DOE Review Release 08/25/2025

Idaho National Laboratory and Argonne National Laboratory-West – Occupational Internal Dose

ORAUT-TKBS-0007-5 Rev. 04
Effective Date: 08/18/2025
Supersedes: Revision 03

Subject Expert(s): Brian P. Gleckler

Document Owner Approval:	Signature on File Brian P. Gleckler, Document Owner	Approval Date:	08/18/2025
Concurrence:	Signature on File Wade C. Morris, Objective 1 Manager	Concurrence Date:	08/18/2025
Concurrence:	Signature on File Scott R. Siebert, Objective 3 Manager	Concurrence Date:	08/18/2025
Concurrence:	Signature on File Jennifer L. Hoff, Project Director	Concurrence Date:	08/18/2025
Approval:	Signature on File Brant Ulsh, Acting Associate Director of Science	Approval Date:	08/18/2025

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

☐ New ☒ Total Rewrite ☐ Revision

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
08/11/2004	00	New Technical Basis Document for the Idaho National Engineering and Environmental Laboratory – Occupational Internal Dose. First approved issue. Initiated by Norman D. Rohrig.
10/12/2004	00 PC-1	Approved page change revision. Deletes references to rigorous radiation program on page 39. Initiated by Norman D. Rohrig.
01/13/2006	01	Approved issue of Revision 01. Revised to change table number format, responds to new information from Worker Outreach Effort and revises MDA tables. Incorporates 1-year decay time for Al fuel in Tables 5-18 and 5-24. Combined “INTEC & Unknown” and “Other Area” categories in Table 5-24 for 1971-80 into “All but ANL-W.” Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig.
06/05/2006	01 PC-1	Page change revision to correct error on page 24 in Table 5-12 in Section 5.5 on sample volume in 1980s. Approved issue of Rev 01 PC-1. No sections were deleted. Incorporates NIOSH formal review comments. This revision results in a reduction in assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig.
07/30/2007	02	Revision initiated to add Purpose, Scope, and Attributions and Annotations sections. Added references. Updated the formatting and made editorial changes. Corrected error in Fecal MDA for Th-230 and Np-237 in Table 5-13. Constitutes a total rewrite of the document. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Task Manager. Initiated by Jo Ann M. Jenkins.
03/02/2010	03	Revision to incorporate the Argonne National Laboratory – West site. In order to resolve several issues identified in an independent audit of this TBD and other issues identified by ORAUT, the approaches for assessing the INEL bioassay data have been completely revised. The potential lung absorption types to be considered have also been expanded for several radionuclides. These changes generally result in lower organ doses. However, for specific scenarios they can result in higher doses to certain metabolic organs such as the thyroid. The layout of this TBD was completely reorganized to better match the layout recommended in ORAUT-PROC-0031. In addition, this revision fully incorporates the Argonne National Laboratory – West back into this TBD and eliminates the ANL-W Occupational Internal Dose TBD. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jo Ann M. Jenkins.

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
08/18/2025	04	Focused revision to incorporate SEC-00219 and SEC-00224 and to incorporate new language for the SEC-00238 discussion in Section 5.1.3. The new language implements INL and ANL-W SECs. Focused revisions made to Sections 5.1.7.3, 5.5.2, and 5.6. Editorial changes were made throughout the document and tables reformatted for Section 508 compliance. Reviewer and editing comments outside the NIOSH-directed scope of incorporating new language for the SEC discussion were archived for a later revision. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Wade C. Morris and authored by Brian P. Gleckler.

TABLE OF CONTENTS

LIST OF TABLES

ACRONYMS AND ABBREVIATIONS

ACC	American Cyanamid Company
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
ARA	Army (later Auxiliary) Reactor Area
AREA	Army Administration and Hot Cell
ARHC	AREA Hot Cell
ATR	Advanced Test Reactor
AWE	Atomic Weapons Employer
B&W	Babcock & Wilcox Company
BBI	Bechtel BWXT Idaho
CADRE	military personnel at the ARA
CAM	continuous air monitor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CFA	Central Facilities Area
C.F.R.	*Code of Federal Regulations*
Ci	curie
cpm	counts per minute
CPP	Chemical Processing Plant
d	day
DAC	derived air concentration
DCAS	Division of Compensation Analysis and Support
DHHS	U.S. Department of Health and Human Services
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DU	depleted uranium
EBR	Experimental Breeder Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ERDA	U.S. Energy Research and Development Administration
EROB	Engineering Research Office Building
ETR	Engineering Test Reactor
F	fast (absorption type)
ft	foot
g	gram
gal	gallon
GCRE	Gas-Cooled Reactor Experiment
H&S	Health and Safety
HEU	highly enriched uranium
HPS	Health Physics Society
hr	hour

HSD	Health and Safety Division
HSL	Health Services Laboratory
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
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IRC	INL Research Center
KPA	kinetic phosphorescence analysis
L	liter
lb	pound
LPTF	Low Power Test Facility
m	meter
M	moderate (absorption type)
MDA	minimum detectable amount
MeV	megaelectron-volt, 1 million electron-volts
MFAPs	mixed fission and activation products
MFC	Materials and Fuels Complex
MFPs	mixed fission products
mi	mile
min	minute
mL	milliliter
ML-1	Mobile Low-Power Reactor No. 1
mo	month
mrem	millirem
mrep	millirep
MTR	Materials Test Reactor
NaI(Tl)	sodium iodide scintillator doped with thallium
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
NU	natural uranium
OMRE	Organic Moderated Reactor Experiment
ORAU	Oak Ridge Associated Universities
ORAUT	ORAU Team
PBF	Power Burst Facility
pCi	picocurie
PER	program evaluation report
PF	photofluorimetry
PPC	Phillips Petroleum Company

RaLa	radioactive lanthanum
RCIMS	Radiation Control Information Management System
RDR	Radiation Dosimetry and Records
RESL	Radiological and Environmental Sciences Laboratory
RFP	Rocky Flats Plant
RWMC	Radioactive Waste Management Complex
S	slow (absorption type)
SDA	Subsurface Disposal Area
SEC	Special Exposure Cohort
SL-1	Stationary Low-Power Reactor No. 1
SMC	Specific Manufacturing Capability
SPERT	Special Power Excursion Reactor Test
SRDB Ref ID	Site Research Database Reference Identification (number)
SS	super S (lung absorption type)
STEP	Safety Test Engineering Program
STPF	Shield Test Pool Facility
TAN	Test Area North
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor Experiment and Test
TRU	transuranic
TSA	Transuranic Storage Area
TSA/TSB	Technical Support Buildings A and B
U.S.C.	*United States Code*
WBC	whole body count
WCB	Willow Creek Building
WERF	Waste Experimental Reduction Facility
WROC	Waste Reduction Operations Complex
yr	year
α	alpha particle
β	beta particle
γ	gamma ray or photon
σ	standard deviation
μCi	microcurie
μg	microgram
μm	micrometer
§	section or sections

5.1 INTRODUCTION

Technical basis documents (TBDs) 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 historical background information and guidance to assist in the preparation of dose reconstructions at particular U.S. Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies), such as changing scientific understanding of operations, processes, or procedures involving radioactive materials. These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of individual dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA).

