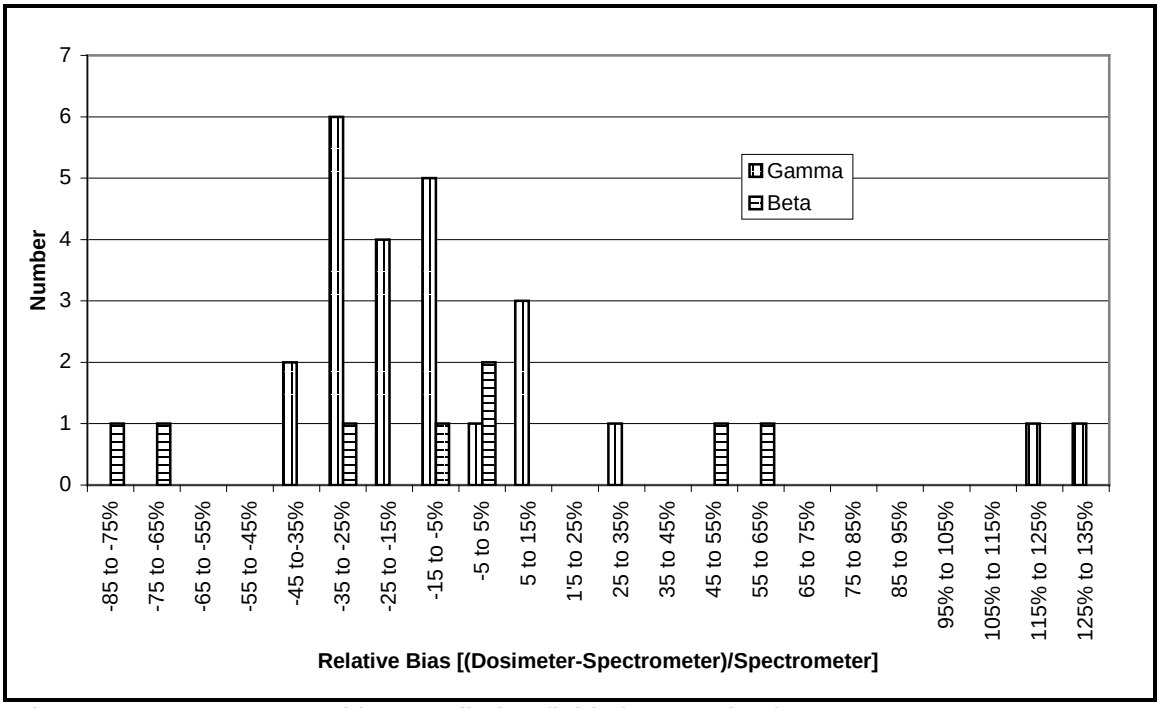


Figure 6-10. Gamma and beta radiation field characterization.

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.

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

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. The radiography facility at TRA has no roof shielding, so dose rates nearby 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 worksite and some from 300-keV ^{192}Ir or 1.25-MeV ^{60}Co.

Essentially all INEEL radiological work areas 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.

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

Process/ buildings	Description	Operations		Radiation type	Energy group (keV)	Percentage
		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			Beta	>15	100
				Photon	30–250	25
					> 250	75
	PBF, TRA, ARA, TAN, EBR, ANL-W, SPERT					
		1952	2003			
Processing plants	Highly dispersed fields of higher energy photon radiation fields from activation and fission product nuclides dominant to most exposure profiles. Potential for higher energy beta radiation during sampling and maintenance work resulting from fission products.			Beta	> 15	100
				Photon	30–250	25
					> 250	75
	ICPP	1952	2003			
Calibrations	Calibration of instruments and dosimeters			Beta	> 15	100
	CF 633, 636	1952	2003	Photon	30–250	25
					> 250	75
Waste handling	Radiation characteristics are highly dependent on source of waste, but typically fission product nuclides (Sr/Y-90, Cs-137) are dominant. For transuranic waste from Rocky Flats contains Am-241 with 59-keV photon.			Beta	> 15	100
				Photon	30–250	25
					> 250	75
	RWMC, WERF	1953	2003			
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.			Beta	> 15	100
				Photon	30–250	90
					> 250	10
	SMC	1985	2003			

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

6.3.4.2 Beta Radiation

Beta radiation fields are usually associated with activation or fission products outside of a container such as in a spill or only lightly shielded or in hot cells. High beta fields were not unusual at the Idaho Nuclear Technology and Engineering Center (INTEC) where large quantities of fission products exist. 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.

