



ORAU TEAM Dose Reconstruction Project for NIOSH

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

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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
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
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
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
mCi	millicurie
MDA	minimum detectable activity
MDL	minimum detectable level
MeV	megavolt-electron, 1 million electron-volts
MFP	mixed fission product
mg	milligram
mi	mile
min	minute
mL	milliliter
mo	month
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPC _a	MPC for airborne activity
mrem	millirem
mrep	millirep
MTR	Materials Test Reactor
Nal(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
PBF	Power Burst Facility
pCi	picocurie
PCS	primary coolant system
POC	probability of causation
RaLa	radioactive lanthanum
RAM	radiation (or remote) area monitor
RCG _a	radioactive concentration guide [for airborne activity]
RCIMS	Radiation Control Information Management System
RDR	Radiation Dosimetry and Records
RESL	Radiological Environmental Sciences Laboratory
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
TLV	threshold limit value
TMI	Three Mile Island
TRA	Test Reactor Area
TREAT	Transient Reactor Experiment and Test
TRU	transuranic
TSA	Transuranic Storage Area
U.S.C.	United States Code
WBC	whole-body counting
WERF	Waste Experimental Reduction Facility
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
α	alpha particle
β	beta particle
γ	gamma
σ	standard deviation
μCi	microcurie
μg	microgram
μm	micrometer
§	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 for 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 causation¹] guidelines established under subsection (c) ...” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation) define “performance of duty” for DOE employees with a covered cancer or restrict the “duty” to nuclear weapons work.

As noted above, the statute includes a definition of a DOE facility that excludes “buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program” [42 U.S.C. § 7384l(12)]. While this definition contains an exclusion with respect to the Naval Nuclear Propulsion Program, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled “Exposure in the Performance of Duty”] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external dosimetry monitoring results are considered valid for use in dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived:

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

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

5.1.1 Purpose

This TBD provides specific information concerning documentation of historical practices at the Idaho National Laboratory (INL).

5.1.2 Scope

The majority of this document provides background information to aid the internal dose reconstructor through increased general understanding, data interpretation, defaults, and so forth. Sections 5.2.1 to 5.2.3 provide facility descriptions, and Section 5.2.4 details the radionuclides of concern. Section 5.3 describes the INL radiological protection program as it evolved over the years, Section 5.4 discusses internal dose control, and Sections 5.5 and 5.6 describe minimum detectable activities (MDAs) and whole-body counting (WBC), respectively. Section 5.7 contains specific information necessary for dose reconstruction when the facility or facilities where the employee worked are known. Sections 5.8 and 5.9 describe the treatment of missed dose and unmonitored workers.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.10.

5.2 HISTORICAL OVERVIEW

In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) and started construction of facilities on a 572,000-acre site approximately 50 mi west of Idaho Falls in southeastern Idaho. The NRTS was later renamed the Idaho National Engineering Laboratory and then the Idaho National Engineering and Environmental Laboratory. In 2005, DOE shortened the name to the Idaho National Laboratory. For convenience, this TBD uses INL where it is unnecessary to distinguish.

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 INL that had a marked influence on the internal dosimetry programs at each of the facilities were:

- The original mission of the NRTS was (as the name implied) highly enriched (over 50% and mostly over 90%) uranium reactor concept development, materials testing through high-flux test reactor operation, and chemical processing of those highly enriched (valuable) uranium fuels. The production of weapons-grade nuclear materials was not a mission.
- The NRTS began operations 8 to 10 yr after Oak Ridge National Laboratory 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 Oak Ridge (ACC 1952) were adopted at the startup of the NRTS facilities.
- Two AEC field offices (Chicago and Idaho) were responsible for and had oversight of the NRTS programs included in this report. In addition, the Nuclear Navy under direction of the Pittsburgh Field Office administered programs and used facilities on the NRTS for training and program development. There were thus three Federal organizations utilizing the NRTS and its infrastructure. This TBD does not apply to naval facilities or 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 NRTS among a large variety of facilities and constantly changing contractors, the AEC established a Health and Safety (H&S) Laboratory at NRTS 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 AEC HSL and each contractor [2].

As a consequence, and in spite of the constant changes at the NRTS, basic assumptions about MDAs or minimum detectable levels (MDLs), missed dose potential, and so forth are relatively consistent across the years [3]. There were differences in the available nuclear materials from facility to facility, but as early as 1955 or 1956 gamma spectral analysis capabilities at the NRTS allowed the significant bioassay results (those which would result in reportable internal dose) to be defined in terms of the specific radionuclides. The practice in the case of a higher urine sample result was to attempt radionuclide identification through gamma spectral analysis and chemical separation. This document describes default assumptions for use in cases when the bioassay records for a worker do not include the radionuclide analyses and only record gross beta or gross alpha results [4].

When an incident occurred, it was the policy to investigate thoroughly and identify all individuals involved in the incident [5]. 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.

5.2.1 Test Reactors

NRTS, which was the primary nuclear reactor development laboratory in the United States, tested or evaluated more than 100 reactor concepts (DOE 1997). Fifty-two test reactors were designed, constructed, and operated (including operation-to-destruction tests) on the NRTS. The INL site 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 Argonne National Laboratory-West (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 INL 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.

5.2.2 Highly Enriched Spent Fuel Chemical Processing

In addition to being the site for experimental test reactors, INL is the home of the Idaho Nuclear Technology and Engineering Center (INTEC), formerly known as the Idaho Chemical Processing Plant (ICPP). INTEC reprocessed highly enriched reactor fuel (^{235}U enrichments of 50% to 93%) for 39 yr from 1952 to 1991. Aged mixed fission products (MFPs) were the predominant internal hazard, although enriched uranium isotopes and plutonium isotopes (^{238}Pu enhanced) were limiting in specific process locations. INTEC 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.

5.2.3 Other Nuclear Facilities and Processes

Other nuclear facilities at INL that resulted in internal dose potential or experience include:

- The Radioactive Waste Management Complex (RWMC) handled radioactive wastes generated by nuclear facilities on INL 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) 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.
- At the Naval Reactors Facility (NRF), the U.S. Navy used the INL for operating reactors and as a naval reactor training center. Because this is not a DOE program and not under the oversight of DOE, NRF is not part of the dose reconstruction and compensation program. However, through the years NRF has participated in limited coordination of radiological protection programs and site support services. It is possible that some workers' internal dose could have resulted from their support work at the NRF [7].

5.2.4 Radionuclides of Concern and Solubility

INL facilities and activities have related primarily to experimental reactor design and development, spent fuel processing, DU parts production, and low- and high-level radioactive waste treatment and disposal. The latest revision of the INL Site Description TBD (ORAUT 2005) describes these activities in more detail. Table 5-1, which lists the internal doses received at the INL from 1992 to 2000, demonstrates that the radionuclides of more recent concern to internal dose are MFPs (primarily aged), uranium and its decay products, and TRU wastes and their decay products. The table lists internal doses as committed effective dose equivalents (CEDEs). This information is from

INEEL (2001) and can be useful to the reconstruction effort primarily as an indicator of intake radionuclide identity and quantities for this recent 10-yr period. Before 1990, workers received higher doses from the radionuclides in Table 5-1, as well as from MAPs (primarily longer-lived, i.e., ^{110m}Ag and others) [8]. Most internal doses have been identified following an incident rather than as a result of routine bioassay measurements (INEEL 2001).

Table 5-1. Recorded internal doses for 1992 to 2000 (INEEL 2001, p 37).^a

Year of dose assignment	Facility	CEDE (mrem)	Radionuclides of concern
2000	SMC	10, 31, 20, 10, 13, & 20	U-238
2000	INTEC	11	Pu-238, Pu-239/240
1999	SMC	15, 48, 13, & 12	U-238
1999	INTEC	16	Pu-239/240
1998	SMC	16	U-238
1997	TRA	10 ^a & 10 ^a	Eu-152 and Eu-154
1997	INTEC	24	Pu-238 and Pu-239/240
1997	TAN	13 ^a	Am-241, Cs-137, Sr-90, U-233, U-238, Pu-238, Pu-239/240
1997	SMC	16 & 20	U-238
1996	RWMC	43	Pu-239
1996	INTEC	15, 87, 136, 652 ^a , 655 ^a , 677 ^a , & 678 ^a	Am-241, Cs-137, Sr-90, Pu-238, Pu-239/240
1996	SMC	10, 10, 12, 16, 17, 18, 20, 20, & 23	U-238
1995	INTEC	10, 13, 14, 15, 23, 28, 29, 42, 45, & 53	Am-241, Cs-137, Sr-90, Pu-238, Pu-239/240
1995	SMC	10, 12, 14, 15, 19, & 26	U-238
1994	INTEC	14, 20, 25, & 29	Am-241, Cs-137, Pu-238, Pu-239/240
1994	SMC	10, 10, 12, & 15	U-238
1993	INTEC	14, 35, 36, 39, 50, & 53	Am-241, Pu-238, Pu-239/240
1993	SMC	11	U-238
1992	RWMC	20 & 205	Am-241, Pu-238, Pu-239/240
1992	SMC	11, 12, 12, 14, 15, 15, 16, 20, 32, & 52	U-238

a. Internal doses determined after a "significant" event when intake occurred. However, even the other intakes were defined as a result of contamination incidents or high airborne measurements.

Table 5-2 lists radionuclides of concern at INL 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 INL program follows the DOE guidance and uses the International Commission on Radiological Protection (ICRP) Publication 30 nomenclature for solubility (ICRP 1979) as noted in the table. In addition, Table 5-2 provides the ICRP Publication 68 (ICRP 1995) recommended absorption type.

Through the years at INL, plotting urine and fecal elimination curves has shown that an effective solubility of moderate (M) to slow (S) per the ICRP Publication 66 system (ICRP 1994) is to be expected for the radionuclides listed in Table 5-2, with the exceptions of strontium, cesium, iodine, mercury, and tritium. Elimination curves for ³H, ⁹⁰Sr, ¹³¹I, ¹³⁷Cs, and ²⁰³Hg show they are typically of a fast absorption type (F). The chemical explanation is that radioactive materials in oxygen atmospheres oxidize rapidly, which forms less soluble compounds. The default assumption of type M or S would be appropriate when based on the result to the organ in question that is more favorable to the claimant.

Table 5-2. Primary radionuclides of concern.