In this document the word “facility” is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an “AWE facility” or a “DOE facility.” The term “AWE facility” is defined in EEOICPA to mean “a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling.” 42 *United States Code* (U.S.C.) § 7384l(5). On the other hand, a DOE facility is defined as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located—(A) in which operations are, or have been, conducted by, or on behalf of, the [DOE] (except for buildings, structures, premises, grounds, or operations … pertaining to the Naval Nuclear Propulsion Program); and (B) with regard to which the [DOE] has or had—(i) a proprietary interest; or (ii) entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services.” 42 U.S.C. § 7384l(12). The DOE determines whether a site meets the statutory definition of an AWE facility and the U.S. Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Under EEOICPA, a Part B cancer claim for benefits must be based on an energy employee’s eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility’s designated time period and location (i.e., a “covered employee with cancer”). After DOL determines that a claim meets the eligibility requirements under Part B of EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. 42 U.S.C. § 7384l(11). Also under EEOICPA, the types of exposure to be included in dose reconstructions for DOE employees are those radiation exposures incurred in the performance of duty. As such, NIOSH includes all radiation exposures received as a condition of employment at DOE facilities in its dose reconstructions for covered employees, which may include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. This is because NIOSH does not determine the fraction of total measured radiation exposure at a DOE facility that is contributed by the Naval Nuclear Propulsion Program at the DOE facility during a specified period of time for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees [NIOSH 2010]:

- Background radiation, including radiation from naturally occurring radon present in conventional structures, and
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons.

5.1.1 Purpose

The purpose of this TBD is to document the internal dosimetry program and practices at the Idaho National Laboratory (INL) and Argonne National Laboratory-West (ANL-W), and to provide the technical basis to evaluate the internal occupational radiation dose for EEOICPA claims. These sites have been overseen by the U.S. Atomic Energy Commission (AEC), the U.S. Energy Research and Development Administration (ERDA), and DOE. In February 2005, ANL-W was merged into INL and subsequently renamed the Materials and Fuels Complex (MFC). For convenience, this TBD uses ANL-W because this was the name for most of the site's history and is the name most familiar to EEOICPA claimants.

5.1.2 Scope

This TBD provides supporting documentation to assist in evaluating occupational internal doses under OCAS-IG-001, *Internal Dose Reconstruction Implementation Guideline* [NIOSH 2002]. For nearly all scenarios and periods, NIOSH considers the available data and methods for performing internal dose reconstruction to be adequate for estimating with sufficient accuracy the internal doses at INL and ANL-W throughout their history. The known exceptions to this are discussed in Section 5.1.3.

As used in this TBD, the term "significant" means "practically significant" as opposed to "statistically significant." In other words, in this report "significant" means that the difference (effect size) is large enough to influence how a parameter is used or treated in practice.

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

5.1.3 Special Exposure Cohort

The Secretary of the U.S. Department of Health and Human Services (DHHS) has added several classes of INL and ANL-W employees to the SEC. The subsections below provide the details regarding those classes. Table 5-1 in Section 5.1.3.3 provides a summary of those SEC classes and the types of doses that cannot be completely reconstructed.

5.1.3.1 Idaho National Laboratory

For the period from January 1, 1963, through December 31, 1980, NIOSH has determined that the internal exposures cannot be adequately reconstructed for the exposures to alpha-emitting radionuclides separated from the mixed fission products (MFPs) that might have occurred at the INL Chemical Processing Plant (CPP) due to poor contamination controls and inadequate personnel monitoring [NIOSH 2017a, p. 251; NIOSH 2017b, p. 41]. That determination was made as part of the evaluations performed for petitions SEC-00219 and SEC-00238 [NIOSH 2017a; NIOSH 2017b].

The infeasibility for internal doses from alpha-emitting radionuclides separated from the MFPs is from January 1, 1963, through December 31, 1974, but was divided into two different periods with different employee class definitions based on NIOSH's recommendations. Those two periods were for January 1, 1963, through February 28, 1970, and March 1, 1970, through December 31, 1974 [NIOSH 2017a, p. 252]. Because NIOSH suspected that the infeasibility associated with petition SEC-00219 went beyond December 31, 1974, petition SEC-00238 (an 83.14 petition) was initiated by NIOSH to further evaluate the period beginning January 1, 1975 [NIOSH 2017a, p. 242]. Due to the evaluation performed for petition SEC-00238, NIOSH recommended that another class be added to the SEC for the period of January 1, 1975, through December 31, 1980 [NIOSH 2017b, p. 42].

As indicated above, the infeasible doses at INL were limited to the internal doses from exposures to alpha-emitting radionuclides that might have occurred at the CPP when they were separated from the MFPs. The areas with alpha-emitting radionuclides separated from the MFPs were limited to a few specific locations at CPP [NIOSH 2017b, p. 20]. In addition, only a relatively small fraction of the CPP workers would have entered the areas where alpha-emitting radionuclides separated from the MFPs were present due to the additional access controls for those areas. Those areas would have had additional access controls due to the presence of special nuclear materials (primarily HEU and plutonium) and the higher radiological hazard levels in those areas. In general, workers normally only enter areas with additional security and radiological protection access controls when necessary. They are not areas that workers randomly visit or wander through.

By policy, all INL workers and visitors entering areas where radioactive materials were handled were required to wear personnel metering instrumentation in the form of a dosimeter [Gammill 1955, p. 2; Cipperley 1958, pp. 29–31; American Cyanamid Company (ACC) 1952, p. 72; Phillips Petroleum Company (PPC) 1959, p. 129; Cipperley 1968, p. 12; Horan and Braun 1993, p. 21; Oak Ridge Associated Universities (ORAU) Team (ORAUT) 2011, p. 12]. At INL, radioactive materials were primarily handled within the major operating areas, which were large fenced-in areas with security access points [AEC 1954–1960, 1957–1975]. The dosimeters were also referred to as badges at INL because they had the wearers' security badge incorporated into them [PPC 1959, p. 129; Cipperley 1968, p. 15].

At the start of radiological operations, INL employed a "one badge, one area" practice for external dosimetry monitoring; when a worker entered a fenced operating area through the security entrance, a dosimeter specific for that operating area was provided, and the worker turned it in before exiting. In March 1970, a deviation from the "one badge, one area" practice was implemented in an attempt to reduce temporary badge use as well as overall program costs [NIOSH 2017a, pp. 156–157]. This is referred to as the "all-area badging" period, during which the external dosimetry records only identify the operating area the worker was primarily assigned to. For that period, there is no longer any information to indicate if and when a worker visited any other operating areas. In December 1974, the dosimetry badge system was returned to the "one area, one badge" practice until January 2000 [NIOSH 2017a, pp. 156–157].