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 better than 15%.

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

Based on the range 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 the INEEL, so a correction factor common for all facilities was estimated.

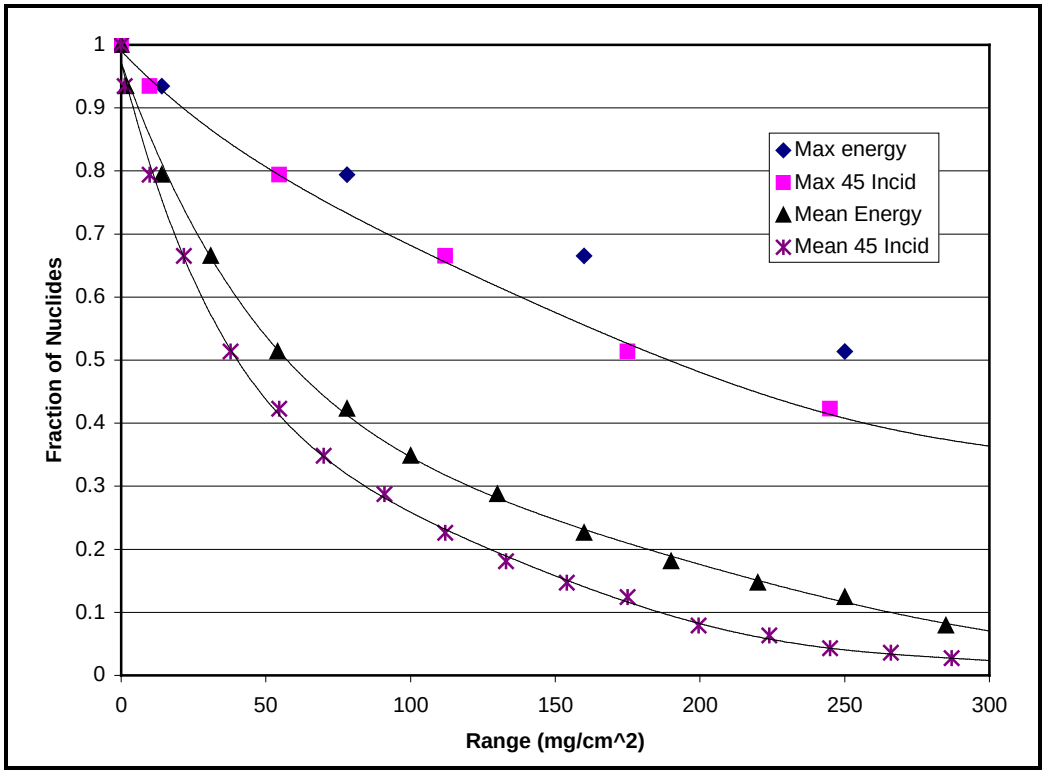


Figure 6-11. Distribution of beta ranges.

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 the INEEL. 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 45º incidence at the maximum energy and the mean energy. These curves of the fraction of nuclides with a larger range essentially show the depth dependence of 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/cm^2, a depth that was chosen in the early 1980s. These curves demonstrate why the beta dose assigned for skin is inappropriate to use for the breast and testes

where much of the organ is at a depth greater than 1 mm or 100 mg/cm^2, and for most of us at depths greater than 1 cm.

