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PUBLICATION RECORD

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ANSI	American National Standards Institute
AP	anterior-posterior
ARA	Auxiliary Reactor Area
ATLAS	Automatic Thermoluminescent Analyzer System
ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory - West
CFA	Central Facilities Area
Ci	curie
cm	centimeter
DOE	U.S. Department of Energy
DOE-ID	DOE Idaho Operations Office
DOELAP	DOE Laboratory Accreditation Program
EDF	Engineering Design File
EG&G	Edgerton, Germeshausen, and Greer Corporation
ERDA	Energy Research and Development Administration
EBR	Experimental Breeder Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
FNCF	facility neutron correction factor
g	gram
hr	hour
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
ID	(DOE) Idaho Operations Office
in	inch
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
IREP	Interactive RadioEpidemiological Program
kerma	kinetic energy released to matter
keV	kilovolt-electron, 1,000 electron volts
LET	Linear Energy Transfer
LiF	lithium fluoride
MeV	megavolt-electron, 1 million electron volts
mg	milligram
mm	millimeter
mR	milliroentgen
mrad	millirad

mrem	millirem
mrep	millirep
MRL	Minimum Reporting Level
MTR	Materials Test Reactor
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurement
NIOSH	National Institute for Occupational Safety and Health
NRTS	National Reactor Testing Station
NTA	nuclear track emulsion, type A
NVLAP	National Voluntary Laboratory Accreditation Program
PoBe	polonium beryllium
POC	probability of causation
R	roentgen
RBE	relative biological effectiveness
rem	roentgen equivalent in man
rep	roentgen equivalent physical
RESL	Radiological and Environmental Services Laboratory
RWMC	Radioactive Waste Management Complex
SL-1	Stationary Low-Power Reactor No. 1
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor Test
U.S.C.	United States Code
wk	week
yr	year
Z	atomic number
ZPPR	Zero Power Physics Reactor
§	Section

6.1 INTRODUCTION

6.1.1 Purpose

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

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

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of 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

6.1.2 Scope

From the start of operations in 1951 until early 2005, Argonne National Laboratory – West (ANL-W) was operated by the University of Chicago under supervision of the Chicago Operations Office of the U.S. Atomic Energy Commission (AEC), the Energy Research and Development Agency (ERDA), and DOE. In 2005, the functions of ANL-W were combined with research activities at the former Idaho National Engineering and Environmental Laboratory (INEEL) to form the Idaho National Laboratory (INL), and ANL-W ceased to exist. A branch of the Idaho Operations Office (DOE-ID) provided external dosimetry resources and services at the INEEL, including ANL-W from the start of operations in 1951 [when it was called the National Reactor Testing Station (NRTS)] until 1989, when DOE transferred that responsibility to the prime operating contractor. Despite the fact that INEEL had several contractors at a time and that contractors changed often, the external dosimetry process has remained under technical management of a single organization with responsibilities for dosimetry development, operational dosimetry, and radiological records, which has provided a stable external dosimetry system. The former INEEL was originally called the National Reactor Testing Station (NRTS) and then the INEL (Idaho National Engineering Laboratory), but for simplicity, references will be to the INEEL.

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

6.2 BASIS OF COMPARISON

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

Horan and Braun (1993), Attix and Roesch (1968), and Meinhold (1975) discuss the history of radiation protection requirements from the 1930s. 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 NCRP issued NCRP Report 7 as National Bureau of Standards Handbook 42, which recommended a permissible dose of 0.3 R/wk (15 R/yr) for occupational workers (NBS 1949, p 6). The term dose was undefined. A *roentgen* (R) was defined as the quantity or dose of X-rays such that the associated ionization per 0.001293 g of air (1 cm^3 at standard temperature and

pressure) produces 1 electrostatic unit of charge of either sign. The unit roentgen only applies to X-rays or gamma rays, whereas the rep was used for all ionizing radiation and is the energy absorbed by tissue from the radiation (93 ergs/g, the same as for 1 R of gamma radiation).

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

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

In 1957, the NCRP introduced an age prorating formula for the maximum allowable dose of 5 rem/yr 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-week whole-body limit of 3 rem with "the provision that not more than one-fourth of the 15 rem maximum permissible yearly dose shall be taken in one-fourth of a year" (AEC 1958).

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

In 1957, National Bureau of Standards (NBS, now the National Institute of Standards and Technology) Handbook 63 specified a dependence of the 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 unit kerma (kinetic energy released in material) in 1962 (ICRU 1962).

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

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

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

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

In 1981, DOE Order 5480.1A, Chapter XI, "Requirements for Radiation Protection" (DOE 1981a), superseded ERDA Manual Chapter 0524 (ERDA 1977). In 1988, DOE Order 5480.11, "Radiation Protection for Occupational Workers" (DOE 1988) superseded DOE Order 5480.1A, Chapter XI. This 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, leading 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 ANSI (HPS 1983b) NVLAP processes. Calibration was to the quantities shallow and deep dose equivalent (H_s and H_d) and shallow absorbed dose (D_s), which are similar to the individual dose equivalent superficial and individual dose equivalent penetrating dose defined in ICRU (1985). These quantities were renamed to the personal dose equivalent $H_p(d)$ (ICRU 1993) where d is the depth in millimeters (0.07 mm for surface and 10 mm for deep) from the surface for which the dose is measured. In 1987, DOE Order 5480.15, "Department of Energy Laboratory Accreditation Program for Personnel Dosimetry," established the DOE Laboratory Accreditation Program (DOELAP) system for dosimetry accreditation (DOE 1987). *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 (2006), depend on neutron energy at the entrance to the body rather than on secondary particle LET where the dose is received. Dose conversion factors for organs and for ambient dose equivalent and personal dose equivalent were generated in ICRP Publication 74 (ICRP 1996) and are referenced in the external dose implementation guide (NIOSH 2006).

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

6.3 DOSE RECONSTRUCTION PARAMETERS

6.3.1 Site Administrative Practices

It was ANL-W policy that personnel expected to receive any radiation dose or personnel whose work was centered at the site were assigned a radiation monitoring badge. These badges were usually stored at the respective operational area entrance security gates for ANL-W facilities. Control badges, which are used to subtract background radiation, have also been and are currently located where the badges are stored (Ruhter 2003). This practice could lead to subtracting environmental radiation from site activities, which would reduce the reported doses (ORAUT 2003). Environmental radiation levels have been monitored for most of the life of ANL-W, originally with film badges and later with thermoluminescent dosimeters (TLDs) (ORAUT 2004). Table 4-6 in the ANL-W environmental dose TBD (ORAUT 2004) lists results of this monitoring at facility fenceline locations near the security gates. A fraction (2000/8766) of these values can be added to an individual dose history or used for unmonitored workers working at the site.

Some individuals who could occasionally have visited site facilities but did little work with radiation had badges at several different facilities [1]. It is not appropriate to base missed doses on the multiple badges issued. Early on, the badge change frequency was not the same for everyone. Workers with low probability of exposure were placed on a longer change cycle than those with more chance of exposure (Cusimano 1972). Therefore, missed doses should be based on the actual change frequency for a person, which can be determined from the individual's data package.

The INEEL dosimetry organization developed a set of basic administrative practices in 1951, which have changed somewhat as the technologies of ionizing radiation dosimetry and recordkeeping have changed (Cipperley 1958).

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

From 1951 to 1958, the INEEL dosimetry staff recorded each worker's dose each day on a dose card (Figure 6-1), rezeroed pencil ionization chambers worn by workers, and entered the weekly badge result on the same card (Cipperley 1958). On this sample, on October 28, November 16, and December 9, 1954, the badges were pulled and read in response to high pencil chamber readings. The personnel monitoring badges have always been considered more reliable than pencil dosimeters; so after the film badge results became available, the daily pencil readings were no longer considered doses of record. However, these values can be recovered from the earliest forms for a worst-case estimate of dose. In Figure 6-1, the pencil readings totaled 815 mR [255+280 (Oct 16)+280 (Nov 9)] and the badges reported 0 mR 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 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.

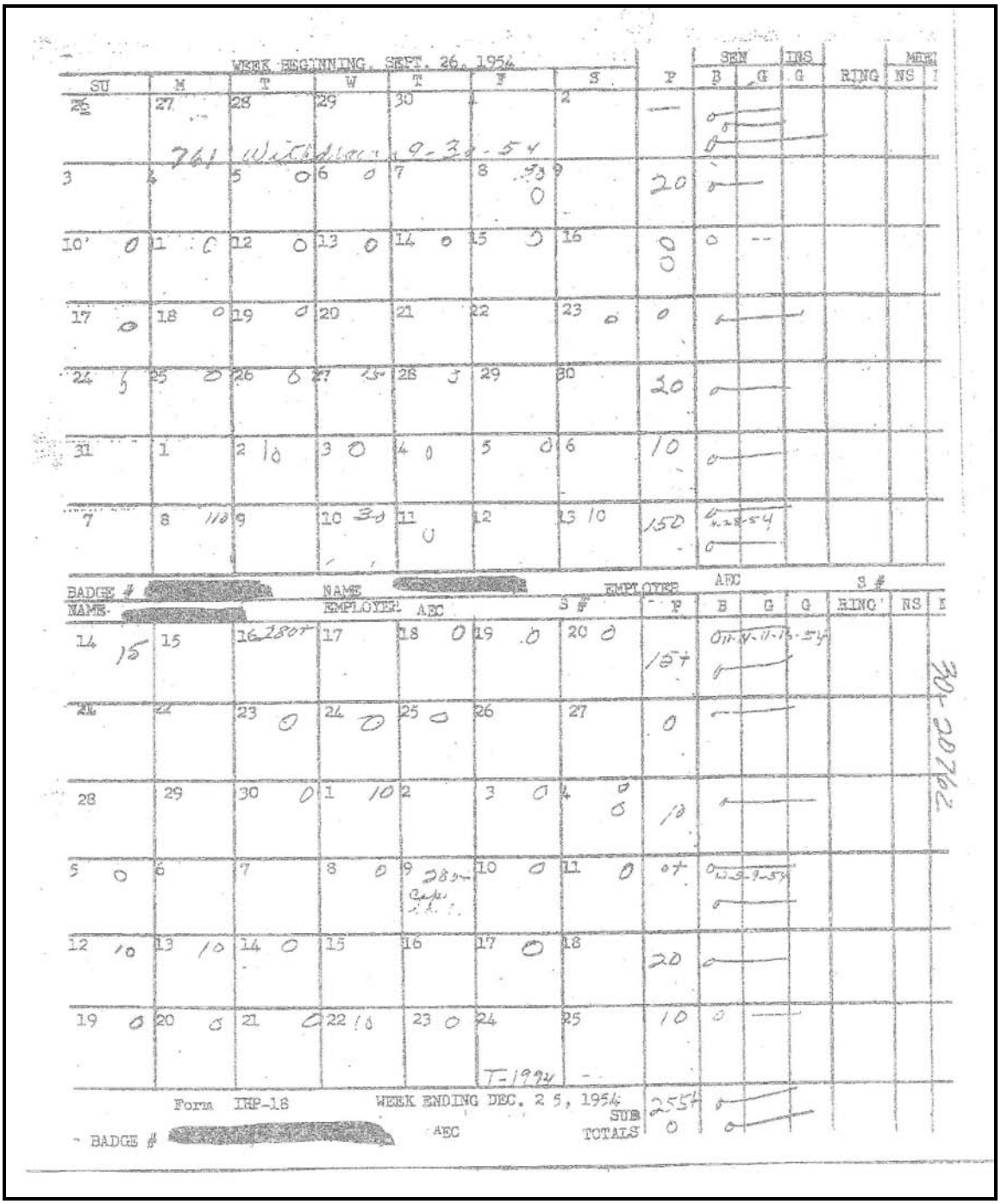
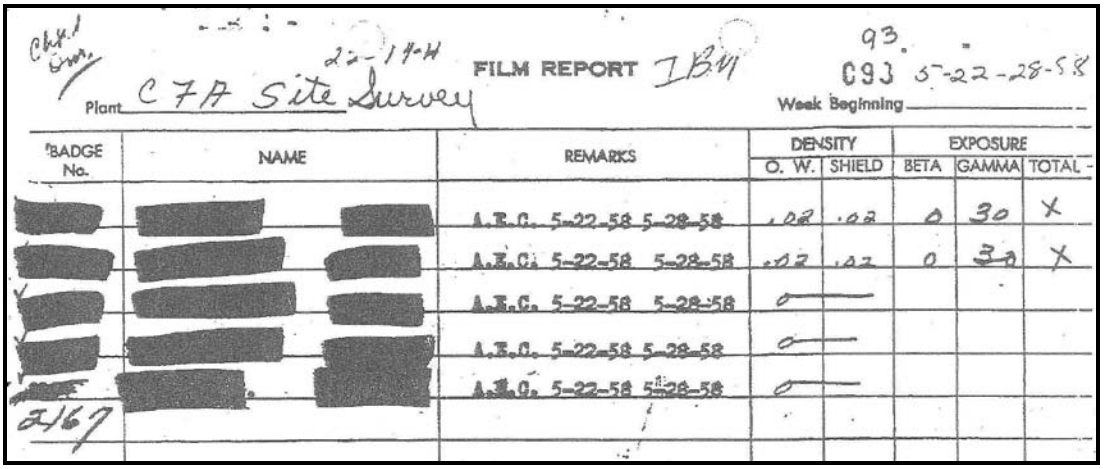