Due to INL's external dose monitoring practices, a worker's external dosimetry records can be used to determine when a worker was at the CPP, except for the "all-area badging" period [NIOSH 2017a, pp. 245–246]. However, there was no method to reliably determine which CPP workers had accessed the areas where alpha-emitting radionuclides separated from the MFPs were present [NIOSH 2017b, p. 20]. Therefore, the employee class definitions could not be limited to a specific category or group of CPP workers and had to be initially expanded to include all CPP workers. As indicated above, an "all-area badging" practice was conducted between March 1970 and December 1974. The "all-area badging" practice primarily made it impossible to identify the workers that visited the CPP operating area. Because all workers that entered the CPP operating area could not be reliably identified during the "all-area badging" period, NIOSH proposed that the employee class definition for petition SEC-00219 be expanded to include all monitored INL workers for the period of March 1, 1970, through December 31, 1974 [NIOSH 2017a, pp. 245–246]. Because INL returned to the "one area, one badge" practice in December 1974, the employee class definition for the period of January 1, 1975, through December 31, 1980 could again be limited to CPP workers [NIOSH 2017a, p. 245; NIOSH 2017b, p. 20].

Based on NIOSH recommendations, the Secretary of the DHHS has added the following classes of INL employees to the SEC.

January 1, 1963, through February 28, 1970

All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Idaho National Laboratory (INL) in Scoville, Idaho, and who were monitored for external radiation at the Idaho Chemical Processing Plant (CPP) (e.g., at least one film badge or TLD [thermoluminescent dosimeter] *dosimeter from CPP) between January 1, 1963, and February 28, 1970, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort* [Azar 2019, p. 3].

March 1, 1970, through December 31, 1974

All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Idaho National Laboratory (INL) in Scoville, Idaho, and were monitored for external radiation at INL (e.g., having at least one film badge or TLD dosimeter) during the period from March 1, 1970, through December 31, 1974, and were employed for a number of work days aggregating at least 250 work days, occurring either solely under employment during the period from March 1, 1970, through December 31, 1974, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort [Burwell 2016a, p. 3].

January 1, 1975, through December 31, 1980

All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Idaho National Laboratory (INL) in Scoville, Idaho, and who were monitored for external radiation at the Idaho Chemical Processing Plant (CPP) (e.g., at least one film badge or TLD dosimeter from CPP) between January 1, 1975, and December 31, 1980, for a number of work days aggregating at least 250 work days, occurring solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort [Hargon 2017, p. 3].

Although NIOSH found that it is not possible to completely reconstruct radiation doses for these classes, NIOSH intends to use any internal and external monitoring data that might become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstruction for workers at CPP during the periods from January 1, 1963 to February 28, 1970 and January 1, 1975 to December 31, 1980, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate. Dose reconstructions for any workers at INL during the period from March 1, 1970 to December 31, 1974, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate [NIOSH 2017a, p. 38; 2017b, p. 238; Azar 2019, p. 5; Burwell 2016a, p. 4; Hargon 2017, p. 5].

5.1.3.2 Argonne National Laboratory-West

For the period from April 10, 1951, through December 31, 1957, with the exception of the occupational medical and environmental exposures, NIOSH has determined that the internal and external personnel exposures cannot be adequately reconstructed for the workers at ANL-W due to missing records [NIOSH 2016, pp. 209, 213–214]. That determination was made as part of the evaluation performed for petition SEC-00224 [NIOSH 2016]. All the radiation exposures during this period would have occurred at the Experimental Breeder Reactor-I (EBR-I) complex, because radiological work at the EBR-II complex did not begin until 1959 [NIOSH 2016, p. 55].

Based on NIOSH's recommendations, the Secretary of the DHHS has added the following class of ANL-W employees to the SEC.

April 10, 1951, through December 31, 1957

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Argonne National Laboratory-West during the time period from April 10, 1951, through December 31, 1957, for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort* [Burwell 2016b, p. 3].

Although NIOSH found that it is not possible to completely reconstruct radiation doses for the class, NIOSH intends to use any internal and external monitoring data that might become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures). Therefore, dose reconstructions for individuals employed at ANL-W during the period from April 10, 1951, through December 31, 1957, but who do not qualify for inclusion in the SEC, may be performed using these data as appropriate [NIOSH 2016, p. 217; Burwell 2016b, pp. 4–5].

5.1.3.3 Summary of SEC Class Designations

Table 5-1 provides a summary of the INL and ANL-W SEC class designations discussed above.

Table 5-1. Summary of INL and ANL-W SEC class designations.

Worker category	SEC period	Doses that *cannot* be completely reconstructed	Source
All ANL-W	04/10/1951–12/31/1957	External and internal doses, except for occupational medical and environmental doses	Based on information in NIOSH [2016, p. 192, 214, 217] and Burwell [2016b, pp. 3–4]
CPP Only	01/01/1963–02/28/1970	Internal doses from exposures to alpha-emitting radionuclides separated from the MFPs	Based on information in NIOSH [2017a, p. 248, 251] and Azar [2019, pp. 3–4]
All INL	03/01/1970–12/31/1974	Internal doses from exposures to alpha-emitting radionuclides separated from the MFPs	Based on information in NIOSH [2017a, p. 248, 251] and Burwell [2016a, pp. 3–4]
CPP Only	01/01/1975–12/31/1980	Internal doses from exposures to alpha-emitting radionuclides separated from the MFPs	Based on information in NIOSH [2017b, p. 41] and Hargon [2017, pp. 3–5].

For partial dose reconstruction purposes, INL and ANL-W workers are primarily identified by their DOL-provided covered employment. As discussed in Section 5.1.3.1, determining when an INL worker was at CPP should primarily be based on the work location information in their external dose records. The work location information in the external dose records can also be used to identify when a worker was at INL and/or ANL-W when there is no indication of INL and/or ANL-W employment in their DOL-provided covered employment, which is often the case for workers that merely visited a site.

SEC classes are established for a specific EEOICPA covered facility, and the dose infeasibilities for those classes do not affect any other EEOICPA covered facility unless specified in a DHHS designation for that other facility. As indicated in 42 *Code of Federal Regulations* (C.F.R.) § 83.9(c)(i), an EEOICPA covered facility could be either an AWE or DOE facility that could "constitute a single building or structure, including the grounds upon which it is located, or a site encompassing numerous buildings or structures, including the grounds upon which it is located" [42 C.F.R. 83, 2023, p. 5]. In

the case of ANL-W and INL, the interplay between both sites and their SECs can be confusing because they were distinct covered facilities under EEOICPA before February 2005 and because some workers from ANL-W also periodically performed work at INL and vice versa. This is further complicated because the same technical basis documents (site profile) are used for both sites. For the INL SEC periods, additional confusion is caused by the fact that INL is a very large site with multiple large operating areas that are separated by distances of more than a mile (1.61 km), the fact that two of the INL SEC periods only apply to workers at one of those operating areas (CPP), and the INL SEC periods include external dosimetry requirements that vary depending on the INL SEC period. Therefore, NIOSH has provided the following supplemental guidance for assigning unmonitored dose for the partial dose reconstructions during the ANL-W and INL SEC periods.