Element	Radionuclides	ICRP 30 ^a retention class	ICRP 68 absorption type	Preferred analysis technique
Hydrogen	H-3 [assume tritiated water vapor (HTO)]	D	F	Urine
Chromium	Cr-51		F,M,S	WBC
Manganese	Mn-54		F,M	WBC
Iron	Fe-59		F,M	WBC
Cobalt	Co-58, Co-60	W,Y	M,S	WBC
Zinc	Zn-65		S	WBC
Strontium	Sr-89, Sr-90/Y-90 ^b	D	F	Urine
Zirconium/niobium	Zr/Nb-95	W,Y	F,M,S	WBC
Molybdenum	Mo-99		F,S	Urine
Technetium	Tc-99		F,M	Urine
Ruthenium	Ru-103, Ru-106		F,M,S	Urine
Silver	Ag-110m		F,M,S	WBC
Antimony	Sb-122, Sb-125		F,M	WBC
Tellurium	Te-132		F,M	Urine, WBC
Iodine	I-131, I-133 (assume elemental I ₂)		F	Urine, thyroid counter
Cesium	Cs-134, Cs-137	D	F	WBC
Barium/lanthanum	Ba/La-140		F/F,M	WBC
Cerium	Ce-141, Ce-144	W,Y	M,S	WBC
Europium	Eu-152, Eu-154, Eu-155	W,Y	M	WBC
Gadolinium	Gd-153		F,M	WBC
Tantalum	Ta-182		M,S	WBC
Mercury	Hg-203 (assume inorganic)		F	Urine, WBC
Protactinium	Pa-233		M, S	WBC
Uranium	U-233, U-234, U-235, U-236, U-238	D,W,Y	F,M, S	Urine, fecal
Neptunium	Np-237		M	Urine, fecal
Plutonium	Pu-238, Pu-239/240	W,Y	M, S	Urine, fecal
Americium	Am-241	W	M	Urine, fecal, lung count

- a. Assigned in the INL internal dosimetry technical basis document (INEEL 2001); ICRP 30 = ICRP 1979; ICRP 68 = ICRP 1995.
- b. Yttrium has an absorption type of M or S; however, in the case of short-lived Y-90, it tracks its parent Sr-90, which is normally a type F at the INL.

5.3 RADIOLOGICAL PROTECTION PROGRAM MANAGEMENT AND SUPPORT

5.3.1 Internal Dosimetry Issues Related to Contractor Changes

The changes in contractors at INL during its 54-yr history [listed in Tables 2-1 and 2-2 of the Site Description TBD of this Site Profile (ORAUT 2005)] resulted in relatively frequent management changes at most of the facilities. (The contract with the University of Chicago to operate the ANL-W facilities has not changed in this 54-yr period.)

The primary oversight for INL, which included most projects on the site and all support functions, was initially assigned to the AEC Idaho Operations Office (IDO) [9]. 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 [10]. The Chicago Operations Office (COO) has provided oversight for ANL-W (contracted by the University of Chicago) programs and facilities, while the U.S. Navy has provided oversight for the NRF program [11]. ANL-W, which is a DOE program, is included in this

TBD. The ANL-W program uses site support services, including internal dosimetry support, but the bioassay results are reported through COO [12].

The INL personnel dosimetry records have been and are documented and permanently maintained by the various organizations over time [13]. 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 report.

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 INL, particularly the bioassay analytical techniques and calculation processes [14]. 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 [15].

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 [16].

Internal dose reconstruction for personnel who worked at a number of INL 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 [17]. 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). In the cases where only gross beta or gross alpha bioassay results were available, the results were normally within 2σ [18]. If it is necessary to evaluate intakes from the gross beta or gross alpha results, the radionuclide defaults should be $^{90}\text{Sr}/^{90}\text{Y}$ and ^{238}Pu , respectively.

5.3.2 Bioassay Programs

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 mo in 1953 [19]. Table 5-3 lists the reconstructed history of routine bioassay frequency.

5.3.3 Internal Dose Records

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 INL 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 INL internal dosimetry program functions and provides quality assurance. The RDR unit functions include documentation and records custodial responsibilities. In 1999, the Radiation Control

Table 5-3. Routine bioassay history summary.^a

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1953–1960	Annual	<i>In vitro</i> urine	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Individual data sheets; Table 5-10; Horan 1959; AEC 1961
1961	Annual	<i>In vitro</i> urine; <i>in vivo</i>	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Table 5-10; Horan 1962
1962–1972	Annual	<i>In vitro</i> urine; <i>in vivo</i>	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Horan 1962 Dodd 1963
1973–1981	Annual When internal intake suspected Annual	<i>In vitro</i> urine Fecal <i>In vivo</i>	Radiation workers	<u>Reporting</u> 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
1982–1987	Annual When internal intake suspected Annual Termination	<i>In vitro</i> urine Fecal <i>In vivo</i>	CPP-603 workers; fuel reprocessing operators Waste reprocessing operators; shift laboratory workers; health physics technicians Selected radiochemistry workers; maintenance workers; denitrator operators. All radiation workers.	<u>Reporting</u> 50-yr CDE >10% quarterly standard in ERDA Manual Chapter 0524 (ERDA 1975).	Staggered to monitor group throughout the year.	Author unknown 1981
1988–1989	1 to 6 mo Annual 18 to 24 mo Termination New hire	<i>In vivo</i> <i>In vitro</i> fecal <i>In vitro</i> urine <i>In vivo</i> <i>In vitro</i> <i>In vivo</i>	All radiation workers. When internal exposure suspected. Depending on review of radiation dose history.	<u>Investigating</u> 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 receives some sort of analysis or sampling every 3 mo.	Tschaeche 1988
1990–1994	Annual 6 mo Annual New hire Termination	<i>In vivo</i> <i>In vitro</i> fecal <i>In vitro</i> urine <i>In vivo</i> <i>In vitro</i> urine; fecal <i>In vivo; in vitro</i>	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. Worked at a facility where gamma-emitting radionuclides were handled. Worked in U manufacturing or recovery facilities; worked with transuranic materials. Any employee suspected of having an internal exposure or on a scheduled monitoring program.	<u>Reporting</u> In accordance with DOE Order 5480.11 (DOE 1988). Workers that could receive 0.1 rem AEDE or 5 rem ADE organ or tissue dose. <u>Investigating</u> AEDE ≥0.01 rem.	Bioassay requested when workplace monitoring program indicates >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

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1995	Appropriate to the facility mission, potential uptakes.	<i>In vivo</i>	All radiation workers that 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.	<u>Reporting</u> In accordance with DOE 5480.11 (DOE 1988) and 10 CFR Part 835: Workers that could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE.	Each facility has a specific Technical Basis Document for Internal Dosimetry.	Andersen, Perry, and Ruhter 1995
	When workplace monitoring indicated significant potential for intakes.	<i>In vitro</i> urine; fecal				
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	<i>In vivo</i> <i>In vitro</i>	Any employee that was on a scheduled monitoring program.			
2001	As developed by individual facilities based on analysis tables developed for each radionuclide.	<i>In vivo</i> <i>In vitro</i> urine fecal	All radiation workers.	<u>Reporting</u> In accordance with DOE 5480.11 (DOE 1988) and 10 CFR Part 835. Workers that 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
	Termination	<i>In vivo</i> <i>In vitro</i>	Any employee that was on a scheduled monitoring program.			

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

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).

5.3.4 Internal Dose Regulations, Investigation Levels, and Data Codes

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 5-4 describes coded information that could appear in records after 1989. Table 5-5 describes internal dose information that could appear in pre-1989 records. Table 5-6 contains analytical nomenclature, and Table 5-7 contains INL codes for various site areas.

Table 5-4. 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 5-5. 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.

Federal regulations about permissible internal dose and formal reporting requirements to the AEC, the U.S. Energy Research and Development Administration (ERDA), and the DOE changed periodically during the site's history. While the regulations influenced the level of calculated internal dose that would be reported, these changes did not significantly affect the analytical programs for the detection of internal intake [20]. For example, the monitoring and analytical programs were designed to initiate, through *in vitro* and/or *in vivo* bioassay analysis, an investigation of any potential internal intake as indicated by positive air sampling, personnel contamination, etc. Most of these recorded analyses did not result in detectable radionuclides [21]. 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). Regulations 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, 1963b, 1968, 1975; ERDA 1975). Changes in the reporting levels did not generally result in changes to the air-monitoring or bioassay programs [22]. Each individual analytical result was documented and placed in individual exposure files regardless of the formal reporting requirements.

Table 5-6. Analytical information that could be in worker dose files.

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

Table 5-7. Area codes that could be in worker dose files.

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

a. See the acronyms and abbreviations list.

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 [23]. Dickson (1977) established official investigation levels (Table 5-8) 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). These procedural limits did not materially affect the bioassay sampling frequency or the recording of even undetectable radioactivity in bioassay samples, although the request for and number of follow-up samples and analyses could have been different as a function of the formal regulations in effect [24].

Table 5-8. Derived investigation levels (μCi) in 1977 for acute exposures (Dickson 1977).

Radionuclide	Inhalation (lung burden)	Ingestion (total activity)
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

5.4 INTERNAL DOSE CONTROL

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 [25]. 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 INL for internal exposures has been preventive in nature [26]. 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 INL 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).

In a related matter, 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 5-9 is a summary of control limits primarily from the *CPP* [Chemical Processing Plant] *Health Physics Manual* (ACC 1952) and current operating procedures.

5.4.1 Air Monitoring

The monitoring of radioactivity in the air in occupied areas was a basic element of the internal exposure prevention program. Beta/gamma continuous air monitors (CAMs) were used from the beginning of all facility and program operations in routinely occupied areas. With the exception of the 1985 SMC Project, the primary contaminant radionuclides by activity were either MFPs or MAPs, which were beta/gamma emitters with maximum permissible concentrations/derived air

Table 5-9. Surface contamination control and MDAs.

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

concentrations (MPCs/DACs) above $1 \times 10^{-9} \mu\text{Ci}/\text{cm}^3$. TRU materials and uranium were available at some of the INL facilities, but they were nearly always well-tagged with beta/gamma activity that allowed beta/gamma-detecting CAMs to be used to warn of possible alpha contamination or internal exposures.

ACC (1952) describes a CAM and three other air-sampling systems. The manual required use of a filter-type respirator when airborne activity exceeded $1 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$ for beta/gamma activity or $1 \times 10^{-11} \mu\text{Ci}/\text{cm}^3$ 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, p IX: 4-1) (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 [27]. 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 [28]. 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 [29]. 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 [30].

In general, workers were asked to submit to bioassay whenever they were in an area where a CAM alarm sounded. In addition, the fixed location and retrospective air-sampling system signaled the need for bioassay if elevated air sample results were detected [31].

