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Subject Expert(s): Brian P. Gleckler

Site Expert(s): Bryce L. Rich, Douglas Wenzel, Norman D. Rohrig, Boyd Levitt, and Henry Peterson

Approval:	Signature on File Jo Ann M. Jenkins, Document Owner	Approval Date:	03/01/2010
Concurrence:	Signature on File John M. Byrne, Objective 1 Manager	Concurrence Date:	03/01/2010
Concurrence:	Signature on File Edward F. Maher, Objective 3 Manager	Concurrence Date:	03/01/2010
Concurrence:	Signature on File Kate Kimpan, Project Director	Concurrence Date:	03/01/2010
Approval:	Signature on File Stuart L. Hinnefeld, Interim Director, DCAS	Approval Date:	03/02/2010

☐ 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
08/11/2004	00	New Technical Basis Document for the Idaho National Engineering and Environmental Laboratory – Occupational Internal Dose. First approved issue. Initiated by Norman D. Rohrig.
10/12/2004	00 PC-1	Approved page change revision. Deletes references to rigorous radiation program on page 39. Initiated by Norman D. Rohrig. Approval: Signature on File 09/30/2004 Norman D. Rohrig, Document Owner Signature on File 10/01/2004 Judson L. Kenoyer, Task 3 Manager Signature on File 10/01/2004 Richard E. Toohey, Project Director Signature on File 10/12/2004 James W. Neton, Associate Director for Science
01/13/2006	01	Approved issue of Revision 01. Revised to change table number format, responds to new information from Worker Outreach Effort and revises MDA tables. Incorporates 1-year decay time for Al fuel in Tables 5-18 and 5-24. Combined "INTEC & Unknown" and "Other Area" categories in Table 5-24 for 1971-80 into "All but ANL-W." Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig.
06/05/2006	01 PC-1	Page change revision to correct error on page 24 in Table 5-12 in Section 5.5 on sample volume in 1980s. Approved issue of Rev 01 PC-1. No sections were deleted. Incorporates NIOSH formal review comments. This revision results in a reduction in assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig. Approval: Signature on File 05/23/2006 John M. Byrne, Document Owner Signature on File 05/23/2006 John M. Byrne, Task 3 Manager Signature on File 05/22/2006 Edward F. Maher, Task 5 Manager Signature on File 05/24/2006 Kate Kimpan, Project Director Signature on File 06/05/2006 James W. Neton, Associate Director for Science
07/30/2007	02	Revision initiated to add Purpose, Scope, and Attributions and Annotations sections. Added references. Updated the formatting and made editorial changes. Corrected error in Fecal MDA for Th-230 and Np-237 in Table 5-13. Constitutes a total rewrite of the document. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Task Manager. Initiated by Jo Ann M. Jenkins.

03/02/2010	03	Revision to incorporate the Argonne National Laboratory - West site. In order to resolve several issues identified in an independent audit of this TBD and other issues identified by ORAUT, the approaches for assessing the INEL bioassay data have been completely revised. The potential lung absorption types to be considered have also been expanded for several radionuclides. These changes generally result in lower organ doses. However, for specific scenarios they can result in higher doses to certain metabolic organs such as the thyroid. The layout of this TBD was completely reorganized to better match the layout recommended in ORAUT-PROC-0031. In addition, this revision fully incorporates the Argonne National Laboratory – West back into this TBD and eliminates the ANL-W Occupational Internal Dose TBD. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jo Ann M. Jenkins.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AEDE	annual effective dose equivalent
AMAD	activity median aerodynamic diameter
ANL-W	Argonne National Laboratory–West (recently renamed the Materials and Fuels Complex [MFC])
ANP	aircraft nuclear propulsion
ATR	Advanced Test Reactor
CADRE	guard force
CAM	continuous air monitor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFA	Central Facilities Area
CFR	Code of Federal Regulations
Ci	curie
COO	Chicago Operations Office
cpm	counts per minute
CPP	Chemical Processing Plant
d	day
DAC	derived air concentration
DE	dose equivalent
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
DU	depleted uranium
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
ETR	Engineering Test Reactor
F	fast absorption type
FCF	Fuel Cutting Facility
ft	foot
g	gram
gal	gallon
GCRE	Gas-Cooled Reactor Experiment
GSD	geometric standard deviation
H&S	Health and Safety
HFEF	Hot Fuel Examination Facility
hr	hour
HSD	Health and Safety Division
HSL	Health Services Laboratory
HTO	tritiated water vapor

ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
IDO	Idaho Operations Office
IET	initial engine test
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
INEEL	Idaho National Engineering & Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
L	liter
LPTF	Low Power Test Facility
m	meter
M	moderate absorption type
MAP	mixed activation product
MDA	minimum detectable activity
MDL	minimum detectable level
MeV	megavolt-electron, 1 million electron-volts
MFC	Materials and Fuels Complex (new name for ANL-W)
MFP	mixed fission product
mi	mile
min	minute
ml	milliliter
mo	month
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
OMRE	Organic Moderated Reactor Experiment
ORAU	Oak Ridge Associated Universities
ORAUT	Oak Ridge Associated Universities Team
PBF	Power Burst Facility
pCi	picocurie
PF	photofluorimetry
POC	probability of causation

RaLa	radioactive lanthanum
RAM	radiation (or remote) area monitor
RCIMS	Radiation Control Information Management System
RDR	Radiation Dosimetry and Records
RESL	Radiological Environmental Sciences Laboratory
RFP	Rocky Flats Plant
RWMC	Radioactive Waste Management Complex
S	slow absorption type
SDA	Subsurface Disposal Area
SL-1	Stationary Low-Power Reactor
SMC	Specific Manufacturing Capability
SPERT	Special Power Excursion Reactor Test
SRDB Ref ID	Site Research Database Reference Identification (number)
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
TRU	transuranic
TSA	Transuranic Storage Area
µCi	microcurie
µg	microgram
µm	micrometer
U.S.C.	United States Code
WBC	whole-body counting/count
WERF	Waste Experimental Reduction Facility
WROC	Waste Reduction Operations Complex
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
α	alpha particle
β	beta particle
γ	gamma ray or photon
σ	standard deviation
3x3	3 inch by 3 inch
§	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, 42 C.F.R. Pt. 82) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work (NIOSH 2007).

The statute also 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 excludes Naval Nuclear Propulsion Facilities from being covered under the Act, 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 occupational radiation exposures are considered valid for inclusion in a 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 (NIOSH 2007):

- Background radiation, including radiation from naturally occurring radon present in conventional structures

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

- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.7.

5.1.1 Purpose

The purpose of this TBD is to document the internal dosimetry program and practices at the Idaho National Laboratory, and to provide the technical basis to be used to evaluate the internal occupational radiation dose for EEOICPA claims.

5.1.2 Scope

This TBD provides supporting documentation to assist in the evaluation of occupational internal doses in accordance with the guidelines described in *Internal Dose Reconstruction Implementation Guideline* (NIOSH 2002). NIOSH considers the available data and methods for performing internal dose reconstruction to be adequate for estimating with sufficient accuracy the internal doses at the INEL Site throughout its entire history.

5.1.3 Historical Overview

In 1949, the U.S. Atomic Energy Commission (AEC) established the NRTS and started construction of facilities on a 572,000-acre site approximately 50 miles west of Idaho Falls in southeastern Idaho. The site has experienced several name changes since its creation. The following are the names the site has been called over the years and the approximate date ranges for those names: NRTS (1949–1973), INEL (1974–1996), INEEL (1997–2004), and INL (2005 to the present). For convenience, INEL is used throughout the remainder of this document where it is unnecessary to distinguish between one of its other names.

Each of the original AEC laboratories was unique in both mission and location. Because 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. Some of the unique characteristics of radiation safety (and internal dosimetry specifically) at the INEL that had a marked influence on the internal dosimetry programs at each of the facilities were:

- The original mission of the INEL was the development of high enriched uranium (HEU) reactor concepts, materials testing through high-flux test reactor operation, and chemical processing of those (valuable) fuels. The U-235 enrichment of the HEU at the INEL was over 50% and mostly over 90%. The production of weapons-grade nuclear materials was not a mission.

- The INEL began operations 8 to 10 years after Oak Ridge National Laboratory (ORNL) and the Hanford Site. During those developmental years significant technical progress in professional skills, instrumentation, analyses, procedures, and techniques was accomplished. Radiation safety programs and techniques from ORNL (ACC 1952) were adopted at the startup of the INEL facilities.

- Two AEC field offices (Chicago and Idaho) were responsible for and had oversight of the INEL programs included in this report. In addition, the Nuclear Navy under direction of the Pittsburgh Field Office administered programs and used facilities on the INEL for training and program development. Thus, there were three Federal organizations using the INEL and its infrastructure. This TBD does not apply to naval facilities or naval personnel, even if those personnel received exposure from AEC operations.

- In addition to the Federal agencies involved at the site, numerous contractors operated the many facilities for the agencies and shared support personnel to various degrees.

- To provide consistency of radiation safety programs at the INEL among a large variety of facilities and constantly changing contractors, the AEC established a Health and Safety (H&S) Laboratory at the INEL 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) technical support in quality assurance of external and internal radiation dose evaluation; (6) maintenance and documentation of personnel dosimetry records; and (7) research and development in these areas of responsibility. The name of this organization changed to Health Services Laboratory (HSL), then to the Health and Safety Division (HSD), then to the Idaho Center for Radiological and Environmental Sciences, and most recently to the Radiological and Environmental Sciences Laboratory (RESL) [1].

- Although the design and administration of the radiation safety programs in the workplace were the responsibility of each facility contractor, AEC conducted oversight. Technical data, information (particularly in the instances of detectable worker intake), and analytical internal dose calculations and evaluations were exchanged between the HSL and each contractor [1].

As a consequence, and in spite of the constant changes at the INEL, basic assumptions about detection levels (e.g. MDAs, MDLs, etc…) and missed dose potential were relatively consistent across the years [2]. There were differences in the available nuclear materials from facility to facility, but by early 1958 gamma spectral analysis capabilities at the INEL allowed the significant bioassay results (those that would result in reportable internal dose) to be defined in terms of the specific radionuclides (Hayden 1958). The practice in the case of a higher urine sample result was to attempt radionuclide identification through gamma spectral analysis and chemical separation. This TBD describes default assumptions for use in cases when the bioassay records for a worker do not include the radionuclide specific analyses and only include records for gross radioactivity measurements.

5.1.3.1 Test Reactor Area

The Test Reactor Area (TRA) complex includes hot cells, a gamma irradiation pool facility, research laboratories, and analytical laboratories. The reactors at the TRA, as well as the others at the INEL, were used for testing materials, experiments, neutron irradiation facilities, and so forth (Stacy 2000). They were not involved in the production of plutonium or any other weapons materials, unlike some reactors at other DOE sites. The MTR was the second operating reactor at the INEL and ran from March 1952 to 1970 in the TRA (Stacy 2000). The TRA has also hosted the Engineering Test Reactor (ETR; 1957 to 1981) and the ATR (1967 to present) along with six reactor-critical facilities that supported the test reactors (Stacy 2000).

The uranium used in the TRA reactors is enriched to 93% U-235, and the fuel is clad in aluminum. The predominant activation product in the cladding is Na-24, which is formed by activation of sodium in the aluminum. Sodium-24 (Na-24) has a half-life of 15 hours and emits a high-energy gamma ray (2.75 MeV). The inhalation dose to personnel from Na-24 is insignificant in comparison to that from the fission products in the fuel [3]. There are minor levels of activation products of stainless steel (Co-58, Co-60, Cr-51, Mn-56, etc.) in the primary coolant system due to corrosion of its stainless-steel components [4].

Several factors contributed to unusual amounts of fission products in the primary coolant systems of the MTR and ETR during early operations. With cladding technology in its infancy, the quality of the cladding was not the best and fission products leaked through it [4]. Another factor was fuel

contaminated with tramp uranium on the outside of the cladding. During reactor operation, fission products in tramp uranium were released directly into the primary coolant system [4]. Reactor operators and the fuel manufacturer resolved these deficiencies over time; by the time the ATR became operational, the primary coolant system of the ATR was considerably less contaminated than that of the MTR or the ETR during their early years of operation. In addition, reactor components such as fuel elements, reactor loop components, and so forth that are removed from the reactor and placed in the canal are a source of contamination in the canal water. If these components are not cleaned adequately before they are removed from the canal, the activity on them can become airborne.

The majority of radioactivity releases from the TRA reactors to areas potentially occupied by workers consisted of noble gases that decayed to short-lived particulates. The principal dose to personnel from releases of noble gases was direct radiation rather than inhalation [5]. Table 5-1 provides a summary of the major airborne incidents that occurred at the TRA.

Table 5-1. Major airborne incidents at the TRA.

Date	Incident	Radionuclide(s) released	Reference
03/28/1954	GE-ANP-1 depressurization	Noble gas	Sommers 1954
12/17/1958	GEH-4 rupture at MTR	Noble gas + iodine	Sommers 1958
1961	Ag-110m spill at ETR	Ag-110m	Horan 1962
06/13/1967	GA-18-1 depressurization at ETR	Ta-182 and Ta-183	Nertney et al. 1967
01/06/1977	Noble gas release at ATR	Noble gas	Sommers 1977

5.1.3.2 Argonne National Laboratory – West

The Argonne National Laboratory near Chicago established a branch at the INEL known as the Argonne National Laboratory – West (ANL-W), where it built and operated several reactors that were of fundamental importance for the development of commercial nuclear power. The 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. In February 2005, the ANL-W was merged into the INEL, and was subsequently renamed the Materials and Fuels Complex (MFC). However, for convenience, this TBD uses ANL-W, that has been this area's name for most of its history and is the name that it is known as by most of the EEOICPA claimants.

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

5.1.3.3 Other Test Reactor Areas

As the primary nuclear reactor development laboratory in the United States, the INEL tested or evaluated more than 100 reactor concepts (DOE 1997). Fifty-two test reactors were designed, constructed, and operated (including operation-to-destruction tests) at the INEL. The INEL has experienced a number of episodic reactor events, both planned and accidental [for example, the military Stationary Low-Power Reactor (SL-1) accident on January 3, 1961 (Stacy 2000); a series of deliberate safety experiments by the ANL-W in which reactors were allowed to go *prompt-critical* with resultant reactor destruction (Stacy 2000); and the Aircraft Nuclear Propulsion (ANP) Program that operated initial engine tests (IETs) with large environmental releases in the 1950s (DOE 1991)]. External and internal doses to workers, both expected and accidental, were associated with these events (Till et al. 2002).

The largest internal exposures at the INEL resulted from accidental intakes associated with episodic events or planned major releases, for which the times and characterizations of the intake materials were well known [6]. These exposures were documented in each exposed employee's file.

5.1.3.4 Idaho Chemical Processing Plant (ICPP)

Initially known as the Chemical Processing Plant (CPP), renamed as the Idaho Chemical Processing Plant (ICPP), and eventually renamed as the Idaho Nuclear Technology and Engineering Center (INTEC); this TBD uses ICPP for convenience where it is unnecessary to distinguish between one of its other names. The ICPP reprocessed highly enriched reactor fuel (U-235 enrichments of 50% to 93%) for 39 years from 1952 to 1991. Aged mixed fission products (MFPs) were the predominant internal hazard, although enriched uranium isotopes and plutonium isotopes (Pu-238 enhanced) were limiting in specific process locations.

