Document Title: **Argonne National Laboratory – West – Occupational Internal Dose**	Document Number: ORAUT-TKBS-0026-5 Revision: 01 Effective Date: 05/23/2007 Type of Document TBD Supersedes: Revision 00

Subject Expert(s): Douglas Wenzel

Site Expert(s): Norman D. Rohrig

Approval:	Signature on File Jo Ann M. Jenkins, Document Owner	Approval Date:	05/11/2007
Concurrence:	Signature on File John M. Byrne, Task 3 Manager	Concurrence Date:	05/11/2007
Concurrence:	Signature on File Edward F. Maher, Task 5 Manager	Concurrence Date:	05/11/2007
Concurrence:	Signature on File Kate Kimpan, Project Director	Concurrence Date:	05/17/2007
Approval:	Brant A. Ulsh Signature on File for James W. Neton, Associate Director for Science	Approval Date:	05/23/2007

☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

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

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
01/20/2005	00	New technical basis document for the Argonne National Laboratory – West – Occupational Internal Dose. First approved issue. Initiated by Norman D. Rohrig.
05/23/2007	01	Approved Revision 01 initiated to include Attributions and Annotations. Incorporates formal internal review comments. This revision results in no change to the assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Jo Ann M. Jenkins.

TABLE OF CONTENTS

LIST OF TABLES

ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AECL	administrative exposure control level
AEDE	annual effective dose equivalent
ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory–West
ANPP	Aircraft Nuclear Propulsion Program
ANSI	American National Standards Institute
CAM	continuous air monitor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFA	Central Facilities Area
CFR	Code of Federal Regulations
cm	centimeter
COO	Chicago Operations Office
CPP	Chemical Processing Plant
cpm	counts per minute
DAC	derived air concentration
DE	dose equivalent
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
EBR	Experimental Breeder Reactor
ECF	Expended Core Facility
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ERDA	U.S. Energy Research and Development Administration
F	fast (absorption type)
FCF	fuel cycle facility
g	gram
GCRE	Gas-Cooled Reactor Experiment
GE	General Electric Company
hr	hour
HFEF	Hot Fuel Examination Facility
HSD	Health and Safety Division
HSL	Health Services Laboratory
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
IDO	Idaho Operations Office
in.	inch
INL	Idaho National Laboratory
L	liter
LPTF	Low Power Test Facility

M	moderate (absorption type)
MDA	minimum detectable activity
MFP	mixed fission product
mi	mile
min	minute
mL	milliliter
ML-1	Mobile Low-Power Reactor No. 1
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
mrem	millirem
mrep	millirep
MTR	Materials Test Reactor
NaI(Tl)	sodium iodide doped with thallium
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
PBF	Power Burst Facility
pCi	picocurie
POC	probability of causation
RDR	Radiation Dosimetry and Records
RWMC	Radioactive Waste Management Complex
S	slow (absorption type)
SL-1	Stationary Low-Power Reactor No. 1
SMC	Specific Manufacturing Capability
SPERT	Special Power Excursion Reactor Test
STEP	Safety Test Engineering Program
STPF	Shield Test Pool Facility
TAN	Test Area North
TBD	technical basis document
TRA	Test Reactor Area
TREAT	Transient Reactor Experiment and Test (facility)
TRU	transuranic
U.S.C.	United States Code
WBC	whole-body counting
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
α	alpha particle
β	beta particle

γ	gamma
σ	standard deviation
µCi	microcurie
µg	microgram
§	section or sections

5.1 INTRODUCTION

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

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

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

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

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

5.1.1 Purpose

This TBD discusses Argonne National Laboratory–West (ANL-W) internal dosimetry data for dose reconstruction and includes guidance for the appropriate use of that information.

5.1.2 Scope

In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) and started construction of facilities on a 572,000-acre site approximately 50 mi west of Idaho Falls in southeastern Idaho. NRTS was later renamed the Idaho National Engineering Laboratory, then the Idaho National Engineering and Environmental Laboratory, and most recently the Idaho National Laboratory (INL). Argonne National Laboratory (ANL) near Chicago established a branch at the NRTS known as ANL-W where it built and operated several reactors that were of fundamental importance for the development of commercial nuclear power. ANL-W was operated under contract to the Chicago Field Office of the AEC/DOE by the University of Chicago from 1951 through January 2005. ANL-W was merged with INL in February 2005, and Battelle Energy Alliance assumed all operations.

Each of the original AEC laboratories was unique in both mission and location. Since the early days of the AEC programs, which represent the beginnings of the nuclear age, significant technical developments were a necessity, not the least of which were developments in radiation safety. Some of the unique characteristics of radiation safety (and internal dosimetry specifically) at ANL-W, which had a marked influence on the conduct of the internal dosimetry programs are as follows:

- The original mission of the ANL-W was uranium reactor concept development. The production of weapons-grade nuclear materials was not a mission.

- The beginning of ANL-W activities was 8 to 10 years later than those at ANL-East, Oak Ridge National Laboratory, and the Hanford Site. During those developmental years significant technical progress in professional skills, instrumentation, analyses, procedures, and techniques were accomplished.

- The Chicago Field Office was responsible for and had oversight of the ANL-W program.

- To provide necessary consistency of radiation safety programs at the NRTS, the AEC through the Idaho Operations Office (IDO) established a Health and Safety Laboratory (HSL) to provide technical support in the areas of (1) environmental surveillance, (2) external dosimetry (personnel dosimeters of all types), (3) portable radiation detection instrumentation inventories, calibration, and maintenance, (4) internal *in vitro* and *in vivo* bioassay analytical laboratories, (5) maintenance and documentation of personnel dosimetry records, and (6) research and development in these areas of responsibility. The name of this organization changed several times to Health and Safety Division (HSD), Idaho Center for Radiological and Environmental Sciences, and most recently to the Radiological and Environmental Sciences Laboratory. Technical data, dosimetry services, information (particularly in the instances of detectable worker intake), and analytical internal dose calculations and evaluations were exchanged between the AEC HSL and ANL-W.

As a consequence, basic assumptions about minimum detectable activities (MDAs) or minimum detectable levels, missed dose potential, etc. are relatively consistent across the years. As early as 1955 or 1956 gamma spectral analysis capabilities allowed the significant bioassay results (those that would result in reportable internal dose) to be defined in terms of the specific radionuclides. The

practice in the case of a higher urine sample result was to attempt radionuclide identification through gamma spectral analysis and chemical separation. This document describes default assumptions for use in cases when the bioassay records for a claimant do not include specific radionuclide analyses but rather only record gross beta or gross alpha results.

The majority of this document provides background information to aid the internal dose reconstructor through increased general understanding, data interpretation, defaults, etc. Section 5.1.3 provides a facility description, and Section 5.1.4 details the radionuclides of concern. Section 5.2 describes the INL radiological protection program as it evolved over the years, Section 5.3 discusses internal dose control, and Sections 5.4 and 5.5 describe MDAs and whole-body counting (WBC), respectively. Sections 5.6 and 5.7 suggest approaches for the treatment of missed dose and unmonitored workers.