5.4.2 Early Technical and Analytical Capabilities at the National Reactor Testing Station

DOE HSL technical reports and annual reports, coupled with facility memoranda and reports, documented the analytical detection capability of NRTS 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). Early analyses for plutonium generally were gross alpha counts on a plutonium separation; later, alpha spectroscopy was used to count and better characterize the results.

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

AEC (1961) outlined a basic philosophy in relation to gamma counting of bioassay samples. Gamma counting would be effective in all situations except for exposure to pure strontium isotopes. To guard against this unlikely possibility, the procedure of performing a strontium analysis for individual workers at risk (radiation workers) every 2 yr and at termination was established. Because of the improbability of finding detectable activity, all activities were to be precipitated by oxalic acid in a weak 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 8×10^{-8} $\mu\text{Ci/mL}$ of ^{90}Sr .

As part of the EEOICPA coworker data program for the INL, images of all bioassay results from before 1986 were provided to the Oak Ridge Associated Universities (ORAU) Team, and the results were entered into a database. This database has over 150,000 bioassay results. The nearly 135,000 urine bioassay results fall into over 80 categories of analytes, a few of which are the result of minor typographical differences. Figure 5-1 shows the time course of the gross beta, gross gamma, and strontium analyses that were performed most often. It appears that the group beta/gamma protocol seems to have followed the beta protocol in the early years and the gamma protocol after 1964. The early 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 [32]. In the early 1970s, *in vitro* bioassay was largely replaced by WBC [33]. In the 1950s and 1960s, several thousand assays were performed each year, well over one per monitored worker per year. The early results were reported in dpm per sample except for certain assays, such as uranium reported in grams U per liter and tritium in dpm per milliliter [34]. About September 1973 the activity units changed to microcuries with a large negative exponent so that picocuries would be a more natural unit. There are some

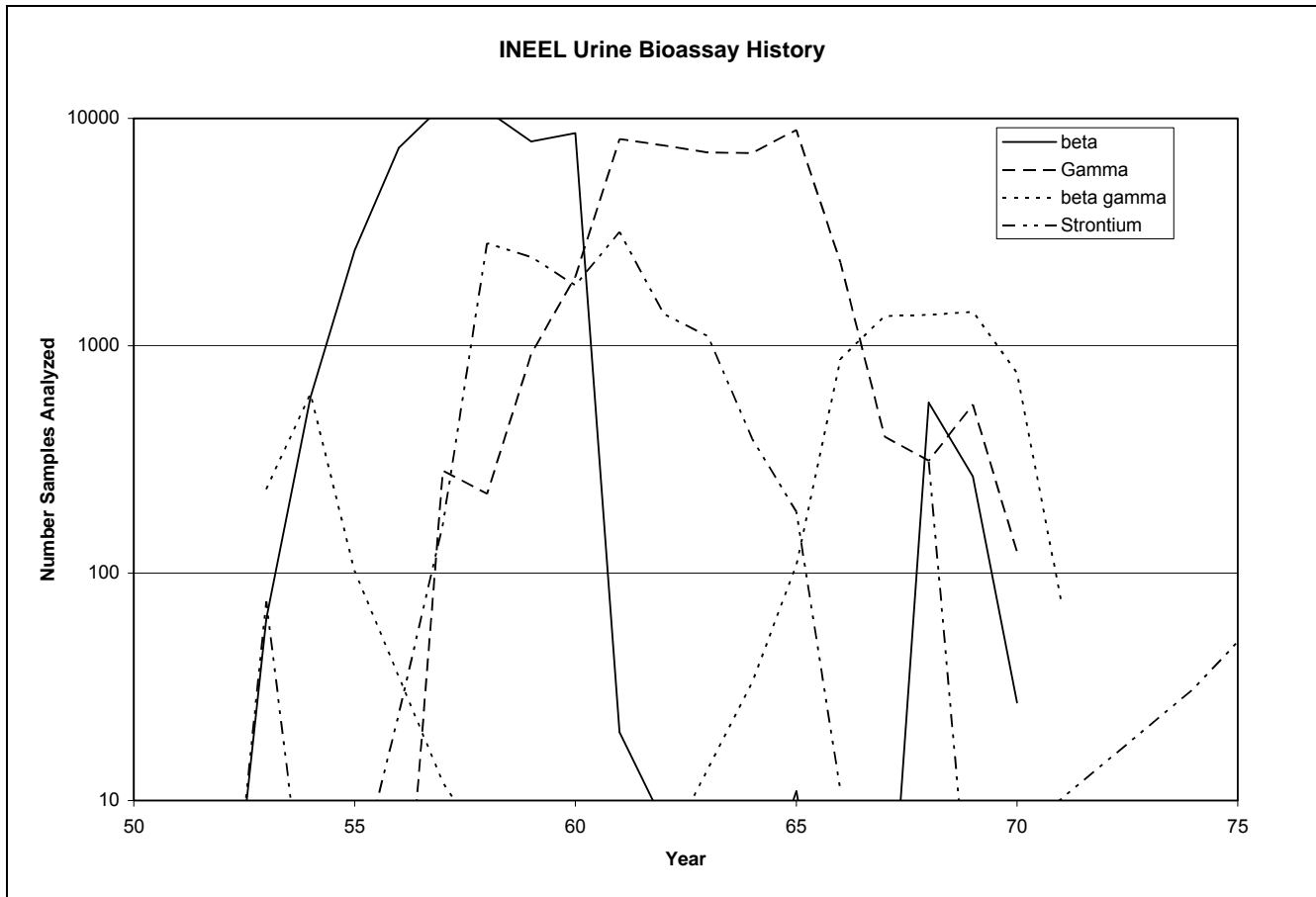


Figure 5-1. History of urine bioassay [35].

confusions in the database record between dpm and microcuries and between per-sample or per-milliliter results, so if the results or uncertainties appear anomalous, that is a likely explanation. The sample size for alpha spectrometry went from the urine sample size (about a liter) to 1 (meaning all of the sample, not 1 mL) in mid-1979.

Table 5-10 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.

Table 5-11 summarizes early detection limits for various environmental water and milk samples between 1953 and 1965. These analytical sensitivities are applicable to analysis of body fluids or substances for the purpose of bioassay (Dodd 1964).

The special and routine bioassay sample analyses 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 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 5-10. 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 \pm 632 dpm/5 mL	992 \pm 40 dpm/5 mL	172 \pm 16 dpm/mL
	Gross γ^c	2,433	2,712	9,120	174	129	-- ^d	7.15	4.76	4	35,972 \pm 310 dpm/5 mL	19,817 \pm 105 dpm/5 mL	1,900,235 \pm 876 dpm/75 mL
Co-60	β	--	--	1	--	--	0	--	--	0	--	--	300 \pm 75 dpm/450 mL
Sr-90	β	3	105	3,248	3	0	2	100	0	2	4.12E-2 dpm/mL	Insignificant	183 \pm 8 dpm/75 mL
Sr-91	β	20	37	2	19	0	--	95	0	0	388 \pm 1.6	Insignificant	4 \pm 8 dpm/mL
I ^e	β	--	9	--	--	2		--	22		0	9992 \pm 80 dpm/mL	--
Cs-137	β	--	--	40	--	--	0	--	--	0	--	--	1,460 \pm 10% dpm/1700 mL
Ba-139	α	20			16	0	--	80	0	--	120 \pm 0.8 dpm/mL	0	--
Th ^f	α	7	0	--	0	0	--	0	0	--	Insignificant	0	Insignificant
U	α	--	--	4	--	--	--	--	--	0	--	--	10 μ g/L
U-233	α	17	3	--	1	0	--	0.06	0	--	180 \pm 4.0 dpm/mL	Insignificant	--
Pu-239	α	18	0	29	0	0		0	0	0	Insignificant	Insignificant	2E-9 μ Ci/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			

- 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.
- If only gross β analyses are available, the default should be Sr-90.
- If only gross γ analyses are available, the default should be Cs-137.
- = no data reported.
- Iodine isotope(s) not identified in references. Assume I-131.
- Thorium isotope(s) not identified in references. Assume Th-228.

Table 5-11. Detection limits applicable to environmental sample analyses, 1953 to 1965 (Dodd 1964, p. 71).

Type of sample	Radiation	Detection limit
Water	Alpha	3E-9 μ Ci/mL
	Beta	6E-9 μ Ci/mL ^a
	Tritium	4E-6 μ Ci/mL
Milk	Iodine-131	10 pCi/L ^b
	Strontium-90	1.5 pCi/L

a. Reduced in October 1962 from 1.5E-7 μ Ci/mL.

b. Reduced in September 1962 from 50 pCi/L.

The individual analytical data were recorded, as were the specific doses for which formal evaluations were required. Aoki (1979) states that before 1977, internal doses were assigned if the internal dose or committed dose was greater than 50% of the yearly body or organ doses allowable per ERDA Manual Chapter 0524 (ERDA 1977), which was the controlling regulation at the time. In 1977, the policy was changed to assign internal doses when the CDE to an organ exceeded one-tenth of the quarterly radiation protection standard (Aoki 1979). Replies to requests for radiation exposure history before 1979 stated that there was “no positive exposure reported” when the dose was below the reporting levels noted above (Aoki 1979). Because individuals remembered receiving some internal exposure, the statement, “no positive exposure reported,” was determined to be misleading and was changed in 1977 to “no reportable levels recorded” (Aoki 1979). In all cases, copies of the bioassay results were placed in the individual’s radiation exposure file and should be in the individual’s dose file in the NIOSH Office of Compensation and Support Claims Tracking System (NOCTS) database.

5.5 MINIMUM DETECTABLE ACTIVITIES

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 N 13.30, *Performance Criteria for Radiobioassay* (HPS 1996), both the *in vitro* and *in vivo* radiobioassay laboratories at INL received DOELAP accreditation in February 1998 (INEEL 2001, p 11). In accordance with this accreditation, MDAs and decision levels at the 95% confidence level (2σ) are performed. Tables 5-12 and 5-13 list current MDAs for urine and fecal sample analysis, respectively, along with values gleaned from historical documents. A large majority of the urine samples taken at the site were single voidings; 24-hr samples were used for special sampling purposes (i.e., follow-up samples, primarily to extend the sensitivity). The MDAs listed are those for the primary samples; the tables include the recommended periods for the MDA values. Table 5-14 lists the current *in vivo* MDAs along with values from historical documents and the recommended periods for use with the MDA values.