To calculate the fraction of dose missed by a dosimeter, the dose reconstructor need only to average the appropriate curve of this nature over the depth of the active detector and compare it to the value at a depth of 7 mg/cm^2. The appropriate curve should be the curve of the mean range for the 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 45º incidence for the mean energy and to add in one-half that value for 45º incidence for the maximum energy. The curves are the result of a polynomial trend line to the data, so averaging the fraction of radionuclides is relatively easy.

Table 6-10 provides the cover and detector thicknesses for the beta badges used at INEEL. 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 non-penetrating result 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. 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.

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 INEEL workers have not been exposed to neutrons and so have not been badged to measure neutrons. Neutron fields at the INEEL 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 time periods where neutron exposure is considered likely.

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).

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

Individuals who have the potential to receive neutron dose currently wear albedo badges, and experience has shown that most do not receive significant doses. In the first 9 months of 1995, 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.

Table 6-11. Facilities and time periods for neutron exposure.

Location	Time 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		TRU 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 & 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

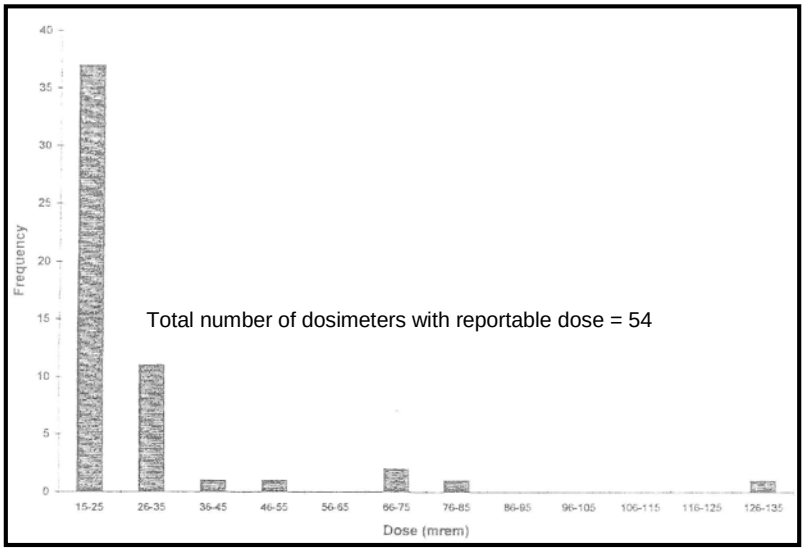


Figure 6-12. Distribution of reportable neutron dose for the first 9 months of 1995. Of 1,461 dosimeters, 1,407 were below the 15 mrem reporting level.

In 1997, several workplace neutron fields were measured with TLDs mounted on a phantom and at nearly the same time a ROSPEC neutron spectrometer (Reilly 1998). The relative biases [(Dosimeter-Spectrometer)/Spectrometer] for the neutron fields are shown in Figure 6-13. These