Figure 6-1. Individual dose reporting form in use until 1958 (Vivian and Rockhold 2003, p 17).

After the pencil ionization chambers were replaced with self-reading pencil dosimeters (also ionization chambers), the ANL-W operational health physics staff would rezero the dosimeters (Horan 1959). The film reading was automated and results were stored in a computer. The form shown in Figure 6-3 reports badge readings for May 1959 when badges were exchanged every 2 weeks. The column under the P of Personnel is an area designator with the code listed under Location at the bottom of

the page (Table 6-1 lists the codes for the areas). The column to the right was unused and dropped somewhat later. The next column to the right contained the reason codes listed in Table 6-2. Computer input card codes are listed in Tables 6-3 and 6-4.



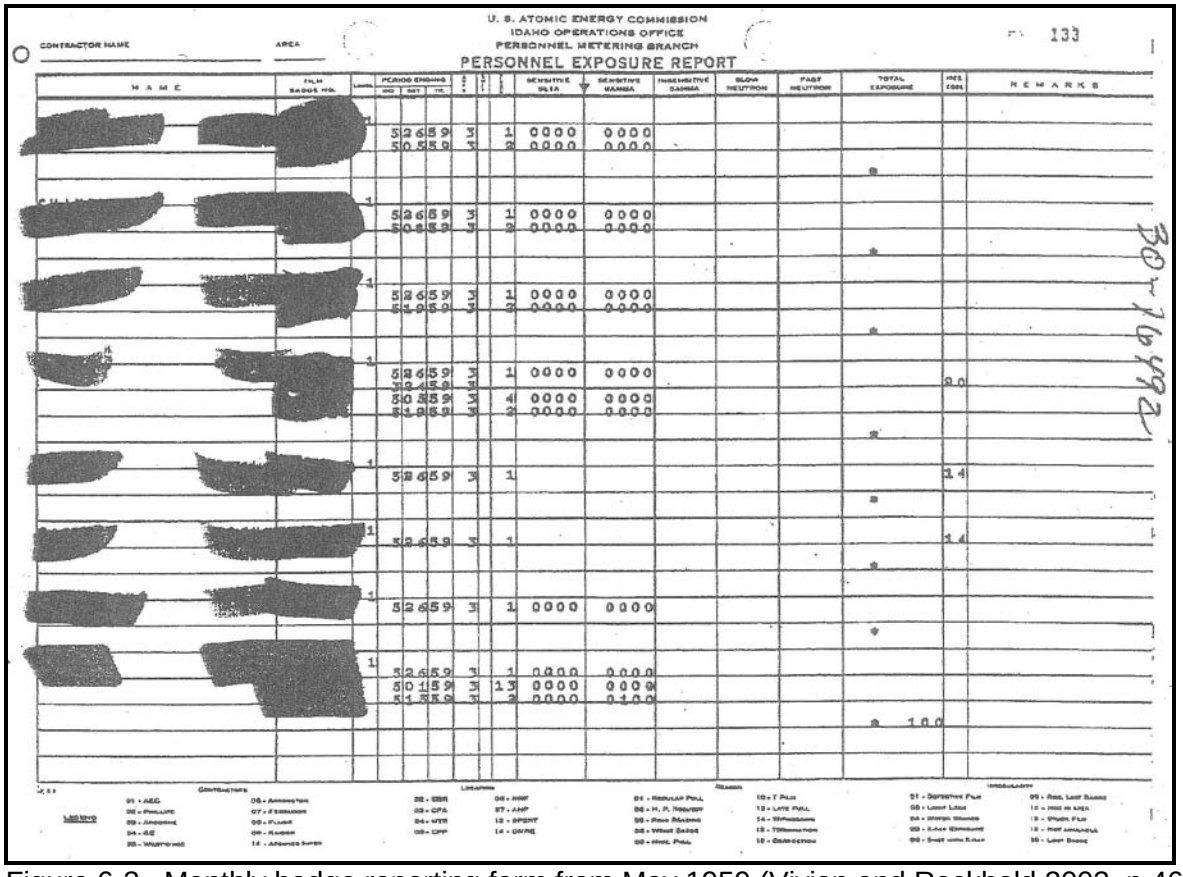
FILM REPORT IBM

Plant C7A Site Survey

Week Beginning 5-22-28-58

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

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



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

PERSONNEL EXPOSURE REPORT

CONTRACTOR NAME ____ AREA ____

Figure 6-3. Monthly badge reporting form from May 1959 (Vivian and Rockhold 2003, p 46).

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.

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

Area code	Description	Area code	Description
1	AEC Headquarters Bldg	20, 261, 264	TREAT
2	EBR – I	25	No information available
12	EX (X is construction) at EBR	26, 263,265	EBR -II
16	SL-1		

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) (Hill-Ellis 2003).

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

Table 6-3. Irregularity codes (columns 70-71) (Hill-Ellis 2003).

01	Defective Film	12	Dropped in Canal or Reactor
02	Impossible to Read	13	
03	Light Leak	14	Not in Area
04	Water Soaked	15	
05		16	
06	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 (Hill-Ellis 2003).

“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 Stationary Low-Power Reactor No. 1 (SL-1) accident. Workers from several areas were pulled into the accident recovery process, and it is notable that one result exceeds the dose limits and that there are few zeros. Figure 6-5 is a follow-up badge report for one result on Figure 6-4.

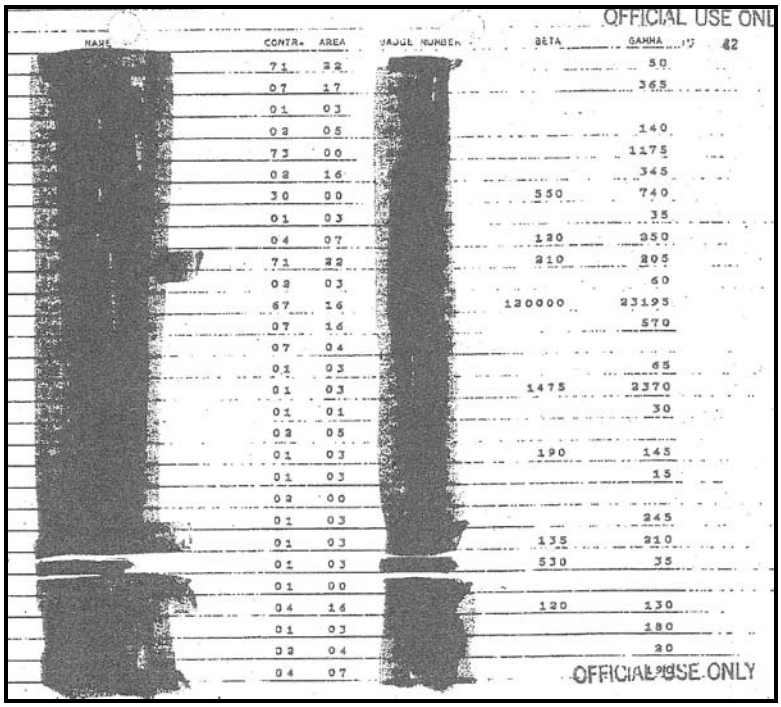


Figure 6-4. Badge pull results from January 1961 for work in recovery from the SL-1 accident (Vivian and Rockhold 2003, p 50).

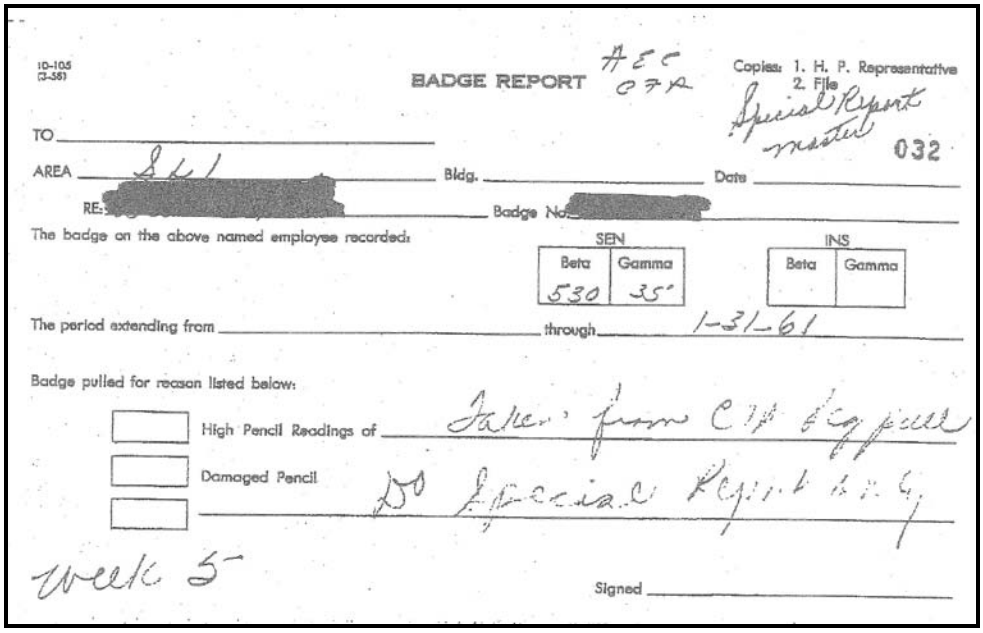


Figure 6-5. Special badge report associated with a high beta reading listed in Figure 6-4 (Vivian and Rockhold 2003, p 57).