5.6 DEVELOPMENT AND IMPACT OF WHOLE-BODY COUNTING

WBC was introduced at INL in 1961. As early as 1961 one of the fundamental conclusions from experience at the NRTS with *in vivo* and *in vitro* internal dosimetry analytical techniques was that a large proportion of the internal exposures to NRTS workers was to insoluble materials. Radionuclides (e.g. ¹²⁵Sb, ^{110m}Ag, ⁶⁵Zn, and ⁹⁵Zr/Nb) 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-hr specimen (Horan 1962; Sill, Anderson, and Percival 1964). WBC was demonstrated to detect activity as low as 0.01 μ Ci in a

Table 5-12. MDAs for urine samples 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	26,000	Ebersole and Flygare 1957 ^b
Gross γ	1957–1964	75	580	10,800	AEC 1961 ^b
	1965–1971	75	205	3,800	Data Sheet
H-3	1972–1994	3	0.5 dpm/mL	700	AEC 1972 ^b , AEC 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	32	AEC 1972 ^b , AEC 1974 ^b
	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	0.14	AEC 1974 ^b
Np-237	1974–present	1,000	0.1	0.14	AEC 1974 ^b
U (FP)	1954–1961	0.1	1E-5 g U/L ^c	14 μ g U ^c	Database
	1962–1971	0.1	5E-6 g U/L ^c	7 μ g U ^c	Database
U (KPA) ^d	1985–present		0.2 μ g/L	0.28 μ g 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	1.3	Dodd 1964 ^b
	1971–1973	1,000	1.03	1.4	AEC 1972 ^b
	1974–1979	1,000	0.47	0.66	AEC 1974 ^b
	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	0.22	AEC 1974 ^b
	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	0.22	AEC 1974 ^b

- a. Based on 1,400-mL daily volume and typical sample size. May need adjustment for larger or smaller sample.
b. MDA calculated from inferred 2σ uncertainty.
c. Smallest reported value. Not MDA.
d. KPA = kinetic phosphorescence analysis.

Table 5-13. MDAs for fecal samples 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	Dodd 1964 ^b
	1974–1994	0.02	AEC 1974
	1995–present	0.03	Andersen, Perry, and Ruhter 1995; Bechtel BWXT 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 inferred 2 σ uncertainty.

10-min count (Horan 1962). This detection level was several orders of magnitude more sensitive than the maximum permissible body burdens (MPBBs) for most beta/gamma fission and activation products.

As a consequence, the *in vivo* counting program was used to count (1) all terminating employees who required physical examinations, (2) employees that were suspected of having a possible internal intake, and (3) selected groups of individuals that were scheduled for semiroutine analyses by health physics supervisors (Sommers 1961). In 1963 approximately 1,650 counts were performed; only those activities greater than 0.1 μ Ci were further quantified. This level was determined to be less than one-tenth of the MPBB for most of the gamma-emitting isotopes.

The 1963 annual summary report (Dodd 1964) describes the year's follow-up analyses for the WBC program. As shown in Table 5-15, many of the individuals were counted multiple times. The maximum activity detected provides an upper bound on how large an activity could be found in someone in earlier years before the whole-body counter was operational.

Table 5-14. *In vivo* MDAs by period.

Radiation/ radionuclide	Period	<i>In vivo</i> MDA (nCi)	Count time (min)	Reference
Cr-51	1962–2000	12	10	Percival and Anderson 1962 ^a
	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	1962–2000	12	10	Percival and Anderson 1962 ^a
	2001–present	2.5	5	INEEL 2001
	2001–present	1.1	10	INEEL 2001
Co-60	1962–1970	12	10	Percival and Anderson 1962 ^a
	1971–1988	5	10	AEC 1972 ^a ; AEC 1974
	1989	7	10	Martin 1989
	1990–1992	2 (lung)		Rich 1990
	1993–2000	7	10	Grothaus 1993; Andersen, Perry and Ruhter 1995
	2001–present	2.5	5	INEEL 2001
	2001–present	1.1	10	INEEL 2001
Zn-65	1962–1988	12	10	Percival and Anderson 1962 ^a
	1989–2000	10	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	4.9	5	INEEL 2001
	2001–present	2	10	INEEL 2001
Sr/Y-90	1968–1977	70 (skull)	10	Voelz 1969 ^a ; AEC 1972 ^a ; AEC 1974
	1978–present	34 (skull)	10	Martin 1989; Grothaus 1993
Zr/Nb-95	1962–1988	12	10	Percival and Anderson 1962 ^a
	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	2.6	5	INEEL 2001
Ru-106	2001–present	27	5	INEEL 2001
	2001–present	7.6	10	INEEL 2001
Ag-110 ^m	1962–present	12	10	Percival and Anderson 1962 ^a
Sb-125	1962–present	14	10	Martin 1989; Grothaus 1993
I-131	1962–1989	12	10	Percival and Anderson 1962
	1990–1992	2 (thyroid)	10	Rich 1990
	1993–2000	0.3 (thyroid)	10	Grothaus 1993
	2001–present	3.8	5	INEEL 2001
	2001–present	0.13 (thyroid)		INEEL 2001
Cs-134	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1990–present	2(lung)		Rich 1990
	2001–present	3	5	INEEL 2001
	2001–present	0.96	10	INEEL 2001
Cs-137	1962–1970	12	10	Percival and Anderson 1962 ^a
	1971–1998	5	10	AEC 1972 ^a ; AEC 1974; Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1999–2000	2. (lung)	10	Rich 1990
	2001–present	3.1	5	INEEL 2001
	2001–present	1.9	10	INEEL 2001
Ba/La-140	1962–present	12	5	INEEL 2001
Ce-141	1962–present	9.9	5	INEEL 2001
	2001–present	3.2	10	INEEL 2001
	2001–present	0.11 (lung)	60	INEEL 2001
Ce-144	1962–2000	50	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001–present	44	5	INEEL 2001
	2001–present	15	10	INEEL 2001
	2001–present	0.44 (lung)	60	INEEL 2001
Eu-152	1962–present	4	10	INEEL 2001
	2001–present	0.18 (lung)	60	INEEL 2001

Radiation/ radionuclide	Period	<i>In vivo</i> MDA (nCi)	Count time (min)	Reference
Eu-154	1962–present	2	10	INEEL 2001
Eu-155	1962–present	1	10	INEEL 2001
Ga-153	1962–present	6.5	10	INEEL 2001
	2001–present	0.096 (lung)	60	INEEL 2001
Hf-181	1962–present	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
Ta-182	1962–present	12	10	Percival and Anderson 1962 ^a
Hg-203	1962–present	12	10	Percival and Anderson 1962 ^a
Th-230	1974–present		1,000	AEC 1974
Th-234	2001–present	1.4 (lung)	60	INEEL 2001
Np-237	1974–present			AEC 1974
U-235	1993	0.2 (wound)	20	Grothaus 1993
	1962–present	0.2 (lung)		Rich 1990
	2001	0.11 (lung)	60	INEEL 2001
U-dep/nat	1989	3 (lung)	60	Martin 1989; Grothaus 1993
Pu-238	1989–1998	26 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	1 (wound)	20	Grothaus 1993
	1999–2000	30 (lung)		Rich 1990
	2001	54 (lung)	60	INEEL 2001
Pu-239/240	1971–1993	30	100	AEC 1972 ^a
	1974–1988	74 (lung)	100	AEC 1974
	1989–1988	80 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	2 (wound)	20	Grothaus 1993
	1990–2000	30 (lung)		Rich 1990
	2001–present	140 (lung)	60	INEEL 2001
Am-241	1989–1999	0.6 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	0.1 (wound)	20	Grothaus 1993
	1990–2000	0.2 (lung)		Rich 1990
	2001–present	0.14 (lung)	60	INEEL 2001

a. MDA calculated from inferred 2σ uncertainty.

5.7 SPECIFIC FACILITIES

Each of the facilities was responsible for its internal dosimetry monitoring program, which was designed 1) to prevent and mitigate internal exposure and 2) to evaluate and document internal dose above the detectable limits [36]. The following descriptions provide insight and default instructions for dose reconstruction if the specific operational facility is known.

5.7.1 Test Area North

General Electric built the first facilities at TAN in 1952 for the ANP Program, which was active during the 1950s and early 1960s before ANP was determined to be impractical [37]. The initial mission was to develop reactors for aircraft propulsion. The large facilities for this program have been used for a number of subsequent INL projects. Approximately 25 different reactor concepts and experiments have been conducted at this location, which features large hot cells, maintenance shops, water storage pools, and waste management areas (Stacy 2000). The IETs in the 1950s were conducted under area controls and radiological monitoring surveillance. The reactors operated in the open, and each test involved the release of large quantities of short-lived radioactive fission product gasses and volatiles. However, workers were protected by enclosures (e.g., control point buildings) and constant monitoring for identification of unanticipated exposures [38].

After termination of the IET programs, the TAN facilities were used to handle, inspect, store, and prepare for disposal the materials from unplanned reactor excursions. Major reactor components of the damaged SL-1 reactor were examined in the large hot cell and prepared for ultimate disposal.

Table 5-15. 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-103-106 ^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-141-144	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.

Fuel from the damaged Three Mile Island (TMI) reactor was brought to and stored in the large water pool facility, where it was examined and prepared for permanent dry storage.

The operation of the hot cells, storage basins, and waste treatment facilities involved aged MFPs (primarily ¹³⁷Cs and ⁹⁰Sr/Y) with periods when ⁹⁵Zr/Nb and ¹⁴⁴Ce were present. Activation products were also encountered, primarily ⁶⁰Co. Alpha emitters (uranium isotopes and TRU radionuclides) were present, but their ratios were at least 50:1 beta:alpha.

Operation of the 15 experimental reactor facilities in this area of the site resulted in short-lived fission products in addition to longer-lived MFPs and some MAPs. A few reactor experiments involved operating the reactor to destruction, with the attendant breach of containment and potential internal exposures.

5.7.1.1 Specific Manufacturing Capability Project

The SMC Project began late in the site history (1985) at the ANP Program site in the large hangar facility. The program uses DU to produce armor packages for the U.S. Army M1-A1 and M1-A2 tanks (Stacy 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 radionuclides of concern at SMC are the isotopes of DU listed in Table 5-16. The mass percentages, relative activities in picocuries per microgram, and the total picocuries per microgram are based on Integrated Modules for Bioassay Analysis (IMBA) default values.

