ORAU Team **Dose Reconstruction Project for NIOSH** Technical Basis Document for the Argonne National Laboratory – West – Occupational Internal Dose	Document Number: ORAUT-TKBS-0026-5 Effective Date: 01/20/2005 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 26
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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AECL	administrative exposure control level
AEDE	annual effective dose equivalent
AMAD	activity median aerodynamic diameter
ANL-W	Argonne National Laboratory-West
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
Ci	curie
cm	centimeter
COO	Chicago Operations Office
CPP	Chemical Processing Plant
cpm	counts per minute
d	day
DAC	derived air concentration
DAC-hr	derived air concentration-hour
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
EBR-I	Experimental Breeder Reactor No. 1
EBR-II	Experimental Breeder Reactor No. 2
ERDA	Energy Research and Development Administration
F	fast absorption type
FASS	Fixed Air Sampling System
FCF	fuel cycle facility
hr	hour
HFEF	Hot Fuel Examination Facility
HP	health physicist
H&S	Health and Safety
HSD	Health and Safety Division
HSL	Health Services Laboratory
ICRP	International Commission on Radiological Protection
IDO	Idaho Operations Office
INEEL	Idaho National Engineering and Environmental Laboratory
INL	Idaho National Laboratory
L	liter
m	meter
M	moderate absorption type
mCi	millicurie

MDA	minimum detectable activity or amount
MDL	minimum detectable level
MFP	mixed fission products
mg	milligram
min	minute
mL	milliliter
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPC_a	MPC for airborne activity
mrem	millirem
mrep	millirep
NaI(Tl)	sodium iodide doped with thallium
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NRTS	National Reactor Testing Station
pCi	picocurie
RAM	radiation area monitor or remote area monitor
RDR	Radiation Dosimetry and Records
RESL	Radiological Environmental Sciences Laboratory
S	slow absorption type
TBD	technical basis document
TLV	threshold limit value
TRU	transuranic
U.S.C.	United States Code
WBC	whole-body counting
yr	year
ZPPR	Zero Power Plutonium Reactor (later Zero Power Physics Reactor)
α	alpha particle
β	beta particle
γ	gamma
σ	standard deviation
µCi	microcurie
µg	microgram
µm	micrometer

5.1 INTRODUCTION AND HISTORICAL OVERVIEW

Technical Basis Documents (TBDs) and Site Profile Documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health (NIOSH) in the completion of the work required for each individual 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 of 2000 [42 U.S.C. § 7384l(5) and (12)].

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 miles west of Idaho Falls in southeastern Idaho. NRTS was later the Idaho National Engineering Laboratory (INEL) and is now the Idaho National Engineering and Environmental Laboratory (INEEL). Argonne National Laboratory (ANL) near Chicago established a branch at the NRTS known as ANL – West (ANL-W) where they built and operated several reactors which 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, operating contractor from 1951 through January 2005. ANL-W will be merged with the research side of the INEEL to form the INL and will be operated by Battelle Energy Alliance beginning in February 2005.

Each of the original AEC Laboratories was unique in both mission and location. Since the early days of the AEC programs represented the beginnings of the nuclear age, significant technical developments were a necessity, not the least of which were developments in radiation safety areas. Some of the unique characteristics of radiation safety (and internal dosimetry specifically) at the NRTS/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 ANL-W beginning was 8-10 years later than ANL-East, Oak Ridge National Laboratory, and Hanford. During those developmental years significant technical progress in professional skills, instrumentation, analyses, procedures, and techniques were accomplished.

- The AEC/DOE Chicago field office was responsible for and had oversight of the ANL-W program.

- To provide needed consistency of radiation safety programs at the NRTS, the AEC through the Idaho field office 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 Services Laboratory (HSL), Health and Safety Division (HSD), Idaho Center for Radiological and Environmental Sciences (ICRES), and now to the Radiological and Environmental Sciences Laboratory (RESL). 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 (MDLs), 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 which 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 and 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 5.1.1 provides a facility description, and Section 5.1.2 details the radionuclides of concern. Section 5.2 describes the INEEL 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.

5.1.1 Argonne National Laboratory-West

ANL-W, started in 1951, continues to conduct nuclear power developmental programs. Although ANL-W receives internal dosimetry support from the INEEL service laboratories, its radiological safety programs operate under authority of the DOE Chicago Operations Office.

Nine experimental reactors under the technical direction of ANL-W were operated at two locations, one on the southwest side of the Site near 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-destruct 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).