5.1.3 Argonne National Laboratory–West

ANL-W, which began operations in 1951, continued to conduct nuclear power developmental programs through January 2005. Although ANL-W received internal dosimetry support from the INL service laboratories, its radiological safety programs operated under authority of the Chicago Operations Office (COO).

Nine experimental reactors under the technical direction of ANL-W were operated at two locations, one on the southwest side of the Site near the Radioactive Waste Management Complex (RWMC) and the others at the current location on the southeast side of the site. Early reactor operations included physics critical experiments; power production; routine unmoderated operation; uranium-fueled, plutonium-fueled, and breeder reactor designs; and self-destruction experiments.

A series of deliberate safety experiments were conducted by ANL-W in which reactors were allowed to go *prompt-critical* with resultant reactor destruction (Stacy 2000). External and internal doses to workers, both expected and accidental, were associated with these activities (RAC 2002).

5.1.4 Radioactive Nuclides of Concern and Solubility

ANL-W facilities and activities related primarily to experimental reactor design and development. The latest revision of the ANL-W Site Description TBD (ORAUT 2004) describes these activities in more detail. The radionuclides of concern as listed in Table 5-1 from an internal dose standpoint are mixed fission products (MFPs) (primarily aged), uranium and decay products, and transuranic (TRU) and decay products. The uranium had various enrichments from depleted to at least 68% enriched, both used in Experimental Breeder Reactor No. 2 (EBR-II). Most internal doses were identified after an incident rather than as a result of routine bioassay measurements. Through the years at ANL-W, plotting urine and fecal elimination curves has shown that many nuclides appear to be eliminated slowly, which indicates a relatively insoluble material. The chemical explanation is that radioactive materials oxidize rapidly and form less soluble compounds. A default assumption of type M or S would be appropriate, where International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995) specifies these for the nuclide. Strontium is an exception; type F should be assumed.

5.2 RADIOLOGICAL PROTECTION PROGRAM MANAGEMENT AND SUPPORT

The contract with the University of Chicago to operate the ANL-W facilities did not change during the 54 years of operations. The COO provided oversight for ANL-W programs and facilities. The ANL-W program used site support services, including internal dosimetry support.

Table 5-1. Radionuclides of concern (Nielsen 1996a).

Radionuclides	Sources/characteristics
H-3 (HTO)	EBR-II Reactor Facility & Sodium Components Maintenance Shop
Mn-54, Fe-59, Co-58, Co-60	EBR-II primary source. Levels low and decreasing since decommissioning.
Sr/Y-90	All facilities handling fission products
Cs-134, Cs-137	All facilities handling fission products
U-234, 235, 238	Hot cell, hoods, glove boxes, waste, reactor fuel, research areas
Pu-238, 239	FCF, HFEF, ZPPR, Analytical Laboratory
Am-241	FCF, HFEF, ZPPR, Analytical Laboratory

The personnel dosimetry records were documented and are permanently maintained. Records about individual facility or contractor field monitoring programs (air-monitoring data, personnel contamination records, etc.) were maintained by ANL-W. The field monitoring data were not available for use in this report.

The HSL had radionuclide identification capabilities from the early 1960s. Positive bioassay results (analyses in which the results exceeded 2σ counting statistics) were normally followed by a confirmatory analysis to identify specific radionuclides. In the cases where only gross beta or gross alpha bioassay results are available, the results are normally within 2σ. However, if it is necessary to evaluate intakes from the gross beta or gross alpha results, the radionuclide defaults should be $^{90}Sr/^{90}Y$ and ^{239}Pu, respectively.

5.2.1 Bioassay Programs

Routine bioassay of radiation workers was conducted from the beginning of operations. However, formal documentation of the ANL-W bioassay programs was not found for periods before 1989. Some of the data sheets on individuals indicate that urine bioassay sampling occurred annually from 1959 to 1966. Whole-body counts began in 1963 and were typically annual. Table 5-2 lists the reconstructed history of routine bioassay frequency.

5.2.2 Internal Dose Records

Formal or *legal* internal dose data were maintained by the DOE HSD in individual hard-copy folders until mid-1990 when all technical support service functions, including those related to internal dosimetry, were transferred to the INL prime contractor. At that time, *in vitro* analytical functions were transferred to an onsite INL analytical laboratory. The *in vivo* counting laboratory provided support directly through the Radiation Dosimetry and Records (RDR) organization, which administered external and internal dosimetry support programs. The subject matter expert reviewed, validated, and prepared official internal dose assessments. The RDR unit functions included documentation and records custodial responsibilities.

5.2.3 Data Codes and Investigation Levels

The changes in standards and regulations influenced the level of internal dose evaluation and documentation, but did not change the fact that all (negative as well as positive) bioassay data were recorded in the individual dosimetry files.

The information that was used in internal dose assessments and analytical data sheets varied through the years. Table 5-3 describes coded information that could appear in records from after mid-1990. Table 5-4 describes internal dose information that could appear in records from before mid-1990. Table 5-5 contains analytical nomenclature. Table 5-6 contains INL codes for various site areas.

Table 5-2. Routine bioassay history summary.

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1953–1960	Annually	*In vitro* urine	Radiation workers		Frequency is inferred from individual data sheets.	Individual data sheets Table 5-9 AEC 1959 AEC 1961
1961–1972	Annually	*In vitro* urine *In vivo*	Radiation workers		Frequency is inferred from individual data sheets.	AEC 1962 AEC 1963
1973–1981	When intake suspected	Fecal	Radiation workers	Reporting Annual DE >10% quarterly standard in ERDA Manual Chapter 0524	Frequency is inferred from individual data sheets.	AEC 1968 AEC 1975 ERDA 1975
	Annually	*In vivo*				
1982–1988	When intake suspected	Fecal	Radiation workers	Reporting CDE >10% quarterly standard in DOE 5480.1A Chapter 11		DOE 1981
	Annual	*In vivo*				
	Termination	*In vivo*	Radiation workers			
1989–1995	Annual	*In vivo*	Radiation workers, available all employees			ES&H 1989
	Radiation Workers based on exposure potential	*In vitro* fecal	Medium to highest risk working with plutonium contamination			
		In vitro urine	Uranium or tritium workers			
	Termination	*In vivo*	When determined by Radiation Fire and Safety Engineering			
		In vitro				
	New hire	*In vitro*				
		In vivo				
1995–2004	Appropriate to the facility mission, potential uptakes.	*In vivo*	Radiation workers that enter radiological buffer areas or areas of greater radiological controls and are likely to receive intakes resulting in a CEDE of 0.1 rem or more. Type of bioassay based on source term. Urine requested when pure beta, uranium, or TRU was of interest. Feces requested primarily for uranium and TRU source terms.	Reporting In accordance with DOE 5480.11 and 10 CFR 835. Workers that could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE. Investigating Internal doses resulting from all confirmed intakes are to be evaluated.	Follow-up for any suspected intake of radionuclides and to more accurately identify and characterize the amount of intake and excretion pattern.	Nielson 1996a DOE 1988
		In vitro				
		Urine				
		Fecal				
	When work place monitoring indicates significant potential for intakes.					
1995–2004	New hire		To determine internal conditions from previous uptakes or to establish baseline for those continuing to work as radiation workers.			
	Termination	*In vivo*	All employees			

CDE = committed dose equivalent; CEDE = committed effective dose equivalent; DE = dose equivalent; ERDA = U.S. Energy Research and Development Administration.