The ICPP processes were remotely controlled but contact maintenance was required; that is, maintenance personnel entered process cells and repaired equipment by hand. The process equipment in the cells, which had walls of 5-ft-thick high-density concrete, were decontaminated by flushing and rinsing with concentrated acids and complexing agents before entry by health physics and maintenance personnel. These occasional operations were well planned, but they had high potentials for internal exposures [7].

Most internal doses experienced at the ICPP were from accidental releases. Table 5-2 lists unusual and episodic events that have occurred at the ICPP. The ICPP experienced not only operational containment barrier failures but also accidental criticality events in 1959, 1961, and 1978. Because the criticality accidents occurred in process vessels in heavily shielded cells, these events resulted in relatively minor worker intakes (Stacy 2000; AEC 1960; Horan 1962). These exposures are documented in the personnel dosimetry files.

A variety of fuel types from a multitude of reactors were processed at the ICPP throughout its operating history. The dates when processing started for each of the major three fuel types were approximately 1953 for aluminum fuels, 1956 for zirconium fuels, and 1965 for stainless-steel fuels (Staiger 2003). There are relatively long half-life fission products that persist for the ICPP source terms. In most cases, the source terms were well tagged with beta-emitting radionuclides, which allowed beta/gamma-detecting CAMs and bioassay methods for beta/gamma-emitting radionuclides to be used at the ICPP with the realization that they would also warn of possible alpha contamination or internal exposures [8].

The irradiated fuel was normally decayed a minimum of 90 days before shipment to the ICPP for processing (Allied Chemical undated), to minimize the radiological safety hazard of the relatively volatile halogens. After being received at the ICPP, the irradiated fuel was kept in storage until it had a minimum of 120 days of cooling; however, the processing of fuels often did not occur until months or years later (Allied Chemical undated). Because of this relatively long decay time, many of the short half-life radionuclides decayed considerably, leaving the actinides to make up a larger percentage of the total radionuclide inventory of the processed fuel [9].

Table 5-2. Major airborne incidents at the ICPP.

Date	Incident	Radionuclides released	Internal dose discussion	Reference
05/15/57	Iodine release to Y-Cell	I-131	Y-Cell modifications resulted in 8 personnel receiving minor thyroid doses in the range of 600 mrem.	Vance 1957
03/20/58	Iodine release	I-131	Radioactive iodine spread through makeup area to operating corridor. Thyroid intake to several health	Rich 1958; Hayden 1958

Date	Incident	Radionuclides released	Internal dose discussion	Reference
			physics technicians and operators in the 40-µCi range.	
10/16/59	Criticality accident - in shielded process system	Short-lived noble gases and I-131, I-132, I-133, etc.	Short-lived radioactive gases released to plant areas; internal doses reported as minimal.	Ginkel et al. 1960
01/25/61	Criticality accident - in shielded process system	Short-lived noble gases & I-131, I-132, I--133, etc.	Short-lived radioactive gases released through process off-gas system to 76-m stack. Internal doses reported as minimal.	Paulus et al. 1961
01/72	Release of ~1.0 Ci Ru-106 from main stack	Ru-106	No internal doses detected.	ERDA 1977
11/17/72	ICPP mass spectrometry Pu contamination incident	Pu-238, Pu-239	An exposure incident involving about a dozen personnel resulted in 50-yr exposure lung doses ranging up to about 4 rem.	Wenzel 1973, 1974
10/17/78	Criticality accident in shielded process system	Short-lived noble gases	Short-lived lived radioactive gases released through process off-gas system to 76-m stack. Internal doses reported as minimal.	Casto 1980
11/85	N-Cell Pu uptake	Pu-238	Internal exposures were far below DOE exposure limits, but showed a weakness in the radiological control program.	Henry and Slagle 1985
10/30/88	Release of ~0.2 Ci of Ru-106 from main stack	Ru-106	No internal doses detected.	Hoff, Mitchell, and Moore 1989

One exception to this planned fuel aging was the radioactive lanthanum (RaLa) process, which operated in L cell of the 601/602 process building from February 1957 to 1963 (Stacy 2000). This process was designed to extract RaLa from green fuel from the MTR with as little decay as manageable (less than 2 days). Fuel was removed from the MTR, transported about 2 miles to the ICPP in a heavily shielded transport container by a straddle carrier, immediately dissolved, and the barium element was extracted. The Ba/La-140 product was shipped immediately to Los Alamos National Laboratory. The INEL's Ra/La process released large quantities of volatile radioactive iodine isotopes, because early designs for the process were inadequate for controlling iodine (Hayden 1958). Several significant internal exposure incidents occurred in which I-131, I-132, and I-133 intakes occurred before personnel could respond to CAM alarms and take protective or corrective actions [10].

5.1.3.5 Other Nuclear Facilities and Processes

Other nuclear facilities at the INEL where intakes occurred include:

- The Radioactive Waste Management Complex (RWMC) handled radioactive wastes generated by nuclear facilities on the INEL Site and was the primary disposal location for materials from the Rocky Flats Plant. Although most waste came to the RWMC in packages, accidents occurred during handling and processing that resulted in releases (Hoff, Mitchell, and Moore 1989). This in turn caused intakes of aged MFPs, uranium isotopes, transuranic (TRU) radionuclides, and aged mixed activation products (MAPs) (INEEL 2001).

- The Specific Manufacturing Capability (SMC) Project is a depleted uranium (DU) specialty-parts production plant built in 1985 in the ANP Program hanger on the Test Area North (TAN) portion of the site. The SMC Project processes metric tons of DU metal for the production of military shielding units (Stacy 2000). The processes of cutting, machining, and handling

uranium metal produce environments in which both chronic and accidental intakes of DU have occurred.

- The U.S. Navy used the Naval Reactors Facility (NRF) for operating reactors and as a naval reactor training center. Because this is not a DOE program and not under the oversight of DOE, the NRF is not part of the dose reconstruction and compensation program. However, through the years the NRF has participated in limited coordination of radiological protection programs and site support services. It is possible that some INEL workers received internal doses from their support of work at the NRF [11].

5.1.4 Radionuclides of Concern and Solubility

The INEL's facilities and activities have related primarily to experimental reactor design and development, irradiated fuel processing, DU parts production, and low- and high-level radioactive waste treatment and disposal. The *Technical Basis Document for the Idaho National Laboratory – Site Description* (ORAUT 2007a) describes these activities in more detail.

Table 5-3 lists the INEL's radionuclides of concern from these programs and as documented in *INEEL M&O Contractor Technical Basis for Internal Dosimetry, General Technical Basis and Facility Specific Documents* (INEEL 2001). These radionuclides are those for which internal doses were determined in the past and/or for which detection methods were developed.

The INEL Site stored and processed nuclear materials from all over the world (ORAUT 2007a). These materials included, but were not limited to, nuclear wastes from other DOE sites and irradiated nuclear fuels from universities, test reactors, commercial power plants, INEL reactors, and U.S. Department of Defense (DOD) Projects. Because of this, it is not possible to limit the material types to a single selection when there are multiple possibilities identified by the International Commission on Radiological Protection (ICRP). In addition, materials change over time as they age, typically becoming more insoluble. Therefore, all material types included in ICRP Publication 68 (ICRP 1995), as well as super-S plutonium, for a given radionuclide should be considered when performing a dose reconstruction and the one yielding the highest dose should be assigned with the following exceptions.

- Because the available information on the INEL Site does not indicate that strontium titanate ($SrTiO_3$) was ever present at the site and because strontium titanate was an uncommon strontium compound, strontium only needs to be assessed as type F material.

- With the exception of I-131 intakes calculated using the approach described in the *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007c), intakes of radioactive iodine need only be assessed as type F material using the inhalation intake route in the IMBA program. Because the various intake route and material type options for iodine in the IMBA program result in relatively insignificant dose differences and the inhalation of type F material results in a bounding dose estimate, it is reasonable to limit all assessments of radioactive iodine intakes to the inhalation of type F material. When using the approach described in the *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*, the intake route and material type options specified for iodine intakes in that document shall be used.

- Radionuclides where decay products or trace atoms are bound in a matrix of another radionuclide. Examples of this include Sr-90 and its decay product Y-90, and trace atoms of Am-241 in a plutonium matrix. In these examples, the Y-90 tracks with the type F Sr-90 even though Y-90 is only associated with type M and S materials, and Am-241 will track with type S plutonium even though Am-241 is only associated with type M material (ORAU 2007b).

- If the assessment of the bioassay data provides an inarguably better fit for a specific material type, only that material type need be used for the dose assessment (ORAUT 2007b).

Table 5-3. Primary radionuclides of concern.

Element	Radionuclides
Hydrogen	H-3 [assume tritiated water vapor (HTO)][a]
Chromium	Cr-51
Manganese	Mn-54
Iron	Fe-59
Cobalt	Co-58, Co-60
Zinc	Zn-65
Strontium/Yttrium	Sr-89, Sr/Y-90
Zirconium/Niobium	Zr/Nb-95
Molybdenum	Mo-99
Technetium	Tc-99
Ruthenium	Ru-103, Ru-106
Silver	Ag-110m
Antimony	Sb-122, Sb-125
Tellurium	Te-132
Iodine	I-129, I-131, I-132, I-133, I-135
Cesium	Cs-134, Cs-137
Barium/Lanthanum	Ba/La-140
Cerium	Ce-141, Ce-144
Europium	Eu-152, Eu-154, Eu-155
Gadolinium	Gd-153
Tantalum	Ta-182
Mercury	Hg-203 (assume inorganic)[a]
Protactinium	Pa-233
Uranium	U-233, U-234, U-235, U-236, U-238
Neptunium	Np-237
Plutonium	Pu-238, Pu-239/240
Americium	Am-241

a. Based on information provided in the document titled *INEEL M&O Contractor Technical Basis for Internal Dosimetry* (INEEL 2001).

5.1.5 Intake Modes and Particle Size Distributions

Unless specific information is provided, the default guidelines regarding intake mode and particle size distribution in the *Technical Information Bulletin: Internal Dose Reconstruction* (ORAUT 2007b) should be followed.

For the SMC Project, specific particle size information is provided in Section 5.5.5.

5.1.6 Internal Dosimetry Program

The radiological protection program was established to provide timely detection of barrier or ventilation failure. The program consisted of continuous and retrospective air and effluent monitoring combined with personnel and surface contamination monitoring [12]. 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 internal intakes. As a consequence of consistent policy to avoid detectable internal exposures, coupled with the time and technical complexity of an internal dose evaluation, the general policy at the INEL for internal exposures has been preventive in nature [13]. In general, radiological materials handled at the site were of relatively low volume and mass and of higher activity concentration rather than metric tons of

materials of low specific activity. The consistent INEL policy and practice was to require respiratory protection on jobs when the possibility of airborne contamination was thought to exist regardless of the actual measured air or surface contamination (ACC 1952).

The changes in contractors at the INEL during its 60 plus year history resulted in relatively frequent management changes at most of the facilities. However, the contract with the University of Chicago to operate the ANL-W facilities did not change until 2005. Summaries of the contractor changes are provided in Tables 2-1 and 2-2 of the *Technical Basis Document for the Idaho National Laboratory – Site Description* (ORAUT 2007a).

The primary oversight for the INEL, which included most projects on the site and all support functions, was initially assigned to the AEC Idaho Operations Office (IDO) [14]. The AEC created the H&S Laboratory to provide a variety of health and safety support functions to the entire site, which included external and internal dosimetry, health physics instrumentation, fire protection, medical services, and environmental surveillance [14]. Once the NRF was constructed, the U.S. Navy provided oversight for that portion of the INEL Site. In addition, the Chicago Operations Office (COO) provided oversight for ANL-W programs and facilities that were contracted by the University of Chicago, until February 2005. During that period, the ANL-W used site support services, including internal dosimetry support, but the contractor reported the dosimetry results to the COO versus the IDO [14]. In February 2005, oversight of the INEL and ANL-W programs and facilities were combined under the same primary IDO contract.

The INEL personnel dosimetry records have been and are documented and permanently maintained by the various organizations throughout the site's history [15]. Records about individual facility or contractor field monitoring programs (air-monitoring data, personnel contamination records, etc.) were maintained by individual contractors and/or site areas and are not maintained in a single recordkeeping system. The field monitoring data were not available for use in this TBD.

In spite of the frequent changes in operational responsibility through the years and the movement of workers among facilities, there has been a basic level of consistency in the internal dosimetry programs at the INEL, particularly the bioassay analytical techniques and calculation processes [16]. The field programs monitored the workplace and identified work groups to be included in the routine bioassay programs and workers who needed special bioassays. Although these programs were implemented by the individual contractors, there was routine interaction with the H&S Laboratory professionals in interpretation of dosimetry results as well as in determination of necessary corrective practices or procedures [17].

Employees were typically assigned to individual facilities and were monitored for specific radiological hazards associated with the work. During periods when a single prime contractor was responsible for programs at most facilities or for site-wide support personnel, workers in certain crafts (e.g., maintenance, specialty operators, and some health physics technicians) worked at several facilities and were exposed to a variety of radioactive materials in a variety of work situations [18].

Internal dose reconstruction for personnel who worked at a number of the INEL facilities should rely on specific bioassay data (radionuclides, quantities, etc.) when available. The procedures and technical capabilities for collecting and analyzing bioassay samples at the different facilities were basically equivalent [19]. In addition, both the individual facilities and 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 (Bhatt 2002).

5.1.6.1 Contamination Control

The contamination control limits for the detection and control of released activity beyond the control boundaries related to instrumentation capabilities and the basic philosophy of acceptance of detectable contamination. 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 personnel were always set close to the MDA, such that *any detectable* contamination was a signal for preventive and follow-up evaluations and actions. Beta/gamma MDAs typically were a factor of 5 below the limits. Table 5-4 is a summary of control limits primarily from an early health physics manual for the ICPP (ACC 1952) and current operating procedures.

Table 5-4. Surface contamination control and MDAs.

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

5.1.6.2 Air Monitoring

The monitoring of radioactivity in the air in occupied areas was a basic element of the internal exposure prevention program. Beta/gamma-detecting CAMs were used from the beginning of all facility and program operations in routinely occupied areas. With the exception of the SMC Project that was started in 1985, the primary contaminant radionuclides by activity were either MFPs or MAPs, which were beta/gamma emitters with maximum permissible concentrations/derived air concentrations (MPCs/DACs) above 1E-09 µCi/cm^3. TRU materials and uranium were available at some of the INEL facilities, but they were nearly always well-tagged with beta/gamma-emitting radioactivity that allowed beta/gamma-detecting CAMs to be used to warn of possible alpha contamination or internal exposures.

The *CPP Health Physics Manual* from 1952 (ACC 1952) describes a CAM and three other air-sampling systems. The manual required use of a filter-type respirator when airborne activity exceeded 1E-08 µCi/cm^3 for beta/gamma activity or 1E-11 µCi/cm^3 for alpha activity (ACC 1952). An army assault-type mask was required when levels exceeded this by a factor of 10. Positive-pressure air masks were required if levels larger by a factor of 1,000 occurred (ACC 1952).

The CAM systems provided real-time air activity evaluations (although it is not clear what the set points for alarms were), and fixed air samplers at several locations provided retrospective data and an average air concentration of beta/gamma emitters in an area or building [20]. The fixed air filter

samples were counted for both beta and alpha activity. Later, alpha CAMs were provided in select facilities where alpha contaminants could be controlling [21]. CAMs were calibrated, and training programs for health physicists were established for interpreting CAM responses for such variations as situations, radionuclides, response times, and filter accumulations [1]. If personnel were required to work in an area or building where known air contamination was present, respirators were worn to reduce internal contamination intake to levels below detectable amounts [1].