When there has been a question about a dose value being assigned to an ANL-W worker, a Personnel Exposure Questionnaire (PEQ) was normally initiated (Figure 6-6 shows a hypothetical case) (Cipperley 1958). Based on this form, a beta dose of 500 mrem and a gamma dose of 350 mrem for a total dose of 850 mrem would override the pocket meter dose of 290 mrem total.

PERSONNEL EXPOSURE QUESTIONNAIRE

Date 1-5-58

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

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

Reason for investigation.

( ) A reportable weekly daily pocket meter reading total of _____

(X) Weekly film total of 300 mr or more.

( ) _____

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

FILM RESULTS

BETA	GAMMA
500	350

EXPOSURE RESUME

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

Remarks Total 850 mrem

Investigation

a. Findings of Health Physics Representative and or Supervisor.

b. Recommendations.

Investigated by _____ Date _____ Noted _____

Health Physics Supervisor

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

6.3.2 Personnel Monitoring Systems Used at INEEL/ANL-W

6.3.2.1 Initial Film Badge

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

was used for beta-gamma dosimetry for most locations; DuPont Type 558 film (Type 508 sensitive and Type 1290 insensitive) was used at two reactor areas (Cipperley 1958).

Gamma calibration was to a radium source, and beta calibration was to a metallic uranium plate. To determine doses, the film densities were read to ±0.02 density unit (Cipperley 1958). A calibration curve was used to convert the cadmium-shielded portion to penetrating gamma exposure in roentgen. The 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 (Cipperley 1958).

Type 552 and 508 films have Minimum Reporting Levels (MRLs) of approximately 30 and 10 mR, respectively, to radium gamma radiation (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 (Cipperley and Gammill 1959). 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 and becomes less than 1 at about 50 keV. The beta particle range is independent of Z; the range depends only on the density (BRH 1970, p 29), so the 1-mm cadmium filter (~900 mg/cm^2) acts like a tissue depth of 9 mm for beta radiation.

Wrist badges used the same package attached to a wrist band. A finger ring used a small piece of film with a silver or cadmium filter (Cipperley 1958). Pencil ionization chambers were used to monitor daily doses and control operational activities. The dosimetry group read and recorded these pencil readings on cards. Film badge readings were written on the same cards to indicate the dose of record (Cipperley 1958). In 1958, the Victoreen 352 pencil ionization chambers being read by the dosimetry group were replaced with self-reading dosimeters that were read and rezeroed by the field health physics technicians (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, resulting in an absorber thickness of 100 mg/cm^2 in the open window, which filtered out beta radiation below 360 keV.

With the four absorbers, it was possible to separate beta radiation from low-energy photon radiation and to determine photon energy to a degree. Figure 6-7 shows photon energy dependence of the darkening behind the four filters for a combination of uranium beta and X-ray irradiation provided by NBS (Cipperley and Gammill 1959). With DuPont Type 508 film, mixed exposures of radium gamma and uranium beta of 10, 20, and 30 mR or mrep were measurable within ±12 mR with 95% confidence. A minimum reporting level of 10 mrem was used for both beta and gamma radiation (Horan 1963).

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

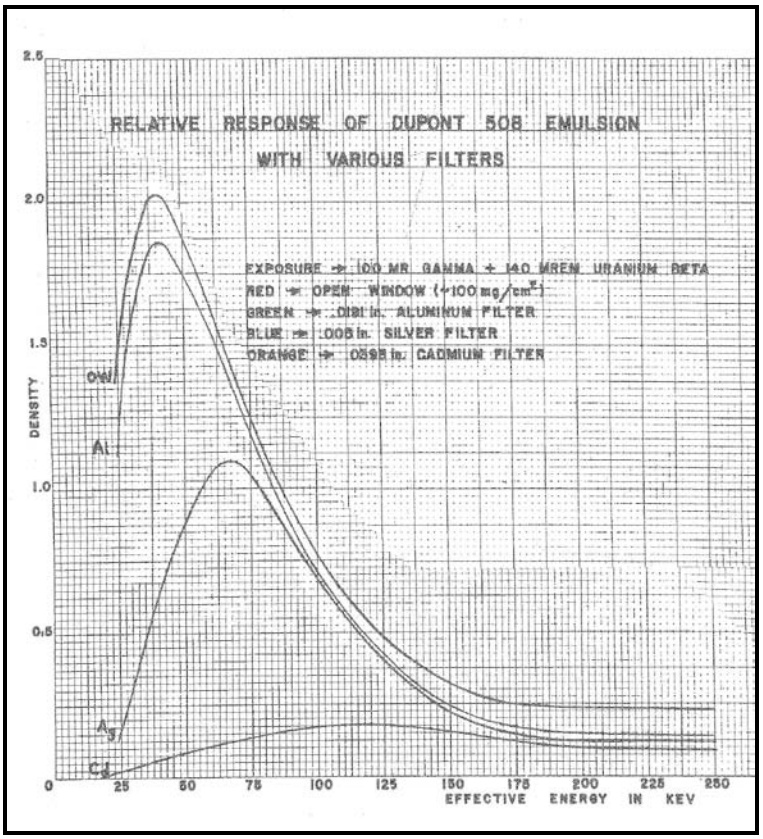


Figure 6-7. Response of DuPont Type 508 film with various filters to 140 mrem uranium beta and 100 mR of different energy photon irradiation provided by NBS. The original badge used the open window and cadmium-shielded films. The multiple filter badge used all three filters plus the open window (Cipperley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

Because about 95% of the weekly badge films had doses less than 30 mrem, in 1958, the badging interval was increased to biweekly or monthly with the exception of the high-dose areas where the weekly schedule continued (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 Stationary Low-Power Reactor No. 1 (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-to-gamma ratio for each particular entry into SL-1 during recovery operations (Cipperley, Henry, and Cusimano 1965).

6.3.2.3 Original Lithium Fluoride Teflon TLD System

Beginning in November and December 1966, individuals projected to receive doses of less than 0.5 rem/yr were placed on 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, date unknown). The disks, manufactured by Teledyne Isotopes, were read with the Teledyne Model 7300 TLD reader. LiF was chosen because the average Z is close to that of air and tissue, resulting in little energy correction for beta or gamma radiation. The badge could read 30 mR on a quarterly basis (Watkins, date unknown), so more small doses were reported (Cusimano and Cipperley 1968). The angular dependence of the gamma response (within 10% to 70º) is superior to film because the material acts like an ionization chamber. For normal monitoring, only the open-window TLD was read and considered penetrating dose unless it read more than 125 mrem, in which case the shielded TLD was also read (Watkins, date unknown).

The pilot tests were successful, and the LiF Teflon TLD system was phased into use in 1966, particularly for individuals who would receive low doses, with longer exchange cycles, typically 3 or 6 months (Vallario, Hankins, and Unruh 1969). 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, which was used for testing (Cusimano and Cipperley 1968).

6.3.2.4 Automatic Thermo Luminescent Analyzer System

Development began in 1969 on a patented Automatic Thermoluminescent Analyzer System (ATLAS). It used LiF in a homogeneous mixture with Teflon and replaced the film in the multielement badge using the same filters. ATLAS became operational for monthly badge changes in February 1974. In June 1974, questions about this system were formalized (Black 1974; Walker 1974) and the system was soon replaced. 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 with ATLAS led to rapid implementation of a two-chip TLD system in May 1975 for ANL. 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 (initially 350 mg/cm^2) of aluminum and the other was under 4 mg/cm^2 of Mylar. The aluminum-covered chip provided penetrating dose at a nominal tissue depth of 1 cm. The beta dose was calculated from the difference between the two chips. Because of the thickness of the Mylar-covered chip, the beta dose was accurate only for the beta energy used in calibration. Field calibrations were used to reduce the problem with beta energy dependence. The thin aluminum filter (density thickness 350 mg/cm^2) allowed higher energy beta radiation to expose the chip used for measuring the penetrating (1,000 mg/cm^2) dose, so it was changed to 2 mm in July 1977 (ERDA ca. 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 were read and the penetrating and nonpenetrating doses were computed (Kalbeitzer 1983).

6.3.2.6 Panasonic Four-Chip System

In 1986, with the advent of DOELAP, INEEL went to a four-element system, the Panasonic 814 AS4 (Gesell, Hall, and Andersen 1992; DOE 2001). Lithium borate ($Li_2B_4O_7$) TLD elements with plastic and aluminum filtration provide an improved measurement of deep dose equivalent and, with a thinner filter, an improved measurement of the shallow dose equivalent. A calcium sulfate ($CaSO_4$) TLD

provides a strong low-energy photon response. The elements are 15 mg/cm^2 thick. Element one has filtration of 16 mg/cm^2, element two has filtration of 58 mg/cm^2 plastic, and elements three ($CaSO_4$) and four have filtration of 550 mg/cm^2 of plastic and 50 mg/cm^2 of aluminum. Although none of the elements are at a depth of 7 mg/cm^2, the specified depth for the shallow dose equivalent, an acceptable response can be obtained by using elements at 16 and 58 mg/cm^2. This system is qualified in the DOELAP beta, photon, and mixture performance categories. The angular dependence of this system has been measured for ^{137}Cs gamma rays and four X-ray energies from 16 to 70 keV (INEEL 2001). For the lower energies, attenuation in the absorbers reduces the measured dose. (The attenuation also reduces the delivered deep dose, but this effect is not incorporated in the results.)

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

6.3.2.7 Nuclear Track Emulsion, Type A for Neutrons

Kodak nuclear track emulsion, Type A (NTA) was used for neutron monitoring when the field radiation protection staff requested it. NTA responds to neutrons with energies above 500 to 800 keV, for which the proton recoil tracks leave enough energy to expose at least four (three in some references) grains of emulsion (Cusimano 1963, Cipperley 1968).

The minimum dose assigned was 14 mrem (Cipperley 1958, p 6). Before 1958, if a proton recoil track was counted in 40 microscope fields, it was read twice more for a total of 120 fields (Cipperley 1958, p 17). 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 (Vivian and Rockhold 2004, p 5). These values indicate that the data sheets support the MRL of 14 mrem.

After 1959, if more than three proton recoil tracks were counted in 40 microscope fields, a different location on the film was counted by two more technicians, providing three independent results (Cipperley 1968). Two tracks or fewer were attributed to background. This resulted in a somewhat higher MRL. In November 1959, a data sheet shows 10- and 20-mrem neutron dose equivalents (Vivian and Rockhold 2004, p 15). In January 1962, a data sheet shows a 20-mrem dose (Vivian and Rockhold 2004, p 6). A data sheet from April 1959 shows neutron dose equivalents of 20, 20, and 40 mrem (Vivian and Rockhold 2004, p 3). These values suggest an MRL of 20 mrem.

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

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

6.3.2.8 Neutron Albedo Dosimetry

Because of the missed dose from neutrons below the NTA energy threshold of 0.5 to 0.8 MeV, particularly at plutonium facilities, and because of the advent of TLD techniques, several development efforts in neutron dosimetry occurred in the early 1970s. The result was several designs using the albedo technique in which scattered neutrons from the wearer's body were absorbed by ^{6}Li in a TLD (Gesell et al. 1996).