Table 5-3. Internal dose assessment information after mid-1990.

Coded information	Description
Name & SS no.	Exposed employee by name and Social Security Number.
Asmt. nos.	This assessment number is the calendar year, and a consecutive numbered assessment for that employee during that specific year.
Intake date	Month/day/year of employee intake.
Radionuclide class & amt.	Specific radionuclide followed immediately by ICRP Publication 30 solubility class symbol D, W, or Y (ICRP 1979). Amount in microcuries or becquerels.
CEDE rem	Calculated CEDE in rem.
Organ (max.)	Organ that received the maximum dose from the specified intake.
Organ CDE rem	CDE calculated for the listed organ in rem.
Employer and exp. location	Abbreviation of DOE site contractor and the plant site of exposure (can include the building number).
Year–total CEDE	CEDE exposures are summed for the year of intake for each employee.
Year–TL organ CDE	Organ CDE total (TL) exposures are summed for the year of intake for each employee.

Table 5-4 Internal dose assessment information before mid-1990.

Dose information	Description
Name, Soc. Sec. no.	Employee name, Social Security Number, and (Contractor Abbreviation/Plant or Facility).
Nuclide	Radionuclide symbol followed by ICRP solubility class (D, W, or Y) (ICRP 1979).
Intake period	Month and year for single exposure or period of time by month and year in which exposure occurred.
CEDE rem	Calculated CEDE in rem.
AEDE rem	Calculated AEDE in rem.
Year	Year for which the AEDE was calculated.

AEDE = annual effective dose equivalent.

Table 5-5. Analytical information that could be in claimant dose files.

Analytical information	Description
Sample no.	Sample log number.
Date and time	Generally clear interpretation.
Sample description	Name of the employee, numerical sample number frequently included, additional special analyses performed (e.g., Sr-90, Y Separation, etc.).
*Anal. for	Generally gross beta and/or gross gamma. Sample aliquots evaporated for gross beta or counted directly in a deep-well NaI scintillation counter. Specific isotopic analysis based upon chemical separation or gamma spectrum also listed in this column.
Quantity used	Size of the sample aliquot, generally in mL.
U^+ or K^+ trans.	Note to indicate analytical correction for natural potassium and uranium.
Count time	Counting either used preset time or preset counts. Time in minutes recorded in either case.
Total count	Total number of counts recorded.
Gross count, cpm	Cpm determined by dividing total counts by time of count.
Bkgd., cpm	Background cpm recorded.
Net count, cpm	Gross cpm minus background cpm.
K-40 corr., cpm	Additional background from K-40 identified. K-40 is not a facility occupational exposure product; ignore from an internal dose reconstruction
Foreign activity, cpm and dpm	Net counts corrected for K-40 and then converted to dpm based upon counter calibration. Uncertainty also included, which is recorded as 1σ based upon counting statistics.
Dpm per a volume	The activity is for the sample volume listed.
Result in µg/L	These results are for uranium whether stated or not.

Table 5-6. Area codes that could be in claimant dose files.

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

Bold indicates ANL-W facilities.

Each individual analytical result was documented and placed in individual exposure files, regardless of the formal reporting requirements. The investigation levels (the levels at which positive bioassay results triggered follow-up sampling to verify that detectable activity had been taken into the body) also changed little from the early years to 2005. Later procedures (DOE 1988) set specific limits on those positive bioassay results that could result in an AEDE of 100 mrem or above as the point at which follow-up and reporting was required. With the DOE *Radiological Control Manual* (DOE 1994), this changed to a CEDE of 100 mrem. In addition, a calculated dose of 10 mrem or above would be recorded as an internal dose. These procedural limits did not materially affect the bioassay sampling frequency or the recording of even undetectable radioactivity in bioassay samples, although the request for and number of follow-up samples and analyses could have been different as a function of the formal regulations in effect.

5.3 INTERNAL DOSE CONTROL

The radiological protection program was designed to detect barrier or ventilation failure in a timely manner. The program consisted of continuous and retrospective air and effluent monitoring combined with personnel and surface contamination monitoring. Detection of barrier failure provided the information for making decisions on personnel evacuation, increasing personal protection equipment (e.g., respirators), and requesting bioassay analyses to identify possible internal intakes.

The contamination control limits for the detection and control of released activity were dependent on instrumentation capabilities. As a result of increased emphasis on exposures that were as low as reasonably achievable, some reduction in acceptable release levels was implemented. The contamination control limits for alpha on plant surfaces and particularly on personnel were set close to the MDA, such that *any detectable* contamination was a signal for preventive action and follow-up evaluations. Beta/gamma MDAs typically were a factor of 5 below the limits. Table 5-7 is a summary of control limits primarily from the *CPP* [Chemical Processing Plant] *Health Physics Manual* (ACC 1952) and the operating procedures in effect in 2005.

Table 5-7. Surface contamination control and MDAs.

Period	Surface location	Detection technique	Control levels	MDA - typical
1952–1960s	Plant/equipment	Smears	500 dpm β & 20 dpm α per 100 cm^2	150 dpm β & 10 dpm α per 100 cm^2
	Personal clothing	Portable survey instruments	1500 dpm β & 500 dpm α per 100 cm^2	1,000 dpm β & 500 dpm α per 100 cm^2
	Personal skin	Portable survey instruments	Any detectable reported, e.g. 1,000 dpm β & 500 dpm α per 100 cm^2	1,000 dpm β & 500 dpm α per 100 cm^2
	Shipments	Smears/portable survey instrument	500 dpm β & 20 dpm α per 100 cm^2 – smears 0.1 mrep/hr β & 500 dpm α per 100 cm^2	150 dpm β & 10 dpm α per 100 cm^2 smears. 0.01 mrep/hr β & 500 dpm α per 100 cm^2
1970s–2005	Plant/equipment surfaces	Smears	300 dpm β & 20 dpm α per 100 cm^2	30 dpm β & 10 dpm α per 100 cm^2
	Personnel	Portable survey instruments	Any detectable reported, e.g. 300 dpm β & 200 dpm α per 80-100 cm^2	300 dpm β & 200 dpm α per 80-100 cm^2

5.3.1 Air Monitoring

The monitoring of radioactivity in the air in occupied areas was a basic element of the internal exposure prevention program. Beta/gamma continuous air monitors (CAMs) were used for beta/gamma emitters with maximum permissible concentrations (MPCs) or derived air concentrations (DACs) above 1×10^{-9} µCi/cm^3. TRU materials and uranium were nearly always well tagged with beta/gamma activity that allowed beta/gamma-detecting CAMs to be used to warn of possible alpha contamination and possible internal exposures.

In general, workers were asked to submit to bioassay whenever they were in an area where a CAM alarm sounded. In addition, the fixed location and retrospective air-sampling system would signal the need for bioassay if elevated air sample results were detected.

At ANL-W fixed air sampling heads were used in the Fixed Air Sampling System beginning in 1976 (Courtney et al. 1980). Concentrations averaged between 1×10^{-15} and 5×10^{-15} µCi/cm^{-3} for alpha emitters and between 0.4×10^{-13} and 4.0×10^{-13} µCi/cm^{-3} for beta emitters in the first 3 years of operation. This information can be used to assign an intake for an unmonitored worker.