ANL-W SEC period – April 10, 1951, through December 31, 1957

- In general, no unmonitored dose can be assigned to account for potential exposures received at ANL-W during the ANL-W SEC period.

- If an individual had exposures at both ANL-W and INL during the ANL-W SEC period, unmonitored dose cannot be assigned for their exposures at ANL-W. However, unmonitored dose can be assigned for the exposures while the worker was at INL.

CPP-only SEC periods – January 1, 1963, through February 28, 1970, and January 1, 1975, through December 31, 1980

- For any given calendar year during the CPP-only SEC periods, if the worker has at least one CPP dosimeter, the worker entered CPP during that year and *the dose infeasibility for CPP applies*. This applies even when the worker had dosimeters from other INL areas during that year. For that calendar year, unmonitored doses *cannot* be assigned to account for the worker's potential internal exposures to alpha-emitting radionuclides during that year.

- For any given calendar year during the CPP-only SEC periods, if a worker has only external dosimetry from other INL areas (no CPP dosimeters), the worker did not enter CPP during that year and *the dose infeasibility for CPP does not apply*. For that calendar year, unmonitored doses can be assigned to account for the worker's potential internal exposures to alpha-emitting radionuclides during that year.

All-INL SEC period – March 1, 1970, through December 31, 1974

- For any given calendar year during the all-INL SEC period, if a worker has at least one INL dosimeter, the worker could have entered CPP during that year and *the dose infeasibility applies*. For that calendar year, unmonitored doses *cannot* be assigned to account for the worker's potential internal exposures to alpha-emitting radionuclides during that year.

- For any given calendar year during the all-INL SEC period, if a worker does not have any INL dosimeters, *the dose infeasibility does not apply* for that year. Because it is very unlikely that an INL worker would be monitored for internal dose when they were not monitored for external dose (excluding baseline and termination bioassays), only environmental internal dose will likely need assessment according to the guidance in Section 5.6.

For the INL SEC periods, it was determined that internal doses from exposures to alpha-emitting radionuclides cannot be adequately reconstructed due to poor contamination controls and inadequate personnel monitoring [NIOSH 2017a, p. 251; NIOSH 2017b, p. 41]. However, the DHHS designations for the INL SEC classes allow NIOSH to use any internal and external monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose

reconstruction processes or procedures) for the partial reconstructions during the INL SEC periods [Azar 2019, p. 5; Burwell 2016a, p. 4; Hargon 2017, p. 5]. Therefore, internal doses during the INL SEC periods can be reconstructed for periods for which monitoring data are available to reconstruct the internal doses from exposures to alpha-emitting radionuclides.

For the ANL-W SEC period, with the exception of the occupational medical and environmental exposures, it was determined that internal and external exposures cannot be adequately reconstructed due to missing records [NIOSH 2016, pp. 209, 213–214]. However, the DHHS designation for the ANL-W SEC class allows NIOSH to use any internal and external monitoring data that may become available for an individual claim (and that can be interpreted using existing NIOSH dose reconstruction processes or procedures) for the partial reconstructions during the ANL-W SEC period [Burwell 2016b, pp. 4–5]. Therefore, internal doses during the ANL-W SEC period can be reconstructed for periods for which sufficient monitoring data are available to reconstruct those specific doses. Additionally, if there are sufficient monitoring data for calculating a worker's ^{90}Sr and ^{137}Cs intakes for a given period, the unmonitored intake approaches in Sections 5.5.1 and 5.5.2 can also be used to account for potential intakes of the other mixed fission and activation products (MFAPs) and the actinides that were potentially present with the ^{90}Sr and ^{137}Cs. However, the default internal missed dose approach in Section 5.6 should not be used for the ANL-W SEC period.

Because no infeasibilities were identified for INL and ANL-W environmental doses, the environmental doses are not affected by the INL and ANL-W SEC classes and can be assigned.

5.1.4 Historical Overview

In 1949, 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 site has experienced several name changes since its creation. Site names over the years and the approximate date ranges for those names include the NRTS (1949 through 1973), Idaho National Engineering Laboratory (INEL; 1974 through 1996), Idaho National Engineering and Environmental Laboratory (INEEL; 1997 through 2004), and INL (2005 through the present). For convenience, INL is used throughout the rest of this document where it is unnecessary to distinguish between one of its other names.

Each original AEC laboratory was unique in both mission and location. Because the early days of AEC programs represented the beginnings of the nuclear age, significant technical developments were a necessity, including developments in radiation safety. Some unique characteristics of radiation safety (and internal dosimetry specifically) at INL that had a marked influence on internal dosimetry programs at each facility were:

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

- INL began operations 8 to 10 years 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 National Laboratory [ACC 1952] were adopted at the startup of INL facilities.

- Two AEC field offices (Chicago and Idaho) were responsible for and had oversight of the INL programs discussed in this report. In addition, the Nuclear Navy under direction of the Pittsburgh Field Office administered programs and used facilities at INL for training and

program development. Thus, there were three federal organizations using INL and its infrastructure. This TBD does not apply to naval facilities or naval personnel, even if those personnel received exposure from AEC operations.

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

- To provide consistency of radiation safety programs at INL among a large variety of facilities and constantly changing contractors, AEC established a Health and Safety (H&S) Laboratory at INL 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) 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 and its successors conducted oversight. Technical data, information (particularly in the instances of detectable worker intake), and analytical internal dose calculations and evaluations were exchanged between HSL and each contractor [2].

As a consequence, and in spite of the constant changes at INL, basic assumptions about detection levels [e.g., minimum detectable amounts (MDAs), minimum detectable levels, etc.] and missed dose potential were relatively consistent across the years [3]. There were differences in the available nuclear materials from facility to facility, but by early 1958 gamma spectral analysis capabilities at INL allowed the significant bioassay results (those that would result in reportable internal dose) to be defined in terms of the specific radionuclides [Hayden 1958]. The practice in the case of a higher urine sample result was to attempt radionuclide identification through gamma spectral analysis and chemical separation. This TBD describes default assumptions for use in cases when the bioassay records for a worker do not include the radionuclide specific analyses and only include records for gross radioactivity measurements.