In the Hankins dosimeter used at the INEEL and ANL-W (Hankins 1973), TLDs (^{6}Li to capture thermal neutrons) are inside a polyethylene case (to lower the neutron energy) inside a cadmium shell (to eliminate thermal neutrons from outside). Lithium-7 TLDs are used to subtract the gamma dose (Gesell et al. 1996). Because the ^{6}Li(n,α)^{3}He reaction has a strong energy dependence, the response does not follow the flux-to-dose-equivalent conversion, so the neutron signal is divided by a facility neutron correction factor (FNCF) (Gesell et al. 1996). An FNCF that converts the TLD gamma equivalent signal to neutron dose equivalent can be generated from the ratio of the dose equivalent measured with a 9-in.-diameter Eberline PNR-4 and the corresponding signal (reads in mrem, 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). The FNCF values listed in Table 6-5 were measured for different fields at INEEL, were tabulated for assigning the dose equivalent from the badge results, and were routinely updated (Cusimano 1981). 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. INEEL facility neutron correction factors from 1981 (Cusimano 1981).

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

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

6.3.3 Calibration

6.3.3.1 Beta-Gamma Radiation

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

Gamma calibration initially used a radium source. Victoreen R meters standardized by NBS were used to measure radiation fields (Horan 1959, p. 132). Uranium metal bars 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 did not use a phantom to provide backscatter (Cipperley 1966).

Table 6-6. Laboratory sources of uncertainty for beta/photon dosimeter calibration parameters [2].

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 photon beam measured as *exposure*.	-5%	For higher energy ^{226}Ra and ^{137}Cs gamma radiation used to calibrate dosimeters, this caused slight (about 3%) **under-response** in recorded dose.
Tissue depth of dose	Historically, INEEL used unspecified depth to estimate deep dose.	±5%	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 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 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. Midcycle calibration minimizes effects.

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

As reported in 1981, an extrapolation chamber was built for the measurement of beta doses (Gupta 1981). The chamber window was polycarbonate, the gas was air, and the thick collecting electrode was Shonka tissue-equivalent plastic. The chamber was used to calibrate a 2.5 Ci ^{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 (Gupta 1981).

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

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

In 1989, the INEEL dosimetry service transferred from RESL to Edgerton, Germeshausen, and Greer Corporation (EG&G), Inc., the prime contractor. Calibrations continued to use DOE RESL sources, and no changes were made to the dosimetry system. The 1991 Tiger Team Review of the INEEL site indicated that the joint use by the INEEL contractor and DOE-ID of the same sources for calibration led to a conflict of interest or an advantage in DOELAP tests (DOE 1991, p 4-178). As a result, EG&G purchased a Shepherd panoramic irradiator with a ^{137}Cs source for badge irradiations. This irradiator does not use a phantom, but was cross-referenced using many TLD irradiations to the DOE source using a phantom (Andersen 1994). In addition, the contractor developed and characterized a uranium slab for beta irradiations (Bean 1995).

6.3.3.2 Neutron Calibration

Table 6-7 lists 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 [3].[a]

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

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

The initial NTA neutron badges were calibrated using a PoBe neutron source (30 Ci in 1958) (Horan 1959). In 1982, an AmBe source was used (Cusimano 1982). Alpha particles from the americium or polonium interact in the $^{9}Be(\alpha,n)^{12}C$ reaction and generate a broad spectrum of neutrons up to about 11 MeV (mean energy about 5 MeV) as shown in Figure 6-8 (Kluge and Weise 1982). The yield of the AmBe source should be only about 3% larger than that for the PoBe source (Anderson 1971). Kluge and Weise (1982) calculate conversion factors of 3.51 to 3.76 × 10^{-8} rem cm^2/n 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 will be about 11% high. Monte Carlo calculations for 5-MeV neutrons show a dose equivalent of about 4.2 × 10^{-8} rem cm^2/n averaged over the 0- to 2-cm shell on a 30-cm-diameter cylindrical phantom (NCRP 1971b).

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

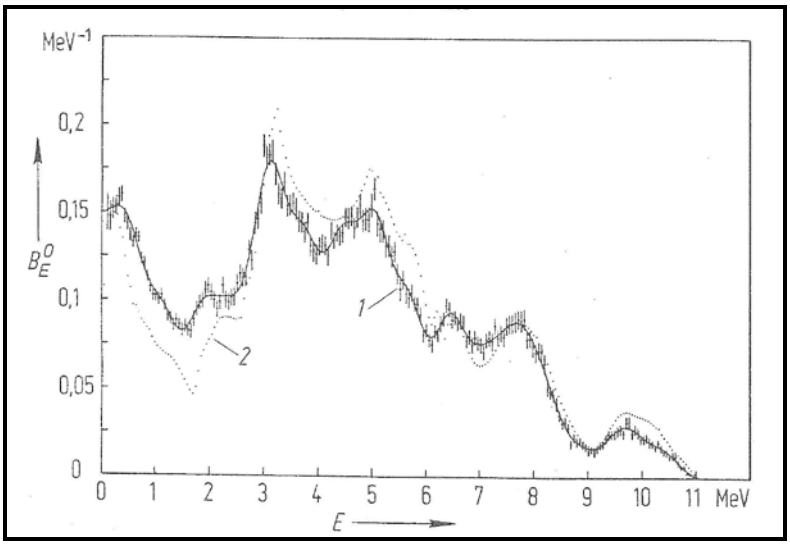


Figure 6-8. Probability density of neutron spectrum from a ^{241}AmBe (α,n) source (Kluge and Weise 1982).

6.3.4 Workplace Radiation Fields

6.3.4.1 Gamma Radiation

In response to a Tiger Team finding (DOE 1991), radiation fields at the INEEL have been characterized by making field measurements with a NaI(Tl) gamma spectrometer and TLDs mounted on a phantom, then comparing the results (Reilly 1998). Figure 6-9 shows the percentage bias for the beta and gamma measurements. Most results lie between +27% and -43%.

The high gamma bias results were for locations at the Radioactive Waste Management Center (RWMC) looking at skyshine (backscattered and thus, low-energy photons) from low-level waste in the Subsurface Disposal Area (Reilly 1998). The doses measured with NaI(Tl) were low (6 and 11 mrem) and the threshold energy on the NaI(Tl) detector was about 100 keV, so some low-energy photons are likely to have been missed (Reilly 1998).

The radiation fields at ANL-W, with a few exceptions, have been generated primarily by mixed fission and activation products. Therefore, most of the photon doses have come from photons with energy greater than 250 keV [4]. The INEEL dosimeters are judged to measure these fields well.

The few exceptions have usually been characterized by low dose rates. Analytical X-ray generators operating below 100 keV have been used in several laboratories. These are easily shielded so the fields have usually been low [5].

A 320-keV and a 160-keV X-ray generator were used for radiography or radiography development studies. Wall shielding has generally been adequate and any transmitted photons would have had energy near the operating voltage because of the hardening caused by the shielding [6].

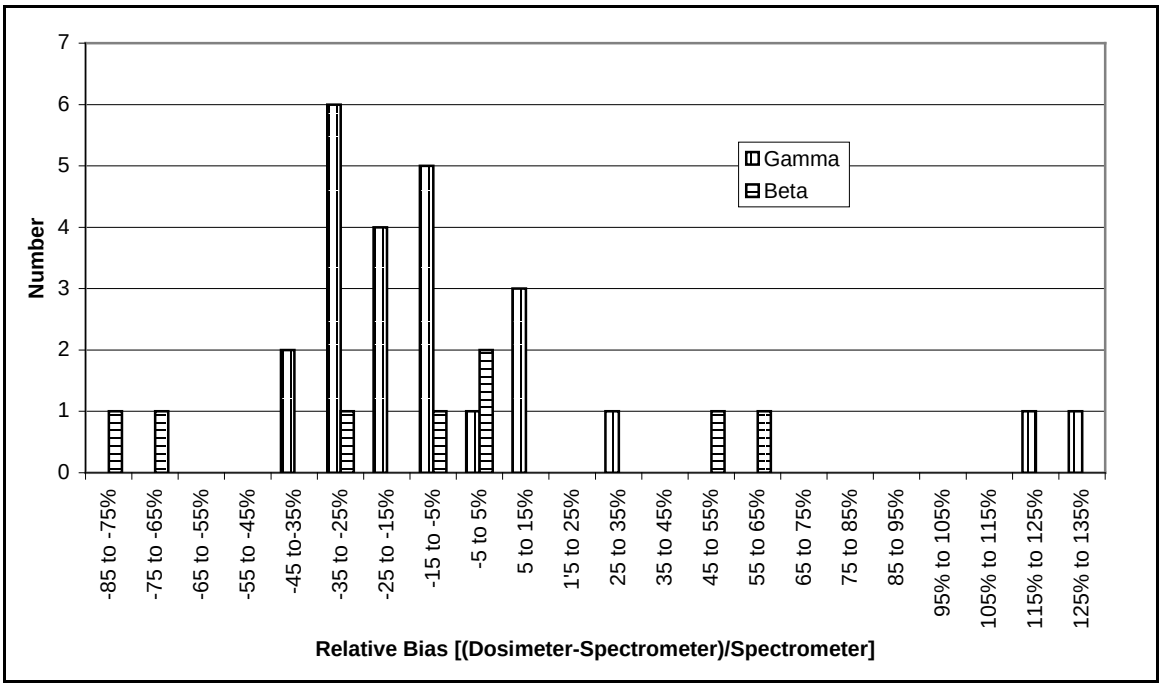


Figure 6-9. Gamma and beta radiation field characterization (Reilly 1998).

Essentially all ANL-W radiological work areas involved beta/photon radiation covering a wide range of energies. These fields can be generally classified according to the IREP codes in Table 6-8.

Table 6-8. Selection of IREP beta and photon energies for ANL-W facilities.

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 could be significant higher energy beta radiation.			Beta Photon	>15 30–250 >250	100 25 75
	EBR, ANL					
		1952	2003			
Processing plants	Highly dispersed fields of higher energy photon radiation fields from activation and fission product nuclides dominant to most exposure profiles. Potential for higher energy beta radiation during sampling and maintenance work resulting from fission products.			Beta Photon	>15 30–250 >250	100 25 75
	ICPP, ANL	1952	2003			
Calibrations	Calibration of instruments and dosimeters			Beta Photon	>15 30 – 250 >250	100 25 75
		1952	2003			

6.3.4.2 Beta Radiation

Beta radiation fields are usually associated with activation or fission product radioactivity that is lightly shielded, handled by workers such as in hot cells, or outside a container such as a spill [7]. The high bias results in Figure 6-9 from comparison of TLD to a phoswich beta spectrometer are for beta sources at contact or at 1 cm, which results in a geometry that is difficult to reproduce. The low-bias results are for large area beta sources for which even the spectrometry results have large variations. The beta occupational radiation fields in Figure 6-9 (only 3) have a bias less than 15%.

Beta field dosimetry became fairly accurate with the definition of DOELAP requirements in the early 1980s. Before then, beta monitoring systems had various flaws, primarily in a detector too thick to give a good surface result or one that was covered with extra material. Calibration was to high-energy betas from either uranium or strontium (see § 6.3.3.1). The dose from low-energy betas can be missed altogether if the beta energy is not sufficient to penetrate the detector cover, and the dose can be under-reported if the beta energy is not sufficient to penetrate the entire detector. The mean beta energy for the spectrum from a particular nuclide is about one-third of the maximum beta energy for that nuclide.