5.3.2 Early Technical and Analytical Capabilities

DOE HSL technical reports and annual reports, along with facility memoranda and reports, documented the analytical detection capability in the 1950s and 1960s. Radionuclide identification by energy spectra was available and used for urine and other bioassay samples. Specific separations (e.g., strontium, iodine) were available to quantify the radioactive components of a variety of samples of interest.

In the early days, a gross beta urine bioassay measurement was made on an evaporated aliquot, or a gamma count was made directly on a liquid sample, or both. Any detectable activity triggered a specific chemical separation analysis (generally for strontium). Early analyses for plutonium generally were gross alpha counts on a plutonium separation; later, alpha spectroscopy was used to count and better characterize the results.

In 1958, the IDO HSD acquired a 256-channel gamma spectrometer with a 3- by 3-in. sodium iodide thallium-doped [NaI(Tl)] detector counting system for analyses of gamma-emitting radionuclides. In 1960, the HSD obtained a 3- by 3-in. well counter for gamma analysis to replace the gross beta counting as the routine analytical procedure for urine samples. AEC (1961, p. 59) states, "Approximately 1.5×10^{-6} µCi/mL of MFPs can be detected in 75 mL of urine in a 5-minute count, which is about the same as was obtained with the gross beta procedure in a 20-minute count."

AEC (1961) outlined a basic philosophy in relation to gamma counting of bioassay samples. Gamma counting would be effective in all situations except for exposure to pure strontium isotopes. To guard against this unlikely possibility, the procedure of performing a strontium analysis for individual workers at risk (radiation workers) every 2 years and at termination was established. Because of the improbability of finding detectable activity, all activities were to be precipitated by oxalic acid in a weak acid solution and counted for gross beta, and the strontium-specific analysis not completed unless a detectable count was obtained on the precipitate. A 100-mL sample of urine permitted the detection of approximately 8×10^{-8} µCi/mL of ^{90}Sr.

The special and routine bioassay sample analyses were performed and documented by the DOE analytical laboratory. The individual analytical data were recorded. At ANL-W, there was a personnel dose file on each individual that contains results of all samplings and notations on all contamination incidents. In all cases, copies of the bioassay results should be in the individual's dose file in the NIOSH-Office of Compensation and Support Claims Tracking System database.

5.3.3 Recent History of Bioassay Results

Table 5-8 provides a recent history of bioassay sampling based on annual skin contamination and exposure reports (Carr 1990; Holson 1991; Kirchner 1992; Marshall 2002; Nielsen 1992a to 1995b, 1995b to 2004; Vroman 2001). Blank entries mean that the information was not available. The skin contaminations tended to be a precursor for positive bioassay, and the number was countable and tracked since at least 1984. The associated activities have not been extremely high. In 1993 there were a large number of skin and clothing contaminations in relation the Fuel Cycle Facility (FCF), which was being refurbished. Almost all workers received a whole-body count until the late 1990s. In 1998, an FCF seal tube repair led to 11 positive counts on five people. The highest calculated CEDE was 1 mrem. The other positive count was from a medical test about 3 years before. The tritium assay was about monthly, so many fewer people were involved and the doses were generally trivial because of the high monitoring sensitivity. The number of tritium assays was cut back because of the history of not recording anything significant in dose from tritium. The plutonium fecal assay led to mostly false positive results for which a subsequent fecal assay failed to show any activity. In 1993 a person received a hand wound and was chelated, which resulted in two positive urine counts but no positive fecal count (Burke 1993). A CEDE of 100 mrem was assigned. This recent history suggests that rather modest internal exposures occurred at ANL-W.

5.4 MINIMUM DETECTABLE ACTIVITIES

In compliance with the November 1998 DOE requirements (10 CFR Part 835) for the DOE Laboratory Accreditation Program (DOELAP), and based on ANSI Standard N13.30, *An American National Standard*, *Performance Criteria for Radiobioassay* (HPS 1996), the *in vitro* and *in vivo* radiobioassay laboratories at INL received DOELAP accreditation in February 1998. In

Table 5-8. Recent history of bioassay sampling.

Year	# Skin contam events	# Clothing contam events	Maximum beta contam	# WBCs	Positive WBCs	# Fecal Pu assay	Pos *in vitro*[a]	# Tritium assay	Pos tritium	Max T dose
1984	20		**cpm until**							**mrem**
1985	17		**1992**							
1986	7		**dpm after**							
1987	5		**1992**							
1988	16									
1989	18		8,000							
1990	10			542	0	44	0			
1991	10		12,000	648	0	60	0	287	29	14
1992	15	7	8,000	609	0	69	0	40	8	<1
1993	23	58	500,000	1,082	0	33	2 urine	48	6	<1
1994	7	21	139,000	837	0	70	1 FP	46	14	<1
1995	3	10	784,000	758	0	106	1 FP	106	10	<1
1996	2	11	186,000	273	0	75	1 FP	139	13	<1
1997	5	10	135,000	329	1	76	2 FP 1 P	138	8	<1
1998	5	21		309	12	56	2 P	152	16	<1
1999	7	5		311	1	105	2 FP	365	84	<1
2000	2	4		318	1	136	0	697	50	<1
2001	1	7		254	1	143	1 P	552	73	<1
2002	4	8		212	1	143	3 P	40	7	<1
2003	2	8		222	0	127	1 FP	9	1	<1

a. FP – false positive P - Positive

accordance with this accreditation, MDAs and decision levels at the 95% confidence level are performed. Tables 5-9 and 5-10 list the MDAs by period for urine and fecal sample analysis, respectively. The earliest reference (Ebersole and Flygare 1957) defines a detection limit as "twice the standard deviation of its determination." The detection limits in early references have therefore been multiplied by 2.33 to determine an MDA similar to the values from the American National Standards Institute (ANSI). A large majority of the urine samples were single voidings; 24-hr samples were used for special sampling purposes (i.e., follow-up samples, primarily to extend the sensitivity). If the urine sample aliquot size was stated, the MDA for urine samples was given in both picocuries per milliliter and picocuries per sample; if the size was not stated, only the reference value was provided. Most bioassay data have units of activity per sample with the units changing from disintegrations per minute to microcuries to becquerels and back again. The listed MDAs are those for the primary samples. The recommended periods for the MDA values are included in the tables. Table 5-11 lists the 2005 *in vivo* MDAs along with values from historical documents and the recommended periods for use of the MDA values.

5.5 DEVELOPMENT AND IMPACT OF WHOLE-BODY COUNTING

WBC was introduced at the INL in 1961. As early as 1961 one of the fundamental conclusions from experience with *in vivo* and *in vitro* internal dosimetry analytical techniques was the fact that a large proportion of the internal exposures was to insoluble materials. Radioactive nuclides (e.g. ^{125}Sb, ^{110m}Ag, ^{65}Zn, and ^{95}Zr/Nb) were detected by an *in vivo* count but were not detected in the urine. Concurrent analyses of fecal and urine excreta demonstrated the main elimination route to be by the feces, with so little voided in the urine as to be undetectable even in a 24-hour specimen (AEC 1962; Sill, Anderson, and Percival 1964). WBC was demonstrated to detect activity as low as 0.01 µCi in a 10-minute count (AEC 1962). This detection level was several orders of magnitude more sensitive than the maximum permissible body burdens (MPBBs) for most beta/gamma fission and activation

Table 5-9. MDAs for urine samples by period.