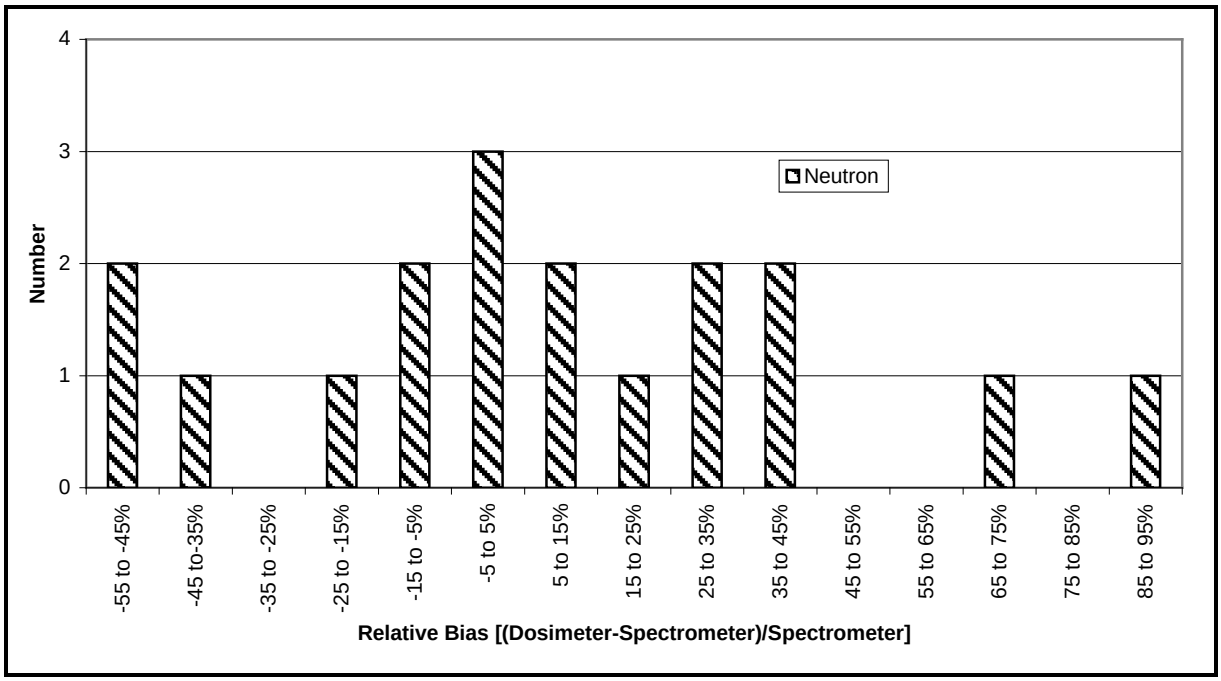


Figure 6-13. Neutron radiation field characterization.

results show a greater dispersion than the gamma results. The two lowest values (-52% and -51%) are for TLD measurements on opposite sides of a phantom where the field is from ^{252}Cf on an overhead filter bank. The phantom attenuates the radiation from each side so the TLDs only see half the radiation field. The next lowest value (-38%) is for the ^{252}Cf instrument calibration source at a distance of 3.5 m where the operator stands. The two highest values (0.94 and 0.71) are for a waste drum that was reanalyzed and a new 9":3" ratio determined because of the unsatisfactory initial result. The report suggests that other waste barrels might have had neutron sources causing interference. The remaining bias values lie between -0.16 and 0.44.

Figure 6-14 provides spectra for the 14-MeV neutron generator as seen through 10 cm of polyethylene shielding typical of the INEEL facilities, ^{252}Cf (fission) neutron source (Ing and Makra

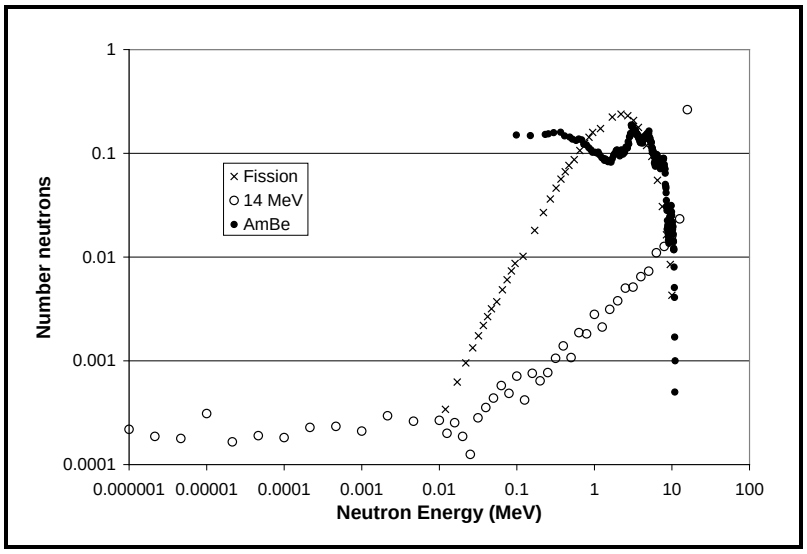


Figure 6-14. Neutron spectra simulating INEEL facilities.