Based on the range energy curve for beta particles and the beta energy distribution of beta emitters (HEW 1970, pp. 90, 91, and 123), the fraction of beta radionuclides with ranges greater than the abscissa is plotted on the ordinate in Figure 6-10. The beta nuclides varied by location and time, so a correction factor common for all facilities was estimated [8]. This analysis used the entire mixture of radionuclides to avoid questions of whether the choice is correct and to reflect the wide variety of radionuclides used. To reflect that the beta spectrum is not monoenergetic because of the energy carried off by the neutrino, a curve is presented for the mean energy (one-third of the maximum energy). To reflect that some beta particles enter the detector at an angle, a curve is provided for 45º incidence at the maximum energy and the mean energy. These curves of the fraction of nuclides with a larger range essentially show the depth dependence of beta dose because the energy loss of electrons does not have much energy dependence. These curves also demonstrate why early dosimeters with thick, sensitive elements failed to correctly report the beta dose at a depth of 7 mg/cm^2, which was chosen in the early 1980s. In addition, these curves demonstrate why the beta dose assigned for skin is inappropriate to use for the breast and testes, in which much of the organ is at a depth greater than 1 mm or 100 mg/cm^2, and for most of persons at depths greater than 1 cm.

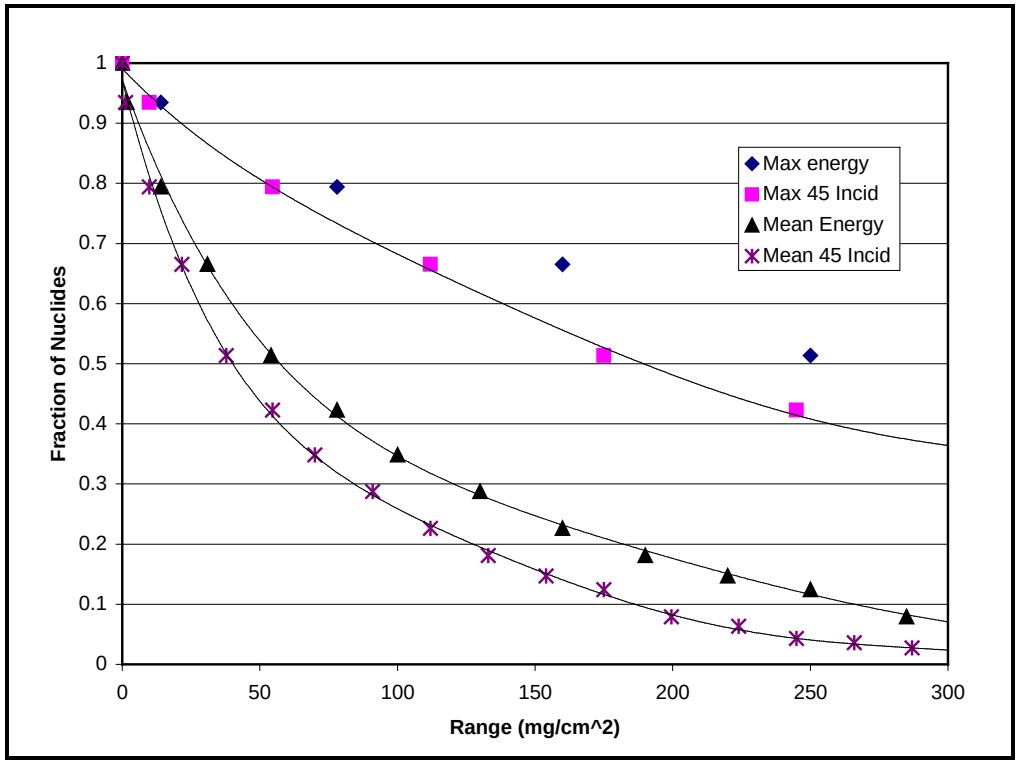


Figure 6-10. Distribution of beta ranges [9].

To calculate the fraction of dose missed by a dosimeter, compare the average of the appropriate curve of this nature over the depth of the active detector to the value at a depth of 7 mg/cm^2. The appropriate curve should be the curve of the mean range for the beta spectrum and the angular distribution of the radiation exposure. To estimate this value, this analysis added the mean energy curve for perpendicular incidence and 1.4 (relative path length) times that for 45º incidence for the mean energy and added one-half that value for 45º incidence for the maximum energy [10]. The curves are the result of a polynomial trend line to the data, so averaging the fraction of radionuclides is relatively easy.

Table 6-9 provides the cover and detector thicknesses for the INEEL beta badges. To determine the corrected beta dose, the measured and missed non-penetrating result should be multiplied by the values in the last column of Table 6-9. 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 [11]. For the Panasonic system, such a correction has already been made (Gesell 1986), so the recommended correction is 1.

Table 6-9. Beta dosimeter thicknesses and associated under-reporting [12].

Dosimeter system	Period	Covers (mg/cm^2)	Detector thickness (mg/cm^2)	Beta correction factor
Two-filter film	1951–1958	50[a]	50[a]	2.0
Multifilter film	1958–1974	100	20[a]	2.8
Low-dose TLD	1969–1974	100	75	3.0
ATLAS	1974–1975	100	100[a]	3.3
Harshaw TLD	1975–1976	104	344	4.8
Harshaw TLD	1976–1985	4	240	2.4
Panasonic TLD	1986–2004	16	15	1.0

a. Estimated.

6.3.4.3 Neutron Radiation

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

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

For calendar year 1979, five people for all of the INEEL received neutron doses between 0.5 and 1 rem, and 79 received measurable neutron doses below 0.5 rem (Jones 1980).

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

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

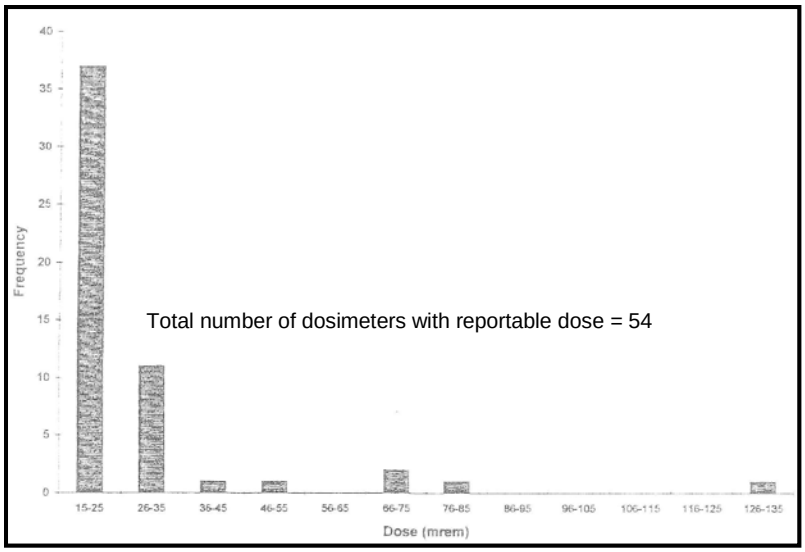


Figure 6-11. Distribution of reportable neutron dose at INEEL for the first 9 months of 1995 (Gesell et al. 1996).

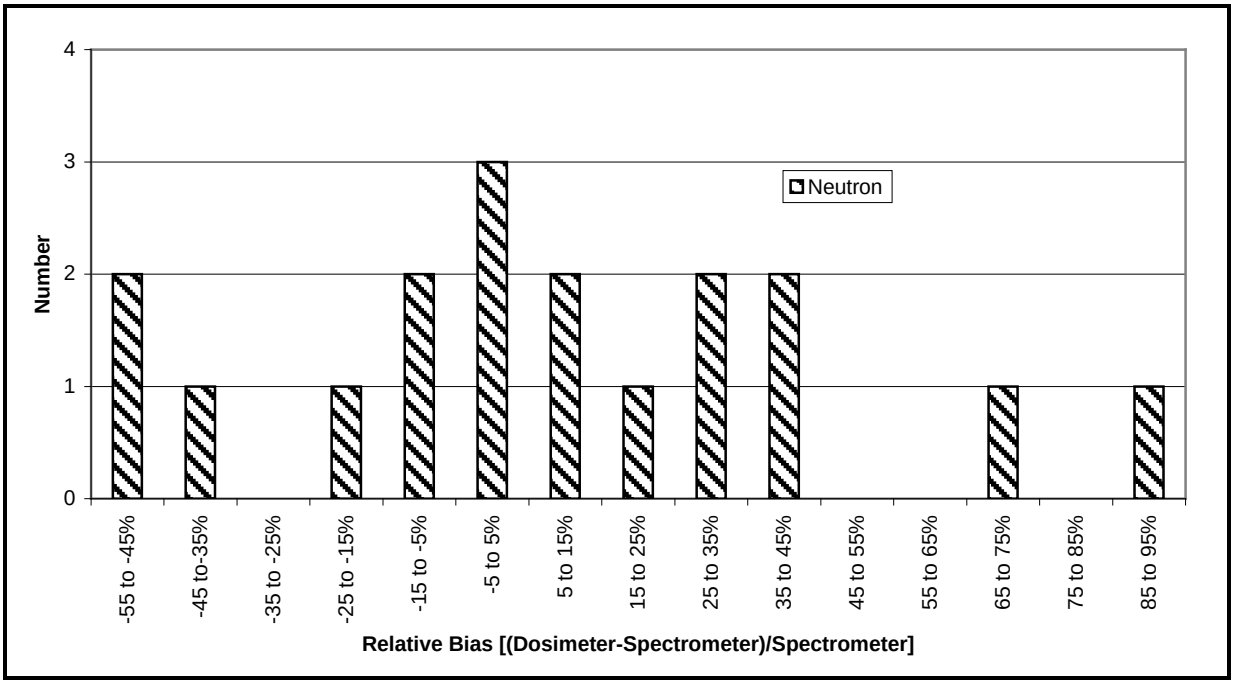


Figure 6-12. Neutron radiation field characterization (Reilly 1998).

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

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

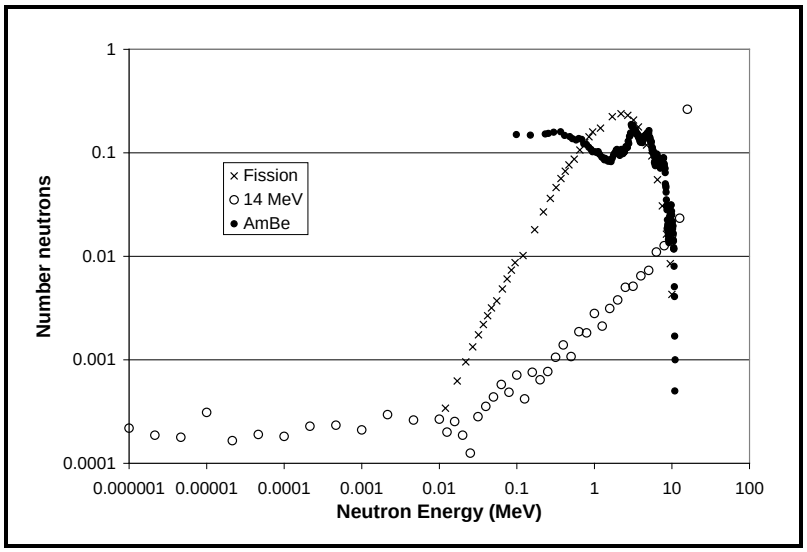


Figure 6-13. Neutron spectra simulating INEEL facilities [13].

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

6.3.4.3.1 Materials Test Reactor (MTR), Zero Power Physics Reactor (ZPPR), and Transient Reactor Test (TREAT) Neutron Radiation

The exceptions to the above discussion are:

- The MTR at NRTS/INEL, which operated from 1952 to 1970 and had beam ports and neutron beams extending onto a research floor.
- The ZPPR at ANL-W, which operated from 1969 to1992. When fueled with plutonium, fairly large fields existed between the halves of the reactor due to (α,n) reactions.
- The TREAT at ANL-W.