Radiation/ radionuclide	Period	Urine (pCi/mL)	Urine (pCi/sample)	Nominal sample volume (mL)	Reference
Gross β	1951–1953	8.4			Data sheet
	1954–1959	9.3	46	5	Ebersole & Flygare 1957[a]
	1960–70	3.5			AEC 1961[a]
	1971–2005	1	5	5	AEC 1972[a]
Gross γ	1960–1964	3.5	260	75	AEC 1961[a]
	1965–1971	2.3			Data Sheet
	1972–2005	4.6	345	75	AEC 1972[a]
H-3	1972–1994	35	105	3	AEC 1972[a], AEC 1974
	1995–2005	1.4			Andersen, Perry, and Ruhter 1995; Rielly 2001
Co-60	1963–2005	0.01			Rich 1990
Sr-90	1953–1959	9.2			Ebersole and Flygare 1957
	1960–1970	0.2	20	100	AEC 1961
	1971–1989	0.023	1.7	75	AEC 1972[a], AEC 1974[a]
	1990–1994	0.01			Rich 1990
	1995–2005		1.9		Andersen, Perry, and Ruhter 1995
I-131	1963–2005	0.01			Rich 1990
Cs-134	1963–2005	0.01			Rich 1990
Cs-137	1963–2005	0.01			Rich 1990
Th-230	1974–2005	4.6 E-5	0.046	1000	AEC 1974[a]
Np-237	1974–2005	4.6 E-5	0.046	1000	AEC 1974[a]
U (FP)	1954–1985	14 µg/L			Assumed value
U (KPA)	1985–2005	0.2 µg/L			Rich 1990
U-233/234	1970–1994	1.E-4			Rich 1990
	1995–2005		0.041		Andersen, Perry, and Ruhter 1995
U-235	1970–1994	1.E-4			Rich 1990
	1995–2005		0.038		Andersen, Perry, and Ruhter 1995
U-238	1970–1994	1.E-4			Rich 1990
	1995–2005		0.030		Andersen, Perry, and Ruhter 1995
Pu-238	1974–1989	2.3 E-5	0.023	1000	AEC 1974[a]
	1990–1994	6 E-5			Rich 1990
	1995–2005		0.022		Andersen, Perry, and Ruhter 1995
Pu-239/240	1964–1970	4.E-4			AEC 1964[a]
	1971–1973	5.E-4	0.5	1000	AEC 1972[a]
	1974–1989	2 E-5	0.02	1000	AEC 1974[a]
	1990–1994	6.E-5			Rich 1990
	1995–2005		0.027		Andersen, Perry, and Ruhter 1995
Am-241	1974–1989	7 E-5	0.07	1000	AEC 1974[a]
	1990–1994	2. E-4			Rich 1990
	1995–2005		0.023		Andersen, Perry, and Ruhter 1995
Cm-244	1974–2005	1.4 E-5	0.014	1000	AEC 1974[a]
Cf-252	1974–2005	1.4 E-5	0.014	1000	AEC 1974[a]

a. MDA calculated from inferred 2σ uncertainty.

Table 5-10. MDAs for fecal samples by period.

Radiation/ radionuclide	Period	Fecal[a] (pCi/sample)	Reference
Co-60	1963–2005	10	Rich 1990
Sr-90	1963–1994	10	Rich 1990
	1995–2005	1.9	Andersen, Perry, and Ruhter 1995; Rielly 2001
Cs-134	1963–2005	10	Rich 1990
Cs-137	1963–1999	10	Rich 1990
	2000–2005	0.3	BBI 2000
Th-230	1974–2005	0.03	AEC 1974
Np-237	1974–2005	0.03	AEC 1974
U-233/234	1970–2002	0.041	Andersen, Perry, and Ruhter 1995; Rielly 2001
	2003–2005	0.05	Bhatt 2003
U-235	1970–2003	0.038	Andersen, Perry, and Ruhter 1995; Rielly 2001
	2003–2005	0.09	Bhatt 2003
U-238	1970–1994	0.5	Rich 1990
	1995–2002	0.03	Andersen, Perry, and Ruhter 1995; BBI 2000; Rielly 2001
	2003–2005	0.09	Bhatt 2003
Pu-238	1974–1994	0.03	AEC 1974
	1995–2002	0.022	Andersen, Perry, and Ruhter 1995; Rielly 2001
	2003–2005	0.02	Bhatt 2003
Pu-239/240	1964–1973	0.4	AEC 1964[b]
	1974–1994	0.02	AEC 1974
	1995–2005	0.03	Andersen, Perry, and Ruhter 1995; BBI 2000; Rielly 2001; Bhatt 2002
Am-241	1974–1994	0.07	AEC 1974
	1995–2001	0.023	Andersen, Perry, and Ruhter 1995; Rielly 2001
	2002–2005	0.04	Bhatt 2002
Cm-244	1974–2005	0.02	AEC 1974
Cf-252	1974–2005	0.02	AEC 1974

a. When sample size is not identified in individual's records, assume the activity is that excreted per day.
b. MDA calculated from inferred 2σ uncertainty.

Table 5-11. *In vivo* MDAs by period.

Radiation/ radionuclide	Period	*In vivo* result (nCi)	Count time (min)	Reference
Cr-51	1962–2000	12	10	Percival and Anderson 1962[a]
	2001–2005	32	5	Rielly 2001
Mn-54	1962–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, Ruhter 1995
	2001–2005	2.6	5	Rielly 2001
	2001–2005	1.3	10	Rielly 2001
Fe-59	1962–2001	4.5	5	Rielly 2001
	2001–2005	1.5	10	Rielly 2001
Co-58	1962–2000	12	10	Percival and Anderson 1962[a]
	2001–2005	2.5	5	Rielly 2001
	2001–2005	1.1	10	Rielly 2001
Co-60	1962–1970	12	10	Percival and Anderson 1962[a]
	1971–1988	5	10	AEC 1972[a]; AEC 1974
	1989	7	10	Martin 1989
	1990–1992	2 (lung)		Rich 1990
	1993–2000	7	10	Grothaus 1993; Andersen, Perry and Ruhter 1995
	2001–2005	2.5	5	Rielly 2001
	2001–2005	1.1	10	Rielly 2001
Zn-65	1962–1988	12	10	Percival and Anderson 1962[a]
	1989–2000	10	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995