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 works reasonably well. Use of small ^{252}Cf sources for research began after albedo badge use began.

Most of the reactors built at the INEEL had no beam ports. Thus the neutrons were generally well contained away from the workplace. The reactor core environment is characterized not only by high neutron levels, but also by very high gamma levels. The gamma shielding is often water and concrete which are also very good neutron shields. The neutron fields in the energy spectrum for reactors (and lower energy) will be attenuated much more quickly in concrete or water than will the gamma fields. This is not true for lead or iron, but these are usually not used as gamma shields where neutrons also exist. Thus neutron fields are 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 individually do not provide all components of the radiation field. Hankins (1961) used 2-, 3-, and 8-in. polyethylene Bonner balls in a cadmium shield to characterize the intermediate and fast neutrons at 21 locations around the MTR floor and measured the thermal neutron component at six other locations. The Hankins data have been reanalyzed (ORAUT 2004b) 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 (2004) reanalysis of the Hankins data.

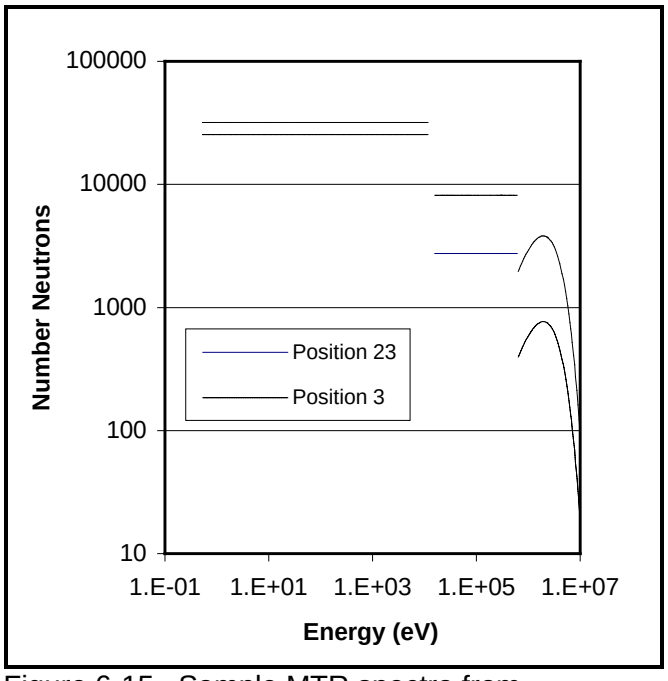


Figure 6-15. Sample MTR spectra from Hankins Bonner measurements.

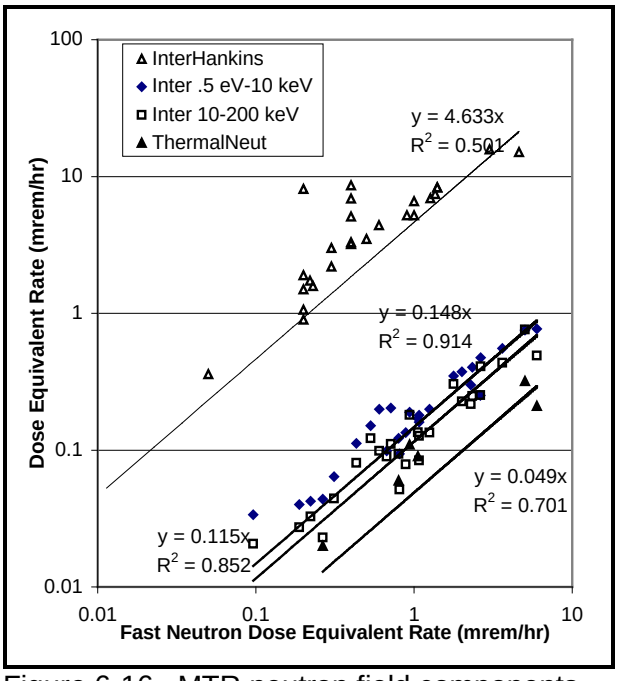


Figure 6-16. MTR neutron field components.