Some neutron surveys of the MTR experimental floor have been recovered (Sommers 1959, 1962; Hankins 1961), but these individually do not provide all components of the radiation field. Hankins (1961) used 2-, 3-, and 8-in. polyethylene Bonner balls in a cadmium shield to characterize the intermediate and fast neutrons at 21 locations around the MTR floor. He made these measurements and also measured the thermal neutron component at six other locations. The Hankins data have

been reanalyzed (ORAUT 2006) using more recent Bonner response curves (Hertel and Davidson 1985). Figure 6-14 shows the resultant neutron spectra for locations 3 and 23, which had higher doses and nearly the maximum low-energy intermediate and fission components, respectively.

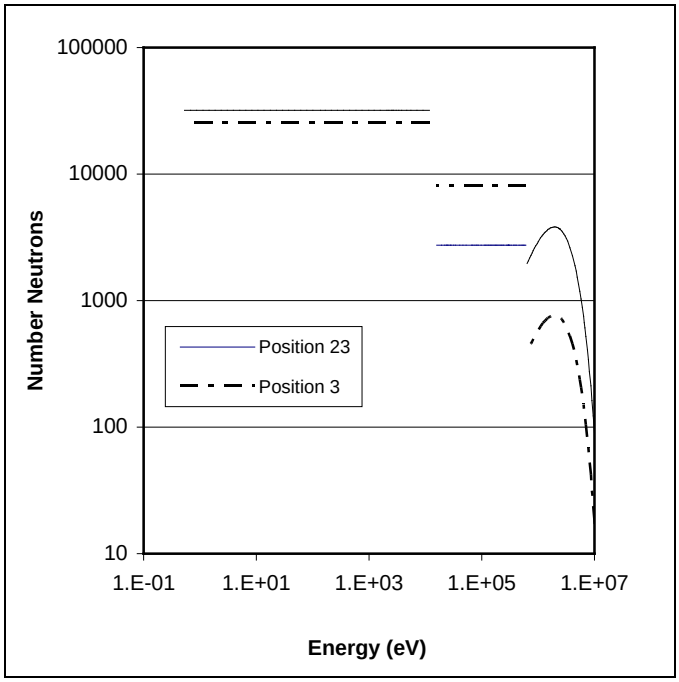


Figure 6-14. Sample MTR spectra from Hankins Bonner measurements (ORAUT 2006).

Figure 6-15 shows the correlations of the thermal and intermediate neutron dose equivalents to the fast neutron dose equivalent for the ORAUT reanalysis of the Hankins data. The trend line for the reanalyzed intermediate-energy neutron dose equivalent has a R^2 value of 0.85 and 0.91 as compared to a R^2 value of 0.50 for the original analysis, which fits the data better. The average ratio of thermal to fast neutron is 0.071 ±0.025; for the low-energy intermediate to the fast, the ratio is 0.177 ±0.057; and for the higher energy intermediate to the fast, the ratio is 0.149 ±0.046, where fast neutrons are taken as those above 0.2 MeV (ORAUT 2006).

The ANL personnel likely to receive neutron dose were assigned NTA film in their dosimetry packets, but the packets would have missed doses below 0.5 to 0.8 MeV. For the MTR spectra, the fraction of neutron dose equivalent above 0.8 MeV has an average value of 0.52 ±0.08 and varies from 35% to 66% depending on the location (ORAUT 2006). To correct for missed dose on ANL reactor experiment floors, the NTA results should be multiplied by 2 ± 0.3 (1/0.52, $0.08/0.52^2$) for a Monte Carlo dose reconstruction (ORAUT 2006).

Sommers (1962) reported thermal and fast neutron dose equivalent rates and gamma dose rates around the MTR beam lines. The thermal measurements near beams are believed to not be representative of the general workplace. Figure 6-16 shows the correlation of fast neutron dose equivalent to the gamma dose for these measurements. The fast neutron component was less than the measurement capability of 1 mrem/hr for several measurements. These values are shown with the triangles at one-half of the smallest measured value. Using the Shapiro-Wilks Normality Test (Gilbert 1987) and including the insignificant fast neutron values at one-half of the minimum reported value suggests that the normal distribution is a slightly better description of the data than a lognormal distribution. The fast neutron dose equivalent is 0.42 ±0.35 of the gamma dose rate for this data set

(ORAUT 2006). Combining these results, the total neutron dose equivalent is 0.58 ±0.48 of the gamma dose equivalent on the MTR experimental floor (ORAUT 2006). Equation 6-1 elaborates this calculation. The variation in the different components of neutron dose rate are so much smaller than the variation between the fast neutron and gamma dose equivalent rate and as such, are unimportant.

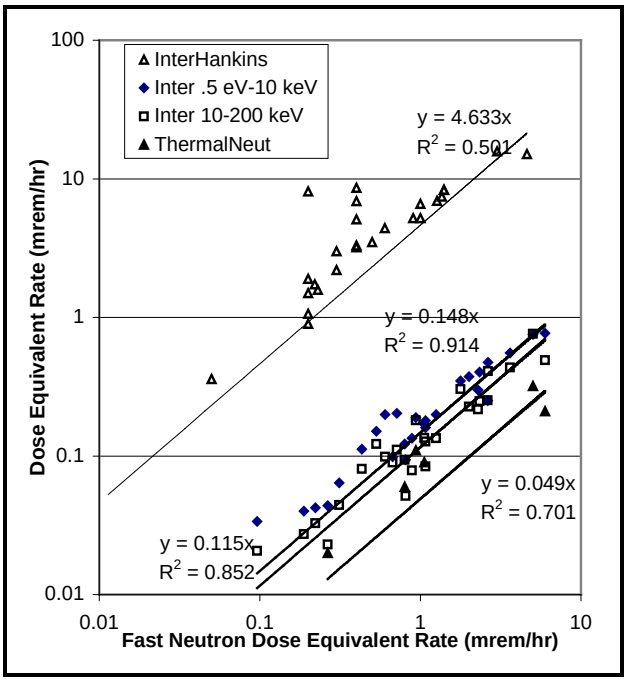


Figure 6-15. MTR neutron field components (ORAUT 2006).

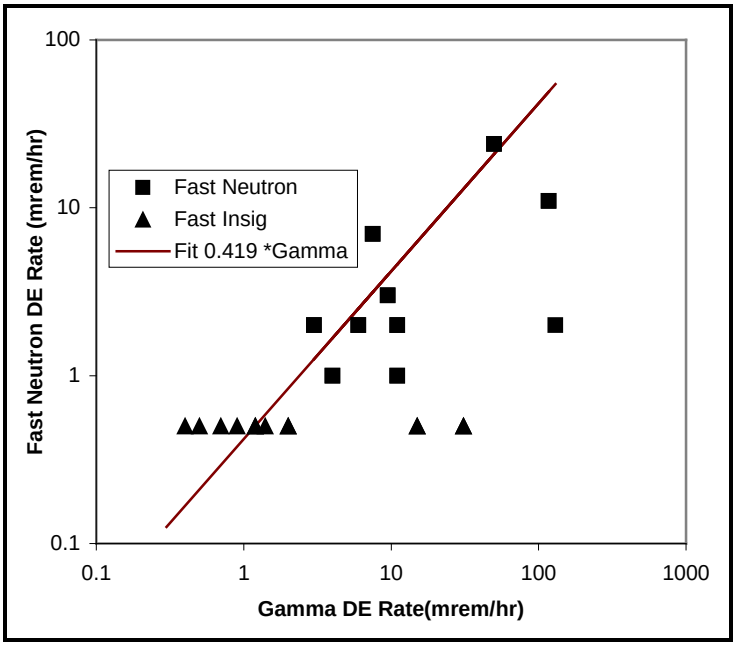


Figure 6-16. Correlation of fast neutron dose equivalent to gamma dose at MTR (ORAUT 2006).

$$\frac{TotalNeutronDose}{GammaDose} = \frac{FastN}{GammaDose}(1 + \frac{Thermal}{FastN} + \frac{LoInter}{FastN} + \frac{HiInter}{FastN}) \qquad \text{(Eq. 6-1) [14]}$$
$$= (0.42 \pm 0.35)(1 + 0.071 + 0.177 + 0.149) = 0.58 \pm 0.48$$

Although this limited data for a small portion of the INEEL with weak correlation might suggest multiplying the gamma dose by 1.6 (i.e., 1+0.58) or 2.1 (i.e., 1+1.06), doing so is probably inappropriate because many of the people wearing NTA film would receive gamma dose at locations other than on the MTR experimental floor while the reactor was operating [15]. For example, health physics technicians would often be covering jobs with only beta-gamma fields. A craftsperson could service pumps carrying radioactive water and not receive any neutron dose. Further extrapolating experience at the MTR at TRA to TREAT or ZPPR at ANL-W is a stretch [16]. While similarities of the neutron spectra and associated corrections are likely valid, the relative amount of gamma and neutron radiation might not be. In addition, dosimetry data sheets demonstrate that small neutron doses were as likely or more likely to be determined as small photon doses, so this sort of correction is not necessary [17].

A fairly large neutron field existed between the halves of the ZPPR when it was off due to ^{240}Pu spontaneous fission decay and subsequent multiplication in the subcritical reactor. Rather than trust the NTA film in use at the time, the practice was to conduct neutron and gamma surveys using a 9" ball detector. Simons, Young, and Thalgott (1972) report spectral measurements (shown in Fig. 6-17) and neutron dose assignments based on time and neutron dose rate. Neutron doses were estimated based on the measured neutron:gamma ratio and gamma doses measured with pencil dosimeters. These results were entered into the dosimetry records (Burke 2006). Shields were also used to minimize the reactor surface open to workers. Figure 6-18 shows the integral distribution of neutron dose with energy for this spectrum. The fraction of dose below the NTA cutoff is quite consistent with the MTR result.