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 yr 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

Table 5-16. Mass and activity ratios of SMC DU isotopes [39].^a

Isotope	Mass %	Activity %	Relative activity (pCi/μg)	Total (pCi/μg)
U-238	99.8	83.42	0.3354	0.402
U-236	0.0031	0.05	0.0002	
U-235	0.20	1.07	0.004	
U-234	0.001	15.46	0.062	

a. In addition to the uranium isotopes, DU contains two beta-emitting radionuclides. Due to short half-lives, Th-234 and Pa-234m reach equilibrium with the U-238 parent within about 6 mo of billet casting. Beta monitoring is an important part of radiological survey and measurements, even though these radionuclides do not contribute significantly to internal doses from intakes of DU.

S 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 [40].

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).

Chemical Toxicity: The threshold limit value (TLV) for inhalation of airborne concentrations of uranium and its compounds (independent of isotopic composition), as reported by the American Conference of Governmental Industrial Hygienists, is 0.2 mg/m³. The TLV is the time-weighted average concentration for a normal 8-hr workday and a 40-hr workweek, to which nearly all workers can be repeatedly exposed, day after day, without adverse effect. Conversion of the mass concentration using the specific activity of DU of 3.81×10^{-7} mCi/mg results in a radioactivity concentration of 7.6×10^{-11} μCi/cm³. The DAC for Type M uranium is 3×10^{-10} μCi/cm³, which is a factor of 4 larger than the chemical TLV. The SMC Project staff has always been aware of the need to consider the chemical toxicity of SMC Project DU exposure in addition to the radiological limit [41].

Natural Background Uranium Excretion: Urine samples submitted by SMC Project nonradiation workers in 1987, 1994, and 1998 were assumed to represent the nonoccupational excretion levels of the SMC Project worker population. The results ranged from 0.04 to 0.33 μg/L with wide fluctuations

in individual measurement, some as high as 1.0 µg/L (King 2001). The average reported uranium concentration was 0.157 ±0.109 µg/L at 1σ uncertainty. Therefore, 0.16 µg/L is used as the nonoccupational component of uranium excretion for SMC Project workers, and is subtracted from each urine result before assessment of occupational internal dose. The bioassay results in the worker files reflect the subtraction of 0.16 µg/L from the value determined in the laboratory bioassay result (King 2001).

5.7.2 Idaho Nuclear Technology and Engineering Center

INTEC, formerly the ICPP, consists of a complex of highly enriched spent fuel storage basins, fuel dissolution and uranium extraction processing facilities, a high-level liquid waste storage tank farm, high-level waste calcining processes, and associated analytical and support capabilities. INTEC was a process facility for the recovery of highly enriched uranium from spent fuels from a variety of national and a few foreign reactors. Because highly enriched uranium was the product, the process vessels had to be small for criticality control. Rather than being a plant with large canyon construction and complete remote control and maintenance, INTEC 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 [42].

Most internal doses experienced at INTEC were from accidental releases. Table 5-17 lists unusual and episodic events that have occurred at INTEC.

Table 5-17. Notable airborne incidents at INTEC.

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 at INTEC	I-131	Radioactive iodine spread through makeup area to operating corridor. Thyroid intake to several health physics technicians and operators in the 40-µCi range.	Rich 1958; Hayden 1958
10/16/59	Criticality accident - in shielded process system	Short-lived noble gases and I-131, -132, -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, -132, -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 INTEC main stack	Ru-106	No internal doses detected.	ERDA 1977
11/17/72	INTEC mass spectrometry Pu contamination incident	Pu-238, -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	Henry and Slagle 1985

			the radiological control program.	
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

5.7.2.1 Highly Enriched Spent Fuel Storage

The original spent fuel storage facility (CPP-603) was a 1.5-million gal, three-basin, 20-ft-deep unlined concrete water pit that operated from 1950 until 1984. Because the basin was unlined, the use of demineralized water was not feasible due to corrosion of the bare concrete. In the late 1970s, ion-exchange columns and a sand filter were installed for water cleanup.

The basin was used for the storage of aluminum-clad, stainless-steel-clad, zirconium-clad, and sodium-bonded stainless-steel fuels. During 34 yr of operation, cases of cladding leaks resulted in MFP contamination of the basin water. Breaks in the stainless-steel cladding of sodium-bonded elements allowed a sodium-water reaction that increased the dispersal of fission products to the pool water. Pool water contamination up to 0.15 $\mu\text{Ci/mL}$ (Rich et al. 1974) existed for periods during which cleanup methods were developed. As a result, during the late 1950s through the 1970s, air activity above the pool area was routinely measured at 10% to 25% (Rich et al. 1974) of the radioactive concentration guide for airborne activity (RCG_a) [$1 \times 10^{-9} \mu\text{Ci/mL}$ for soluble ^{90}Sr (AEC 1968)] and was one of the very few operations at INL in which operators were allowed to work in fractional MPC_a levels without respiratory protection for several hours a day. Routine bioassay sampling was increased for those personnel. The primary contaminants were aged MFPs, primarily $^{90}\text{Sr}/^{90}\text{Y}$ and ^{137}Cs . For workers with established work histories in the Building 603 storage facility for extended periods, and if specific bioassay analyses are either not available or insufficient, a default intake of 250 $\text{RCG}_a\text{-hr/yr}$ should be assumed (i.e., 1,000 hr/yr at 0.25 RCG_a), which is favorable to the claimant [43]:

$$20,000 \text{ cm}^3/\text{min} \times 60 \text{ min/hr} \times 1,000 \text{ hr/yr} \times 0.25 \times 1 \times 10^{-9} \mu\text{Ci/cm}^3 \times 1 \times 10^6 \text{ pCi}/\mu\text{Ci} \\ = 3 \times 10^5 \text{ pCi/yr.}$$

Based on the activity mix for stainless-steel fuel after 3 yr of decay provided in Table 5-18, an annual intake of 54,000 pCi of ^{147}Pm , 48,000 pCi of ^{144}Ce , 30,000 pCi of ^{137}Cs , 26,000 pCi each of ^{90}Sr and ^{90}Y , 9,300 pCi of ^{106}Ru , 90 pCi of ^{239}Pu , and 33 pCi of ^{234}U can be assumed to account for over 95% of the dose. The remaining ~56,000 pCi are spread over many nuclides and cause less than 5% of the dose [44].

Table 5-18. Radiologically significant radionuclides for INTEC-processed fuels^a [45].

Nuclide	Half-life	Absorption type	Aluminum-clad fuel (decayed 1 yr)		Stainless steel-clad fuel (decayed 3 yr)		Zirconium-clad fuel (decayed 5 yr)	
			Relative activity	% inhalation dose	Relative activity	% inhalation dose	Relative activity	% inhalation dose
Sr-90	28.78 yr	F	2.4E-02	13.1	8.6E-2	14.1	2.0E-1	6.3
Y-91	58.51 d	S	2.6E-02	1.1	9.7E-6	0.0	8.0E-10	0.0
Zr-95	63.98 d	S	4.0E-02	1.2	3.3E-5	0.0	5.6E-9	0.0
Ru-106	368.2 d	S	2.3E-02	5.5	3.1E-2	5.7	4.5E-3	0.2
Cs-137	30.07 yr	F	2.5E-02	1.2	9.3E-2	3.2	2.1E-1	1.4
Ce-144	284.3 d	M	3.0E-01	47.4	1.6E-1	18.4	2.2E-2	0.5
Pm-147	2.623 yr	M	5.6E-02	1.3	1.8E-1	3.2	8.4E-2	0.3
U-234	245,500 yr	S	2.9E-09	0.0	1.1E-4	4.0	7.2E-8	0.0
Pu-238	87.71 yr	M	1.3E-04	26.1	7.6E-6	1.2	3.0E-3	90.0
Pu-239	24,110 yr	M	4.1E-07	0.1	3.0E-4	50.0	1.6E-6	0.1
Total ^b				97.0		99.0		98.4
Mass fraction U-234 to total U				3.4E-5		5.3E-5		1.4E-4

- a. Assume exposure to aluminum-clad fuel activity from the beginning through 1970, zirconium-clad fuel activity from 1971 to present, and stainless-steel-clad fuel activity when there is indication that the Pu-239 activity exceeds the Pu-238 activity.
- b. The total percent of the inhalation dose is less than 100% because other radionuclides not included in the table contribute small amounts of dose.

In 1984, the Fluorinel Dissolution Process and Fuel Storage Facility was completed. It features a large stainless-steel-lined pool with ion-exchange cleanup systems and other features for improved contamination control and the reduction of chronic internal exposure potential.

5.7.2.2 High-Level Wastes

The high-level waste storage tank farm consists of a series of 500,000-gal underground stainless-steel tanks, each of which is in a reinforced concrete bunker with alarmed sumps. Several major underground spills occurred in the tank farm, primarily at stainless-steel to mild-steel connections. These spills resulted in extremely high levels of contaminated soil that were removed with remote equipment. These operations had high release potential, and were planned and executed with personal protective equipment and monitoring. The contaminated soil was kept wet when being handled to minimize airborne contamination. Both the *in vitro* and *in vivo* data are documented in the personnel dosimetry files [46].

5.7.2.3 High-Level Waste Calcination

High-level waste calcination operations, which began in 1963, consisted of the production of a high-temperature calcine by spraying high-level liquid waste into a fluidized bed of calcine. The nitric oxides were vented up the stack following high-efficiency filtering, which left the radionuclides high-fire coated on calcine granules. The potential for release of high-level MFP, MAP, and TRU particulate activity was significant, but it was recognized and monitored with an extensive array of air and radiation (or remote) area monitors (RAMs) [47]. Facility operators were placed on routine as well as special bioassay schedules [48].

5.7.2.4 Process Analytical Facilities

The Remote Analytical Facilities and lower-level Process Sample Analytical Laboratories analyzed samples critical to process controls. The potential for radioactive material release and internal exposure was significant in these facilities. Some internal exposure incidents occurred in the laboratories from loss of control of process samples in hoods and other causes. These incidents frequently involved loss of control of TRU materials during analytical procedures in hoods or benchtop confinement-type operations [49].

5.7.2.5 Spent Fuel Processing and INTEC Most-Limiting Radionuclides

Highly enriched fuel processing, which began in 1953, consisted of nitric acid dissolution of aluminum-clad elements, electrolytic nitric acid dissolution of stainless-steel-clad elements, hydrofluoric acid dissolution of zirconium-clad elements, and nitric acid dissolution of graphite fuel following graphite burning. Uranium was extracted from the dissolved elements (using variations of the plutonium-uranium extraction process, which produced a uranyl nitrate product). The first-, second-, and third-cycle raffinates contained high levels of MAPs and MFPs along with TRU radionuclides. These products, in highly corrosive matrices, were difficult to contain in the confinement barriers of the process piping, process off-gas, etc., and they resulted in routine leaks and spills. Most of the leaks were confined to process cells, but they occasionally occurred in occupied process control and equipment areas [50]. For this reason, the extensive network of CAMs and RAMs was essential for timely detection of loss of confinement.