The trend line for the reanalyzed intermediate-energy neutron dose equivalent has an R^2 value 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 neutron is 0.071 ±0.025, for the low-energy intermediate to the fast is 0.177 ±0.057, and for the higher energy intermediate to the fast is 0.149 ±0.046 where fast neutrons are taken as 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 from 35% to 66% depending on the location. 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 ±0.2 (1/0.52, 0.08/0.52) for a Monte Carlo dose reconstruction.

Sommers (1962) reported thermal and fast neutron dose equivalent rates and gamma dose rates around the MTR beam lines. The thermal measurements near beams are believed not to be representative of the general workplace. Figure 6-17 shows the correlation of fast neutron dose equivalent to the gamma dose for these measurements.

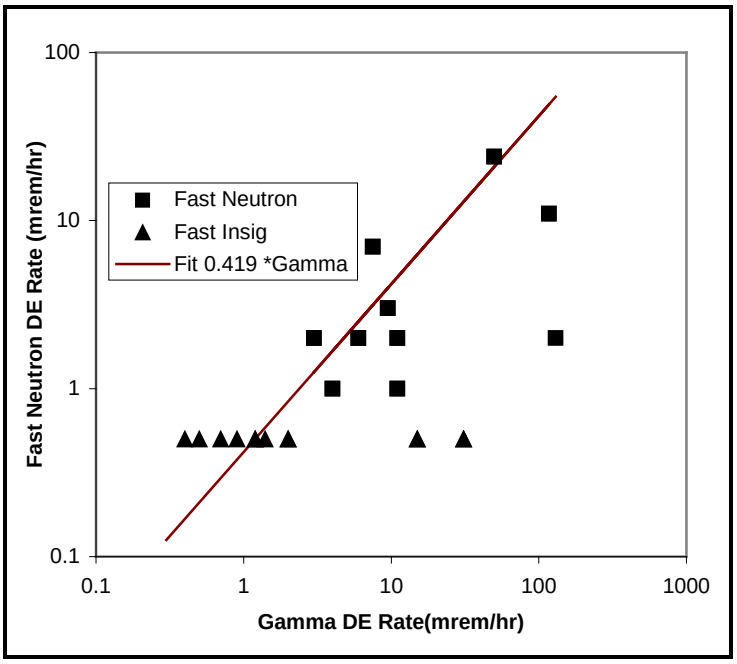


Figure 6-17. Correlation of fast neutron dose equivalent to gamma dose at MTR.

The fast neutron component was insignificant for several of the measurements. These values are shown with the triangles at one-half of the smallest measured value. Using the Shapiro-Wilks Normality Test (Gilbert 1987) and including the insignificant fast neutron values at one-half of the minimum reported value suggests that the normal distribution is a slightly better description of the data than a 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:

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

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.

Many of the people wearing NTA film would receive gamma dose at locations other than on the MTR experimental floor while the reactor was operating. For example, health physics technicians would often have been covering jobs with only beta/gamma fields. A craftsperson might have serviced pumps carrying 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.

6.3.4.5 Test Area North Fuel Storage Casks

As noted in the INEEL 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 wk while temperature measurements were made before they were moved outside onto the storage pad. The casks contain irradiated oxide fuel in which $^{18}O(\alpha,n)$ reaction generates neutrons in the range of a few MeV. 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 office most affected was for an operations person. 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.

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 related to $H_p(10)$ performance of the neutron dosimeters is the difference between calibration and workplace neutron energy spectra.