In general, workers were asked to submit to bioassay measurements whenever they were in an area where a CAM alarm sounded. In addition, the fixed location and retrospective air-sampling system signaled the need for bioassay if elevated air sample results were detected [22].

5.1.6.3 Bioassay Monitoring

Routine bioassay of radiation workers has occurred since the beginning of site operations. However, formal documentation of the bioassay programs was not found for periods before 1981. Some of the data sheets on individuals indicate that bioassay sampling occurred routinely every 6 months in 1953 [23]. Table 5-5 lists the reconstructed history of routine bioassay frequency. The INEL's monitoring and analytical programs were also designed to initiate an investigation of any potential internal intake as indicated by off-normal workplace indicators such as positive air sampling, personnel contamination, etc (INEEL 2001). Those investigations often included performing non-routine bioassay measurements for the potentially exposed workers. In addition, most of the recorded bioassay analyses performed at the INEL did not result in detectable radionuclides (ORAUT 2010a, 2010b).

Table 5-5. Routine Bioassay History Summary.[a]

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1953–1960	Annual	*In vitro* urine	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Individual data sheets; Table 5-10; Horan 1959; AEC 1961
1961	Annual	*In vitro* urine; *in vivo*	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Table 5-10; Horan 1962
1962–1972	Annual	*In vitro* urine; *in vivo*	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Horan 1962 Dodd 1963
1973–1981	Annual	*In vitro* urine	Radiation workers	Reporting Annual DE >10% quarterly standard in ERDA Manual Chapter 0524 (ERDA 1975).	Frequency is inferred from individual data sheets.	AEC 1968; AEC 1975; ERDA 1975
	When internal intake suspected	Fecal				
	Annual	*In vivo*				
1982–1987	Annual	*In vitro* urine	CPP-603 workers; fuel reprocessing operators	Reporting 50-yr CDE >10% quarterly standard in ERDA Manual Chapter 0524 (ERDA 1975).	Staggered to monitor group throughout the year.	Author unknown 1981
	When internal intake suspected	Fecal	Waste reprocessing operators; shift laboratory workers; health physics technicians			
	Annual	*In vivo*	Selected radiochemistry workers; maintenance workers; denitrator operators.			
	Termination	*In vivo*	All radiation workers.			

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1988–1989	1 to 6 mo	*In vivo*	All radiation workers.	Investigating Lung, 50-yr CDE >0.5–1.0 rem. Bone surface, 50-yr CDE >1.0–2.0 rem. Other organs, 50-yr CDE >0.5-1.0 rem.	Staggered so that worker received some sort of analysis or sampling every 3 mo.	Tschaeche 1988
	Annual	*In vitro* fecal				
	18 to 24 mo	*In vitro* urine				
	Termination	*In vivo*	When internal exposure suspected.			
		In vitro				
	New hire	*In vitro*	Depending on review of radiation dose history.			
		In vivo				
1990–1994	Annual	*In vivo*	All radiation workers where exposure to surface or airborne radioactive contamination could give at least 0.1-mrem AEDE from occupational sources, or give an organ or tissue DE >5 rem annual.	Reporting In accordance with DOE Order 5480.11 (DOE 1988). Workers who could receive 0.1 rem AEDE or 5 rem ADE organ or tissue dose.	Bioassay requested when workplace monitoring program indicated >0.02 annual limit of intake. Follow-up triggered by positive results from the workplace monitoring program, positive routine bioassay sample, or in response to incidents involving suspected intakes.	King 1990; Rich 1990
	6 mo	*In vitro* fecal				
	Annual	*In vitro* urine				
	New hire	*In vivo*	Worked at a facility where gamma-emitting radionuclides were handled.	Investigating AEDE ≥0.01 rem.		
		In vitro urine; fecal	Worked in U manufacturing or recovery facilities; worked with transuranic materials.			
	Termination	*In vivo; in vitro*	Any employee suspected of having an internal exposure or on a scheduled monitoring program.			
1995	Appropriate to the facility mission, potential uptakes.	*In vivo*	All radiation workers who enter radiological buffer areas or areas of higher 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 (DOE 1988) and 10 CFR Part 835: Workers who could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE.	Each facility had a specific Technical Basis Document for Internal Dosimetry.	Andersen, Perry, and Ruhter 1995
	When workplace monitoring indicated significant potential for intakes.	*In vitro* urine; fecal	Random sampling was performed to demonstrate the adequacy of the radiological controls in limiting the internal intake of radionuclides. Employees were selected at random from both non-radiation workers and a radiation worker population.	Investigating Internal doses resulting from all confirmed intakes were 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.	
1995–2000	New hire		Based on screening to determine internal conditions from previous uptakes or to establish baseline for those continuing to work as radiation workers.			
	Termination	*In vivo*	Any employee who was on a scheduled monitoring program.			
		In vitro				
2001	As developed by individual facilities based on analysis tables developed for each radionuclide.	*In vivo*	All radiation workers.	Reporting In accordance with DOE 5480.11 (DOE 1988) and 10 CFR Part 835. Workers who could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE.	Bioassay is mandatory when an employee or visitor is involved in an event where the internal uptake of radionuclides was likely to have occurred.	INEEL 2001
		In vitro urine fecal				
	Termination	*In vivo*	Any employee who was on a			

		In vitro	scheduled monitoring program.	Investigating Uranium, >1.0 µg/L *In vitro* activity detected >2σ. *In vivo* >2.33 σ. Default trigger levels exceeded.		

a. AEDE = annual effective dose equivalent; CDE = committed dose equivalent; ERDA = U.S. Energy Research and Development Administration.

DOE HSL technical reports and annual reports, coupled with facility memoranda and reports, documented the analytical detection capability of the INEL in the early 1950s and 1960s. Internal monitoring programs were in place when facility operations began in late 1951. For example, during ANP Program-IET activity in 1956, particulate and liquid caustic filter samples of effluent were analyzed with gamma spectroscopy and specific chemical separations of the identified radionuclides (Ebersole 1956). This analytical capability to identify radionuclides 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 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). The gross beta assay used a 5 mL volume, but that was replaced with the gross gamma assay supplemented with strontium analyses that typically used a 75 mL volume [24]. 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 3x3 (3 inch by 3 inch) sodium iodide thallium-doped [NaI(Tl)] detector counting system for analyses of gamma-emitting radionuclides. In 1960, the HSD obtained a 3x3 well counter for gamma analysis, which replaced gross beta counting as the routine analytical procedure for urine samples. The *Annual Report of Health and Safety Division 1960* (AEC 1961, p. 59) indicates that approximately 1.5E-06 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.

The *Annual Report of Health and Safety Division 1960* (AEC 1961) outlined a basic philosophy in relation to gamma counting of bioassay samples. Gamma counting would be effective in all situations except exposures 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 solution, gross beta counted, and the strontium analysis not completed unless a detectable count was obtained on the precipitate. A 100 mL sample of urine permitted the detection of approximately 8E-08 µCi/mL of Sr-90.

Special and routine bioassay measurements were performed and documented by the DOE analytical laboratory. Puphal (1994) reported on the procedures used for bioassay in the Analytical Chemistry Branch beginning in 1960. These procedures were collected and placed into a procedures manual in 1982 for periodic revision (Bodnar and Percival 1982). There was another version of the procedures after the analytical work was transferred to Westinghouse Idaho Nuclear Company (INEEL 2002).

Table 5-6 reproduces reports of the urinalysis results for 1959, 1960, and 1961 as obtained from Table 4 of AEC (1960), Table IV of AEC (1961), and Table XIII of Horan (1962). These results are not identical but quite similar to those from the newer database. The practice was to perform a gross beta or gamma analysis and identify specific radionuclides if the gross counts indicated activity above background levels. The total number of urinalyses in 1959 was 11,066; 3,524 people had radiation badges; and 715 received external doses above 500 mrem. These numbers demonstrate that

workers provided urine samples multiple times during the year. The 1963 annual summary report (Dodd 1964) describes the year's follow-up analyses for the whole-body counting program. As shown in Table 5-7, many of the individuals were counted multiple times.

Images of the INEL's urine sample results through 1986 and *in vivo* measurements results through early 1996 were provided to the Oak Ridge Associated Universities Team (ORAUT), and the results were entered into a database. This database has more than 140,000 *in vitro* bioassay measurement results and over 95,000 *in vivo* bioassay measurement results (ORAUT 2010a, ORAUT 2010b). Of the *in vitro* bioassay measurements, over 61% of them were gross beta or gross gamma radioactivity in urine measurements (ORAUT 2010a). Of the *in vivo* bioassay measurements, over 70% of them were whole body counts (ORAUT 2010b). Prior to 1961, the predominant type of bioassay measurement being performed at the INEL was urine sampling with a gross beta radioactivity analysis. In December 1960, the gross beta radioactivity analysis was discontinued entirely (AEC 1961). Starting in 1961, the gross gamma radioactivity analysis replaced the gross beta radioactivity analysis as the predominant type of analysis performed on urine samples. After 1960, urine samples were still being periodically analyzed for strontium radioactivity; however, the strontium analysis was never the predominant analysis performed on the urine samples (ORAUT 2010b). In 1961, whole body counting was introduced and was often performed along with *in vitro* bioassay, until the early 1970's. In the early 1970s, *in vitro* bioassay was largely replaced by whole-body counting [1].

Significant intakes of radioactivity were uncommon at the INEL. This determination is supported by an evaluation of the bioassay data in ORAUT's database. Of the more than 89,000 urine samples that were analyzed for gross beta radioactivity and gross gamma radioactivity, less than 2% of those measurements were above the MDA values provided in Table 5-14 (ORAUT 2010a). Of more than 69,000 whole body counts, which is typically a more sensitive measurement than urinalysis, less than 10% of those measurements were reported as having results above their respective detection levels (ORAUT 2010b). In addition, over 92% of the whole body counts results that were above their respective detection levels were still below the 0.1 µCi reporting level (ORAUT 2010b), which indicates that large intakes were uncommon at the INEL.

During the earlier years at the INEL, workers were not routinely bioassayed for plutonium. Because plutonium was not separated from the spent nuclear fuel at the INEL, the plutonium was always present with the more readily detectable mixed fission products that were also in the irradiated fuel. Therefore, in the vast majority of the plutonium exposure scenarios, the plutonium would have been present with the product and waste streams containing mixed fission products, and any intakes of radioactivity would have been more readily detectable by performing bioassay measurements for mixed fission products. The available bioassay data indicates that certain ANL-W and ICPP workers started receiving routine plutonium bioassays starting around the late 1970's.

Table 5-6. Urinalysis results in 1959, 1960, and 1961 (AEC 1960, 1961; Horan 1962).

Nuclide/ element of interest	Type activity	Total number performed			Statistically significant						Highest result[a]		
					Number			Percent					
		1959	1960	1961	1959	1960	1961	1959	1960	1961	1959	1960	1961
	Gross β[b]	8,546	8,546	30	65	15	5	0.76	0.18	17	18,820 ±632 dpm/5 mL	992 ±40 dpm/5 mL	172 ±16 dpm/mL
	Gross γ[c]	2,433	2,712	9,120	174	129	--[d]	7.15	4.76	4	35,972 ±310 dpm/5 mL	19,817 ±105 dpm/5 mL	1,900,235 ±876 dpm/75 mL
Co-60	β	--	--	1	--	--	0	--	--	0	--	--	300 ±75 dpm/450 mL
Sr-90	β	3	105	3,248	3	0	2	100	0	2	4.12E-2 dpm/mL	Insignificant	183 ±8 dpm/75 mL
Sr-91	β	20	37	2	19	0	--	95	0	0	388 ±1.6	Insignificant	4 ±8 dpm/mL
I[e]	β	--	9	--	--	2		--	22		0	9992 ±80 dpm/mL	--
Cs-137	β	--	--	40	--	--	0	--	--	0	--	--	1,460 ±10% dpm/1700 mL
Ba-139	α	20			16	0	--	80	0	--	120 ±0.8 dpm/mL	0	--
Th[f]	α	7	0	--	0	0	--	0	0	--	Insignificant	0	Insignificant
U	α	--	--	4	--	--	--	--	--	0	--	--	10 µg/L
U-233	α	17	3	--	1	0	--	0.06	0	--	180 ±4.0 dpm/mL	Insignificant	--
Pu-239	α	18	0	29	0	0		0	0	0	Insignificant	Insignificant	2E-9 µCi/mL
Am-241	α	2	0	--	0	0	--	0	0	--	Insignificant	0	--
Totals		11,066	11,352	12,494	278	146	--	2.51	1.29	4			

a. All except two I-131 exposures in 1961 listed under gross gamma activity are less than 10% of the permissible body burden for the radionuclide of interest.
b. If only gross β analyses are available, the default should be Sr-90.
c. In many instances positive gross gamma results have been attributed to radioactive isotopes of iodine.
d. -- = no data reported.
e. Iodine isotope(s) not identified in references. Assume I-131.
f. Thorium isotope(s) not identified in references. Assume Th-228.

Table 5-7. Summary statistics from the 1963 WBC program (Dodd 1964, p. 19).

Radionuclide	Times reported	Number of individuals	Maximum activity (µCi)
Cr-51	15	10	1.2
Co-60/Fe-59	848	387	1.5
Mn-54	98	51	0.16
Co-58	62	50	0.03
Zn-65	505	171	1.20
Zr/Nb-95[a]	427	232	1.66
Ru-103/Ru-106[b]	93	75	0.22
Ag-110m	583	186	0.93
Sb-122	2	2	0.08
I-131	110	82	5.0
Cs-134	361	168	0.14
Cs-137	2,332	573	1.32
Ba/La-140	90	51	0.07
Ce-141/Ce-144	59	49	0.16
Ta-182	50	36	0.02
Hg-203	28	6	0.16
Pa-233	13	10	0.48
Np-239	1	1	1.68
Sb-125	3	3	0.1
Mo/Tc-99	8	5	0.72
I-132	8	7	<0.1
I-133	3	3	<0.1
Te-132	6	6	<0.1
Hg-197	7	3	0.7

a. Consider Zr-95.
b. Consider Ru-106.

5.1.7 Recordkeeping

Formal or *legal* internal dose data were maintained by the DOE HSD in individual hard-copy folders until 1989 when all technical support service functions, including those related to internal dosimetry, were transferred to the INEL Site's prime contractor. At that time, *in vitro* analytical functions were transferred to an onsite 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 current contractor's subject matter expert reviews, validates, and prepares official internal dose assessments. A DOE staff member at RESL is responsible for oversight of the INEL's internal dosimetry program functions and provides quality assurance. The RDR unit functions include documentation and records custodial responsibilities. In 1999, the Radiation Control Information Management System (RCIMS) database was placed in service to support the radiation protection program, including internal dosimetry. RCIMS lists reported internal doses as CEDE when an individual's dose history is prepared (INEEL 2001).

The following information is important to internal dose reconstruction because the worker files from DOE can contain a variety of internal dose information including the calculated internal doses as well as the *in vitro* and *in vivo* individual bioassay results. The changing 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 5-8 describes internal dose information that could appear in pre-1989 records. Table 5-9 describes coded information that could appear in records after 1989. Tables 5-10 and 5-11 provide the column descriptions for the sample record sheets that were used for the urine sample data. Column descriptions are provided for the sample record sheets used before February 1955 and

for the sheets used during the period of February 1955 and later. Table 5-12 provides the INEL's location codes for various work areas.