Source terms for the various INTEC operations are based on the types of fuel that were reprocessed. All of the reprocessed fuels were highly enriched with ^{235}U enrichments ranging from 50% to 93%. Most of the INTEC reprocessing involved aluminum-clad fuels from test reactors, stainless-steel clad fuels from the Experimental Breeder Reactor II (EBR-II), and zirconium-clad fuels from various reactors. There were other minor campaigns for processing minor amounts of fuel such as graphite fuel from the ROVER (space nuclear propulsion) program. In addition, some fuels that had no burnup or very little burnup were reprocessed in a hands-on operation known as Custom Processing.

To minimize the radiological safety hazard of the relatively volatile halogens, the fuel was normally decayed a minimum of 120 d before shipment to INTEC for processing. Processing of fuels often did not occur until years later. 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 [51]. To give an indication of the radionuclide inventory in fuels that were processed at INTEC, the ORIGEN2 software (Croff 1980) was used to determine the decay of previously calculated fuel inventories (Wenzel 2000) to typical decay times before processing. Aluminum fuel based on fuel from the Advanced Test Reactor (ATR) was decayed 1 yr. Stainless-steel fuel based on fuel from EBR-II was decayed 3 yr. Zirconium fuel based on various reactors was decayed 5 yr. These decay times represented the minimum decay times that occurred before processing of the fuels. Table 5-18 lists the radiologically significant nuclides and the percent of the inhalation dose from inhaling activity from these three types of fuel. Several short-lived progeny (^{90}Y , ^{106}Rh , $^{137\text{m}}\text{Ba}$, and ^{144}Pr) have over one-third of the activity, which will be followed in IMBA calculations and are not shown. The inhalation dose percentages were calculated using ICRP Publication 68 dose conversion factors for AMADs of $5\ \mu\text{m}$ (ICRP 1995).

Aluminum fuels were processed between 1953 and 1986, stainless-steel fuels primarily between 1977 and 1981, and zirconium fuels between 1972 and 1988 (Staiger 2003). There are relatively long half-life fission products that persist for INTEC source terms. In most cases, the source terms were well tagged with beta-emitting radionuclides, which allowed beta/gamma-detecting CAMs to be used at INTEC with the realization that they would also warn of possible alpha contamination or internal exposures [52].

Table 5-18 contains too many radionuclides for efficient dose reconstruction. Rather than include all of the radionuclides in the default summary table for missed dose (Table 5-24 later in this document), only ^{90}Sr , ^{137}Cs , ^{144}Ce , and ^{238}Pu are included for aluminum and zirconium fuels. For stainless-steel fuels, the ^{238}Pu is replaced by ^{239}Pu [53]. Cesium-137 was selected because it is most commonly reported in the *in vivo* results rather than for its dose contribution. For the pre-1960 period, ^{91}Y and ^{95}Zr are included, but after that the fuel being processed had decayed sufficiently so that these 58- and 64-d half-life radionuclides would have decayed away.

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 Materials Test Reactor (MTR) with as little decay as manageable (less than 2 d). Fuel was removed from the MTR, transported about 2 mi to INTEC in a heavily shielded transport container by a straddle carrier, immediately dissolved, and the barium element was extracted. The $^{140}\text{Ba}/\text{La}$ product was shipped immediately to Los Alamos National Laboratory. This process released large quantities of volatile radioactive iodines, which have a higher potential for escaping confinement systems. Several significant internal exposure incidents occurred in which ^{131}I , ^{132}I , and ^{133}I thyroid intakes occurred before personnel could respond to CAM alarms and take protective or corrective actions [54].

5.7.3 Argonne National Laboratory–West

ANL-W, based at INL since 1951, continues to conduct nuclear programs. Although ANL-W receives internal dosimetry support from the INL service laboratories, its radiological safety programs operate under the DOE COO. However, over the 52 yr of ANL-W experience, informal program coordination has occurred to foster technical consistency with the other INL facility internal dosimetry programs.

Nine experimental reactors under the technical direction of ANL-W were operated at two INL locations, one on the southwest side of the site near RWMC and the others at the current location on the southeast side of the site. Early reactor operations included physics critical experiments; power production; routine unmoderated operation, uranium-fueled, plutonium-fueled, and breeder reactor designs; and self-destruct experiments. As listed in Table 5-19, the radionuclides of concern ranged the spectrum of MFP, MAP, uranium, and TRU nuclides.

Table 5-19. Radionuclides of concern and MDAs for ANL-W locations (Nielsen 1996).

Radionuclide	Absorption types	Sources/characteristics ^a	MDAs		
			Urine (pCi/mL)	<i>In vivo</i> ^b , 5-min count (nCi)	<i>In vivo</i> ^c , 10-min count (nCi)
H-3 (HTO)	F	EBR-II Reactor Facility & Sodium Components Maintenance Shop	2.0	N/A ^d	N/A
Mn-54	F, M	EBR-II primary source. Levels low and decreasing	N/A	2.6	1.3
Fe-59			N/A	4.5	2.4
Co-58			N/A	2.5	1.3
Co-60			N/A	2.5	1.4
Sr/Y-90	F	All facilities handling fission products	0.02 ^e	N/A	N/A
Cs-134	F	All facilities handling fission products	N/A	2.7	1.3
Cs-137			N/A	3.0	1.5
U-235	F, M, S	Hot cell, hoods, glove boxes, waste, reactor fuel, research areas	0.02 ^e	0.1	N/A
U-238			0.02 ^e	1.6	N/A
Pu-238	M, S	FCF, HFEF, ZPPR, Analytical Laboratory	0.02 ^e	60.0	N/A
Pu-239			0.02 ^e	161.0	N/A
Am-241	M	FCF, HFEF, ZPPR, Analytical Laboratory	N/A	0.1	N/A

- See the acronyms and abbreviations list.
- Using a Canberra WBC with a 5-min count.
- Using a large NaI WBC detector with a 10-min count.
- N/A = not applicable.
- Assumes a 100-mL sample.

5.7.4 Radioactive Waste Management Complex

The RWMC has supported INL operations as a waste management complex since 1952 and has received large quantities of TRU waste from Rocky Flats 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 the Subsurface Disposal Area (SDA) for permanent disposal of low-level waste and some early TRU waste (which will eventually be exhumed and repackaged); the 58-acre Transuranic Storage Area (TSA) for temporary storage, examination, and certification before shipment to the Waste Isolation Pilot Plant; the operations area; and 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 (there were two in 1992 and one in 1996) [55]. Therefore, bioassay is conducted randomly at the current time (INEEL 2001).

Tables 5-20 and 5-21 summarize the major radionuclides in the RWMC waste inventory. TRU radionuclides are the primary contaminants in the TSA waste; all but ²⁴¹Pu 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.

Because the waste has all chemical and physical characteristics by nature, the absorption type most favorable to claimants should be assumed unless data indicate that another type fits the data better. Exposures in later years are most likely from contaminated soil or corrosion products, and a default particle size distribution of 5- μ m AMAD should be assumed as recommended by ICRP Publication 68 (ICRP 1995).

Table 5-20. 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

Table 5-21. 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
			Ni-63	3.3E-1	7.4
			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

5.7.5 Waste Reduction Operations Complex

The Waste Reduction Operations Complex 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 INL 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.

Because the types of radioactive materials processed at WERF varied depending on the area shipping the waste to WERF, the assumption favorable to claimants is that radioactive materials came from zirconium fuel as processed at INTEC (see Table 5-18, columns 7 and 8).

5.7.6 Test Reactor Area

5.7.6.1 Test Reactor Area Reactors

The MTR was the second operating reactor at INL and ran from March 1952 to 1970 in the Test Reactor Area (TRA) (Stacy 2000); 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 TRA complex includes hot cells, a gamma irradiation pool facility, research laboratories, and analytical laboratories. The reactors at TRA, as well as the others at the INL, 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 DOE reactors.

The uranium in the TRA reactors is enriched to 93% ^{235}U , and the fuel is clad in aluminum. The predominant activation product in the cladding is ^{24}Na , which is formed by activation of sodium in the aluminum. Sodium-24 has a half-life of 15 hr and emits a high-energy gamma ray (2.75 MeV). The inhalation dose to personnel from ^{24}Na is insignificant in comparison to that from the fission products in the fuel [56]. There are minor levels of activation products of stainless steel (^{58}Co , ^{60}Co , ^{51}Cr , ^{56}Mn , etc.) in the primary coolant system (PCS) due to corrosion of the stainless-steel PCS components [57]. To be favorable to claimants, this TBD recommends direct use of any *in vivo* counting data in the worker files. If applicable data are absent, the *in vivo* MDLs can be assumed consistent with information in Tables 5-14 and 5-24 (see Section 5.8).

Several factors contributed to unusual amounts of fission products in the PCSs 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 [58]. Another factor was *tramp* fuel, which was a contaminant on the outside of the cladding. During reactor operation, fission products in tramp uranium were released directly to the PCS [59]. Reactor operators and the fuel manufacturer resolved these deficiencies over time; by the time the ATR became operational, the PCS of the ATR was considerably less contaminated than that of the MTR or the ETR during their early years of operation. The radiologically significant radionuclides for aluminum-clad fuel processed at INTEC in Table 5-18 are applicable to this PCS contamination [60].

The majority of radioactivity releases from the TRA reactors to areas potentially occupied by workers consisted of noble gases that promptly decayed to short-lived particulates. The principal dose to personnel from releases of noble gases was direct radiation rather than inhalation [61]. The direct radiation caused instruments to alarm, which resulted in immediate evacuation of the affected areas. The position favorable to claimants is that there could have been some halogens and particulate radionuclides released along with the noble gases. The ORIGEN2 code (Croff 1980) was used to calculate a radionuclide inventory for ATR fuel at the time of reactor shutdown. The radionuclide inventory was then fractionated according to the release percentages often used to assess hypothetical reactor accidents (DiNunno et al. 1962) as 100% noble gases, 50% halogens, and 1% solids. A halogen:solid ratio of 50:1 was therefore used to determine the radionuclides that could significantly contribute to inhalation dose. Table 5-22 contains the relative amounts of halogens and particulate radionuclides from ATR fuel that contribute significantly to inhalation dose, but there are too many for efficient dose reconstruction. The iodine isotopes contribute the large majority (90.3%) of the dose from a test reactor gaseous release. Therefore, ^{131}I was selected to be the representative

radionuclide for missed dose (in Table 5-24), and the dose from ^{131}I was weighted by a factor of 1.6 to account for the total iodine dose [62]. When there is measured ^{131}I for any facility, the dose reconstructor can choose to apply this factor of 1.6 to the ^{131}I dose to account for short half-life iodines and can choose to ratio ^{144}Ce , ^{89}Sr , ^{91}Y , and ^{95}Zr activities to the ^{131}I activity using the values in Table 5-22.