5.1.4.1 Test Reactor Area

The Test Reactor Area (TRA) complex includes hot cells, a gamma irradiation pool facility, research laboratories, and analytical laboratories. The reactors at TRA, as well as the others at INL, were used for testing materials, experiments, and neutron irradiation facilities [Stacy 2000]. The Materials Test Reactor (MTR) was the second operating reactor at INL and ran from March 1952 to 1970 in TRA [Stacy 2000, pp. 278–279]. TRA has also hosted the Engineering Test Reactor (ETR; 1957 to 1981) and the Advanced Test Reactor (ATR; 1967 to present) along with six reactor-critical facilities that supported the test reactors [Stacy 2000].

The uranium used in the TRA reactors is enriched to 93% ^{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 hours and emits a high-energy gamma ray (2.75 MeV). The inhalation dose to personnel from ^{24}Na is insignificant in comparison with that from the fission products in the fuel [4]. There are minor levels of activation products of stainless steel (^{58}Co, ^{60}Co, ^{51}Cr, ^{56}Mn, etc.) in the primary coolant system due to corrosion of its stainless-steel components [5].

Several factors contributed to unusual amounts of fission products in the primary coolant systems of 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 [6]. Another factor was fuel contaminated with tramp uranium on the outside of the cladding. During reactor operation, fission products in tramp uranium were released directly into the primary coolant system [7]. Reactor operators and the fuel manufacturer resolved these deficiencies over time. By the time ATR became operational, its primary coolant system was considerably less contaminated than that of MTR or ETR during their early years of operation. Other sources of contamination are the reactor components such as fuel elements, control elements, irradiation experiments, reactor loop components, pressure tubes, and irradiated hardware that are routinely removed from the reactors and placed in their canals. The TRA reactor canals are similar to the spent fuel storage pools at other types of reactors; however, they also include other facilities for processing the irradiation experiments and other materials. If those irradiated components were not cleaned adequately before their removal from the canal, the activity on them could become airborne.

Most radioactivity releases from TRA reactors to areas potentially occupied by workers consisted of noble gases that decayed to short-lived particulates. The principal dose to personnel from releases of noble gases was direct radiation rather than inhalation [8]. Table 5-2 provides a summary of the major airborne incidents that occurred at TRA.

Table 5-2. Major airborne incidents at TRA.

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

5.1.4.2 Argonne National Laboratory-West

The Argonne National Laboratory near Chicago established a branch at INL known as ANL-W, where it built and operated several reactors that were of fundamental importance for the development of commercial nuclear power. ANL-W was operated under contract to the Chicago Field Office of AEC/ERDA/DOE by the University of Chicago from 1951 through January 2005. In February 2005, ANL-W was merged into INL and subsequently renamed MFC. As stated in Section 5.1.1, this TBD uses ANL-W because this was the name for most of the site's history and is the name most familiar to EEOICPA claimants.

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

5.1.4.3 Other Test Reactor Areas

As the primary nuclear reactor development laboratory in the United States, INL tested or evaluated more than 100 reactor concepts [DOE 1997]. Fifty-two test reactors were designed, constructed, and operated (including operation-to-destruction tests). INL has experienced a number of episodic reactor events, both planned and accidental; examples include the military Stationary Low-Power Reactor (SL-1) accident on January 3, 1961 [Stacy 2000]; a series of deliberate safety experiments by 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 [9]. These exposures were documented in each exposed employee's file.

5.1.4.4 Chemical Processing Plant

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

CPP 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 [10].

Most internal doses at CPP were from accidental releases. Table 5-3 lists unusual and episodic events that have occurred at CPP. CPP 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 [AEC 1960; Horan 1962; Stacy 2000]. These exposures are documented in the personnel dosimetry files.

Table 5-3. Major airborne incidents at CPP.

Date	Incident	Radionuclides released	Internal dose discussion	Sources
05/15/1957	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
03/20/1958	Iodine release	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
10/16/1959	Criticality accident in shielded process system	Short-lived noble gases and I-131, I-132, I-133, etc.	Short-lived radioactive gases released to plant areas; internal doses reported as minimal.	Ginkel et al. 1960
01/25/1961	Criticality accident in shielded process system	Short-lived noble gases & I-131, I-132, I-133, etc.	Short-lived radioactive gases released through process off-gas system to 76-m stack. Internal doses reported as minimal.	Paulus et al. 1961
01/1972	Release of ~1.0 Ci Ru-106 from main stack	Ru-106	No internal doses detected.	ERDA 1977
11/17/1972	CPP mass spectrometry Pu contamination incident	Pu-238, Pu-239	An exposure incident involving about a dozen personnel resulted in 50-yr exposure lung doses ranging up to about 4 rem.	Wenzel 1973, 1974
10/17/1978	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
11/1985	N-Cell Pu uptake	Pu-238	Internal exposures were far below DOE exposure limits, but showed a weakness in the radiological control program.	Henry and Slagle 1985
10/30/1988	Release of ~0.2 Ci of Ru-106 from main stack	Ru-106	No internal doses detected.	Hoff et al. 1989

A variety of fuel types from a multitude of reactors were processed at CPP throughout its operating history. The dates when processing started for each of the major three fuel types were approximately 1953 for aluminum fuels, 1956 for zirconium fuels, and 1965 for stainless-steel fuels [Staiger 2003]. There are relatively long half-life fission products that persist for CPP source terms. In most cases, the source terms were well tagged with beta-emitting radionuclides, which allowed beta/gamma-detecting continuous air monitors (CAMs) and bioassay methods for beta/gamma-emitting radionuclides to be used at CPP with the realization that they would also warn of possible alpha contamination or internal exposures [11].

The irradiated fuel was normally decayed a minimum of 90 days before shipment to CPP for processing [Allied Chemical Corporation, no date], to minimize the radiological safety hazard of the relatively volatile halogens. After being received at CPP, the irradiated fuel was kept in storage until it had a minimum of 120 days of cooling; however, the processing of fuels often did not occur until months or years later [Allied Chemical Corporation, no date]. 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 [12].

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 MTR with as little decay as manageable (less than 2 days). Fuel was removed from MTR, transported about 2 mi to CPP in a heavily shielded transport container by a straddle carrier, immediately dissolved, and the barium element was extracted. The ^{140}Ba/La product was shipped immediately to Los Alamos National Laboratory. The INL RaLa process released large quantities of volatile radioactive iodine isotopes because early designs for the process were inadequate for controlling iodine [Hayden 1958]. Several significant internal exposure incidents occurred in which ^{131}I, ^{132}I, and ^{133}I intakes occurred before personnel could respond to CAM alarms and take protective or corrective actions [13].

5.1.4.5 Other Nuclear Facilities and Processes

Other nuclear facilities at INL where intakes occurred include:

- RWMC handled radioactive wastes from nuclear facilities on the site and was the primary disposal location for materials from the Rocky Flats Plant (RFP). Although most waste came to RWMC in packages, accidents occurred during handling and processing that resulted in releases [Hoff et al. 1989]. This in turn caused intakes of aged MFAPs, uranium isotopes, and transuranic (TRU) radionuclides [Bechtel BWXT Idaho (BBI) 2001, pp. 81–93].