Table 5-8. Internal dose assessment information before 1989.

Dose information	Description
Name, Social Security #	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 by month and year in which exposure occurred.
Organ (max.)	Organ that received the maximum dose from the specified intake.
Organ CDE rem	CDE calculated for the listed organ in rem.
CEDE rem	Calculated CEDE in rem.
AEDE rem	Calculated AEDE in rem.
Year	Year for which the AEDE was calculated.

Table 5-9. Internal dose assessment information after 1989.

Coded information	Description
Name & Social Security #	Exposed employee by name and Social Security number.
Asmt. nos.	This assessment number is the calendar year (e.g., 83) 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.
AEDE rem	Calculated AEDE in rem.
Year	Year for which the AEDE was calculated.
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 exposures are summed for the year of intake for each employee.

Table 5-10. Column descriptions for urine sample record sheets before February 1955.

Analytical information	Description
Sample no.	Sample log number.
Date and hour	Generally clear interpretation.
Sample description	Name of the employee, numerical sample number frequently included, additional special analyses performed (Sr-90, Y Separation, etc.).
Sampling data	This column is subdivided in to three separate columns: Rate, Time, and Quant. The columns appear to be applicable to air sampling data and are not used for urine samples.
Anal. for	Means analyzed for. Generally gross beta-gamma and/or gross gamma. The gross beta-gamma analyses are typically denoted by either a "beta-gamma", "beta", "B", "B-I", "β", or "βγ" in this column. The gross gamma analyses are typically denoted by either a "gamma", "I", or "γ" in this column. Note that an "I" in this column does not indicate an iodine analysis unless specifically labeled as such. For gross beta-gamma analyses, a 5 ml sample aliquot was typically evaporated and counted using an end-window Geiger-Mueller detector. For the gross gamma analyses, a 50 ml sample aliquot was typically counted directly in a deep-well NaI scintillation detector. Identifiers for specific isotopic analyses, based on chemical separations or gamma spectrometry were also listed in this column.
Quantity used	Size of the sample aliquot in whatever units are indicated. Typically, the aliquot sizes are provided in ml.
Date or time	For urine samples the header for this column is usually crossed out and replaced with "(K^+) Trans", which indicates that an aliquot of the sample was also analyzed for K-40. When this is the case, the reported value in this column is the percent transmission reading from a flame photometer analysis that was performed on an aliquot of the sample that was analyzed for its potassium content. The percent transmission value is then used to determine the amount of the radioactivity in the urine sample that is attributable to naturally occurring K-40. However, before February 1955, the sample results appear to have never been corrected for the radioactivity that was attributable to K-40.
Count time	Counting methods used either preset times or preset counts. The sample count time was typically recorded in minutes in either case. For some early gross beta-gamma analyses, the sample count time was recorded in seconds.
Inst. read.	This is the reading from the counting instrument which is recorded in the units indicated. Typically, this is the total number of counts recorded for the sample.
Corr.	This column is used for the background count rate. A 1σ counting error is sometimes also reported with the background count rate.
Corr. read.	This column is used for the net count rate, which is calculated by subtracting the background count rate from the calculated gross count rate.
Mass equiv.	This column is often used to report the amount of K-40 in dpm that the sample contains. Before February 1955, there is no indication that the sample results were corrected to account for the K-40 activity.
Corrected value	This column is used to report the net counting results in units of activity, usually dpm. When the results are reported in dpm, it should be assumed that they are actually representing units of dpm/sample, unless indicated otherwise.

Table 5-11. Column descriptions for urine sample record sheets from February 1955 and later.

Analytical information	Description
Sample no.	Sample log number.
Date and hour	Generally clear interpretation.
Sample description	Name of the employee, numerical sample number frequently included, additional special analyses performed (Sr-90, Y Separation, etc.).
Anal. for	Means analyzed for. Generally gross beta-gamma and/or gross gamma. The gross beta-gamma analyses are typically denoted by either a "beta-gamma", "beta", "B", "B-I", "β", or "βγ" in this column. The gross gamma analyses are typically denoted by either a "gamma", "I", or "γ" in this column. Note that an "I" in this column does not indicate an iodine analysis unless specifically labeled as such. For gross beta-gamma analyses, a 5 ml sample aliquot was typically evaporated and counted using an end-window Geiger-Mueller detector. For the gross gamma analyses, a 50 ml sample aliquot was typically counted directly in a deep-well NaI scintillation detector. Identifiers for specific isotopic analyses, based on chemical separations or gamma spectrometry were also listed in this column.
Quantity used	Size of the sample aliquot in ml, unless other units are otherwise indicated.
U^+ or K^+ trans.	Indicates that an analytical correction for natural uranium or potassium might be applied to the recorded results. The value in this column is almost always associated with a correction for natural potassium and the natural uranium correction is relatively rare. The reported value in this column is the percent transmission reading from a flame photometer analysis that was performed on an aliquot of the sample. The percent transmission value is then used to determine the amount of the radioactivity in the urine sample that is attributable to natural uranium or potassium.
Count time	Counting methods used either preset times or preset counts. The sample count time was typically recorded in minutes in either case. For some early gross beta-gamma analyses, the sample count time was recorded in seconds.
Total count	Total number of counts recorded for the sample.
Gross count, cpm	The gross count rate was determined by dividing total counts by the count time.
Bkgd., cpm	Background count rate recorded. A 1σ counting error is sometimes also reported with the background count rate.
Net count, cpm	The net count rate, which is calculated by subtracting the background count rate from the gross count rate.
K-40 corr., cpm	Starting in 1955 and when the value in this column is reported in cpm, a K-40 correction has likely been applied to the recorded analytical results. The K-40 correction is applied by subtracting the value in this column from the recorded net count rate and then dividing the adjusted net count rate by the detector's counting efficiency.
Foreign activity, cpm and dpm	Unless stated otherwise, the results in this column are reported in units of cpm and dpm. Net count rate, which is sometimes corrected for K-40, and then converted to dpm based on the detector's counting efficiency. Uncertainty also included, which is recorded based on 1σ counting statistics. When the results are reported in dpm, it should be assumed that they are actually representing units of dpm/sample, unless indicated otherwise. At times the dpm units are crossed out and the results are recorded in other units. The other units will then be specified, and are typically units of dpm/ml or µCi/ml. In some instances the units may be in µg/L, which indicate the results for a uranium analysis whether or not a uranium analysis was previously specified.

Table 5-12. Location codes that could be in worker dose files.

Location code	Description[a]	Location code	Description[a]
1	AEC Headquarters Building	28	Onsite site survey
2	ANL-W - EBR-I	29	Offsite site survey
3, 34, 35	CFA	30	SL-1 - ANP program
4, 42, 45	MTR, TRA	31	STPF
5, 53, 55	ICPP	33	CFA Laundry
6, 32, 63, 64, 65, 764, 765	NRF	66	Non-security
		67	Division of Compliance
7	TAN - ANP program	68	STEP
8	Services	69	LPTF (Phillips and AEC)
9	NX (X is construction) at NRF	71	CADRE (guard force)
10, 105	AX at TAN	125	PBF construction
11, 113, 115	CX at ICPP	133, 135	PBF/WROC
12	EX at EBR	160, 161	IRC
13	SPERT	162	WCB
14	OMRE	163	EROB
15	SX at SPERT	164	TSA/TSB
16	SL-1	165	WAC
17, 333	MX at MTR	244, 245	ARA I
18, 814, 815	WP, RWMC	344, 345	ARA II
19, 772, 775	TAN (Phillips and AEC)	354, 355	ARA III
20, 261, 264	TREAT	555	Guards
21	LX	754/755	DOE – ID/RESL
22	GCRE	773	SMC (B&W cask)
23	OX	774, 776	SMC
24	ARHG	825	RWMC construction
25	No information available	835	RWMC storage
26, 263, 265	ANL-W	845	Pit 9
27	ML-1		

a. See the acronyms and abbreviations list.

Federal regulations about permissible internal dose and formal reporting requirements to the AEC, ERDA, and DOE changed periodically during the site's history. As the internal dose limits were reduced over time, the frequency of monitoring typically increases, new analytical methods are developed for existing types of bioassay to detect lower quantities, and new types of bioassay are developed and implemented (e.g., whole body counting, chest counting, routine fecal sampling, etc…). As a result, a site's ability to detect intakes was progressively enhanced over time, because of the changes in the permissible levels of internal dose. During the early years internal dose was usually considered separately from external dose in terms of meeting specific exposure limits, and the calculated dose was reported and documented only if specific dose levels were exceeded (Aoki 1979). AEC and ERDA policies required periodic urinalyses or *in vivo* counting or evaluation of air concentrations if the whole-body dose or committed dose could exceed 300 mrem in a calendar quarter (AEC 1958, 1963, 1968, 1975; ERDA 1975). 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 [22]. Dickson (1977) established official investigation levels (Table 5-13) for acute uptakes of radionuclides corresponding to one-tenth of the quarterly radiation standard. Later procedures (DOE 1988) set specific limits on those positive bioassay results that could result in 100-mrem AEDE or above as the point at which follow-up and reporting was required. With the DOE *Radiological Control Manual* (e.g., 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 (DOE 1994).

Table 5-13. Derived investigation levels in 1977 for acute exposures (Dickson 1977).

Radionuclide	Inhalation lung burden (µCi)	Ingestion total activity (µCi)
Cr-51	20	500
Mn-54	0.4	30
Co-57	2	90
Co-60	0.09	9
Zn-65	0.6	30
Zr-95	0.3	20
Ru-106	0.06	3
Sb-125	0.3	30
Cs-134	0.1	3
Cs-137	0.1	4
Ce-144	0.06	3
Pu-239	Whenever detected	
Am-241	Whenever detected	
Sr-90 (bone)	When detected by skull counting	
I-131 (thyroid)	Initial content 0.27	Not provided

5.1.7.1 Radiological Incident Records

When an incident occurred, it was the policy to investigate thoroughly and identify all individuals involved in the incident [26]. Therefore, when there is no evidence in the incident file or the individual's dosimetry file that an individual was involved and no other evidence supporting that an individual was involved in the incident, dose reconstructors should assume that the individual was not involved.

5.2 *IN VITRO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PROTOCOLS

No bioassay data appear to have been collected before 1952; however, very few of the radiological facilities at the INEL were operational before 1952.

In compliance with the November 1998 Code of Federal Regulations requirement (10 CFR Part 835) for the DOE Laboratory Accreditation Program (DOELAP), and based on American National Standards Institute N13.30, *Performance Criteria for Radiobioassay* (HPS 1996), both the *in vitro* and *in vivo* radiobioassay laboratories at the INEL received DOELAP accreditation in February 1998 (INEEL 2001, p 11). In accordance with this accreditation, MDAs and decision levels at the 95% confidence level are performed. Tables 5-14 through 5-17 and 5-18 through 5-21 list the current MDAs for the various bioassay methods employed at the INEL along with values from historical documents and the recommended periods of use for the MDA values.

5.2.1 Urine Sample Analyses

The majority of the urine samples taken at the site were single voidings; 24 samples were used for special sampling purposes (i.e., follow-up samples, primarily to extend the sensitivity). Urine sample results are typically reported in units of activity per sample along with the total sample and/or aliquot size. Table 5-14 provides the known MDAs for this *in vitro* bioassay method.

Table 5-14. Urine sample MDAs by period.

Radiation/ radionuclide	Period	Typical volume (ml)	Typical MDA (dpm/sample)	Typical daily[a] MDA (dpm/d)	Reference
Gross β	1951–1953	5	86	24,000	Data Sheet
	1954–1960	5	93[b]	26,000	Ebersole and Flygare 1957
Gross γ	1957–1964	75	580[b]	10,800	AEC 1961
	1965–1971	75	205	3,800	Data Sheet
H-3	1972–1994	3	0.5 dpm/ml[b]	700	AEC 1972, 1974
	1995–present		3 dpm/ml	4,200	Andersen, Perry, and Ruhter 1995, INEEL 2001
Co-60	1957–1958	50	51	1,400	Database
Sr-90	1953–6/14/62	75	37	700	Database
	6/15/62–1970	75	20	370	Database
	1971–1989	75	1.7[b]	32	AEC 1972, 1974
	1990–present	500 min	1.9	5	Andersen, Perry, and Ruhter 1995
I-131	1957–1970	75	370	6,900	Database
Cs-134	1974–present	400	2	7	Database
Cs-137	1961–present	400	410	1,435	Database
Th-230	1974–present	1,000	0.1[b]	0.14	AEC 1974
Np-237	1974–present	1,000	0.1[b]	0.14	AEC 1974
U (PF)	1954–1961	0.1	1E-5 g U/L[c]	14 µg U[c]	Database
	1962–1971	0.1	5E-6 g U/L[c]	7 µg U[c]	Database
U (KPA)	1985–present		0.2 µg U/L	0.28 µg U	Rich 1990
U-233/234	1979–1986	700	0.52	1.0	Database
	1995–present	500 min	0.091	0.25	Andersen, Perry, and Ruhter 1995
U-235	1970–1979	1,000	0.22	0.31	Rich 1990
	1980–1994	700	0.13	0.26	Database
	1995–present	500 min	0.084	0.24	Andersen, Perry, and Ruhter 1995
U-238	1970–1979	1,000	0.22	0.31	Rich 1990
	1980–1994	700	0.21	0.42	Database
	1995–present	500 min	0.067	0.19	Andersen, Perry, and Ruhter 1995
Pu-238	1981–1984	700	0.072	0.14	Database
	1990–1994	1,000	0.13	0.18	Rich 1990
	1995–present	500 min	0.049	0.14	Andersen, Perry, and Ruhter 1995
Pu-239/240	1964–1970	1,000	0.93[b]	1.3	Dodd 1964
	1971–1973	1,000	1.03[b]	1.4	AEC 1972
	1974–1979	1,000	0.47[b]	0.66	AEC 1974
	1980–1989	700	0.073	0.14	Database
	1990–1994	1,000	0.060	0.084	Rich 1990
	1995–present	500 min	0.060	0.17	Andersen, Perry, and Ruhter 1995
Am-241	1977–1979	1,000	0.16[b]	0.22	AEC 1974
	1980–1989	700	0.29	0.6	Database
	1990–1994	1,000	0.2	0.28	Rich 1990
	1995–present	500 min	0.051	0.14	Andersen, Perry, and Ruhter 1995
Cm-244	1974–present	1,000	0.155[b]	0.22	AEC 1974

a. Based on 1,400-mL daily volume and typical sample size, and would need to be adjusted for different sample volumes.
b. MDA calculated from an inferred 2σ uncertainty.
c. Smallest reported value. Not MDA.

5.2.2 Fecal Sample Analyses

At times, fecal samples were collected from workers to assess potential intakes of certain radionuclides. Table 5-15 provides the known MDAs for this *in vitro* bioassay method.

Table 5-15. Fecal sample MDAs by period.