Radiation/ radionuclide	Period	*In vivo* result (nCi)	Count time (min)	Reference
	2001–2005	4.9	5	Rielly 2001
	2001–2005	2	10	Rielly 2001
Sr/Y-90	1968–1977	70 (skull)	10	AEC 1969[a]; AEC 1972[a]; AEC 1974
	1978–2005	34 (skull)	10	Martin 1989; Grothaus 1993
Zr/Nb-95	1962–1988	12	10	Percival and Anderson 1962[a]
	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–2005	2.6	5	Rielly 2001
Ru-106	2001–2005	27	5	Rielly 2001
	2001–2005	7.6	10	Rielly 2001
Ag-110^{m}	1962–2005	12	10	Percival and Anderson 1962[a]
Sb-125	1962–2005	14	10	Martin 1989; Grothaus 1993
I-131	1962–1989	12	10	Percival and Anderson 1962
	1990–1992	2 (thyroid)	10	Rich 1990
	1993–2000	0.3 (thyroid)	10	Grothaus 1993
	2001–2005	3.8	5	Rielly 2001
	2001–2005	0.13 (thyroid)		Rielly 2001
Cs-134	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1990–2005	2(lung)		Rich 1990
	2001–2005	3	5	Rielly 2001
	2001–2005	0.96	10	Rielly 2001
Cs-137	1962–1970	12	10	Percival and Anderson 1962[a]
	1971–1998	5	10	AEC 1972[a]; AEC 1974; Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1999–2000	2. (lung)	10	Rich 1990
	2001–2005	3.1	5	Rielly 2001
	2001–2005	1.9	10	Rielly 2001
Ba/La-140	1962–2005	12	5	Rielly 2001
Ce-141	1962–2005	9.9	5	Rielly 2001
	2001–2005	3.2	10	Rielly 2001
	2001–2005	0.11 (lung)	60	Reilly 2001
Ce-144	1962–2000	50	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–2005	44	5	Rielly 2001
	2001–2005	15	10	Rielly 2001
	2001–2005	0.44 (lung)	60	Rielly 2001
Eu-152	1962–2005	4	10	Rielly 2001
	2001–2005	0.18 (lung)	60	Rielly 2001
Eu-154	1962–2005	2	10	Rielly 2001
Eu-155	1962–2005	1	10	Rielly 2001
Ga-153	1962–2005	6.5	10	Rielly 2001
	2001–2005	0.096 (lung)	60	Rielly 2001
Hf-181	1962–2005	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
Ta-182	1962–2005	12	10	Percival and Anderson 1962[a]
Hg-203	1962–2005	12	10	Percival and Anderson 1962[a]
Th-230	1974–2005		1,000	AEC 1974
Th-234	2001–2005	1.4 (lung)	60	Rielly 2001
Np-237	1974–2005			AEC 1974
U-235	1993	0.2 (wound)	20	Grothaus 1993
	1962–2005	0.2 (lung)		Rich 1990
	2001	0.11 (lung)	60	Rielly 2001
U-dep/nat	1989	3 (lung)	60	Martin 1989; Grothaus 1993
Pu-238	1989–1998	26 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	1 (wound)	20	Grothaus 1993
	1999–2000	30 (lung)		Rich 1990

Radiation/ radionuclide	Period	*In vivo* result (nCi)	Count time (min)	Reference
	2001	54 (lung)	60	Rielly 2001
Pu-239/240	1971–1993	30	100	AEC 1972[a]
	1974–1988	74 (lung)	100	AEC 1974
	1989–1988	80 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	2 (wound)	20	Grothaus 1993
	1990–2000	30 (lung)		Rich 1990
	2001–2005	140 (lung)	60	Rielly 2001
Am-241	1989–1999	0.6 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	0.1 (wound)	20	Grothaus 1993
	1990–2000	0.2 (lung)		Rich 1990
	2001–2005	0.14 (lung)	60	Rielly 2001

a. MDA calculated from inferred 2σ uncertainty.

products. As a consequence, the *in vivo* counting program was used to count (1) all terminating employees, (2) employees suspected of having a possible internal intake, and (3) any interested employee on an annual basis. Only those activities greater than 0.1 µCi were further quantified. This level was determined to be less than one-tenth of the MPBB for most of the gamma-emitting isotopes.

5.6 DEFAULT FOR MISSED DOSE

Based on the ANL-W operational environment characteristics, a number of missed dose default assumptions have been derived. The breeder reactors have a fast neutron spectrum and stainless-steel clad fuel. Rather than include all of the radionuclides in the default summary table for missed dose, only ^{90}Sr, ^{137}Cs, ^{144}Ce, and ^{239}Pu are included for stainless-steel fuels. Cesium-137 was selected because it is most commonly reported in the *in vivo* results rather than for its dose contribution. The potential missed inhalation dose from the other radionuclides is accounted for by weighting the dose from these selected radionuclides by the weighting factors. This gives an equivalent to 100% of the dose from the radionuclide distributions for the three types of fuels. Table 5-12 is a summary of these recommended defaults.

Table 5-12. Default table for missed dose.

Period	Based on	Recommendation	Basis
Start up date through 1960	Urine gross β	Calculate chronic Sr-90 intake that results in a urine activity of 0.4 × gross β	Typical β activity is 0.33 Sr-90, 0.33 Y-90 and 0.33 Cs-137. Use of 0.4 is favorable to claimants.
		Cs-137 intake = Sr-90 intake	Half-lives and fission yields of Cs-137 are Sr-90 are approximately equal.
		Pu-239 intake = 0.004 × Sr-90 intake	Pu:Sr-90 ratio of 0.003 weighted by a factor of 1.2.
		Ce-144 intake = 2.4 × Sr-90 intake	Ce-144:Sr-90 ratio of 2 weighted by a factor of 1.2.
1961–1980	*In vivo* Cs-137	Calculate chronic Cs-137 intake that results in the *in vivo* measurement	Not applicable.
		Sr-90 intake = Cs-137 intake when no *in vitro* measurement	Half-lives and fission yields are approximately equal.
		Pu-239 intake = 0.004 × Cs-137 intake when no *in vitro* measurement	Pu:Cs-137 ratio of 0.003 in stainless-steel fuel weighted by a factor of 1.2.
		Ce-144 intake = 2.4 × Cs-137 assigned intake when no measurement	Ce-144:Cs-137 ratio of 2 in stainless-steel fuel weighted by a factor of 1.2.
1981–2005	Bioassay	Use bioassay results.	Not applicable.

For most of the history of the ANL-W facilities, personnel dosimeters were issued to all workers who entered the security access control points at each facility, regardless of work assignments. For example, administrative and clerical personnel were required to wear these radiation-monitoring dosimeters even though they were not exposed to elevated background levels or internal dose potential. Moreover, WBC was made available to any worker regardless of the likelihood of exposure to radiation. If exposure was likely, the worker was requested to have a whole-body count (ES&H 1989).

5.7 UNMONITORED WORKERS

If no detectable external dose or WBC information was recorded and no *in vitro* samples were recorded, the person should be considered an unmonitored worker for internal dose purposes and only the environmental dose should be included.

As noted above, ANL-W personnel dosimeters were issued to all workers. Many of these workers, due to the nature of their work environment, would not have inhaled or ingested radioactivity and therefore would not have been subject to routine bioassay.

Most radioactivity encountered was well tagged with beta/gamma activity, which produced measurable direct radiation doses on a personnel dosimeter. Therefore, the probability that a worker with no external dose received a significant unmonitored internal intake of radioactive material is very low. Individuals who were not issued a personal dosimeter and have no record of internal dose monitoring should be assigned only the environmental dose for the facility.