- The Specific Manufacturing Capability (SMC) Project is a depleted uranium (DU) specialty-parts production plant built in 1985 in the ANP Program hangar on the Test Area North (TAN) portion of the site. The SMC Project processes metric tons of DU metal for the production of military shielding units [Stacy 2000]. The processes of cutting, machining, and handling uranium metal produce environments in which both chronic and accidental intakes of DU have occurred.

- The U.S. Navy used the Naval Reactors Facility (NRF) for operating reactors and as a naval reactor training center. Because this is not a DOE program and not under DOE oversight, 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 INL workers received internal doses from their support of work at NRF [14].

5.1.5 Radionuclides of Concern and Solubility

INL facilities and activities were primarily involved with experimental reactor design and development, irradiated fuel processing, DU parts production, and low- and high-level radioactive waste treatment and disposal. ORAUT-TKBS-0007-2, *Idaho National Laboratory and Argonne National Laboratory-West – Site Description* [ORAUT 2010a], describes these activities in more detail.

Table 5-4 lists the INL radionuclides of concern from these programs and as documented in *INEEL M&O Contractor Technical Basis for Internal Dosimetry* [BBI 2001]. These radionuclides are those for which internal doses were determined in the past and/or for which detection methods were developed.

Table 5-4. Primary radionuclides of concern.

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

a. Based on information in BBI [2001].

INL stored and processed nuclear materials from all over the world [ORAUT 2010a]. These materials included, but were not limited to, nuclear wastes from other DOE sites and irradiated nuclear fuels from universities, test reactors, commercial power plants, INL reactors, and U.S. Department of Defense Projects. Because of this, it is not possible to limit the material types to a single selection when there are multiple possibilities identified by the International Commission on Radiological Protection (ICRP). In addition, materials change over time as they age, typically becoming more insoluble. Therefore, all material types included in ICRP Publication 68 [ICRP 1995], as well as Type SS plutonium, for a given radionuclide should be considered when performing a dose reconstruction and the one yielding the highest dose should be assigned with the following exceptions:

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

- With the exception of ^{131}I intakes calculated using the approach described in ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* [OTIB-0054; ORAUT 2015], intakes of radioactive iodine need only be assessed as Type F material using the inhalation intake route in the Integrated Modules for Bioassay Analysis (IMBA) program. Because the various intake route and material type options for iodine in the IMBA program result in relatively insignificant dose differences and the inhalation of Type F material results in a bounding dose estimate, it is reasonable to limit all assessments of radioactive iodine intakes to the inhalation of Type F material. When using the

approach in OTIB-0054, use the intake route and material type options specified for iodine intakes in that document.

- Radionuclides where decay products or trace atoms are bound in a matrix of another radionuclide. Examples of this include ^{90}Sr and its decay product ^{90}Y, and trace atoms of ^{241}Am in a plutonium matrix. In these examples, the ^{90}Y tracks with the Type F ^{90}Sr even though ^{90}Y is only associated with Type M and S materials, and ^{241}Am will track with Type S plutonium even though ^{241}Am is only associated with Type M material. Section 3.2.2 in ORAUT-OTIB-0060, *Internal Dose Reconstruction* [OTIB-0060; ORAUT 2018, pp. 20–23], provides additional guidance on this.

- If the assessment of the bioassay data provides an inarguably better fit for a specific material type, only that material type need be used for the dose assessment [ORAUT 2018].

5.1.6 Intake Modes and Particle Size Distributions

Unless specific information is provided, the default guidelines for intake mode and particle size distribution in OTIB-0060 should be followed.

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

5.1.7 Internal Dosimetry Program

The radiological protection program was established to provide timely detection of barrier or ventilation failure. The program consisted of continuous and retrospective air and effluent monitoring combined with personnel and surface contamination monitoring [15]. 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 [16]. 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 airborne contamination was thought possible regardless of the actual measured air or surface contamination [ACC 1952].

The changes in contractors at INL during its 60-plus year history resulted in relatively frequent management changes at most of the facilities. However, the contract with the University of Chicago to operate ANL-W facilities did not change until 2005. Summaries of the contractor changes are provided in Tables 2-1 and 2-2 of ORAUT-TKBS-0007-2 [ORAUT 2010a].

The primary oversight of INL, which included most projects on the site and all support functions, was initially assigned to the AEC Idaho Operations Office (IDO) [17]. 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 [18]. Once NRF was constructed, the U.S. Navy provided oversight for that portion of the INL site. In addition, the Chicago Operations Office provided oversight for ANL-W programs and facilities that were contracted by the University of Chicago until February 2005. During that period, ANL-W used site support services, including internal dosimetry support, but the contractor reported the dosimetry results to the Chicago Operations Office versus IDO [19]. In February 2005, oversight of INL and ANL-W programs and facilities were combined under the same primary IDO contract.

INL personnel dosimetry records have been and are documented and permanently maintained by the various organizations throughout the site's history [20]. Records about individual facility or contractor field monitoring programs (air monitoring data, personnel contamination records, etc.) were maintained by individual contractors and/or site areas and are not maintained in a single recordkeeping system. The field monitoring data were not available for use in this TBD.

In spite of the frequent changes in operational responsibility through the years and the movement of workers among facilities, there has been a basic level of consistency in the internal dosimetry programs at INL, particularly the bioassay analytical techniques and calculation processes [21]. 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 [22].

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 [23].

Internal dose reconstruction for personnel who worked at a number of the 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 [24]. In addition, both the individual facilities and the H&S laboratories had radionuclide identification capabilities from the early 1960s. Positive bioassay results (analyses in which the results exceeded 2σ counting statistics) were normally followed by a confirmatory analysis to identify specific radionuclides [Bhatt 2002].

5.1.7.1 Contamination Control

The contamination control limits for detecting and controlling released activity beyond the control boundaries were dependent on instrumentation capabilities and the basic philosophy of acceptance of detectable contamination. Due to an increased emphasis on maintaining exposures as low as reasonably achievable, some reduction in acceptable release levels was implemented. The contamination control limits for alpha on plant surfaces and particularly on personnel were always set close to the MDA, such that *any detectable* contamination was a signal for preventive and followup evaluations and actions. Beta/gamma MDAs typically were a factor of 5 below the limits. Tables 5-5 and 5-6 summarize control limits primarily from an early health physics manual for CPP [ACC 1952] and current operating procedures.

Table 5-5. Surface contamination control and MDAs, 1952 through 1960s.