Radiation/ radionuclide	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; INEEL 2001
Cs-134	1963–present	10	Rich 1990
Cs-137	1963–1999	0.01	Rich 1990
	2000–present	0.3	BBI 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; INEEL 2001
	2003–present	0.05	Bhatt 2003
U-235	1970–2003	0.038	Andersen, Perry, and Ruhter 1995; INEEL 2001
	2003–present	0.09	Bhatt 2003
U-238	1970–1994	0.5	Rich 1990
	1995–2002	0.03	Andersen, Perry, and Ruhter 1995; BBI 2000; INEEL 2001
	2003–present	0.09	Bhatt 2003
Pu-238	1974–1994	0.03	AEC 1974
	1995–2002	0.022	Andersen, Perry, and Ruhter 1995; INEEL 2001
	2003–present	0.02	Bhatt 2003
Pu-239/240	1964–1973	0.4[b]	Dodd 1964
	1974–1994	0.02	AEC 1974
	1995–present	0.03	Andersen, Perry, and Ruhter 1995; BBI 2000; INEEL 2001; Bhatt 2002
Am-241	1974–1994	0.07	AEC 1974
	1995–2001	0.023	Andersen, Perry, and Ruhter 1995; INEEL 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 an individual's records, assume the activity is that excreted per day.
b. MDA calculated from an inferred 2σ uncertainty.

5.3 *IN VIVO* MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PROTOCOLS

5.3.1 Whole-Body Counting

5.3.1.1 General Information on Whole-Body Counting

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

As a consequence, the *in vivo* counting program was used to count (1) all terminating employees who required physical examinations, (2) employees who were suspected of having a possible internal intake, and (3) selected groups of individuals who were scheduled for semi-routine analyses by health physics supervisors (Sommers 1961). In 1963 approximately 1,650 counts were performed; 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 5-16. Whole-body count MDAs by period.

Radiation/ radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
Cr-51	1961–2000	5	10	Percival and Anderson 1962
	2001–present	32	5	INEEL 2001
Mn-54	1962–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, Ruhter 1995
	2001–present	2.6	5	INEEL 2001
	2001–present	1.3	10	INEEL 2001
Fe-59	1962–2001	4.5	5	INEEL 2001
	2001–present	1.5	10	INEEL 2001
Co-58	1961–2000	5	10	Percival and Anderson 1962
	2001–present	2.5	5	INEEL 2001
	2001–present	1.1	10	INEEL 2001
Co-60	1961–2000	5	10	Percival and Anderson 1962
	1971–1988	5[a]	10	AEC 1972; 1974
	1989–2000	7	10	Martin 1989; Grothaus 1993; Andersen, Perry and Ruhter 1995
	2001–present	2.5	5	INEEL 2001
	2001–present	1.1	10	INEEL 2001
Zn-65	1961–2000	5	10	Percival and Anderson 1962
	1989–2000	10	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	4.9	5	INEEL 2001
	2001–present	2	10	INEEL 2001
Zr/Nb-95	1961–2000	5	10	Percival and Anderson 1962
	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	2.6	5	INEEL 2001
Ru-106	2001–present	27	5	INEEL 2001
	2001–present	7.6	10	INEEL 2001
Ag-110m	1961–2000	5	10	Percival and Anderson 1962
Sb-125	1961–1988	5	10	Percival and Anderson 1962
	1989–present	14	10	Martin 1989; Grothaus 1993
I-131	1961–2000	5	10	Percival and Anderson 1962
	2001–present	3.8	5	INEEL 2001
Cs-134	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	3	5	INEEL 2001
	2001–present	0.96	10	INEEL 2001
Cs-137	1961–1970	5	10	Percival and Anderson 1962
	1971–1988	2	10	AEC 1972, AEC 1974, Tschaeche 1988
	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	3.1	5	INEEL 2001
	2001–present	1.9	10	INEEL 2001
Ba/La-140	2001–present	12	5	INEEL 2001
Ce-141	2001–present	9.9	5	INEEL 2001
	2001–present	3.2	10	INEEL 2001

Radiation/ radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
Ce-144	1989–2000	50	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	44	5	INEEL 2001
	2001–present	15	10	INEEL 2001
Eu-152	2001–present	4	10	INEEL 2001
Eu-154	2001–present	2	10	INEEL 2001
Eu-155	2001–present	1	10	INEEL 2001
Ga-153	2001–present	6.5	10	INEEL 2001
Hf-181	1989–present	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
Ta-182	1961–1962	5	10	Percival and Anderson 1962
Hg-203	1961–1962	5	10	Percival and Anderson 1962

a. MDA calculated from an inferred 2σ uncertainty.

5.3.1.2 Attributing Positive WBCs to Cs-137 Fallout

Fallout affected everyone in North America, and body burdens of Cs-137 measurable in the whole-body counters were common in the 1960s and 1970s. National Council on Radiation Protection and Measurements (NCRP) Report 94 (NCRP 1987) provides mean body burdens of Cs-137 for the United States for 1953–1977, which are summarized in Table 5-17.

Table 5-17. Mean body burdens of Cs-137 from fallout in the United States.

Year	Body burden (nCi)	Year	Body burden (nCi)
1953	0.27	1966	9.7
1954	1.1	1967	5.6
1955	2.2	1968	3.5
1956	4.3	1969	2.7
1957	5.1	1970	2.7
1958	6.5	1971	2.7
1959	8.1	1972	2.7
1960	6.8	1973	2.7
1961	4.6	1974	1.6
1962	6.0	1975	1.1
1963	11	1976	1.6
1964	19	1977	1.1
1965	16		

a. Values in this table were obtained from NCRP Report 94(NCRP 1987).

If WBC results show detection of only K-40 and Cs-137, and the Cs-137 result is less than the values in Table 5-17, the dose reconstructor can attribute the positive Cs-137 results to fallout and not assign doses based on those results. If other radionuclides are present during the same measurement time period in the WBC or are detected by another bioassay method, it should be assumed that the Cs-137 is entirely from occupational sources. The values in Table 5-17 are only intended to attribute an entire WBC's result to fallout Cs-137 and are not intended to reduce the calculated intakes by subtracting the contribution attributable to fallout Cs-137. Therefore, the values in Table 5-17 should not be subtracted from WBC results that are greater than the Table 5-17 values.

5.3.2 Lung Counts

At times, lung counts were performed on workers to assess potential intakes of certain radionuclides. Table 5-18 provides the known MDAs for this *in vivo* bioassay method.

Table 5-18. Lung count MDAs by period.

Radionuclide	Period	*In vivo* MDA[a] (nCi)	Count time (min)	Reference
Ce-141	2001–present	0.11	60	INEEL 2001
Ce-144	2001–present	0.44	60	INEEL 2001
Eu-152	2001–present	0.18	60	INEEL 2001
Ga-153	2001–present	0.096	60	INEEL 2001
Th-234	2001–present	1.4	60	INEEL 2001
U-235	1990–2000	0.2		Rich 1990
	2001–present	0.11	60	INEEL 2001
U-dep/nat	1989	3	60	Martin 1989; Grothaus 1993
Pu-238	1984–1988	16		Tschaeche 1988
	1989–1995	26	60	Martin 1989; Rich 1990; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1996–present	54	60	INEEL 2001
Pu-239/240	1974–1983	74	100	AEC 1974
	1984–1988	20		Tschaeche 1988
	1989–1995	80	60	Martin 1989; Rich 1990; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1996–present	140	60	INEEL 2001
Am-241	1984–1988	0.15		Tschaeche 1988
	1989–1995	0.6	60	Martin 1989; Rich 1990; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1996–present	0.14	60	INEEL 2001

a. If an MDA value is not available in this table, it can be inferred from the reported 1σ uncertainty as being approximately equal to the 2σ value.

5.3.3 Thyroid Counts

At times, thyroid counts were performed on workers to assess potential intakes of radioactive iodine. Table 5-19 provides the known MDAs for this *in vivo* bioassay method.

Table 5-19. Thyroid count MDAs by period.

Radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
I-131	1971-1974	1E-05	10	AEC 1972, 1974
	1990–1992	2	10	Rich 1990
	1993–2000	0.3	10	Grothaus 1993
	2001–present	0.13		INEEL 2001

5.3.4 Skull Counts

At times skull counts were performed to assess potential intakes of Sr/Y-90. The skull counts measured Bremsstrahlung radiation that is emitted from the skull after the bone has had an uptake of radioactive strontium. Table 5-20 provides the MDAs for this *in vivo* bioassay method.

Table 5-20. Skull count MDAs by period.

Radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
Sr/Y-90	1968–1977	70[a]	10	Voelz 1969; AEC 1972, 1974
	1978–present	34	10	Martin 1989; Grothaus 1993

a. MDA calculated from an inferred 2σ uncertainty.

5.3.5 Wound Counts

Wound counts have been performed at the INEL to assess potential intakes from contaminated wounds. Table 5-21 provides the known MDAs for this *in vivo* bioassay method.

Table 5-21. Wound count MDAs by period.

Radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
U-235	1993	0.2	20	Grothaus 1993
Pu-238	1993	1	20	Grothaus 1993
Pu-239/240	1993	2	20	Grothaus 1993
Am-241	1993	0.1	20	Grothaus 1993

5.4 INTERFERENCES AND UNCERTAINTIES

5.4.1 Interferences

5.4.1.1 Contamination of Samples

The most common type of interference encountered with bioassay measurements is due to contaminated *in vitro* samples and contaminated workers for *in vivo* measurements. This type of interference can produce erroneously high bioassay measurement results that can sometimes be identified as outliers. Notes regarding suspected contamination are sometimes found in the bioassay records. When no such notes are found in the bioassay records, a dose reconstructor may still be able to determine if a bioassay measurement is an outlier by using the subsequent bioassay measurement results and information regarding the affected worker's work locations and the activities being performed at those locations before the elevated measurement. However, a significant amount of caution should be exercised when determining if an INEL bioassay measurement result is an outlier, because the most common significant intakes at the INEL Site were attributable to radionuclides that are quickly eliminated from the body, which often do not show up in subsequent bioassay measurements. Therefore, bioassay results above their detection or reporting limits should be considered to be real, unless there is conclusive information to the contrary.

5.4.1.2 Naturally Occurring Uranium in Uranium Bioassay Measurements

Uranium is present in secular equilibrium throughout the Earth. Consequently, there is a continuous source of intake of naturally occurring uranium via ingestion of drinking water and food and inhalation of particulate matter, which can interfere with bioassay measurements being used to detect intakes of uranium from occupational exposures. Because the uranium bioassay measurements performed for the ICPP workers who were exposed to HEU were isotopic analyses and because the mixture of uranium isotopes in recycled HEU is significantly different from natural uranium, positive sample results attributable to naturally occurring uranium can be readily distinguished from results attributable to HEU. However, the Kinetic Phosphorescence Analysis (KPA) that is performed on SMC Project urine samples cannot distinguish between different types of uranium, be it naturally occurring, depleted, or enriched. Therefore, the uranium in urine sample results for SMC Project workers have been adjusted to discount the interference from naturally occurring uranium. Consequently, the

reported bioassay results in the worker files reflect the subtraction of 0.16 µg U/L from the value determined in the laboratory bioassay result (King 2001).

The 0.16 µg U/L adjustment value is based on measured background levels of uranium in urine samples submitted in 1987, 1994, and 1998 by SMC Project workers who were not radiological workers. Those measurements were assumed to represent the non-occupational excretion levels of the SMC Project worker population, because performing background measurements using members of the public was not feasible. The background measurement results ranged from 0.04 to 0.33 µg U/L with wide fluctuations in individual measurements, some as high as 1.0 µg U/L (King 2001). The average reported uranium concentration was 0.157 ±0.109 µg U/L at 1σ uncertainty. Therefore, 0.16 µg U/L is used as the non-occupational component of uranium excretion for SMC Project workers, and is subtracted from each urine result before assessment of occupational internal dose. ICRP Publication 23 (ICRP 1975) lists the daily intake of naturally occurring uranium as 1.9 µg U/d and a daily urine excretion rate of 1.4 L for Reference Man. Using a uranium intake of 1.9 µg U/d and a daily urine excretion rate of 1.4 L, the typical urinary concentration ranges between 0.4 and 0.5 µg U/L for naturally occurring uranium (King 2001). Therefore, the adjustment value used at the INEL is consistent with the ICRP reference values for naturally occurring uranium.

5.4.1.3 Cesium-137 from Fallout

As indicated above, fallout from the atmospheric testing of nuclear weapons affected everyone in North America and body burdens of Cs-137 measurable in whole-body counters throughout the United States were common in the 1960s and 1970s. Even though the United States stopped its atmospheric testing of nuclear weapons by 1963, Cs-137 intakes from fallout were still often detected after that because of the Cs-137 that was deposited on soil and vegetation. Significantly higher Cs-137 intakes from fallout are sometimes identified in persons who consume game meats. This is due to specific game animals whose diets include plants that tend to concentrate the Cs-137 in the environment.

Because the doses attributable to fallout Cs-137 are not considered to be occupational doses, Cs-137 doses attributable to fallout did not need to be assigned by a site. However, no information has been found to indicate that the INEL adjusted its WBC results to eliminate the contribution from non-occupational Cs-137 intakes. Therefore, the dose reconstructor can assume that the reported WBC results are the total measured Cs-137 results.

As indicated above, the doses attributable to fallout Cs-137 are not considered to be occupational doses. Therefore, under the circumstance discussed in Section 5.3.1, the dose reconstructor can attribute positive WBC results to fallout Cs-137 and not assign any intakes or doses based on those WBCs.

5.4.1.4 Intakes of Radionuclides for Medical Reasons

Bioassay measurements at the INEL periodically detected intakes of radionuclides that were received for medical reasons. In such cases, there are notes in the worker's bioassay records identifying the intake as a medical intake. Because medical intakes are not occupational intakes, they are not covered under the EEOICPA and should not be included in the dose reconstructions.

5.4.2 Uncertainties

The uncertainty values for all types of bioassay measurements are typically included in the INEL's bioassay records that are provided by the DOE. When measurement-specific uncertainty values are available, those values are preferred for the data analysis over generic values. When the uncertainty values are not included with the bioassay records, the uncertainty values to be used for the data

analysis should be determined in accordance with the recommendations in the *Technical Information Bulletin: Internal Dose Reconstruction* (ORAUT 2007b).

5.5 INTAKE AND INTERNAL DOSE ASSESSMENT FOR MONITORED WORKERS

This section is intended to apply to the periods of employment that an INEL worker was monitored for internal dose. For INEL workers who were not monitored for some periods of their employment, the recommendations in Section 5.6 should be followed to assess the potential internal doses associated with the unmonitored periods of their employment.

When bioassay data are used to assess the intakes and internal doses for monitored INEL workers, the recommendations in the *Technical Information Bulletin: Internal Dose Reconstruction* (ORAUT 2007b) should be followed. This technical information bulletin is also known as OTIB-0060, and shall be referred to as OTIB-0060 throughout the remainder of Section 5.5. In addition, when workers have monitoring data for specific radionuclides, the radionuclide-specific data are the preferred data for estimating intakes and internal doses for those radionuclides. Because the majority of INEL workers were only monitored for beta- and/or gamma-emitting radionuclides and because the reported bioassay results typically do not provide any indication of the specific radionuclides involved with the intake or potential intake, intakes of specific radionuclides will typically need to be assigned based on ratios that are applied to the estimated Sr-90 and/or Cs-137 intakes. The details on when this needs to be done and on how to do it are provided in the following subsections.

5.5.1 Activation and Fission Product Intakes

Because the majority of the bioassay measurements performed for INEL workers did not determine the potential mixtures of the activation and fission products that the workers were exposed to and because those potential mixtures have a significant impact on the workers' doses, a method for estimating the mixtures of the activation and fission products needed to be selected for the INEL Site. As a result, the method described in the *Technical Information Bulletin: Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* (ORAUT 2007c) was determined to be appropriate and applicable for the INEL Site, because some INEL reactor data were used to develop that method. This technical information bulletin is also known as OTIB-0054, and shall be referred to as OTIB-0054 throughout the remainder of Section 5.5.