5.1.2 Radioactive Nuclides of Concern and Solubility

ANL-W facilities and activities have been related primarily to experimental reactor design and development. The ANL-W Site Description TBD (ORAU 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 product (MFP) (primarily aged), uranium and decay products, and transuranic (TRU) and decay products. The uranium has had various enrichments from depleted to at least 68% enriched, both used in EBR II. Most internal doses have been identified following 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, indicating a relatively insoluble material. The chemical explanation is that radioactive materials oxidize rapidly, forming less soluble compounds. A default assumption of M or S would be appropriate, where ICRP 68 specifies these for the nuclide. Strontium is an exception; type F should be assumed.

Table 5-1. Radionuclides of concern for ANL-W (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

5.2 RADIOLOGICAL PROTECTION PROGRAM MANAGEMENT AND SUPPORT

The contract with the University of Chicago to operate the ANL-W facilities has not changed in this 54-yr period. The Chicago Operations Office (COO) provided oversight for ANL-W programs and facilities. The ANL-W program used site support services, including internal dosimetry support.

The personnel dosimetry records have been and are documented and permanently maintained. Records about individual facility/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 H&S Laboratories 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 has been conducted since 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, typically annually. 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 Health and Safety Division (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 INEEL prime contractor. At that time, *in vitro* analytical functions were transferred to an onsite INEEL analytical laboratory. The *in vivo* counting laboratory provides support directly through the Radiation Dosimetry and Records (RDR) organization, which administers external and internal dosimetry support programs. The subject matter expert reviews, validates, and prepares official internal dose assessments. The RDR unit functions include documentation and records custodial responsibilities.

Table 5-2. ANL-W 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 1962a AEC 1963a
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.	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.			Investigating Internal doses resulting from all confirmed intakes are to be evaluated.		
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			

AEDE = annual effective dose equivalent; ERDA = Energy Research and Development Administration.

5.2.3 Data Codes and Investigation Levels

The changing 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 used in internal dose assessments and analytical data sheets has 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 INEEL codes for various site areas.

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.

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) have also changed little from the early years to the present. Later procedures (DOE 1988) set specific limits on those positive bioassay results that could result in 100 mrem annual effective dose equivalent or above as the point at which followup and reporting was required. With the DOE *Radiological Control Manual* (DOE 1994), this changed to 100 mrem CEDE. 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 and the recording of even nondetectable radioactivity in bioassay samples, although the request for and number of followup 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 evacuating personnel, increasing personnel protection equipment (e.g., respirators), and requesting bioassay analyses to identify possible internal intake.

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. INEEL and ANL-W area codes that could be in claimant dose files.

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

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 preventative 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 current operating procedures.

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-present	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/derived air concentrations (MPCs/DACs) above 1×10^{-9} µCi/cm^3. TRU materials and uranium are/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 (FASS) beginning in 1976 (Courtney 1980). Concentrations have averaged between 1 and 5×10^{-15} µCi cm^{-3} for alpha emitters and between 0.4 and 4.0×10^{-13} µCi cm^{-3} for beta emitters in the first three years of operation. This information could 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, coupled with facility memoranda and reports, documented the analytical detection capability in the 1950s and 1960s. Radionuclide identification by their 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 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 H&S Division 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, replacing the previously used 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 yr 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, gross beta counted, 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 is a personnel dose file on each individual which 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 (NOCTS) 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; Nielsen 1992b; Nielsen 1993a; Nielsen 1993b; Nielsen 1994a; Nielsen 1994b; Nielsen 1995a; Nielsen 1995b; Nielsen 1996b; Nielsen 1996c; Nielsen 1997a; Nielsen 1997b; Nielsen 1998a; Nielsen 1998b; Nielsen 1999a; Nielsen 1999b; Nielsen 2000a; Nielsen 2000b; Nielsen 2003; Nielsen 2004; Vroman 2001). Blank spaces mean that the information was not available. The skin contaminations tend to be a precursor for positive bioassay and the number has been countable and tracked since at least 1984. The activities associated have not been extremely high. In 1993 there were a large number of skin and clothing contaminations associated with the Fuel Cycle Facility (FCF) which was being refurbished. Almost everybody received a whole body count until the late 1990s. In 1998, a FCF seal tube repair led to 11 positive WB counts on 5 people. The highest calculated CEDE was 1 mrem. The other positive count was from a medical test about 3 years previous. 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 seeing anything significant in dose from tritium. The Pu fecal assay led to mostly false positive results where a subsequent fecal assay failed to show any activity. In 1993 a person received a hand wound and was chelated resulting in two positive urine counts but no positive fecal count (Burke 1993). A 100 mrem CEDE was assigned. This recent history suggests rather modest internal exposures occurred at ANL-W.