Surface location	Detection technique	Control levels	Typical MDA
Plant/equipment	Smears	500 dpm β and 20 dpm α per 100 cm^2	150 dpm β and 10 dpm α per 100 cm^2
Personal clothing	Portable survey instruments	1500 dpm β and 500 dpm α per 100 cm^2	1,000 dpm β and 500 dpm α per 100 cm^2
Personal skin	Portable survey instruments	Any detectable reported, e.g., 1,000 dpm β and 500 dpm α per 100 cm^2	1,000 dpm β and 500 dpm α per 100 cm^2
Shipments	Smears	500 dpm β and 20 dpm α per 100 cm^2	150 dpm β and 10 dpm α per 100 cm^2
Shipments	Portable survey instruments	0.1 mrep/hr β and 500 dpm α per 100 cm^2	0.01 mrep/hr β and 500 dpm α per 100 cm^2

Table 5-6. Surface contamination control and MDAs, 1970s through present.

Surface location	Detection technique	Control levels	Typical MDA
Plant/equipment surfaces	Smears	300 dpm β and 20 dpm α per 100 cm^2	30 dpm β and 10 dpm α per 100 cm^2
Personnel	Portable survey instruments	Any detectable reported, e.g., 300 dpm β and 200 dpm α per 80-100 cm^2	300 dpm β and 200 dpm α per 80–100 cm^2

5.1.7.2 Air Monitoring

Monitoring airborne radioactivity in occupied areas was a basic part of the internal exposure prevention program. Beta/gamma-detecting CAMs were used from the beginning of all facility and program operations in routinely occupied areas. With the exception of the SMC Project that started in 1985, the primary contaminant radionuclides by activity were MFAPs, which were beta/gamma emitters with maximum permissible concentrations or derived air concentrations (DACs) above 1×10^{-9} µCi/cm^3. TRU materials and uranium were available at some INL facilities, but they were nearly always well tagged with beta/gamma-emitting radioactivity that allowed beta/gamma-detecting CAMs to be used to warn of possible alpha contamination or internal exposures.

The *CPP Health Physics Manual* from 1952 [ACC 1952] describes a CAM and three other air sampling systems. The manual required use of a filter-type respirator when airborne activity exceeded 1×10^{-8} µCi/cm^3 for beta/gamma activity or 1×10^{-11} µCi/cm^3 for alpha activity [ACC 1952]. An army assault-type mask was required when levels exceeded this by a factor of 10. Positive-pressure air masks were required if levels larger by a factor of 1,000 occurred [ACC 1952].

The CAM systems provided real-time air activity evaluations (although it is not clear what the set points for alarms were), and fixed air samplers at several locations provided retrospective data and an average air concentration of beta/gamma emitters in an area or building [25]. 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 [26]. 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 [27]. 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 [28].

In general, workers were asked to submit to bioassay measurements whenever they were in an area where a CAM alarm sounded. In addition, the fixed location and retrospective air sampling system signaled the need for bioassay if elevated air sample results were detected [29].

5.1.7.3 Bioassay Monitoring

Routine bioassay of radiation workers has occurred since the beginning of site operations. However, formal documentation of the bioassay programs was not found for periods before 1981. Some of the data sheets on individuals indicate that bioassay sampling occurred routinely every 6 months in 1953 [30]. Table 5-7 lists the reconstructed history of routine bioassay frequency. The INL monitoring and analytical programs were also designed to initiate an investigation of any potential internal intake as indicated by off-normal workplace indicators such as positive air sampling, personnel contamination, etc. [BBI 2001]. Those investigations often included performing nonroutine bioassay measurements for the potentially exposed workers. In addition, most of the recorded bioassay analyses performed at INL did not result in detectable radionuclides [ORAUT 2010b,c].

DOE HSL technical reports and annual reports, coupled with facility memoranda and reports, documented INL's analytical detection capability 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 and iodine) were available to quantify the radioactive components of a variety of samples of interest.

In the early days a gross beta measurement was made on an evaporated aliquot or a gamma count was made directly on a liquid sample, or both. Any detectable activity triggered a specific chemical separation analysis (generally strontium). The gross beta assay used a 5-mL volume, but that was replaced with the gross gamma assay supplemented with strontium analyses that typically used a 75-mL volume [31]. 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, IDO HSD acquired a 256-channel gamma spectrometer with a 3x3 (3 inch by 3 inch) sodium iodide thallium-doped [NaI(Tl)] scintillation detector and counting system for analyzing gamma-emitting radionuclides. In 1960, HSD obtained a 3x3 NaI(Tl) well detector for gamma analysis, which replaced gross beta counting as the routine analytical procedure for urine samples. The *Annual Report of Health and Safety Division 1960* [AEC 1961, p. 69] indicates that approximately 1.5×10^{-6} µCi/mL of MFPs can be detected in 75 mL of urine in a 5-minute count, which is about the same as was obtained with the gross beta procedure in a 20-minute count.

Table 5-7. Routine bioassay history summary.[a]

Period, frequency, and type	Groups analyzed/sampled	Investigating level	Comments	Sources
1953–1960 Annual In vitro urine	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Individual data sheets; Table 5-10; Horan 1958; AEC 1961
1961–1972 Annual In vitro urine In vivo	Radiation workers	Unknown	Frequency is inferred from individual data sheets.	Table 5-10; Horan 1962; Dodd 1963
1973–1981 Annual In vitro urine In vivo When internal intake suspected Fecal	Radiation workers	*Reporting* Annual dose equivalent >10% quarterly standard in *ERDA Manual* Chapter 0524 [ERDA 1975]	Frequency is inferred from individual data sheets.	AEC 1968, 1975; ERDA 1975
1982–1987 Annual In vitro urine	CPP-603 workers; fuel reprocessing operators	*Reporting* 50-yr CDE >10% quarterly standard in *ERDA Manual* Chapter 0524 [ERDA 1975]	Staggered to monitor group throughout the year	ICPP 1981
1982–1987 When internal intake suspected Fecal	Waste reprocessing operators; shift laboratory workers; health physics technicians	*Reporting* 50-yr CDE >10% quarterly standard in *ERDA Manual* Chapter 0524 [ERDA 1975]	Staggered to monitor group throughout the year	ICPP 1981
1982–1987 Annual In vivo	Selected radiochemistry workers; maintenance workers; denitrator operators	*Reporting* 50-yr CDE >10% quarterly standard in *ERDA Manual* Chapter 0524 [ERDA 1975]	Staggered to monitor group throughout the year	ICPP 1981
1982–1987 Termination In vivo	All radiation workers	*Reporting* 50-yr CDE >10% quarterly standard in *ERDA Manual* Chapter 0524 [ERDA 1975]	Staggered to monitor group throughout the year	ICPP 1981