Another potential source of inhalation dose was the pressurized-loop experiments in the TRA reactors. These experiments sometimes contained silver and tantalum, which became activated and produced $^{110\text{m}}\text{Ag}$ and ^{182}Ta . When a depressurizing incident occurred in the loop, these activation products were sometimes released to work areas (Nertney et al. 1967). One of the major airborne incidents at TRA listed in Table 5-23 involved the release of ^{182}Ta . The position favorable to claimants is to assign a missed dose based on the MDA for *in vivo* counting (see Table 5-14) for $^{110\text{m}}\text{Ag}$ and ^{182}Ta .

Table 5-22. Radiologically significant radionuclides for ATR fuel gaseous releases [63].

Nuclide	Half- life	Absorption type	Fraction activity	Percent inhalation dose	Ratio activity to I-131 activity
Sr-89	50.5 d	S	0.0028	1.0	3.5E-02
Y-91	58.51 d	M/S	0.0033	1.3	4.1E-02
Zr-95	63.98 d	S	0.0036	0.9	4.4E-02
I-131	8.041 d	F	0.0816	55.7	Not applicable
I-132	2.3 h	F	0.1298	1.6	Not applicable
I-133	20.8 h	F	0.2040	26.6	Not applicable
I-134	0.876 h	F	0.2225	1.1	Not applicable
I-135	6.61 h	F	0.1855	5.3	Not applicable
Ce-144	284.3 d	S	0.0012	1.7	1.5E-02
Total			0.834	95.2	Not applicable
Dose percentage from all iodine isotopes				90.3	
Weighting factor for I-131 to account for iodine dose				1.6	

Table 5-23. Major airborne incidents at TRA.

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

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 radiologically significant radionuclides listed in Table 5-18 for aluminum fuel processed at INTEC are applicable to this TRA contamination.

5.7.6.2 Test Reactor Area Laboratories

The wing buildings of the MTR at TRA house chemistry and other laboratories. Over the years, experiments in the laboratories have resulted in contamination and airborne activity incidents. These experiments have involved various isotopes of plutonium and uranium along with other radionuclides.

The radiologically significant radionuclides listed in Table 5-18 for aluminum-clad fuel processed at INTEC are applicable for the experiments in the TRA laboratories.

Beginning in 1980, many of the laboratories were involved in various studies of TMI reactor fuel from the accident in March 1979. Because the TMI core had only the equivalent of 100 effective full-power d at the time of the accident, the TRU content would not be as high as that for a typical power reactor core, which is routinely operated for a year or so before the fuel is changed. Use of the radiologically significant nuclides for zirconium fuel processed at INTEC in Table 5-18 is favorable to claimants for evaluating inhalation dose from exposure to contamination from the TMI fuel.

5.8 DEFAULT FOR MISSED DOSE

Based on the INL characteristics and circumstances, a number of missed dose default assumptions have been derived. Table 5-24 is a summary of these recommended defaults.

Table 5-24. Default table for missed dose.

Period	Based on	Site area	Recommendation	Basis
Startup–1960	Urine gross β	All	Calculate chronic Sr-90 intake that results in a urine activity of $0.4 \times$ gross β .	Typical β activity is 0.33 Sr-90, 0.33 Y-90 & 0.33 Cs-137. Use of 0.4 is favorable to claimants.
			Cs-137 intake = Sr-90 intake	Half-lives and fission yields of Cs-137 & Sr-90 are approximately equal.
			Pu-238 intake = $0.005 \times$ Sr-90 intake	Pu:Sr-90 ratio of 0.005 in Al-clad fuel.
			Ce-144 intake = $12.4 \times$ Sr-90 intake	Ce-144:Sr-90 ratio of 12.4 in Al-clad fuel.
			Y-91 intake = $1.1 \times$ Sr-90 intake	Y-91:Sr-90 ratio of 1.1 in Al-clad fuel.
		Zr-95 intake = $1.2 \times$ Sr-90 intake	Zr-clad-95:Sr-90 ratio of 1.2 in Al-clad fuel.	
		TRA ^a	Annual I-131 intake = 25 μ Ci	Assumes I-131 in thyroid at 3 d is 5 μ Ci Vapor Type F, the maximum <i>in vivo</i> measured in 1963.
1961–1970	<i>In vivo</i> Cs-137	All	Calculate chronic Cs-137 intake that results in the <i>in vivo</i> measurement.	
			Sr-90 intake = Cs-137 intake when no <i>in vitro</i> measurement.	Half-lives and fission yields are approximately equal.
			Pu-238 intake = $0.005 \times$ Cs-137 intake when no <i>in vitro</i> measurement.	Pu:Cs-137 ratio of 0.005 in Al-clad fuel.
			Ce-144 intake = $12.4 \times$ Cs-137 intake when no measurement.	Ce-144:Sr-90 ratio of 12.4 in Al-clad fuel.
		TRA	Annual I-131 intake = 60 nCi when no measurement.	Based on MDA for I-131 of 12 nCi at 3 d in thyroid Vapor Type F.
			One acute intake of Ag-110m = 71 nCi when no measurement.	Based on MDA for Ag-110m of 12 nCi at 3 d type M. Known incident involving Ag-110m.
One acute intake of Ta-182 = 80 nCi when no measurement.	Based on MDA for Ta-182 of 12 nCi at 3 d type M. Known incident involving Ta-182.			

Period	Based on	Site area	Recommendation	Basis
1971–1980	<i>In vivo</i> Cs-137	All	Sr-90 intake = Cs-137 intake when no <i>in vitro</i> measurement.	Half-lives and fission yields of Cs-137 & Sr-90 are approximately equal.
		All but ANL-W	Pu-238 intake = $0.02 \times$ Cs-137 intake when no <i>in vitro</i> measurement	Pu:Sr-90 ratio of 0.02 in Zr-clad fuel. See Table 5-18, column 9.
			Ce-144 intake = $1.05 \times$ Cs-137 intake when no measurement	Ce-144:Sr-90 ratio of 1.05 in Zr-clad fuel.
		ANL-W	Pu-239 intake = $0.003 \times$ Cs-137 intake when no <i>in vitro</i> measurement	Pu:Cs-137 ratio of 0.003 in stainless-steel-clad fuel.
			Ce-144 intake = $2 \times$ Cs-137 assigned intake when no measurement	Ce-144:Cs-137 ratio of 2 in stainless-steel-clad fuel.
		TRA	Annual I-131 intake = 94 nCi when no measurement.	Based on MDA for I-131 of 12 nCi at 3 d thyroid Vapor Type F weighted by a factor of 1.6.
1981–present	Bioassay	All except ANL-W, INTEC ^b , & SMC	Calculate associated radionuclide intakes based on Al-clad fuel distribution	Distribution calculated for Al-clad fuel. See Table 5-18, column 3.
		ANL-W	Calculate associated radionuclide intakes based on stainless-steel-clad fuel distribution	Distribution calculated for stainless steel fuel. See Table 5-18, column 4.
		INTEC ^b & unknown ^c	Calculate associated radionuclide intakes based on Zr-clad fuel distribution	Distribution calculated for Zr-clad fuel. See Table 5-18, column 5.
		SMC	Calculate associated intakes based on known isotope distribution	Isotope distribution, see Table 5-16.

- a. The dose recommendations for specific areas (TRA, INTEC, ANL-W) are to be added to recommendations for “All”.
- b. Formerly known as ICPP.
- c. When the worker’s area is not known, or if the employee worked in many different areas, use the dose recommendations for INTEC.

For most of the history of the INL, personnel dosimeters were issued to all workers, regardless of work assignment, who entered the security access control points at each facility. For example, administrative and clerical personnel were required to wear dosimeters even though they were not exposed to elevated backgrounds or internal dose potential. Dose reconstructors should determine the appropriate default categories from Table 5-24 as follows:

- If the worker’s file includes positive external dosimeter readings, the worker should be treated as a radiation worker and the default internal missed dose should be applied as outlined in the table.
- If no detectable external or internal dose information is recorded, only the environmental dose should be included. For example, a worker at TRA in 1975 should be assigned the intakes for “All,” for “All but ANL-W,” and for “TRA” [64].

5.9 UNMONITORED WORKERS

As noted above, INL personnel dosimeters were issued to all workers at facilities that handled radioactive material. Many of these workers, due to the nature of their work, would not have been exposed to internal activity and would not have been subject to routine bioassay [65].

The INL radiation protection program to detect the presence and spread of radioactive contamination included:

- Areas with potential airborne contamination were monitored with alarming CAMs.
- The spread of contamination was monitored through the use of smears and monitoring instrumentation.

INL policy was to screen radiation workers using urine samples and WBC for internal contamination many (often four or more) times a year.

Most of the encountered activity was well tagged with beta/gamma activity, which would have produced measurable direct radiation doses on a personnel dosimeter [66]. Therefore, the probability that a worker received a significant unmonitored internal intake of radioactive material is very low. It is recommended that individuals who were not issued a personnel dosimeter and have no record of internal dose monitoring be assigned only the environmental dose for the facility.

INL often used construction workers. Each construction job was evaluated to determine if radiation exposure or internal dose could be received [67]. When construction work occurred in an area with potential radiation exposure, including internal dose exposure, construction workers were monitored in the same fashion as radiation workers. Construction workers who were issued personnel dosimeters should be treated the same as facility employees who were issued personnel dosimeters. Construction workers who were not issued a personnel dosimeter should be assigned the environmental dose for the facility.

5.10 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, 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 document, 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 INL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner, who is fully responsible for the content of this document, including all findings and conclusions. Mr. Rohrig continues to serve as a Site Expert for this document because he possesses or is aware of information relevant for reconstructing radiation doses experienced by claimants who worked at the site. In all cases where such information or prior studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source.