NOTE: Because the radionuclides associated with work performed for the SMC Project were limited to isotopes of uranium and the impurities found in recycled uranium, the approach described in this section is not applicable to work performed for the SMC Project. To estimate intakes and internal doses for SMC Project workers, refer to Section 5.5.5 below.

NOTE: Because the radioactive materials associated with some of the work areas at the RWMC did not have radionuclide compositions that were completely represented by OTIB-0054 (ORAUT 2007c), the intakes received at the RWMC need to be evaluated on a case-by-case basis. To estimate intakes and internal doses for RWMC workers, refer to Section 5.5.6 below.

Section 3.0 of OTIB-0054 (ORAUT 2007c) identifies two exclusions that were potentially applicable to the ICPP. The following paragraphs describe why those exclusions do not apply to the ICPP.

Even though radioactive lanthanum (RaLa) work was performed at the ICPP between 1956 and 1963, this exclusion in OTIB-0054 (ORAUT 2007c) does not apply to the ICPP because the irradiated nuclear fuel that was reprocessed at the ICPP as part of RaLa operations accounted for less than 1% of the total amounts of irradiated reactor fuel reprocessed at the ICPP during each of those years. Therefore, the radioactive materials associated with the RaLa operations only account for an insignificant fraction of the total amount of irradiated reactor fuels that were reprocessed at the ICPP

during each of the years between 1956 and 1963, and those materials are not considered to be representative of the radioactive materials that the majority of workers were exposed to. However, there were a limited number of ICPP workers who did receive intakes of radioactive materials due to the RaLa operations that were performed. Based on the bioassay results for those exposure incidents, the gamma spectrometry results indicated that the intakes from the RaLa operations were usually limited to radioactive isotopes of iodine versus a mixture of all the activation and fission products present in the source term for the RaLa operations. When the dosimetry records for an ICPP worker indicates that they received an intake from the RaLa operations or when there are positive bioassay results during the years of 1956-1963 that include radioactive isotopes of iodine, those potential intakes should be assessed in accordance with OTIB-0060 (ORAUT 2007b) and Section 5.5.4 below.

The available information on the ICPP indicated that some activation and/or fission products might have been extracted and concentrated at the ICPP. However, there is no information to indicate this was a routine process or that the activation and/or fission products were separated in quantities that were capable of causing a significant alteration to the radioactive source term that the workers were exposed to. Therefore, this exclusion does not apply to the ICPP.

5.5.1.1 Assessment of Missed Intakes

The majority of the bioassay measurements at the INEL were performed to detect the predominant beta/gamma-emitting activation and fission products that were found in the irradiated nuclear reactor fuels. As a result, urine samples were typically only analyzed for gross beta, gross gamma, and/or strontium radioactivity. When the urine samples are only analyzed for gross beta, gross gamma, and/or strontium radioactivity, missed Sr-90 and/or Cs-137 intakes should be assessed in accordance with OTIB-0054 (ORAUT 2007c) and OTIB-0060 (ORAUT 2007b). Similarly, missed Cs-137 intakes should be assessed using the WBC data, in accordance with OTIB-0060 (ORAUT 2007b).

NOTE: For some of the strontium analyses only a preliminary count of the gross beta radioactivity was performed on the portion of the sample that was precipitated by oxalic acid, because the samples were below the detection limits. These sample results were typically labeled as "β(ox)" results in the INEL's bioassay records, and should be treated as chemically processed beta samples when following the OTIB-0054 recommendations. When the preliminary β(ox) result was above the detection limits the processing for the strontium analysis was completed and only the completed strontium analysis results were reported.

5.5.1.2 Assessment of Fitted Intakes

Fitted intakes are assessed when there are positive bioassay measurement results. A bioassay measurement is considered to be positive when its result is greater than the reporting level (this could be the MDA, detection level, or some other value that the site used), and is considered to be negative when its result is less than or equal to the reporting level (ORAUT 2007b).

When a positive bioassay result can be attributed to specific radionuclides, the fitted intake should be assessed for those specific radionuclides, in accordance with OTIB-0060 (ORAUT 2007b).

Unfortunately, the radionuclides responsible for many positive urine sample results cannot be determined when only a gross radioactivity analysis was performed on them. In those instances, a Sr-90 or Cs-137 intake should be calculated in accordance with the recommendations provided in OTIB-0054 (ORAUT 2007c) and OTIB-0060 (ORAUT 2007b). Even though no information was provided in the dosimetry records to indicate that the positive urine sample results were attributable to either Na-24 or radioactive iodine, it may be possible to determine if a radioactive iodine or Na-24

intake occurred. Sections 5.5.4 and 5.5.6 provides guidance on how to determine if an intake was likely attributable to Na-24 or an isotope of radioactive iodine.

5.5.1.3 Assessment of Unmonitored Intakes

When missed and/or fitted intakes are calculated for Sr-90 and/or Cs-137, the unmonitored intakes for all other dosimetrically significant activation and fission products should then be calculated using the ratios provided in Table 7-3 of OTIB-0054 (ORAUT 2007c), unless a bioassay result was reported for any of the radionuclides listed in Table 7-3. The use of actual bioassay data (e.g. radionuclide-specific urine sample, fecal sample, or *in vivo* measurement results) is preferred any time radionuclide specific results are provided.

5.5.1.4 Decay Period Selections

The decay periods to be used for the selection of the appropriate urine activity fractions and associated radionuclide ratios for the intake estimates are to be based on a combination of the recommendations in Table 5-12 of OTIB-0054 (ORAUT 2007c) and the available site-specific information.

For operating reactors, a decay period of 10 days is appropriate, until the reactor is shut down permanently. Once a reactor has been shut down permanently for a year, a decay period of 1 year can be used for the intake estimates. Because of the uncertainties with some of the reactor operation end dates (e.g. sometimes only the year of the shutdown is known), the use of the 40 and 180 day decay periods is not recommended for the shutdown period for the INEL reactors.

Because the minimum decay periods for the spent and/or irradiated reactor fuel storage facilities and fuel examinations facilities at ANL-W and TAN could not be determined at the time this TBD was prepared, the recommendations for the INEL reactors also apply to these irradiated or irradiated reactor fuel storage and/or examination facilities, unless documentation supporting different decay periods is located for those facilities.

For the ICPP, intake estimates based on the 40 day, 180 day, and 1 year decay periods should each be assessed, and the decay period resulting in the highest internal dose for a given organ should be used for the dose assignment. Because the ICPP reprocessed irradiated fuels with varying decay periods, this approach is considered to be reasonable for all types of claims regardless of the compensability decision. At the ICPP, some irradiated fuels were reprocessed with less than a 40 day decay period because of the RaLa operations. However, a decay period shorter than 40 days is not justified because of the relatively small amount of reactor fuel involved with the RaLa operations. The RaLa operations performed at the ICPP between 1957 and 1963 involved processing irradiated fuel with only a 2 day decay period; however, as indicated above, this fuel represented less than 1% of the fuel that was reprocessed during each of those years. Irradiated fuels were also stored at the ICPP's Fuel Storage Facility (Building CPP-603). Table 5-12 of OTIB-0054 (ORAUT 2007c) recommends a 10 day decay period for spent fuel storage facilities when a more appropriate decay period is unknown. However, site-specific information indicates that, with the exception of the RaLa operations, the irradiated reactor fuels had a minimum of 90 days of cooling before being received at the ICPP's Fuel Storage Facility (Allied Chemical undated). Therefore, the decay period recommendations for the ICPP also apply to the ICPP's Fuel Storage Facility.

For the INEL's waste management facilities, a decay period of 1 year is appropriate for the intake estimates.

5.5.2 Actinide Intakes

Because uranium was the only actinide that was routinely separated from the irradiated nuclear fuels at the INEL Site and because the much more readily detectable fission products were present along with the actinides, the potential intakes and doses attributable to the actinides can be estimated by applying ratios to the estimated Sr-90 and/or Cs-137 intakes for a worker. In the event that a worker was monitored for uranium intakes or intakes of other actinides, the actual bioassay results should be used to estimate the worker's internal doses for those radionuclides.

NOTE: The recommendations in this section apply to all INEL workers with the exception of the workers on the SMC Project. For SMC Project workers, the recommendations in Sections 5.5.4 should be followed to account for those workers' potential actinide intakes and doses.

Actinide to Sr-90 and actinide to Cs-137 ratios are provided in Tables 5-22 and 5-23 for the three major fuel types that were present at the INEL Site. For the ANL-W reactor facilities, the actinide ratios for stainless-steel fuels should be used. With the exception of the ANL-W reactor facilities, the actinide ratios for aluminum fuels should be used for the remaining reactor sites. Even though some reactors did not use aluminum fuel types, the aluminum fuel actinide ratios for those reactors likely provide a reasonable overestimate of the actinides in the other fuel types, because the fuels from non-aluminum-fueled reactors typically had much lower burnups than what was assumed for the aluminum fuel calculations.

For the ICPP, the actinide ratios for the aluminum fuels likely provide a reasonable overestimate of the actinides present at the ICPP before 1971, because the non-aluminum fuels that were reprocessed at the ICPP before 1971 had much lower burnups than what was assumed for the aluminum fuels. Because the ICPP was reprocessing a combination of different reactor fuels that had higher burnups by 1971 and because the ICPP workers might have been simultaneously exposed to an unknown mixture of those fuel types, the maximum ratio between the three major fuel types (i.e., aluminum, zirconium, and stainless-steel fuels) for a given actinide should be used for ICPP workers after 1970.

For the INEL's waste management facilities, the maximum actinide ratios should be used for the intake estimates.

The radioactive source term that was used to generate the values in Tables 5-22 and 5-23 was based on the information provided in an INEL document titled *Determination of the Normalized Mass of Individual Radionuclides in the Dissolver Product for Aluminum, Zirconium and Stainless Steel Fuels Previously Processed at INTEC* (Wenzel 2000). This document indicated that the source term values were generated using the ORIGEN2 computer model and that calculations were only performed for the three main types of irradiated reactor fuel that were reprocessed at the ICPP (also known as INTEC). In addition, only a single source term calculation was performed for each of the reactor fuel types using the typical composition, configuration, nuclear fuel burnup, and post-irradiation decay periods that were applicable to those reactor fuels. Because no uncertainty information was provided for the ORIGEN2 calculations, the source term information and the actinide ratios that were calculated for this TBD will be treated as constants.

Because the number of favorable to claimant simplifying approaches or assumptions used to develop the ratios for the actinides was minimal, the distribution(s) associated with the indicator radionuclide intakes will need to be accounted for. Therefore, the distribution associated with the unmonitored actinide intakes and internal doses should be indicative of the distributions associated with the indicator radionuclide intakes that they were based on. For a fitted dose calculation, the distribution should be lognormal with a geometric standard deviation (GSD) of 3.0 (ORAUT 2007b). For a missed dose calculation, the distribution should be triangular (ORAUT 2007b).

Table 5-22. Actinide-to-Sr-90 ratios.[a]

Actinide	Reactor fuel types[b]			
	Al	Zr	SS	Max
Ac	8.0E-12 Ac-227	1.3E-11 Ac-227	2.3E-10 Ac-227	2.3E-10 Ac-227
Th	2.4E-08 Th-228	6.4E-08 Th-228	2.3E-07 Th-228	2.3E-07 Th-228
Pa	1.2E-10 Pa-231	1.1E-10 Pa-231	3.8E-09 Pa-231	3.8E-09 Pa-231
U	5.6E-05 U-234	6.2E-06 U-236	1.4E-03 U-234	1.4E-03 U-234
Np	3.4E-06 Np-237	3.7E-06 Np-237	6.8E-07 Np-237	3.7E-06 Np-237
Pu	8.7E-03 Pu-238	1.5E-02 Pu-238	3.7E-03 Pu-239	1.5E-02 Pu-238
Am	1.4E-04 Am-241	3.9E-06 Am-241	9.0E-08 Am-241	1.4E-04 Am-241
Cm	4.9E-05 Cm-244	1.8E-06 Cm-244	1.1E-10 Cm-242	4.9E-05 Cm-244

a. The values in this table were obtained from the MS Excel workbook titled *INEL – Actinide Ratios* (ORAUT 2009a).
b. Al = aluminum; Zr = zirconium; SS = stainless-steel; Max = maximum; The actinide isotopes to use for the dose calculations are provided below each actinide ratio; The actinide isotopes listed are the predominant alpha-emitting actinides in the source term for a given fuel type.

Table 5-23. Actinide-to-Cs-137 ratios.[a]

Actinide	Reactor fuel types[b]			
	Al	Zr	SS	Max
Ac	7.6E-12 Ac-227	1.3E-11 Ac-227	2.1E-10 Ac-227	2.1E-10 Ac-227
Th	2.3E-08 Th-228	6.2E-08 Th-228	2.1E-07 Th-228	2.1E-07 Th-228
Pa	1.2E-10 Pa-231	1.1E-10 Pa-231	3.5E-09 Pa-231	3.5E-09 Pa-231
U	5.3E-05 U-234	6.0E-06 U-236	1.3E-03 U-234	1.3E-03 U-234
Np	3.2E-06 Np-237	3.5E-06 Np-237	6.2E-07 Np-237	3.5E-06 Np-237
Pu	8.3E-03 Pu-238	1.4E-02 Pu-238	3.4E-03 Pu-239	1.4E-02 Pu-238
Am	1.3E-04 Am-241	3.7E-06 Am-241	8.3E-08 Am-241	1.3E-04 Am-241
Cm	4.7E-05 Cm-244	1.7E-06 Cm-244	1.0E-10 Cm-242	4.7E-05 Cm-244

a. The values in this table were obtained from the MS Excel workbook titled *INEL – Actinide Ratios* (ORAUT 2009a).
b. Al = aluminum; Zr = zirconium; SS = stainless-steel; Max = maximum; The actinide isotopes to use for the dose calculations are provided below each actinide ratio; The actinide isotopes listed are the predominant alpha-emitting actinides in the source term for a given fuel type.

5.5.3 Contaminant Intakes for ICPP Workers Who Handled Recycled HEU

If an ICPP worker has bioassay data for uranium, the worker likely handled recycled HEU that was separated from the irradiated reactor fuels. Because the actinide ratios for the recycled HEU were significantly different than the ratios for the irradiated reactor fuel, a different set of ratios needs to be

used to assess unmonitored radionuclide intakes for periods that a worker handled recycled HEU. Because the majority of the ICPP's HEU product was shipped to the Y-12 Plant, the activity ratios for the ICPP HEU are based on information provided for recycled uranium in the *Technical Basis Document for the Y-12 National Security Complex – Occupational Internal Dose* (ORAUT 2006). The default ratios to estimate the unmonitored radionuclide intakes are provided in Table 5-24. The unmonitored radionuclide intakes are calculated by multiplying the default ratios by the total uranium intake that was calculated based on an assessment of the bioassay data.

Table 5-24. Default ratios for recycled HEU intakes.[a]

Radionuclide	Default ratio (pCi/µg total U)
Tc-99	1.33
Th-228	0.05
Np-237	0.06
Pu-238	0.02

a. The values in this table are based on the values provided in Table 5-8 of the Y-12 Plant TBD (ORAUT 2006). The values have also been adjusted in accordance with that TBD for use on all types of cases, including best estimate cases.