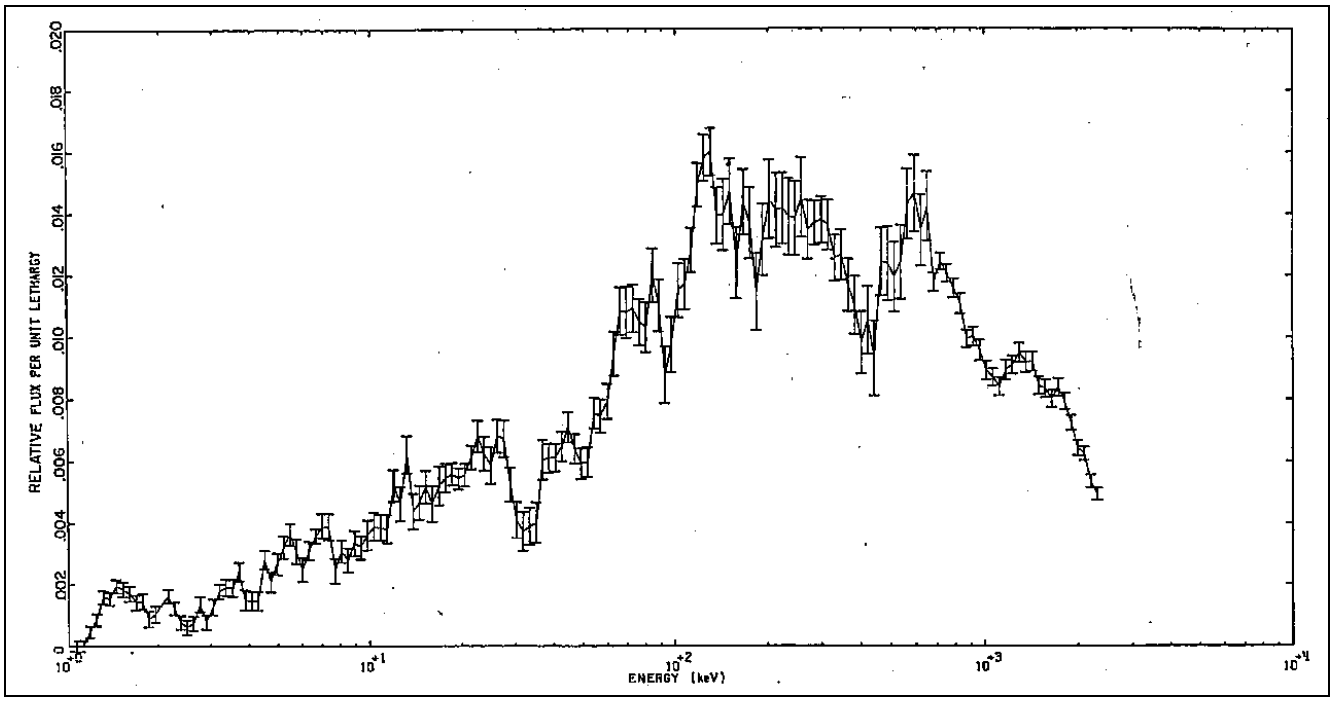


Figure 6-17. ZPPR Assembly-3 Phase-1A Fast Neutron Spectrum Measured Midway between Halves with Both Shields Withdrawn (Simons, Young, and Thalgott 1972).

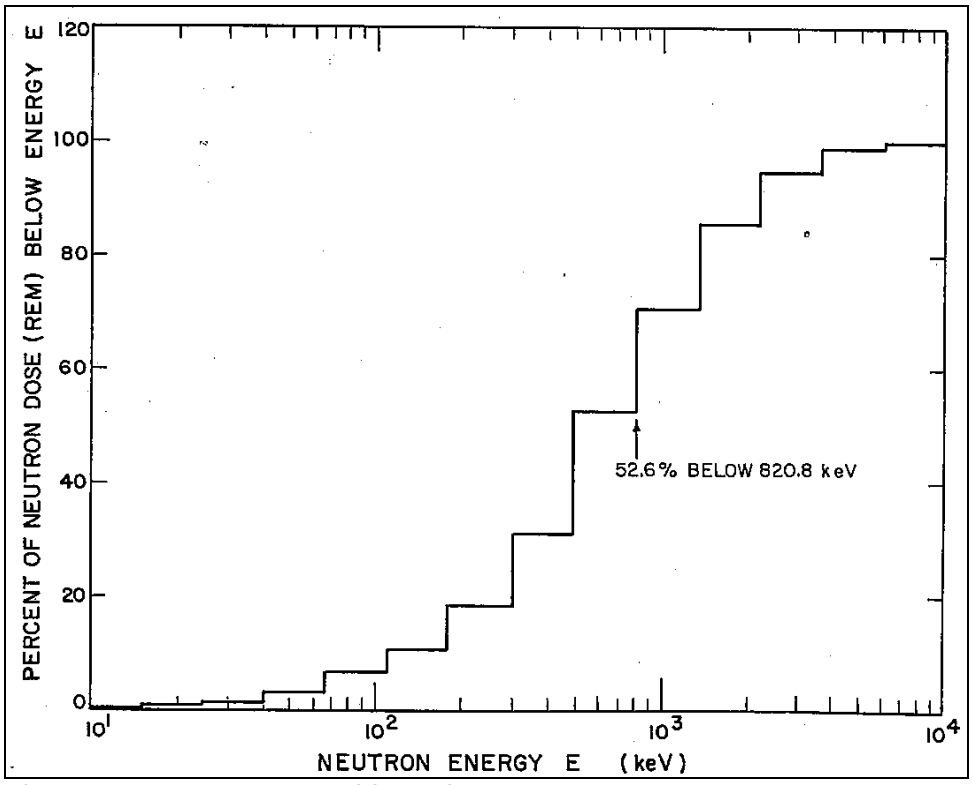


Figure 6-18. ZPPR Assembly-3 Phase-1A Fast Neutron Dose Spectrum Midway between Halves with Both Shields Withdrawn (Simons, Young, and Thalgott 1972).

6.3.4.3.2 Typical Workplace Neutron Dosimeter Hp(10) Performance

Table 6-10 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 [18]. Table 6-11 summarizes the locations at ANL-W where neutron dose is credible.

Table 6-10. Typical workplace neutron dosimeter $H_p(10)$ performance [19].[a]

Parameter	Description	Potential workplace bias[b]
Workplace neutron energy spectra	NTA dosimeter response decreases and TLD response increases with decreasing neutron energy.	Depends on workplace neutron spectra. NTA recorded dose of record likely too low because of high 500-keV threshold for detection of neutrons.
Exposure geometry	NTA dosimeter response increases with increasing exposure angle and TLD response decreases with increasing exposure angle.	NTA recorded dose likely too high because dosimeter response is higher at angles other than AP. TLD recorded dose is lower at angles other than AP. Effect is highly dependent on neutron energy.
Missed dose	Doses less than MRL recorded as zero.	Recorded dose of record is likely too low. The impact of missed dose is greatest in early years because of higher MRLs and shorter exchange cycles of 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 $H_p(10)$.

Table 6-11. Facilities and time periods for neutron exposure [20].

Location	Time period	Comments
Programmatic		
TREAT	1958-1994	
ZPPR	1969-1992	Between reactor halves during maintenance
ANL Neutron Radiography Facility	1977-2006	
Sealed sources		
ZPPR	1969-1992	For instrument calibration

6.4 ADJUSTMENTS TO RECORDED DOSE

6.4.1 Neutron Weighting Factor

All dose equivalents measured at ANL-W and reported in this document used the quality factors based on the LET of the ionizing secondary particles established in the 1950s and used since by U.S. regulatory agencies [21]. In 1990, the ICRP developed new dose concepts that have been used by NIOSH. The quality factor, Q, as a function of LET, was replaced with a radiation weighting factor, $w_{R,}$ which is a function of the neutron energy (ICRP 1990, Table 1).

A correction needs to be made to the ANL-W reported data to change from dose equivalent under NCRP 38 (NCRP 1971b) to a newer dose quantity (NIOSH 2006). ICRP Publication 74 (ICRP 1996) tabulates the ambient dose equivalent (dose equivalent at 10 mm depth in a 30-cm diameter sphere) for neutrons. NIOSH (2006) tabulates the ratios of organ dose equivalents to ambient dose equivalent, so this quantity is to be used for the conversion. Ambient dose equivalent is an ICRU quantity, so it uses a revised Q(L) rather than a $w_{R,}$ so the correction factors are not as large as in other TBDs. The dose equivalent for a spectrum of particle energies is the result of an integral of the fluence spectrum, Φ(E), times a dose equivalent conversion factor, DECF(E), which also depends on energy over the range of energies considered:

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

These factors are incorporated into statements of dose equivalent values and calibrations following generally accepted principles. The conventional dose conversion factors are most clearly and correctly stated in ICRP Publication 21 (ICRP 1973). NCRP Report 38 tabulates a neutron flux density associated with the annual dose limit that is proportional to the reciprocal of the dose conversion factor (NCRP 1971b). Conventionally, the primary geometry considered is from one direction with the maximum dose in the body tabulated. More recent references (ICRU 1985; ICRP 1987, 1996) consider the dose to individual organs for different irradiation geometries, so the more recent tabulations give results lower by factors up to about 10 from attenuation in the human body. Dosimeters are designed to respond to radiation entering the body on the side where they are located and work best for an AP irradiation geometry with the dosimeter on the front of the body [22].

Equation 6-2 also applies to ambient dose equivalent, except 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 [23].

Table 6-12 lists the calculated fractions of dose equivalent in the IREP energy groups and the conversion factors from dose equivalent to equivalent dose for ANL-W spectra. The ratios of average radiation weighting factor to average quality factor for the IREP energy groups have some variation, particularly for the 10-to-100-keV group where the energy dependence of the fluence is radically different for the fission and 14-MeV source than for the reactor spectrum [24]. The lower part of the table lists the recommended default values for the dose equivalent fractions and quality factor corrections. These values are combined in Table 6-13. This correction should be applied to the measured, the missed, and the unmonitored neutron doses either separately or after they are added together.

Table 6-12. Dose equivalent fractions and quality factor corrections, estimated and recommended [25].

IREP energy interval	<10 keV	10-100 keV	100 keV-2 MeV	2-20 MeV
Spectrum calculated values				
Dose equivalent fractions				
Bare fission		4.4E-05	0.20	0.80
AmBe			0.15	0.85
14 MeV 10 cm poly	2.4E-08	3.1E-06	1.5E-03	1.00
MTR exp floor avg	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 avg	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

Table 6-13. Recommended IREP neutron energy fractions and correction factors [26].

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 MISSED DOSE

6.5.1 Photon Missed Dose

Missed photon dose for ANL-W workers would occur where (1) there was no recorded dose because workers were not monitored or the dose was otherwise unavailable, or (2) 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 [27]. 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 [28].

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. ANL-W began using the Harshaw dosimeter 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.

6.5.2 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 INEEL 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 6-9 for the appropriate period.

6.5.3 Missed Neutron Dose

Neutron radiation was present only at the ANL-W reactors, and for calibration of criticality alarms, etc. Most ANL-W workers were not exposed to neutrons and were not monitored for them [29]. For other locations, unmonitored neutron dose is very unlikely because of the very low potential for neutron exposure. The badge of a worker normally monitored for neutron dose might not have been read out and thus, the dose would be unmonitored. This would be indicated by a blank under neutrons on the badge report at a time when other workers have zeros [30]. A reasonable dose to assign for such a situation is the average dose for that worker for nearby monitoring periods [31].

To calculate the missed dose, the reconstructor must first determine if the person worked near neutrons and which category of neutrons. This is best accomplished by review of the work location(s) and whether a worker or others in the badge reporting group were assigned any neutron dose equivalent. For many years, individuals who were assigned NTA film, which was read with minimal dose, have zeros in the record [32]. A blank means that that individual did not have NTA film or it was not read because neutron exposure was believed to be unlikely [33]. It is not certain that this practice was followed for all time [34]. The work location code for TREAT is 20, but ZPPR did not have a unique code. If no neutron dose was assigned to the individual in question or to coworkers for several months, the dose reconstructor should assume that the worker was not exposed to neutrons [35]. Therefore, there would not be any unmonitored neutron dose.

If a worker was likely to have been exposed to neutrons, the dose reconstructor should assign missed neutron dose equivalent using Table 6-15 for the times when workers did not have reported neutron dose [36]. For the period when NTA film was used, the recorded neutron dose should be multiplied using Table 6-16 by 1.25 for all facilities except the TREAT or ZPPR and by 2 for the TREAT or ZPPR when they were operating [37]. The dose equivalent is then apportioned into the IREP groups using Table 6-13.