At ANL-W each construction job was evaluated to determine if radiation exposure or internal dose could be received. When construction work was done in an area with potential radiation exposure, including internal dose exposure, construction workers were monitored in the same fashion as radiation workers (Andersen, Perry and Ruhter 1995 and Reilly 2001). Construction workers who were issued personnel dosimeters should be treated the same as facility employees who were issued personnel dosimeters. Construction workers who were not issued a personnel dosimeter should be assigned only the environmental dose for the facility.

5.8 ATTRIBUTIONS AND ANNOTATIONS

All information that requires identification was addressed via references integrated into the reference section of this document.

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. His work involved management, direction, or implementation of radiation protection and/or health physics program policies, procedures, or practices in relation 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 to reconstruction of radiation doses that were experienced by claimants who worked at the site. In all cases where such information or previous studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source.

REFERENCES

ACC (American Cyanamid Company), 1952, *CPP Health Physics Manual*, IDO-14055, Atomic Energy Division, Idaho Falls, Idaho.

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

AEC (U.S. Atomic Energy Commission), 1961, *Annual Report of Health & Safety Division, 1960,* IDO-12019, Idaho Operations Office, Idaho Falls, Idaho.

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

AEC (U.S. Atomic Energy Commission), 1963, *Annual Progress Report 1962, Health & Safety Division,* IDO-12033, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1964, *Health & Safety Division Annual Progress Report 1963,* IDO-12037, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1968, *Standards for Radiation Protection*, AEC Manual Chapter 0524 and Appendix, Washington, D.C., November 8.

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

AEC (U.S. Atomic Energy Commission), 1972, *Annual Report 1971, Health Services Laboratory,* IDO-12076, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1974, *Detection Limits*, Idaho Operations Office, Health Services Laboratory, Analytical Chemistry Branch, Idaho Falls, Idaho, December.

AEC (U.S. Atomic Energy Commission), 1975, *Occupational Radiation Exposure Information*, AEC Manual Chapter 0525, Washington, D.C., January 7.

Andersen, B. L., O. R. Perry, and P. E. Ruhter, 1995, *Technical Basis for Internal Dosimetry at LITCO*, EDF # RDRU-95-3.0, U.S. Department of Energy, Idaho Falls, Idaho, January 31.

BBI (Bechtel BWXT Idaho), 2000, *Quality Assurance Project Plan for The Analytical Laboratories Department Radioanalytical Section,* PLN-153, Rev. 1, Idaho Falls, Idaho, August 17.

Bhatt, A. R., 2002, *Bioassay Sample Analysis Technical Basis Document,* EDF # In-Vitro-1, July 15, 2002, Project File # INEEL-In-Vitro-1, U. S. Department of Energy, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.

Bhatt, A. R., 2003, *In Vitro Analysis Measurement Techniques*, INEEL/INT-03-00702, Rev. 0, U.S. Department of Energy, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.

Burke, L. L., 1993, “Estimation of [Privacy Act information]’s Dose from the Plutonium Contaminated Hand Wound,” memorandum to D. S. Kirschner (sic), ANL-W-ESWM-(LLB)-93-24, May 11.

Carr, G., 1990, "ANL-W Skin Contaminations for 1989," memorandum to C. S. Abrams, ANL-W-SSS-(GC)-90-06, March 6.

Courtney, J. C., J. P. Madison, C. E. Holson, R. L. Black, F. L. Dilorenzo, J. B. Anderson, E. Hylsky, and L. D. Lau, 1980, "Sampling of Airborne Radioactivity in a Hot Fuel Examination Facility," *Health Physics*, volume 39, pp. 736–750.

DOE (U.S. Department of Energy), 1981, *Requirements for Radiation Protection*, Order 5480.1A, Chapter XI, Washington, D.C., April 29.

DOE (U.S. Department of Energy), 1988, *Radiation Protection for Occupational Workers*, Order 5480.11, Washington, D.C., December 21.

DOE (U.S. Department of Energy), 1994, *Radiological Control Manual*, DOE/EH-0256T, Rev. 1, Office of Environment, Safety and Health, Washington, D.C.

Ebersole, E. R., and J. K. Flygare, Jr., 1957, *Determination of Gross Beta Activity in Urine with K-40 Correction*, presented at the First Annual Meeting of the Health Physics Society, June 25 to 27.

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

ES&H (Environment Safety and Health) Manual, Section VII, Chapter 9, *ANL-W Routine Bioassay Program*, Rev. 2, 2/1/89.

Grothaus, G. E., 1993, "MDAs For In-Vivo Counting," letter to C. Bloom, EG&G, ES&Q, February 19.

Holson, C. E., 1991, "1990 Radiation Exposure Summary," memorandum to G. C. Marshall, ANL-W-SES-(CEH)-91-42, February 20.

HPS (Health Physics Society), 1996, *An American National Standard, Performance Criteria for Radiobioassay*, ANSI N13.30-1996, American National Standards Institute, New York, New York.

ICRP (International Commission on Radiological Protection), 1979, *Limits for Intakes of Radionuclides by Workers, Part 1*, Publication 30, Pergamon Press, Oxford, England.

ICRP (International Commission on Radiological Protection), 1995, *Dose Coefficients for Intakes of Radionuclides by Workers,* Publication 68, Pergamon Press, Oxford, England.

Kirchner, D. S., 1992, "1991 Radiation Exposure Summary," memorandum to G. C. Marshall, ANL-W-ESWM-(DSK)-92-34, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 3.

Marshall, G. C., 2002, "2001 Radiation Exposure & Personnel Contamination Summary," memorandum to Distribution, February 13.

Martin, D. B., 1989, "MDAs For In-Vivo Counting," letter to C. Bloom, EG&G, RWMC, November 16.

Nielsen, C. W., 1992a, "ANL-W Skin Contaminations for 1990," memorandum to G. C. Marshall, ANL-W-ESWM-(CWN)-92-03, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 27.

Nielsen, C. W., 1992b, "ANL-W Skin Contaminations for 1991," memorandum to G. C. Marshall, ANL-W-ESWM-(CWN)-92-04, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 31.

Nielsen, C. W., 1993a, "ANL-W Skin Contaminations for 1992," memorandum to G. C. Marshall, ANL-W-ESWM-(CWN)-93-03, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 11.

Nielsen, C. W., 1993b, "1992 Radiation Exposure Summary," memorandum to G. C. Marshall, ANL-W-ESWM-(CWN)-93-004, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 11.

Nielsen, C. W., 1994a, "ANL-W Skin Contaminations for 1993," memorandum to G. C. Marshall, ANL-W-ESWM-(CWN)-94-10, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 27.

Nielsen, C. W., 1994b, "1993 Radiation Exposure Summary," memorandum to G. C. Marshall, ANL-W-ESWM-(CWN)-94-1, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 31.

Nielsen, C. W., 1995a, "ANL-W Skin Contaminations for 1994," memorandum to G. C. Marshall, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 9.

Nielsen, C. W., 1995b, "1994 Radiation Exposure Summary," memorandum to G. C. Marshall, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 25.

Nielsen, C. W., 1996a, "Internal Dosimetry Program," Part 3 of *Argonne National Laboratory – West, Radiation, Fire, and Safety Procedures Manual*, Argonne National Laboratory-West, Idaho Falls, Idaho, December.