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Neutron Weighting Factor

All dose equivalents measured at the INEEL and reported in this document used the quality factors based on the LET of the ionizing secondary particles established in the 1950s and used since by U.S. regulatory agencies. In 1990 the ICRP developed new dose concepts that have been used by

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

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

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

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 INEEL reported data requires correction to change from dose equivalent (pre-ICRP 60) to the newer dose quantity (ICRP 1991; NIOSH 2002). ICRP 74 (1996) tabulates the ambient dose equivalent (dose equivalent at 10 mm depth in a 30-cm-diameter sphere) for neutrons. The ratios of organ to ambient dose equivalents are tabulated in the NIOSH *External Dose Reconstruction Implementation Guideline* (NIOSH 2002), 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 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 range of energies considered:

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

Error! Bookmark not defined. These factors are incorporated into statements of dose equivalent values and calibrations following generally accepted principles. The conventional dose conversion factors are most clearly and correctly stated in ICRP Publication 21 (ICRP 1973). NCRP Report 38 tabulates a neutron flux density associated with the annual dose limit that is proportional to the reciprocal of the dose conversion factor (NCRP 1971b). The primary geometry is conventionally considered from one direction with the maximum dose in the body tabulated. More recent references (ICRU 1985; ICRP 1988, 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 work best for an AP irradiation geometry with the dosimeter on the front of the body.

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 for calculating the correction factors.

Table 6-13 summarizes the locations at the INEEL 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 dose equivalent to equivalent dose for INEEL spectra. The ratios of average radiation weighting factor to average quality factor for the IREP energy groups have some variation, particularly for the 10- to 100-keV group where the energy dependence of the fluence is radically different for the fission and 14-MeV source than for the reactor spectrum. The lower part of the table lists the recommended default values for the dose equivalent fractions and quality factor corrections.

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

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

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; Worker Outreach meeting, April 28, 2004). The latest revision of *Use of Coworker Dosimetry Data for External Dose Assignment* (ORAUT 2005b) provides guidance to assist dose reconstructors in evaluating 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 INEEL workers would occur where 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. The missed dose is calculated as

described in NIOSH (2002) using MRL/2 multiplied by the number of zero dose results. Table 6-14 lists the potential missed photon dose 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.

Table 6-14. Beta/photon dosimeter period of use, type, MRL, exchange frequency, and potential annual missed dose.

Period of use[a]	Dosimeter	Exchange frequency	MRL[b] (mrem)		Annual missed dose (mrem)[c]	
			Photon	Beta	Photon	Beta
August 1951–March 1958	INEEL Initial Film 552 DuPont Film	Weekly (n=52)	30	30	780	780
		Monthly (n=12)			180	180
	Reactor Areas DuPont 558 Film	Weekly (n=52)	10	30	260	780
March 1958 – December 1966	INEEL Multi-Element DuPont 508 Film	Weekly (n=52)	10	30	260	780
		Biweekly (n=26)			130	390
		Monthly (n=12)			60	180
December 1966–February 1974	INEEL Multi-Element DuPont 508 Film	Weekly (n=52)	10	30	260	780
		Biweekly (n=26)			130	390
		Monthly (n=12)			60	180
	INEEL 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]	INEEL 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]	INEEL 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	INEEL Panasonic Four-chip TLD	Monthly (n=12)	15[e]	15[f]	90	90
		Quarterly (n=4)			30	30
		Monthly (n=12)	10[e]	30[f]	60	180
		Quarterly (n=4)			20	60

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

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 the INEEL had essentially no separated plutonium, the electron dose is the difference between the shallow and deep doses. All nonpenetrating dose for the 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 will generally be 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.

6.5.4 Missed Neutron Dose

Neutron radiation was present at the INEEL 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 INEEL Research Center, in calibration laboratories CF 633 and 636, and for calibration of criticality alarms at TRA and ICPP. 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 reconstructor must first determine if the person worked near neutrons and which category of neutrons. This can be done best 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 (also 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.

If a worker was likely to have been exposed to neutrons, the reconstructor should assign missed neutron dose equivalent using Table 6-15 for the times when workers did not have reported neutron dose. 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. Then the dose equivalent is apportioned into the IREP groups using Table 6-16.

Table 6-15. Neutron dosimeter type, period of use, exchange frequency, laboratory minimum detectable limit, and maximum annual missed dose.