Table 5-8. ANL-W Recent History of Bioassay Sampling.

Year	# Skin Contam Events	#Clothng Contam Events	Maximum Beta Contam	# Whole Body Counts	Positive WBC	# Fecal Pu Assay	Pos *in vitro*	# Tritium Assay	Pos Tritium	Max T Dose
1984	20		**cpm till**							**mrem**
1985	17		**1992**							
1986	7		**dpm after**							
1987	5		**1992**							
1988	16									
1989	18		8000							
1990	10			542	0	44	0			
1991	10		12000	648	0	60	0	287	29	14
1992	15	7	8000	609	0	69	0	40	8	<1
1993	23	58	500000	1082	0	33	2 Urine	48	6	<1
1994	7	21	139000	837	0	70	1 FP[a]	46	14	<1
1995	3	10	784000	758	0	106	1 FP	106	10	<1
1996	2	11	186000	273	0	75	1 FP	139	13	<1
1997	5	10	135000	329	1	76	2 FP 1 P[a]	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

5.4 MINIMUM DETECTABLE ACTIVITIES

In compliance with the November 1998 Code of Federal Regulations requirement (10 CFR 835) for DOE Laboratory Accreditation Program (DOELAP), and based on American National Standards Institute (ANSI) N 13.30, *Performance Criteria for Radiobioassay* (ANSI 1996), both the *in vitro* and *in vivo* radiobioassay laboratories at INEEL received DOELAP accreditation in February 1998. In accordance with this accreditation, MDAs and decision levels at the 95% confidence level are performed. Tables 5-9 and 5-10 list current MDAs for urine and fecal sample analysis, respectively, along with values gleaned from historical documents. The earliest reference (Ebersole and Flygare 1957) defines a detection limit as “twice the standard deviation of its determination”. Thus the detection limits in early references have been multiplied by 2.33 to determine an MDA similar to the ANSI defined values. A large majority of the urine samples taken were single voidings; 24-hr samples were used for special sampling purposes (i.e., followup samples, primarily to extend the sensitivity). The MDA for urine sample is given in both pCi/ml and in pCi/sample if the sample aliquot size is stated, and if the size is not stated only the reference value is provided. Most bioassay data has units of activity/sample with the units of the activity changing from dpm to µCi to Bq and back again. The MDAs listed are those for the primary samples. The recommended time periods for the MDA values have been included in the tables. Table 5-11 lists the current in vivo MDAs along with values gleaned from historical documents and the recommended periods for use of the MDA values.

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

Radiation/ radionuclide	Time 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-present	1	5	5	AEC 1972[a]
Gross γ	1960-1964	3.5	260	75	AEC 1961[a]
	1965-1971	2.3			Data Sheet
	1972-present	4.6	345	75	AEC 1972[a]
H-3	1972-1994	35	105	3	AEC 1972[a] , AEC 1974
	1995-present	1.4			Andersen, Perry, and Ruhter 1995, Rielly 2001
Co-60	1963-present	0.01			Rich 1990
Sr-90	1953-1959	9.2			Ebersole 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-present		1.9		Andersen, Perry, and Ruhter 1995
I-131	1963-present	0.01			Rich 1990
Cs-134	1963-present	0.01			Rich 1990
Cs-137	1963-present	0.01			Rich 1990
Th-230	1974-present	4.6 E-5	0.046	1000	AEC 1974 [a]
Np-237	1974-present	4.6 E-5	0.046	1000	AEC 1974 [a]
U (FP)	1954-1985	14 µg/L			Assumed value
U (KPA)	1985-present	0.2 µg/L			Rich 1990
U-233/234	1970-1994	1.E-4			Rich 1990
	1995-present		0.041		Andersen, Perry, and Ruhter 1995
U-235	1970-1994	1.E-4			Rich 1990
	1995-present		0.038		Andersen, Perry, and Ruhter 1995
U-238	1970-1994	1.E-4			Rich 1990
	1995-present		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-present		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-present		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-present		0.023		Andersen, Perry, and Ruhter 1995
Cm-244	1974-present	1.4 E-5	0.014	1000	AEC 1974 [a]
Cf-252	1974-present	1.4 E-5	0.014	1000	AEC 1974 [a]

a. MDA calculated from inferred 2σ uncertainty.