For example, if in 1970 a person was an experimenter at the ZPPR, and seven of the weekly badges recorded a total of 300 mrem neutron dose equivalent and the other sheets all showed zeros, the missed dose would be 450 mrem [(52 – 7) × 20 ÷ 2] so the total dose by the badges would be 750 mrem. Note that if some reports had blanks rather than zeros (and zeros were entered for other individuals), this would be an unmonitored dose that must be estimated by the first paragraph of this section. Although both ZPPR and TREAT did not operate on full-time schedules, so neutrons would only exist occasionally, it is favorable to claimants to assume that they did [38]. Because the badge only responds to about one-half the ZPPR or TREAT neutron dose equivalent (from Section 6.3.4.2 and Table 6-16), the total dose equivalent is 1.5 rem (this multiplication should also be done for the unmonitored dose). To convert the 1.5 rem received from neutrons at the ZPPR to equivalent dose,

multiply the total dose equivalent by the last column of Table 6-13 to get 300 mrem to the <10-keV group, 90 mrem to the 10-to-100-keV group, 1,050 mrem to the 0.1-to-2-MeV group, and 420 mrem to the above-2-MeV group, for a total equivalent dose of 1.86 rem.

Table 6-15. Neutron dosimeter type, period of use, exchange frequency, laboratory minimum detectable limit, and maximum annual missed dose [39].

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

Table 6-16. Spectrum correction for measured and missed neutron dose [40].

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

The neutron missed dose is divided into two historical periods in the following discussion. The first is before 1976 when only NTA film dosimeters were used with supplemental recording of thermal neutron doses from B-10 pencil dosimeters [41]. The second period is after 1976 when only Hankins albedo dosimeters were used [42]. The estimated MRLs for these neutron dosimeters are summarized in Table 6-15. It is possible to estimate the missed neutron dose using the MRLs because the neutron dosimeters were calibrated with neutron sources that had energies similar to those encountered in the workplace and because most of the neutrons of the calibration source had energies greater than the 500-to-800-keV threshold of the NTA dosimeters [43]. There was, of course, no threshold energy for the measurements using neutron albedo TLD badges.

6.5.3.1 Before October 1976

The use of NTA films for neutron dosimetry before 1976 is documented in various INEEL reports (Cusimano 1963; Cipperley 1958, 1968). As noted above, it is possible to estimate the missed dose using the MRLs. There are many recorded zeros in the neutron dose data for INEEL workers because an NTA film indicated a neutron dose equivalent that was less than the 14-mrem MRL [44]. When the MRL for NTA film is used in estimating the missed neutron dose, it should be multiplied by 1.25 for most workers and by 2 for workers on the TREAT or ZPPR experimental floor [45].

6.5.3.2 After October 1976

Since October 1976, the neutron dose has been measured using the Hankins albedo-type TLD. The characteristics of this dosimeter are well documented (Gesell et al. 1996), and the MRL to be used in estimating missed dose is 15 mrem. A location-specific FNCF has been applied to convert the reading to dose equivalent (Gesell et al. 1996). The adjustment to equivalent dose provided in Table 6-13 must then be made.

6.6 ORGAN DOSE

The dose values should be used to calculate organ doses of interest using NIOSH (2002). It is recommended that the AP geometry should be assumed for the irradiation geometry and for conversion to organ dose [46]. The calculated neutron doses in each energy group should be multiplied by the conversion factors from ambient dose equivalent to organ dose for AP irradiation from Appendix B of NIOSH (2006). For photons before 1981, the conversion factor from exposure to organ dose should be used [47]. For 1981 and after, the conversion factor from deep dose equivalent to organ dose should be used [48].

6.7 UNCERTAINTY

Measurement uncertainties arise from many sources. For gamma rays, the standards for exposure have existed with only minor changes since the 1930s as required for medical uses of radiation. The INEEL used ionization chambers standardized by NBS 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 [49]. The over-response of the multielement film badge to deep dose in tissue arises from their calibration to exposure, which is somewhat more at low energies than the deep dose [50]. The INEEL environment did not have a significant low-energy photon field like a plutonium finishing plant, so the nonpenetrating component was attributed to beta radiation. A realistic estimate of total uncertainty for photon dosimetry is about 35% at 1 sigma [51]. This is roughly consistent with the results in Figure 6-9. For those measurements, the standardization instrument contributed some significant uncertainty.

To determine beta radiations in relation to skin cancer etc., the dose reconstructor should divide the reported nonpenetrating dose by the fraction of dose measured shown in Table 6-9. The uncertainty for beta radiation is somewhat larger than that for gamma rays at an estimated 50% at 1 sigma [52]. This large uncertainty is driven by the uncertainties in field geometry and because beta radiation is often stopped by thin materials such as clothing and air [53]. 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 [54]. The difference between a point source irradiation and a planar source can confuse an algorithm [55]. Earlier techniques did not provide a thin detector with minimal covering, which is important for simulating the skin for beta dosimetry.

The determination of uncertainty for neutron radiations is more complex. The NTA films used before 1975 did not see low-energy neutrons below 0.5 to 0.8 MeV (Cusimano 1963, Cipperley 1968). Corrections are described in Section 6.5.3 for handling this issue. The TLD albedo system provides a very indirect way of measuring dose equivalent to a person. Dose to people is primarily due to hydrogen recoils rather than ^{6}Li(n,α) reactions [56]. The response of the 9-in. PNR-4 detector used to standardize the TLD measurements arises from a different process than dose deposition in the human body. The total uncertainty for neutrons is probably 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 [57].

The cause of the greatest uncertainty for neutrons is the variation of dose caused by the position of the organ within the body. For 1-MeV neutrons, the dose facing the source is about 1,000 higher than the dose on the back side of a 30-cm-diameter sphere of tissue-like material [58]. In a work environment, the 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, this analysis assumed that the geometry of exposure is AP [59]. The apparent discrepancy in Figure 6-

11, in which the dosimeters showed about one-half of the spectrometer readings, occurred because the spectrometer does not simulate the attenuation of the body which, in the measurement, was simulated by the phantom, so the reading was high by a factor of 2.

6.8 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in the preceding text, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here again in the Attributions section of the document, with information provided to identify the source and justification for each associated item. Conventional references are provided in the next section of this document, linking data, quotations, and other information to documents available for review on the ORAU Team servers.

Norman Rohrig served as the initial Document Owner for this document. Mr. Rohrig was previously employed at INL which shared boundaries with ANL-W and used the same dosimetry systems and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner, who is fully responsible for the content of this document, including all findings and conclusions. Mr. Rohrig continues to serve as a Site Expert for this document because he possesses or is aware of information relevant for reconstructing radiation doses experienced by claimants who worked at the site. In all cases where such information or prior studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source. Mr. Rohrig's Disclosure Statement is available at www.oraucoc.org.

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

[2] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is a requirement from ORAUT-PROC-0031, "DOE Technical Basis Document Development, Review, and Approval Process" and uses values based on experience and from the Hanford Site Profile.

[3] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is adopted to INL from the Hanford Site Profile and uses values based on experience at the INL.

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

[5] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The contact radiation field on these devices is less than 0.5 mrem/hr (phone conversation with Larry Burke, Radiological Engineer at ANL-W and INL,11/06/06, Rohrig 2006).

[6] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Because of the energy dependence of photon cross sections, low energy photons are absorbed more strongly than high energy photons, so spectral hardening (reduction of low energy photons) results.

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

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

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

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

[11] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This correction factor is unlikely to have been applied during the calibration factor. The net correction is the ratio of corrections for the calibration spectrum and the field spectrum, so the net effect will be smaller than the corrections in Table 6-9.

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

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

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

[15] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Based on the judgement of Mr. Rohrig, using the gamma dose to estimate the neutron dose is not recommended because the two fields may not be directly related to each other, they often have different sources.

[16] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Since the gamma and neutron fields do not have a common source, there is no reason to believe that the ratios from one facility would apply at the other one.

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

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

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

[20] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is based on facility operating history cited in ANL-W 2 and the potential for worker exposure to neutron radiation based on the facility type and processes judgment of Mr. Rohrig.

[21] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on guidance in OTIB-0055.

[22] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
For neutrons, shielding by a human is significant (factor of 1000 for 1 MeV neutrons) and thus dosimeters need to be facing the radiation source to read reliably.

[23] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is based on guidance in OTIB-0055.

[24] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The values in Table 6-12 are different, namely 1.08 and 1.46 or 1.47.

[25] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table was derived with spectral data in an EXCEL spreadsheet, as shown in Figure 6-13 and 6-14, with the associated energy dependent quality factor and dose effectiveness factors. The last four rows are averages of the associated calculated values higher in the table.

[26] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This table is derived from the values in Table 6-12.

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

[28] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The values in this table are required to calculate the missed dose per ORAUT-PROC-0031 and are based on judgment of Mr. Rohrig.

[29] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This statement is made based on understanding of the type and variety of work performed at ANL-W.

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

[31] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This judgment is made by Mr. Rohrig based on reviews of historical INL dosimetry information and the reference Watson, et al., 1994.

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

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

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

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

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

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

[38] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Assuming a non-zero value when the machine is off is favorable to claimants.

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

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

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

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

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

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

[45] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This correction is listed in Table 6-16.

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

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

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

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

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

[51] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is an estimate, partially based on Figure 6-9 and partially based on uncertainties of many error processes which contribute uncertainty.

[52] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
This is an estimate, partially based on Figure 6-9 and partially based on uncertainties of many error processes which contribute uncertainty.

[53] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
The range of beta particles is shown in Figure 6-10.

[54] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
These estimated corrections are provided in Table 6-9, and they are not small.

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

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

[57] Rohrig, Norman D. ORAU Team. Health Physicist. June-October 2003.
Most of this uncertainty is from the Facility Neutron Correction Factor (FCNF) for the field of the neutron exposure.

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

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

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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 (β)
See *beta radiation*.

beta radiation
Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron. Most of the direct fission products are (negative) beta emitters. Exposure to large amounts of beta radiation from external sources can cause skin burns (erythema), and beta emitters can be harmful inside the body. Thin sheets of metal or plastic can stop beta particles.

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

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

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

DOE Laboratory Accreditation Program (DOELAP)
Program for accreditation by DOE of DOE site dosimetry programs based on performance testing with independent review every 2 years.

dosimeter
Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual.

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
(1) In general, the act of being exposed to ionizing radiation. See *acute exposure* and *chronic exposure*. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

exposure-to-dose-equivalent conversion factor for photons (Cx)
Ratio of exposure in air to the dose equivalent at a specified depth in a material of specified geometry and composition. C_x factors are a function of photon energy, material geometry (e.g., sphere, slab, or torso), and material composition (e.g., tissue-equivalent plastic, soft tissue without trace elements, or soft tissue with trace elements).

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

minimum reporting level (MRL)
Level below which an analytical dose is not recorded in the worker's dose record, usually based on a site-specific policy decision. The recording level is not necessarily the same as the minimum detectable amount or activity for that measurement. Also called less-than value, minimum reportable dose, minimum recordable or recording dose, recording level, and reporting level.

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

pencil dosimeters
See *pocket ionization chamber*.

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

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

pocket ionization chamber
Cylindrical monitoring device commonly clipped to the shirt or laboratory coat pocket to measure ionizing radiation. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

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

ROSPEC
Model name for a rotating neutron spectrometer marketed by Bubble Technology Industries. The system uses four pressurized proportional counters at different pressures to distinguish neutron reactions from gamma events and estimate a neutron energy distribution from 50 keV to 4.5 MeV.

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

shallow absorbed dose (Ds)
Absorbed dose at a depth of 0.007 centimeters (7 milligrams per square centimeter) in a material of specified geometry and composition.

shallow dose equivalent (Hs)
Dose equivalent in units of rem or sievert at a depth of 0.007 centimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

tissue rad
Absorbed dose in tissue.