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) × 14 ÷ 2] so the total dose by the badges would be 500 mrem. Because the badge only sees about one-half 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 6-16, 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. The second period is after September 1976 when only Hankins albedo dosimeters were used. Table 6-14 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 encountered in the workplace and because most of the neutrons to which workers were normally exposed had energies greater than the 500- to 800-keV threshold of the NTA film dosimeters. There was, of course, no threshold energy for the measurements using neutron albedo TLD badges.

Table 6-16. Recommended IREP neutron energy fractions and correction factors.

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 reactors	Experiment floor and adjacent rooms during operation	1953	1970	< 10 keV	20	1	0.20
				10 -100 keV	5	1.1	0.06
				0.1-2 MeV	50	1.4	0.7
				2-20 MeV	25	1.1	0.28

6.5.4.1 Before October 1976

The use of NTA films for neutron dosimetry before 1976 is documented in various INEEL reports (Cusimano 1963; Cipperley 1958, 1968). As noted above, it is possible to estimate the missed dose using the MRLs. There are many recorded zeros in the neutron dose data for INEEL workers for two reasons: (1) An NTA film was developed and not read per standard criteria, or (2) an NTA film indicated a neutron dose equivalent that was less than the film's 14-mrem MRL. 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.

An estimate of the missed neutron dose in some facilities could also be attainable through use of neutron-to-photon dose ratios (NIOSH 2002). However, for the INEEL facilities there are several other sources of gamma exposure with no associated neutron exposure, so that approach would be erroneous.

6.5.4.2 After September 1976

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

For a couple months in the mid-1980s a few office workers without neutron dosimetry were exposed to neutrons at TAN (see Section 6.3.4.5). For these individuals, 50 mrem of neutron dose should be assigned for each period of exposure.

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

6.7 UNCERTAINTY

Measurement uncertainties arise from many sources. A lognormal distribution should generally be assumed unless stated otherwise. For gamma rays, the standards for exposure have existed with only minor changes from the 1930s as required for medical uses of radiation. The INEEL used ionization chambers standardized by NBS (now the National Institute of Standards and Technology) for their calibrations. Use of a phantom for dosimeter irradiation began in the early 1980s, but backscatter only causes a minor change for high-energy photon dosimetry. The 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. The INEEL 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. 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. This is driven by uncertainties in the field geometry and in the fact that beta radiation is often stopped by thin materials such as clothing and air. Algorithms are used to estimate the dose at a depth of 7 mg/cm^2 from dosimeters at depths of 15 to 250 mg/cm^2, and such depth differences can change the signal significantly. The difference between a point- and planar-source irradiation can confuse an algorithm. Earlier techniques did not provide a thin detector with minimal covering, which is important for simulating the skin for beta dosimetry.

For neutron radiations, the situation is more complex. The NTA films used before 1975 did not react to low-energy neutrons below 0.5 to 0.8 MeV. Corrections are described for handling this issue. The TLD albedo system provides a very indirect way of measuring dose equivalent to a person. Dose to workers is primarily due to hydrogen recoils rather than the ^{6}Li(n,α) reactions. 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 GSD of about 1.6. The uncertainty of the base curve for calculating FNCF is a GSD of about 1.35 (Gesell et al. 1996) although this uncertainty does not contribute in the DOELAP process, resulting in a total GSD of 1.7 for albedo neutron measurements.

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. 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). Note

that in Figure 6-12 the discrepancy in which the dosimeters are reporting about one-half of the spectrometer result is because the spectrometer does not simulate the attenuation of the body, so it read high by a factor of two.

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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.

DOELAP
The DOE Laboratory Accreditation Program (DOELAP) accredits DOE site dosimetry programs based on performance testing and onsite reviews performed on a 2-yr cycle.

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

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

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

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

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

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

non-penetrating 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 INEEL film or elements plus neutron dose equivalent. Essentially, personal dose equivalent $H_p(10)$.

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

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

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

tissue rad
Absorbed dose in tissue.