Nielsen, C. W., 1996b, "ANL-W Skin Contaminations for 1995," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 12.

Nielsen, C. W., 1996c, "1995 Radiation Exposure Summary," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 9.

Nielsen, C. W., 1997a, "ANL-W Skin Contaminations for 1996," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 12.

Nielsen, C. W., 1997b, "1996 Radiation Exposure Summary," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 27.

Nielsen, C. W., 1998a, "ANL-W Skin Contaminations for 1997," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, January 29.

Nielsen, C. W., 1998b, "1997 Radiation Exposure Summary," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 2.

Nielsen, C. W., 1999a, "ANL-W Skin Contaminations for 1998," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 2.

Nielsen, C. W., 1999b, "1998 Radiation Exposure Summary," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 10.

Nielsen, C. W., 2000a, "ANL-W Skin Contaminations for 1999," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 3.

Nielsen, C. W., 2000b, "CY 1999 Radiation Exposure Summary," memorandum to W. R. Vroman, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, March 30.

Nielsen, C. W., 2003, "Calendar Year 2002 Radiation Exposure & Personnel Contamination Summary," memorandum to Distribution, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, April 22.

Nielsen, C. W., 2004, "Calendar Year 2003 Radiation Exposure & Personnel Contamination Summary," memorandum to Distribution, University of Chicago, Argonne National Laboratory–West, Idaho Falls, Idaho, February 6.

ORAUT (Oak Ridge Associated Universities Team), 2004, *Technical Basis Document for the Argonne National Laboratory - West (ANL-W) Site Description*, ORAUT-TKBS-0026-2, Rev. 00, Oak Ridge, Tennessee, September 9.

Percival, D. R., and Anderson, J. I., 1962, *Whole-Body Counting at the National Reactor Testing Station*, U.S. Atomic Energy Commission, Health and Safety Division, Idaho Falls, Idaho.

RAC (Radiological Assessment Corporation), 2002, *Final Report, Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory*, U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, Atlanta, Georgia.

Rich, B. L., 1990, *Internal Dosimetry and Control*, Informal Report EGG-ESQ-8808, U.S. Department of Energy, Idaho National Engineering Laboratory, Idaho Falls, Idaho, May.

Rielly, P. S. 2001, *INEEL M & O Contractor Technical Basis for Internal Dosimetry, General Technical Basis and Facility Specific Documents*, Idaho Operations Office, Idaho Falls, Idaho, May 30.

Sill, C. W., J. I. Anderson, and D. R. Percival, 1964, "Comparison of Excretion Analysis with Whole-Body Counting for Assessment of Internal Radioactive Contaminants," *Proceedings of International Conference*, pp. 217–229.

Stacy, S. M., 2000, *Proving the Principle - A History of the Idaho National Engineering and Environmental Laboratory 1949 - 1999*, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

Vroman, W. R., 2001, "2000 Radiation Exposure & Personnel Contamination Summary," memorandum to Distribution, February 22.

GLOSSARY

absorption
As used in this internal dosimetry section, absorption refers to the material being transported to fluids and other organs as well as radiation energy being imparted.

activation
The process of inducing radioactivity by irradiation.

becquerel
A unit of radioactivity equal to one-disintegration per second.

beta radiation
Radiation consisting of electrons or positrons emitted at high velocity from the nuclei of certain radioactive elements. Most direct fission products emit beta radiation.

breeder reactor
A nuclear reactor concept in which the operation produces a net increase in fissionable material.

cladding
The outer layer of material encasing a reactor fuel element (e.g., aluminum or zirconium). Cladding promotes the transfer of heat from the fuel to the coolant and contains fission products and activation products that result from the fissioning of the fuel.

contamination, radioactive
Radioactive material where it does not belong.

core
That part of the reactor consisting of the fuel and some of the control elements for reactor operation.

curie
A unit of radioactivity equal to 3.7×10^{10} disintegrations per second.

decontaminate
Removing a contaminant, such as a radioactive material, from an undesired location.

dosimeter
A device used to measure accumulated radiation exposure.

dosimetry
The science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external or internal sources of radiation.

fission
A nuclear transformation characterized by the splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy.

fission product
Radionuclides resulting from fission.

fuel reprocessing
A chemical process, usually involving several steps, that recovers ^{235}U and other fissionable products from spent fuel.

gamma rays
Short wave length electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture) in an energy range of 10,000 electron volts to 9 million electron volts.

half-life
The time it takes for one-half of any given number of unstable atoms to decay (disintegrate).

ionizing radiation
Electromagnetic or particulate radiation capable of producing charged particles through interactions with matter.

in vitro
In glass. Outside the living body and in an artificial environment. Internal bioassay sampling, such as fecal samples or urine samples.

in vivo
In the living. In the living body of a plant or animal. Bioassay sampling by whole-body counting.

isotope
Nuclides having the same number of protons in their nuclei (same atomic number), but having a differing number of neutrons (different mass number).

millirem
A unit of radiation dose equivalent (or equivalent dose) equal to one-thousandth of a rem (see rem).

microcurie
A measure of radioactivity equal to one-millionth of a curie.

mixed waste
Waste that contains hazardous and radioactive materials.

natural uranium
Uranium that has not been through an enrichment process.

neutron
A basic particle in a nuclear reaction, electrically neutral, with nearly the same mass as a ^{1}H atom.

nuclear waste
A general term used for the byproduct unusable material resulting from nuclear reactions, including high-level, intermediate, low-level, mixed and TRU waste.

nucleus
That part of an atom consisting of the total positive electrical charge and most of the mass.

quality factor, Q
A modifying factor used to derive dose equivalent from absorbed dose.

radiation
Energy transferred through air or some other media in the form of particles or waves (see ionizing radiation).

radioactivity
The spontaneous emission of radiation, generally alpha or beta particles, gamma or X-rays, or neutrons from unstable atoms.

radionuclide
A radioactive species of an atom characterized by the constitution of its nucleus specified by atomic number (the number of protons), and the mass number (equal to the number of protons plus neutrons).

rem
A unit of dose equivalent, equal to the product of the absorbed dose and the quality factor. The word derives from *roentgen equivalent in man*.

shielding
Any material or obstruction that absorbs (or attenuates) radiation to protect personnel or materials.

spent nuclear fuel
Reactor fuel containing fission and activation products that can no longer economically sustain a chain reaction.

type
Refers to the rate of absorption from lung to blood of inhaled radioactive materials and includes types F (fast), M (moderate), and S (slow).

transuranic (TRU) waste
Contaminated waste materials with nuclides having an atomic number greater than 92, a half-life over 20 yr and concentration greater than or equal to 100 nCi/g.

U.S. Atomic Energy Commission
An agency established by the U.S. Government for oversight of nuclear weapons and power production; a predecessor to the U.S. Department of Energy.

X-ray
Ionizing electromagnetic radiation of extranuclear (outside the nucleus) origin.

zirconium
A metallic element highly resistant to corrosion and often used to make cladding for nuclear fuel. It is sometimes alloyed in small amounts in the fuel itself.