It should be noted that the worker might have also been exposed to the radionuclides in the irradiated reactor fuels prior to the uranium being separated. This would be indicated by bioassay data for those workers, which would include either urine sample data analyzed for gross beta, gross gamma, and/or Sr-90 radioactivity or a WBC data for the same period that the uranium bioassay data were provided. In those cases, the potential unmonitored radionuclide exposures should also be assessed in accordance with the recommendations in Sections 5.5.1 and 5.5.2 (i.e., assessed in addition to the unmonitored intakes assessed for recycled HEU handlers).

5.5.4 Radioactive Iodine Intakes

5.5.4.1 General Guidance on Identified Radioactive Iodine Intakes

At the INEL, the results of the bioassay measurements may be dominated by radioactive iodine. When this is the case, it is unlikely that there was a detectable intake of the other activation and fission products. In those instances, the positive bioassay results only need to be assessed for the radioactive isotopes of iodine that might have been present.

The INEL Site often performed a gamma spectrometry analysis to identify and/or confirm the radionuclides that were associated with a bioassay result that INEL dosimetry personnel considered to be significant or suspected to be associated with a radioactive iodine intake. In some instances, the determination that an intake was an iodine intake might have been based on process knowledge or other workplace indicators. In those instances, the isotopes involved with the intake are written next to the positive bioassay result, which is usually a gross gamma in urine result and sometimes might be a gross beta in urine result. However, isotopic mixtures or the quantities of the specific radionuclides involved with the intake are not provided in most circumstances. One common exception to this is when the intake was significant enough to warrant an incident investigation and the generation of an incident report. Because WBCs are a gamma spectrometry type of analysis, the specific radionuclides that are above the detection limits or reporting levels are always reported. When those results are above the reporting levels, which were typically 0.1 µCi, the measured quantities of each radionuclide are provided.

Isotope-specific information provided with positive bioassay results should be used to assess the intakes that are attributable to radioactive iodine. If multiple iodine isotopes are identified and no information regarding the specific mixture is provided, the bioassay results should be assessed

assuming that the intake was attributable solely to each iodine isotope and the isotope generating the highest dose estimate shall be used for the assigned dose. When there is no isotopic information provided for a radioactive iodine intake, the dose reconstructor should assume that the iodine isotope is solely I-131, unless there is information to indicate that the worker's intake occurred during RaLa operations at the ICPP.

If a worker's radioactive iodine intake occurred during RaLa operations at the ICPP, the isotopic mixture provided in Table 5-25 should be used to assess the radioactive iodine intakes and doses. The isotopic fractions for the iodine isotopes provided in Table 5-25 are based on an ORIGEN computer program run for the MTR fuel with a 2 day decay period, which was the fuel type and typical decay period used for the RaLa operations. The isotopes listed in Table 5-25 account for more than 99% of the iodine radioactivity.

Table 5-25. RaLa iodine.

Iodine isotope	Isotopic fractions
I-131	0.3245
I-132	0.4553
I-133	0.2203

5.5.4.2 Special Considerations for Thyroid Cancer Cases

Because thyroid cancer cases are significantly impacted by intakes of radioactive iodine, some special considerations regarding the assessment of the missed and/or unmonitored iodine doses need to be made for those types of cases. For intakes based on positive bioassay results, the recommendations provided in the previous sections are still applicable.

Because a WBC is capable of detecting most short-lived isotopes of iodine, a missed intake and dose for I-131 can be calculated from negative WBCs in lieu of the unmonitored I-131 intakes that would be calculated in accordance to the recommendations in Section 5.5.1.3 above.

When using the OTIB-0054 (ORAUT 2007c) approach for assigning unmonitored iodine intakes after a 40 day decay period, intakes of the much longer-lived I-129 need to be accounted for in the unmonitored internal doses to the thyroid, because the short-lived isotopes have completely decayed away and are no longer present. Until the much shorter lived isotopes of iodine decayed away, the internal doses to the thyroid that were attributable to I-129 were considered to be insignificant relevant to the shorter lived isotopes. It should also be noted that I-129's contribution to the internal doses to organs other than the thyroid are still considered to be insignificant relative to the other activation and fission products that might have been present; therefore, internal doses attributable to I-129 only need to be considered for thyroid cancers.

Because OTIB-0054 (ORAUT 2007c) does not provide I-129-to-indicator nuclide ratios and because the aged iodine is likely in a less volatile form, an approach similar to the approach being used for assessing unmonitored actinide intakes in Section 5.5.2 will be used to assess unmonitored I-129 intakes. Table 5-26 provides I-129 ratios for indicator radionuclides that can be used to estimate the unmonitored I-129 intakes when the applicable decay period exceeds 40 days. Because elemental iodine is very reactive and because it has had over 40 days to react with other materials, I-129 should only be assessed as an inhalation intake and a material having type F lung absorption properties.

Table 5-26. I-129-to-indicator nuclide ratios.[a]

Indicator nuclide	Ratio
Sr-90	4.2E-07
Cs-137	3.9E-07

a. The values in this table were obtained from the MS Excel workbook titled *INEL – I-129 Ratios for Internal TBD* (ORAUT 2009b).

The radioactive source term that was used to generate the values in Table 5-26 was based on the information provided in an INEL document titled *Determination of the Normalized Mass of Individual Radionuclides in the Dissolver Product for Aluminum, Zirconium and Stainless Steel Fuels Previously Processed at INTEC* (Wenzel 2000). This document indicated that the source term values were generated using the ORIGEN2 computer model and that calculations were only performed for the three main types of irradiated reactor fuel that were reprocessed at the ICPP (also known as INTEC). In addition, only a single source term calculation was performed for each of the reactor fuel types using the typical composition, configuration, nuclear fuel burnup, and post-irradiation decay periods that were applicable to those reactor fuels. Because no uncertainty information was provided for the ORIGEN2 calculations, the source term information and the actinide ratios that were calculated for this TBD will be treated as constants.

Because the number of favorable to claimant simplifying approaches or assumptions used to develop the ratios for I-129 was minimal, the distribution(s) associated with the indicator radionuclide intakes will need to be accounted for. Therefore, the distribution associated with the unmonitored I-129 intakes and internal doses should be indicative of the distributions associated with the indicator radionuclide intakes that they were based on. For a fitted dose calculation, the distribution should be lognormal with a geometric standard deviation (GSD) of 3.0 (ORAUT 2007b). For a missed dose calculation, the distribution should be triangular (ORAUT 2007b).

5.5.5 Uranium Intakes for SMC Project Workers

The SMC Project was located at TAN at the old ANP Program facilities (ORAUT 2007a). In 1985, the SMC Project started manufacturing DU armor for the U.S. Army (ORAUT 2007a). As of June 2000, records show that the SMC Project has received 10,129,000 lb of DU for processing (Lewis et al. 2000). Of this, 4,726,000 lb were received from the DOE Feed Materials Production Center (FMPC) at Fernald, Ohio, and 5,403,000 lb were received from the RFP (Lewis et al. 2000). The records also indicate that the DU being used by the SMC Project was recycled DU, and contained other radioactive impurities (Lewis et al. 2000).

The processing of DU at the SMC Project consists of rolling and cutting billets. Recasting is also performed and the SMC Project requires that only metallic DU be used for the recasting process. No chemical processing beyond recasting takes place at the SMC Project. The transuranic impurities in the DU are neither concentrated nor diluted by the recasting process (Lewis et al. 2000). During DU parts fabrication, small quantities of finely divided uranium metal and oxides present inhalation and ingestion potential, as indicated by routine positive personnel bioassays (King 2001).

Air monitoring is the primary method used at the SMC Project to evaluate the potential for exposure to airborne DU. Fixed-head air sampling throughout the plant, supplemented by CAMs, provides the routine information to evaluate the effectiveness of control programs and to indicate potential internal intake. Exposures to concentrations above 0.1 DAC generally indicate the use of respiratory protection and require bioassay follow-up (King 2001).

The uranium values provided in Table 5-27 are representative of the typical composition of recycled DU. In addition to the uranium isotopes, DU contains two beta-emitting radionuclides. Average values for the radiologically significant impurities in the SMC Project's DU are provided in Table 5-28.

Table 5-27. Uranium mass and activity ratios for SMC project DU.[a]

Uranium isotope	Mass %	Activity %	Total (pCi/µg DU)
U-234	0.0010	15.46	0.402
U-235	0.1991	1.07	
U-236	0.0003	0.05	
U-238	99.7996	83.42	

a. The values in this table are the default values for DU that are provided in the IMBA computer program, because the site-specific information (King 2001) did not account for the U-236 that is in recycled DU.

Table 5-28. SMC Project DU impurities.

Nuclide	Average impurity level	
	(pCi/g DU)[a]	(pCi/pCi DU)
Np-237	1.82	4.90E-06
Pu-238	0.272	7.33E-07
Pu-239/240	0.406	1.09E-06
Am-241	2.78	7.49E-06
Tc-99	154	4.15E-04

a. These values were obtained from INEEL/EXT-2000-00959 (Lewis et al. 2000).

Inhalation Absorption Type: Respirable particulates associated with SMC Project operations are probably a mixture of metal and metal oxides. The actual exposures are undoubtedly due to mixtures of absorption types. During the 18 years of operation, much bioassay data have been collected on a large number of individuals. The overall elimination patterns are consistent with type M but probably are a mixture of all types. It could be too simplistic to assume a pure absorption type when the chemical form is not known for certain. The dose reconstructor should assume either type M or type S uranium to maximize the dose to the organ of concern. Exposure to significant quantities of type F uranium at the SMC Project is not considered credible [27].

Particle Size: Detailed particle size analyses of representative samples from the various operations indicate that an activity median aerodynamic diameter (AMAD) of 2.4 µm is appropriate for typical SMC Project operations. This site-specific value of 2.4 µm AMAD is used for assessments of intakes at the SMC Project and is the default particle size distribution (INEEL 2001).

5.5.6 Intakes for RWMC Workers

The RWMC has supported INEL operations as a waste management complex since 1952 and has received large quantities of TRU waste from the RFP and other DOE facilities. Improved operations have resulted in a decrease in internal dose potential. The original disposal techniques (dumping waste in open trenches) were relatively vulnerable to airborne release in comparison to current total-containment practices. The four major areas in the RWMC facility are: (1) the Subsurface Disposal Area (SDA) for permanent disposal of low-level waste and some early TRU waste (which will eventually be exhumed and repackaged); (2) the 58-acre Transuranic Storage Area (TSA) for temporary storage, examination, and certification before shipment to the Waste Isolation Pilot Plant; (3) the operations area; and (4) an administrative area where no radioactive waste is permitted.

The comprehensive radiation protection program for RWMC includes extensive air monitoring, personnel monitoring, and surface contamination surveillance. Although infrequent, there have been instances of inadvertent intakes (based on Table 5-1, there were two in 1992 and one in 1996). Therefore, bioassay is conducted randomly at the present time (INEEL 2001).

Because of the variety of radiological materials that were present at the RWMC, more than one mixture of radioactive materials may need to be considered for the intake and internal dose calculations for RWMC workers. For RWMC workers who were likely exposed to materials bearing

irradiated fuels or for instances when the material that the worker was exposed to was unknown, the guidance information in Sections 5.5.1 and 5.5.2 should be followed.

Tables 5-29 and 5-30 summarize the major radionuclides in the RWMC waste inventory for the SDA and TSA, respectively. TRU radionuclides are the primary contaminants in the TSA waste; all but Pu-241 are alpha emitters. Because the materials are not homogeneous, it should not be assumed that failing to detect one radionuclide in the inventory invalidates detection of other radionuclides.

Table 5-29. Radioactive low-level waste inventory in the active pits in the SDA (INEEL 2001).

Waste type	Volume (m^3)	Total Ci	Radionuclide	Concentration (Ci/m^3)	Percentage
Low-level waste	75,600	3.35E+05	Co-60	4.1E+0	92.00%
			Ni-63	3.3E-1	7.40%
			Sr-90	9.7E-3	0.22%
			Cs-137	9.7E-3	0.22%
			H-3	5.8E-3	0.13%
			C-14	8.9E-4	0.02%

Table 5-30. Radioactive waste inventory in the TSA (INEEL 2001).

Waste type	Volume (m^3)	Total Ci	Radionuclide	Concentration (Ci/m^3)	Percentage
Stored contact-handled TRU waste	65,000	4.06E+5	Pu-241	2.5E+00	44.1%
			Am-241	1.4E+00	24.7%
			Pu-238	9.7E-01	17.1%
			Pu-239	6.3E-01	11.1%
			Pu-240	1.5E-01	2.6%
			U-233	1.4E-02	0.2%
			Cm-244	0.8E-02	0.1%

5.5.7 Intakes for Waste Reduction Operations Complex Workers

The Waste Reduction Operations Complex (WROC) includes several reactor facilities that operated from the 1950s to the late 1960s and the Power Burst Facility (PBF) reactor, which operated from 1972 to 1985. These currently inactive facilities are in a common control area. In addition, a low-level waste incinerator called the Waste Experimental Reduction Facility (WERF) burned waste from all INEL facilities from 1982 to 2001. The WERF, which is undergoing decontamination and decommissioning, was a low-level waste incinerator, and its operations included some mixed waste treatment (Stacy 2000).

The waste at WERF was in the form of burnable containers and the resultant high-fired and solidified ash. The radioactive wastes at the mixed waste storage facility and the reactors were the sources of the radioactivity inventory. The ashes were removed remotely to a glovebox and solidified in 55-gal drums.

The radiological protection program included CAMs, fixed air-sampling systems, RAMs, surface and personnel contamination surveillance, and effluent monitors.

To assess potential exposures to activation and fission products for WROC workers, the recommendations in Section 5.5.1 should be followed. Because the types of radioactive materials processed at WROC varied depending on the area shipping the waste to WROC, the assumption favorable to claimants is that potential actinide exposures were associated with zirconium fuel as processed at the ICPP. Therefore, the recommendations in Section 5.5.2 for zirconium fuels should be used to assess potential actinide exposures at the WROC.

5.5.8 Sodium-24 Intakes at Liquid-Sodium-Cooled Reactors

Because liquid sodium metal was used as a coolant for the INEL's fast reactors, significant quantities of Na-24 were produced by neutron reactions in the coolant. However, unmonitored Na-24 intakes at the fast reactor sites are unlikely for the following reasons.

Sodium metal would not have been in areas routinely accessed by personnel until most of the Na-24 had decayed due to the intense gamma radiation from the Na-24. Given that the half-life of Na-24 is only 15 hours, a decay period of only about 6.25 days would be required to eliminate nearly all of the Na-24 in the sodium metal. Significant intakes of Na-24 from sodium metal in the solid or liquid forms are also unlikely, due to its reduced dispersion potential in those forms. It is also unlikely that unknown accidental exposures to Na-24 would have occurred, because sodium metal is highly reactive and reacts violently with water. Because of the significant hazards associated with sodium metal, any exposures to the metal would have been minimized and would have also been known when they occurred. When an exposure was known to have occurred, the radiological protection practices being performed throughout the INEL Site would have likely required that bioassay measurements capable of detecting Na-24 intakes be performed on the exposed workers. Therefore, no special assessments need be performed for workers who were not monitored for Na-24 exposures at the fast reactors.