5.5 DEVELOPMENT AND IMPACT OF WHOLE-BODY COUNTING (WBC)

WBC was introduced at the INEEL in 1961. As early as 1961 one of the fundamental conclusions from the 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 and 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-hr specimen (AEC 1962a;

Sill, Anderson, and Percival 1964). WBC was demonstrated to detect activity as low as 0.01 µCi in a 10-min count (AEC 1962a). This detection level was several orders of magnitude more sensitive than the maximum permissible body burdens (MPBB) for most beta/gamma fission and activation 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.

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

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

Radiation/ radionuclide	Time Period	*In vivo* result (nCi)	Count time (min)	Reference
Cr-51	1962-2000	12	10	Percival and Anderson 1962[a]
	2001-present	32	5	Rielly 2001
Mn-54	1962-2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, Ruhter 1995
	2001-present	2.6	5	Rielly 2001
	2001-present	1.3	10	Rielly 2001
Fe-59	1962-2001	4.5	5	Rielly 2001
	2001-present	1.5	10	Rielly 2001
Co-58	1962-2000	12	10	Percival and Anderson 1962[a]
	2001-present	2.5	5	Rielly 2001
	2001-present	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

Radiation/ radionuclide	Time Period	*In vivo* result (nCi)	Count time (min)	Reference
	1989	7	10	Martin 1989
	1990-1992	2 (lung)		Rich 1990
	1993-2000	7	10	Grothaus 1993; Andersen, Perry and Ruhter 1995
	2001-present	2.5	5	Rielly 2001
	2001-present	1.1	10	Rielly 2001
Zn-65	1962-1988	12	10	Percival 1962[a]
	1989-2000	10	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001-present	4.9	5	Rielly 2001
	2001-present	2	10	Rielly 2001
Sr/Y-90	1968-1977	70 (skull)	10	AEC 1969[a]; AEC 1972[a]; AEC 1974
	1978-present	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-present	2.6	5	Rielly 2001
Ru-106	2001-present	27	5	Rielly 2001
	2001-present	7.6	10	Rielly 2001
Ag-110^{m}	1962-present	12	10	Percival and Anderson 1962[a]
Sb-125	1962-present	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-present	3.8	5	Rielly 2001
	2001-present	0.13 (thyroid)		Rielly 2001
Cs-134	1989-2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1990-present	2(lung)		Rich 1990
	2001-present	3	5	Rielly 2001
	2001-present	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-present	3.1	5	Rielly 2001
	2001-present	1.9	10	Rielly 2001
Ba/La-140	1962-present	12	5	Rielly 2001
Ce-141	1962-present	9.9	5	Rielly 2001
	2001-present	3.2	10	Rielly 2001
	2001-present	0.11 (lung)	60	Reilly 2001
Ce-144	1962-2000	50	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001-present	44	5	Rielly 2001
	2001-present	15	10	Rielly 2001
	2001-present	0.44 (lung)	60	Rielly 2001
Eu-152	1962-present	4	10	Rielly 2001
	2001-present	0.18 (lung)	60	Rielly 2001
Eu-154	1962-present	2	10	Rielly 2001
Eu-155	1962-present	1	10	Rielly 2001
Ga-153	1962-present	6.5	10	Rielly 2001
	2001-present	0.096 (lung)	60	Rielly 2001
Hf-181	1962-present	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
Ta-182	1962-present	12	10	Percival and Anderson 1962[a]
Hg-203	1962-present	12	10	Percival and Anderson 1962[a]
Th-230	1974-present		1,000	AEC 1974
Th-234	2001-present	1.4 (lung)	60	Rielly 2001
Np-237	1974-present			AEC 1974
U-235	1993	0.2 (wound)	20	Grothaus 1993

Radiation/ radionuclide	Time Period	*In vivo* result (nCi)	Count time (min)	Reference
	1962-present	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
	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-present	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-present	0.14 (lung)	60	Rielly 2001

a. MDA calculated from inferred 2-σ uncertainty.

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.

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 backgrounds or internal dose potential. Moreover, whole body counting was made available to any worker irregardless of their likely exposure to radiation. If exposure was likely, the worker was requested to have a whole body count (ES&H 89).

5.7 UNMONITORED WORKERS

If no detectable external dose or whole body count information is 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 subjected to routine bioassay.

Most radioactivity encountered was well tagged with β/γ 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 a radiation worker. 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 the environmental dose for the facility.

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 & 0.33 Cs-137. Use of 0.4 is claimant favorable
		Cs-137 intake = Sr-90 intake	Half-lives and fission yields of Cs-137 & 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	
		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 - present	Bioassay	Use bioassay results.	

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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.

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.

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.

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.