Bryce Rich served as one of the initial Subject Experts for this document. Mr. Rich was previously employed at INL 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 INL 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 INL 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 INL 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. This statement by Mr. Rich is based on a review of the referenced AEC annual reports.
- [3] 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 the HSD, allowed for a consistency in assumptions about the internal dosimetry program because RESL was a constant even though there was a large number of varying facilities and continually changing contractors at INL. This is supported by the referenced AEC annual reports.
- [4] Rich, Bryce, and Jenkins, JoAnn M. Consulting Health Physicist and Senior Health Physicist. M. H. Chew & Associates and Dade Moeller & Associates. November 2004. To assign an internal dose from a radionuclide intake, the specific radionuclide must be known. In some cases worker bioassay records only provide results for overall beta- and alpha-emitting radionuclides and do not identify the specific radionuclides. In these cases, it is

necessary to provide guidance to the dose reconstructor on what the radionuclides that are represented by the gross beta and alpha analyses are likely to be so that internal doses can be assigned. These default assumptions are based on reviews of bioassay and facility data by Mr. Rich.

- [5] 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 INL incident reports. It is common industry practice to conduct incident investigations in a timely manner.
- [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 knowledge of the NRF from working at INL. 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.
- [8] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement is based on a review of the referenced AEC annual reports and employee bioassay and dosimetry data.
- [9] 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).
- [10] 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).
- [11] 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).
- [12] 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).
- [13] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on a review of the INL 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).
- [14] 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).

- [15] 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 INL internal dosimetry program and is supported by Puphal (1994).
- [16] 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 INL 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.
- [17] 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 INL internal dosimetry program and is supported by the referenced internal dosimetry TBDs.
- [18] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This conclusion is based on the fact that confirmatory analyses that identified specific radionuclides were generally performed on bioassay samples whose initial results were in excess of 2σ . It can be surmised that if additional analyses were not performed on a sample, then the initial gross alpha and gross beta results were within 2σ . This conclusion is supported by facility bioassay data and Puphal (1994).
- [19] 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).
- [20] JoAnn M. Jenkins, Senior Health Physicist, Dade Moeller & Associates. March 2007. Federal regulations provide reporting requirements for internal dose. Internal dose is typically calculated based on bioassay monitoring results. Changes in federal regulations about the threshold at which dose is required to be reported and the annual dose limits affect the level at which calculated dose is reported; they do not generally affect the bioassay methodology and analytical programs for collection of the raw data for dose calculations.
- [21] 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.
- [22] JoAnn M. Jenkins, Senior Health Physicist, Dade Moeller & Associates. March 2007. Federal regulations provide reporting requirements for internal dose. Internal dose is typically calculated based on bioassay monitoring results. Changes in federal regulations about the threshold at which dose is required to be reported and the annual dose limits affect the level at which calculated dose is reported, they do not generally affect the bioassay methodology and

analytical programs for collection the raw data for dose calculations or the facility air monitoring programs.

- [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 the referenced AEC annual reports and internal dosimetry TBDs.
- [24] JoAnn M. Jenkins, Senior Health Physicist, Dade Moeller & Associates. March 2007. The DOE set guidelines for positive bioassay results that require follow-up and reporting. These guidelines changed over time, but the changes in guidelines did not affect bioassay sampling frequency and the recording of results because the changes affected reporting requirements and not the collection of raw data for calculation of internal doses. The number of bioassay samples that required follow-up analyses varied over time as a result of the changing regulatory guidance about the need for follow-up sampling.
- [25] 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 INL 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.
- [26] 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 INL internal dosimetry program and is supported by the referenced AEC annual reports and internal dosimetry TBDs.
- [27] 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.
- [28] 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.
- [29] 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.
- [30] 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.
- [31] 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.

- [32] 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 INL bioassay records.
- [33] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on information from a review of the referenced AEC annual reports.
- [34] 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 INL bioassay records.
- [35] 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.
- [36] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on a review of the referenced internal dosimetry TBDs.
- [37] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on his knowledge of the INL and work experience at the facility and is supported by Stacy (2000).
- [38] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on information in Stacy (2000).
- [39] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. The information in the table is from King (2001).
- [40] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement is based on a review of INEEL (2001).
- [41] 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.
- [42] 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 INL internal dosimetry program. Although the maintenance operations that required entry into the process cells were planned extensively before execution and the cells were decontaminated, the potential for internal exposures was still high because not all of the contaminants could be cleaned out of the cells.
- [43] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. The assumption of 1,000 hr/yr working in the pool area is based on the fact that individuals worked in various areas throughout the facility and not at a single location.

- [44] Rohrig, Norman. Consulting Health Physicist. Intrepid Technology and Resources. November 2004.
These numbers are the product of the relative activity and the total activity (3×10^5 pCi/yr) to obtain the activity of each radionuclide.
- [45] Wenzel, Douglas, and Rohrig, Norman. Principal Health Physicist and Consulting Health Physicist. Intrepid Technology and Resources. November 2004.
This table was compiled by Mr. Wenzel from his knowledge of the fuel history he obtained from his work experience at the INL and of ORIGEN calculations. Percent inhalation dose calculations were performed by Mr. Rohrig using the ORIGEN output and ICRP Publication 68 dose conversion factors (ICRP 1995).
- [46] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on a review of INL bioassay records and the previously mentioned bioassay database.
- [47] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on information from a review of the referenced internal dosimetry TBDs.
- [48] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on information from a review of the referenced internal dosimetry TBDs.
- [49] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement by Mr. Rich is based on information from a review of the referenced internal dosimetry TBDs.
- [50] Levitt, Boyd. Senior Health Physicist. Intrepid Technology and Resources. November 2004.
An example is a leak in the process control corridor of Building 601 that resulted in radiation levels of over 100 rad/hr.
- [51] 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.
- [52] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
The fuel at INTEC 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.
- [53] Wenzel, Douglas. Principal Health Physicist. Intrepid Technology and Resources. November 2004.
Table 5-18 identifies the activity of both plutonium isotopes. The importance of the different plutonium isotopes varies with the type of fuel. These radionuclides were selected by Mr. Wenzel from information from his work experience at the INL and from an extensive review of the referenced AEC annual reports and internal dosimetry TBDs. To simplify missed dose calculations, the following four representative radionuclides were selected for the stated reasons: ^{90}Sr based on the availability of radioanalytical techniques, ^{137}Cs based on

confidence that it was easily detectable, ^{144}Ce due to its large dose contribution, and ^{238}Pu to represent the TRU elements. In the case of stainless-steel fuel, ^{239}Pu should be substituted for ^{238}Pu because it is more prevalent. Together these radionuclides make up 87.8% of the committed effective inhalation dose. Percent committed effective inhalation dose values were taken from Table 5-18. This table was compiled by Mr. Wenzel from his knowledge of both the fuel history he obtained from his work experience at the INL and ORIGEN calculations. Percent committed effective inhalation dose calculations were performed by Mr. Rohrig using the ORIGEN output and ICRP Publication 68 dose conversion factors (ICRP 1995).

- [54] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement by Mr. Rich is based on Hayden (1958).
- [55] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. See Table 5-1.
- [56] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. This statement is based on the fact that the ^{24}Na is not liberated from the aluminum cladding and that ^{24}Na has a short half-life.
- [57] 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.
- [58] 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.
- [59] 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.
- [60] 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.
- [61] 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.
- [62] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004. The dose from ^{131}I makes up 55.7% of the total dose. It is the largest contributor to dose in the iodine family. The total committed effective dose contribution due to iodines is 90.3 percent. Applying a factor of 1.6 to the ^{131}I dose accounts for the dose from the additional iodine isotopes. Doses were taken from Table 5-22. This table was compiled by Mr. Wenzel from his knowledge of the fuel history from his work experience at INL and of ORIGEN calculations. Ratio of activity to ^{131}I activity was performed by Mr. Rohrig using the ORIGEN output.

- [63] Wenzel, Douglas, and Rohrig, Norman. Principal Health Physicist and Consulting Health Physicist. Intrepid Technology and Resources. November 2004.
This table was compiled by Mr. Wenzel from his knowledge of the fuel history from his work experience at INL and of ORIGEN calculations. Ratio of activity to ^{131}I activity was performed by Mr. Rohrig using the ORIGEN output.
- [64] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement is an example of the use of Table 5-24.
- [65] 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 INL internal dosimetry program.
- [66] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement is based on a review of the referenced internal dosimetry TBDs and AEC annual reports.
- [67] Rich, Bryce. Consulting Health Physicist. M. H. Chew & Associates. November 2004.
This statement is based on a review of the referenced internal dosimetry TBDs.

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GLOSSARY

absorption

In relation to health physics, process in which radiation energy is imparted to and radionuclides are transported to bodily fluids, tissues, and organs.

absorption type

Categories for materials according to their speed of absorption in the body, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F for fast absorption (formerly inhalation class D), M for moderate absorption (formerly inhalation class W), and S for slow absorption (formerly inhalation class Y). Also called solubility type.

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 (3.7×10^{10}) becquerels, which is approximately equal to the activity of 1 gram of pure ^{226}Ra .

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. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain DU. DU can be blended with highly enriched uranium to make reactor fuel or used as a raw material to produce plutonium.

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

Of or relating to a process that takes place under artificial conditions or outside a living organism (e.g., in the laboratory). From Latin meaning *in glass*.

in vivo

Of or relating to a process that takes place in a living organism. From Latin meaning *in life*.

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.

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., ^{234}U , ^{235}U , and ^{238}U). Isotopes have very nearly the same chemical properties but often have different physical properties.

milli-

Prefix that divides a unit by 1,000 (multiplies by 1×10^{-3}).

micro-

Prefix that divides a unit by 1 million (multiplies by 1×10^{-6}).

mixed waste

Unwanted material containing both radioactive and hazardous components.

natural uranium (U, U-nat, NU)

Uranium as found in nature, approximately 99.27% ^{238}U , 0.72% ^{235}U , and 0.0054% ^{234}U by weight. The specific activity of this mixture is 2.6×10^7 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

Disintegration of certain elements (e.g., radium, actinium, uranium, and thorium) accompanied by the emission of alpha, beta, gamma, and/or neutron radiation from unstable nuclei. See *radionuclide*.

radionuclide

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

radioactive lanthanum (RaLa)

(1) Isotope ^{140}La (a fission product). (2) A lanthanum recovery process at INTEC for development of weapons.

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 spent 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 nuclear fuel

Fuel that has been in a reactor long enough to become ineffective because the proportion of fissile material has dropped below a certain level. Spent nuclear fuel contains fission and activation products.

spent fuel storage basin

Pool or pit of reinforced concrete filled with water for storage of spent 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

Penetrating electromagnetic radiation (photons) of short wavelength (0.001 to 10 nanometers) and energy less than 250 kiloelectron-volts. X-rays usually come from excitation of the electron field around certain nuclei. Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside 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.