If it can be determined that a worker received an intake of Na-24 or was monitored for a Na-24 exposure, fitted and/or missed Na-24 intakes and doses may need to be assessed. When bioassay measurements were used to monitor a worker for Na-24 exposures, the potential Na-24 intakes and doses are assessed in accordance with the OTIB-0060 (ORAUT 2007b).

5.6 INTAKE AND INTERNAL DOSE ASSESSMENT FOR UNMONITORED WORKERS

This section is intended to apply to the periods of employment that an INEL worker was not monitored for internal dose.

For most of the history of the INEL, personnel dosimeters were issued to all workers who entered the security access control points at each facility, regardless of their work assignment. For example, administrative and clerical personnel were required to wear dosimeters even though they were not exposed to external or internal doses that were above the onsite ambient levels. Dose reconstructors should determine the appropriate unmonitored dose assignment in accordance with the following recommendations or the recommendations for unmonitored workers in the *Technical Information Bulletin: Application of Internal Doses Based on Claimant-Favorable Assumptions for Processing as Best Estimates*, also known as OTIB-0033 (ORAUT 2005a).

- If the worker's file includes positive external dosimeter readings during a given calendar year that the worker was not monitored for internal dose, the worker should be treated as a radiation worker and a default missed internal dose (i.e., internal missed doses based on hypothetical bioassay data) should be assessed. The default missed internal dose is assessed by assuming that the most likely type of bioassay measurement was performed on the last day of the calendar year with a positive external dose and that bioassay measurement was negative. For the years from INEL's startup through 1960 the most likely type of bioassay method was a urine sample analyzed for gross beta radioactivity; after 1960 the most likely type of bioassay method was a WBC. The missed intake and dose calculations are the same as described in Section 5.5 above. For readily non-compensable claims, a simpler but favorable to claimant alternative to assigning a default missed internal dose is to assign OTIB-0018 (ORAUT 2005b) based doses in accordance with the recommendations in OTIB-0033 (ORAUT 2005a) for unmonitored workers who were routinely exposed.

- If a worker was not monitored for internal dose and had no detectable external doses reported for a given calendar year, only the environmental internal doses need to be assessed for that calendar year.

The approaches outlined in this section are based on the determination that significant unmonitored intakes of radioactive material are unlikely at the INEL. This determination was based on evaluations of the most common types of bioassay that were employed throughout the INEL's history and an evaluation of the results for those bioassay measurements. The INEL's bioassay records indicate that over 90% of those bioassay results were below their respective detection limits (ORAUT 2010a, 2010b).

5.7 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this TBD, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this TBD, link data, quotations, and other information to documents available for review on the Project's Site Research Database.

Norman Rohrig served as the initial Document Owner for this document. Mr. Rohrig was previously employed at the INEL and his work involved management, direction, or implementation of radiation protection and/or health physics program policies, procedures, or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner, who is fully responsible for the content of this TBD, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source.

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

Boyd Levitt served as one of the initial Subject Experts for this document. Mr. Levitt was previously employed at the INEL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Levitt, those materials are fully attributed to the source.

Douglas Wenzel served as one of the initial Subject Experts for this document. Mr. Wenzel was previously employed at the INEL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Wenzel, those materials are fully attributed to the source.

Henry Peterson served as one of the initial Subject Experts for this document. Mr. Peterson was previously employed at the INEL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Peterson, those materials are fully attributed to the source.

[1] Rich, Bryce. Principal Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of the referenced AEC annual reports.

[2] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
The existence of the RESL, also known as the H&S Laboratory, HSL, and HSD, allowed for a consistency in assumptions about the internal dosimetry program because RESL was a constant even though there were a large number of varying facilities and continually changing contractors at the INEL. This is supported by the referenced AEC annual reports.

[3] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement is based on the fact that the Na-24 is not liberated from the aluminum cladding and that Na-24 has a short half-life.

[4] Peterson, Henry. Principal Health Physicist. Intrepid Technology and Resources. November 2004.
This statement is based on Mr. Peterson's experience as a radiological engineer at TRA.

[5] Jenkins, JoAnn M. Senior Health Physicist. Dade Moeller & Associates. March 2007.
Because noble gases are inert, they do not interact with the lung when they are inhaled. The inhalation dose due to noble gases is due to the amount of gas that is inhaled and then exhaled. Because the volume of noble gases in the semi-infinite cloud around an individual is much greater than the inhaled volume, the external dose from noble gases is generally greater.

[6] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of dosimetry and incident records. Typically, large internal exposures can be related to unplanned events or planned releases, both of which are usually well documented.

[7] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on information from King (2001). It is important to use caution in considering worker controls based solely on radiological limits because these limits might not provide adequate protection from an industrial hygiene perspective.

[8] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
The fuel at ICPP contained long-lived fission products with both alpha- and beta-emitting radionuclides. This allowed for the use of beta/gamma CAMs. It was understood that if beta/gamma airborne radioactivity was detected by the CAMs, the possibility of airborne alpha activity also existed. This is supported by the data in Table 5-18 that demonstrate that most of the isotopes were beta and gamma emitters.

[9] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
The delay before processing fuel allowed the short-lived radioisotopes to decay significantly, which left only the longer-lived radioisotopes as significant contributors to radiation dose.

[10] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on Hayden (1958).

[11] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on knowledge of the NRF from working at INEL. Because the NRF had operating reactors, the potential for internal exposure existed and, therefore, individuals who worked at the facility could have received internal doses.

[12] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on his experience as a member of the H&S management team that supported the INEL internal dosimetry program and is supported by the referenced AEC annual reports and internal dosimetry TBDs. It is common industry practice to take a proactive approach to the detection of ventilation failure. By monitoring the work area with continuous and retrospective air monitors, and using personnel and contamination monitoring, the facility H&S team provided real-time notification of ventilation failures and a means to assess personnel exposures.

[13] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on his experience as a member of the H&S management team that supported the INEL internal dosimetry program and is supported by the referenced AEC annual reports and internal dosimetry TBDs.

[14] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on a review of the referenced AEC annual reports. It is further supported by Stacy (2000).

[15] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on a review of the INEL dosimetry and bioassay records. It is further supported by the creation of the RCIMS database, which has a bioassay tracking function that went into effect in June 1999 and is described in Bhatt (2002).

[16] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This conclusion is based on a review of Andersen, Perry, and Ruhter (1995), INEEL (2001), King (1990) and Puphal (1994).

[17] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on his experience as a member of the H&S management team that supported the INEL internal dosimetry program and is supported by Puphal (1994).

[18] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. Certain crafts were required site-wide. Examples of these are maintenance and construction workers such as pipefitters, plumbers, electricians, and health physics technicians. When a single contractor managed all or most of the INEL site, it was common for these types of personnel to perform work all over the site instead of at a specific facility. In these cases, personnel could have been exposed to a variety of radioactive material due to their varying work locations and situations.

[19] Rich, Bryce, and Jenkins, JoAnn M. Consulting Health Physicist and Senior Health Physicist. M. H. Chew & Associates and Dade Moeller & Associates. November 2004. Internal doses are most accurately calculated when they are based on specific knowledge of exposure conditions such as specific radionuclides and quantities. When this information is available, it should be used in the dose reconstruction. When an employee worked at various locations, the specific bioassay data for each location should be used to calculate their internal

dose. The statement by Mr. Rich about the equivalency of the bioassay programs at the different facilities is based on his professional experience as a member of the H&S management team that supported the INEL internal dosimetry program and is supported by the referenced internal dosimetry TBDs.

[20] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of the referenced AEC annual reports and internal dosimetry TBDs. CAMs measure the amount of radioactivity in the air and an alarm occurs when a preset level is reached. Retrospective air monitors also provide information on the levels of airborne activity, but the information is obtained after the fact when the air samples are analyzed.

[21] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of the referenced AEC annual reports and internal dosimetry TBDs.

[22] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of facility bioassay data and is supported by the referenced AEC annual reports and internal dosimetry TBDs.

[23] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of facility bioassay data and is supported by Puphal (1994).

[24] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on information from an extensive review of INEL bioassay records.

[25] Rohrig, Norman, Consulting Health Physicist. Intrepid Technology and Resources. November 2004.
This plot was generated from noting the number of samples of different types in the database derived by the ORAU Team from copies of the bioassay results.

[26] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of dosimetry records. It is also supported by the INEL incident reports. It is common industry practice to conduct incident investigations in a timely manner.

[27] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement is based on a review of the document titled *INEEL M&O Contractor Technical Basis for Internal Dosimetry* (INEEL 2001).

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GLOSSARY

absorption
In external dosimetry, process in which radiation energy is imparted to material. In internal dosimetry, movement of material to blood regardless of mechanism.

absorption type
Categories for materials according to their rate of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively in the respiratory tract (slow solubilization). Also called solubility type. See *inhalation class*.

activation
Creation of a radioisotope by interaction of a stable (nonradioactive) element with neutrons, protons, or other types of radiation.

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

breeder reactor
Nuclear reactor that makes more new fissionable material than it consumes.

calcine
(1) Dry solid (grainy or granular) product of a chemical process that removes liquids from a solution. (2) Process for creating the chemical reaction that removes liquids from a solution.

cladding
The outer layer of metal that encases a reactor fuel element or fissile material of the pit of a nuclear weapon, often made with aluminum or zirconium. In a reactor, cladding promotes the transfer of heat from the fuel to the coolant, and it builds up fission and activation products over time from the fission of the fuel.

contamination
Radioactive material in an undesired location including air, soil, buildings, animals, and persons.

core
Central region of a nuclear reactor where fission of the fuel takes place.

criticality
State of a radioactive mass (e.g., the core of a nuclear reactor) when the fission reaction becomes self-sustaining. Nuclear reactors *go critical* when started.

curie (Ci)

Traditional unit of radioactivity equal to 37 billion (3.7E+10) becquerels, which is approximately equal to the activity of 1 gram of pure Ra-226.

decontamination

Reduction or removal of radioactive material from a structure, area, object, or person. Decontamination can occur through (1) treating the surface to remove or decrease the contamination or (2) allowing natural radioactive decay to occur over a period of time.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium.

dosimeter

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

dosimetry

Measurement and calculation of internal and external radiation doses.

enriched uranium

Uranium in which processing has increased the proportion of ^{235}U to ^{238}U to above the natural level of 0.7%. Reactor-grade uranium is usually about 3.5% ^{235}U; weapons-grade uranium contains greater than 90% ^{235}U.

fission

Splitting of the nucleus of an atom (usually of a heavy element) into at least two other nuclei and the release of a relatively large amount of energy. This transformation usually releases two or three neutrons.

fission product

(1) Radionuclides produced by fission or by the subsequent radioactive decay of radionuclides. (2) Fragments other than neutrons that result from the splitting of an atomic nucleus.

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma rays are very penetrating, but dense materials such as lead or uranium or thick structures can stop them. Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

half-life

Time in which half of a given quantity of a particular radionuclide disintegrates (decays) into another nuclear form. During one half-life, the number of atoms of a particular radionuclide decreases by one half. Each radionuclide has a unique half-life ranging from millionths of a second to billions of years.

hot cell

Shielded laboratory for handling of radioactive materials with the aid of remotely operated manipulators. The walls and windows are made of materials that protect workers from radiation.

***in vitro* bioassay**
Measurements to determine the presence of or to estimate the amount of radioactive material in the excreta or in other biological materials removed from the body.

***in vivo* bioassay**
The measurements of radioactive material in the human body utilizing instrumentation that detects radiation emitted from the radioactive material in the body.

ionizing radiation
Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons.

irradiated fuel
Nuclear reactor fuel that has been exposed to neutron radiation in a nuclear reactor and has undergone fission reactions. Irradiated fuel often contains hazardous levels of activation and fission products that make handling it difficult. Nuclear reactor fuel that has been irradiated to the point where it is no longer useful in sustaining a nuclear reaction is typically referred to as spent fuel. See *spent fuel*.

isotope
One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., U-234, U-235, and U-238). Isotopes have very nearly the same chemical properties but often have different physical properties.

micro-
Prefix that divides a unit by 1 million (multiplies by 1E-06).

milli-
Prefix that divides a unit by 1,000 (multiplies by 1E-03).

mixed waste
Unwanted material containing both radioactive and hazardous components.

natural uranium (U, U-nat, NU)
Uranium as found in nature, approximately 99.27% U-238, 0.72% U-235, and 0.0054% U-234 by weight. The specific activity of this mixture is 2.6E+07 becquerel per kilogram (0.7 picocuries per gram).

neutron (n)
Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

nucleus
Central core of an atom, which consists of positively charged protons and, with the exception of ordinary hydrogen, electrically neutral neutrons. The number of protons (atomic number) uniquely defines a chemical element, and the number of protons and neutrons is the mass number of a nuclide. The plural is nuclei.

nuclide

Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

photon

Basic unit of electromagnetic radiation. Photons are massless "packages" of light energy that range from low-energy microwave photons to high-energy gamma rays. Photons have energies between 10 and 100 kiloelectron-volts.

proton

Basic nucleic particle with a positive electrical charge and mass slightly less than that of a neutron. There are protons in the nuclei of every atom, and the number of protons is the atomic number, which determines the chemical element.

radiation

Subatomic particles and electromagnetic rays (photons) that travel from one point to another, some of which can pass through or partly through solid materials including the human body. See *ionizing radiation*.

radioactivity

A property possessed by some isotopes of some elements (e.g. K-40, C-14, Cs-137, etc…). The process of undergoing spontaneous transformation of the nucleus, generally with the emission of alpha or beta particles, often accompanied by gamma rays. This term is also used to designate radioactive materials. See *radionuclide*.

radionuclide

Radioactive nuclide. See *radioactivity* and *nuclide*.

radioactive lanthanum (RaLa)

Lanthanum-140 was used in diagnostic tests of the implosion mechanism of a nuclear fission bomb. Inside the high-explosive shell, a core of La-140 replaced the usual plutonium core, and gamma radiation from the lanthanum provided information on the course of the implosion.

radioactive waste

Radioactive solid, liquid, and gaseous materials for which there is no further use. Wastes are generally classified as high-level (with radioactivity as high as hundreds of thousands of curies per gallon or cubic foot), low-level (in the range of 1 microcurie per gallon or cubic foot), intermediate level (between these extremes), mixed (also contains hazardous waste), and transuranic.

rem

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The average American receives 360 millirem a year from background radiation. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

reprocessing

Mechanical and chemical processing of irradiated nuclear fuel to separate useable fissionable products (i.e., uranium and plutonium) from waste material. Reprocessing was discontinued in the United States in 1992.

shielding

Material or obstruction that absorbs ionizing radiation and tends to protect personnel or materials from its effects.

spent fuel

Nuclear reactor fuel that has been irradiated in a nuclear reactor to the point where it is no longer useful in sustaining a nuclear reaction. See *irradiated fuel*.

spent fuel storage basin

Pool or pit of reinforced concrete filled with water for storage of irradiated nuclear fuel. The water is shielding and coolant.

transuranic (TRU) waste

Radioactive waste that contains transuranic elements and has radioactivity of 100 or more nanocuries per gram.

U.S. Atomic Energy Commission (AEC)

Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The U.S. Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

X-ray radiation

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). The only difference between X-rays and gamma rays is their point of origin. X-rays originate in the electron shells of an atom, whereas, gamma rays originate in the nucleus of an atom.

zirconium

Metallic element with atomic number 40. Zirconium is highly resistant to corrosion, and it is alloyed with aluminum to make cladding for nuclear fuel and sometimes in small amounts with the fuel itself.