Period, frequency, and type	Groups analyzed/sampled	Investigating level	Comments	Sources
1988–1989 1 to 6 mo In vivo Annual In vitro fecal 18 to 24 mo In vitro urine	All radiation workers	*Investigating* Lung, 50-yr CDE >0.5–1.0 rem Bone surface, 50-yr CDE >1.0–2.0 rem Other organs, 50-yr CDE >0.5–1.0 rem	Staggered so that worker received some sort of analysis or sampling every 3 mo	Guinn and Mansfield 1988
1988–1989 Termination In vivo In vitro	When internal exposure suspected	*Investigating* Lung, 50-yr CDE >0.5–1.0 rem Bone surface, 50-yr CDE >1.0–2.0 rem Other organs, 50-yr CDE >0.5–1.0 rem	Staggered so that worker received some sort of analysis or sampling every 3 mo	Guinn and Mansfield 1988
1988–1989 New hire In vitro In vivo	Depending on review of radiation dose history	*Investigating* Lung, 50-yr CDE >0.5–1.0 rem Bone surface, 50-yr CDE >1.0–2.0 rem Other organs, 50-yr CDE >0.5–1.0 rem	Staggered so that worker received some sort of analysis or sampling every 3 mo	Guinn and Mansfield 1988
1990–1994 Annual In vivo In vitro urine 6 mo In vitro fecal	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 dose equivalent >5 rem annual	*Reporting* In accordance with DOE Order 5480.11 [DOE 1988] Workers who could receive 0.1 rem AEDE or 5 rem ADE organ or tissue dose	Bioassay requested when workplace monitoring program indicated >0.02 annual limit of intake. Followup 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

Period, frequency, and type	Groups analyzed/sampled	Investigating level	Comments	Sources
1990–1994 New hire In vivo	Worked at a facility where gamma-emitting radionuclides were handled	*Investigating* AEDE ≥0.01 rem	Bioassay requested when workplace monitoring program indicated >0.02 annual limit of intake. Followup 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
1990–1994 New hire In vitro urine, fecal	Worked in U manufacturing or recovery facilities; worked with TRU materials	Not applicable	Bioassay requested when workplace monitoring program indicated >0.02 annual limit of intake. Followup 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
1990–1994 Termination In vivo In vitro	Any employee suspected of having an internal exposure or on a scheduled monitoring program	Not applicable	Bioassay requested when workplace monitoring program indicated >0.02 annual limit of intake. Followup 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

Period, frequency, and type	Groups analyzed/sampled	Investigating level	Comments	Sources
1995 Termination In vivo	All radiation workers who enter radiological buffer areas or areas of higher radiological controls and are likely to receive intakes resulting in a CEDE of 0.1 rem or more. Type of bioassay based on source term. Urine requested when pure beta, uranium, or TRU was of interest. Feces requested primarily for uranium and TRU source terms.	*Reporting* In accordance with DOE 5480.11 [DOE 1988] and 10 C.F.R. Part 835 [10 C.F.R. 835, 1999]: Workers who could receive 0.1 rem CEDE Declared pregnant workers when embryo/fetus could receive 0.05 rem dose equivalent	Each facility had a specific TBD for internal dosimetry.	Andersen et al. 1995
1995 When workplace monitoring indicated significant potential for intakes In vitro urine, fecal	Random sampling was performed to demonstrate the adequacy of the radiological controls in limiting the internal intake of radionuclides. Employees were selected at random from both non-radiation workers and a radiation worker population.	*Investigating* Internal doses resulting from all confirmed intakes were to be evaluated.	Followup for any suspected intake of radionuclides and to more accurately identify and characterize the amount of intake and excretion pattern	Andersen et al. 1995
1995–2000 New hire Dependent on screening	Based on screening to determine internal conditions from previous uptakes or to establish baseline for those continuing to work as radiation workers	Not applicable	None	Andersen et al. 1995
1995–2000 Termination In vivo In vitro	Any employee who was on a scheduled monitoring program	Not applicable	None	Andersen et al. 1995

Period, frequency, and type	Groups analyzed/sampled	Investigating level	Comments	Sources
2001–present As developed by individual facilities based on analysis tables developed for each radionuclide In vivo In vitro urine, fecal	All radiation workers	*Reporting* In accordance with DOE 5480.11 [DOE 1988] and 10 C.F.R. Part 835 [10 C.F.R. 835, 1999]: Workers who could receive 0.1 rem CEDE Declared pregnant workers when embryo/fetus could receive 0.05 rem dose equivalent *Investigating* Uranium, >1.0 µg/L In vitro activity detected >2σ In vivo >2.33σ Default trigger levels exceeded	Bioassay is mandatory when an employee or visitor is involved in an event where internal uptake of radionuclides likely occurred.	BBI 2001
2001–present Termination In vivo, In vitro	Any employee who was on a scheduled monitoring program	*Investigating* Uranium, >1.0 µg/L In vitro activity detected >2σ In vivo >2.33σ Default trigger levels exceeded	Bioassay is mandatory when an employee or visitor is involved in an event where internal uptake of radionuclides likely occurred.	BBI 2001

a. AEDE = annual effective dose equivalent; CDE = committed dose equivalent; CEDE = committed effective dose equivalent.

The *Annual Report of Health and Safety Division 1960* [AEC 1961] outlined a basic philosophy for gamma counting of bioassay samples. Gamma counting would be effective in all situations except exposures to pure strontium isotopes. To guard against this unlikely possibility, the procedure to perform a strontium analysis for individual workers at risk (radiation workers) every 2 years and at termination was established. Because of the improbability of finding detectable activity, all activities were to be precipitated by oxalic acid in a weak solution, gross beta counted, and the strontium analysis not completed unless a detectable count was obtained on the precipitate. A 100-mL sample of urine permitted the detection of approximately 8×10^{-8} µCi/mL of ^{90}Sr.

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

Table 5-8 provides a summary of the in vitro bioassay measurements performed for INL and ANL-W workers during the years 1953 through 1975.

Table 5-8. In vitro bioassay measurements at INL and ANL-W, 1953 through 1975.[a,b]

Year	Gross alpha	Gross beta or beta/gamma	Gross gamma	Cs[c]	Sr[d]	U[e]	Pu[f]
1953	NA	455	NA	NA	NA	1	NA
1954	6	1,400	2	NA	NA	52	NA
1955	NA	2,774	1	NA	NA	42	3
1956	6	4,924	1	NA	89	17	NA
1957	NA	7,572	242	1	1	95	2
1958	1	7,398	240	NA	4	133	3
1959	1	7,010	1,048	NA	37	57	21
1960	NA	6,213	1,794	NA	69	40	9
1961	NA	40	7,006	39	2,096	2	8
1962	8	8	6,125	2	1,579	13	18
1963	NA	19	5,890	NA	937	124	135
1964	1	2	5,401	NA	2,228	60	40
1965	NA	10	5,610	3	1,079	50	58
1966	1	NA	1,823	NA	1,099	11	11
1967	NA	20	414	NA	395	17	31
1968	NA	447	302	NA	328	NA	15
1969	NA	148	581	2	182	1	19
1970	1	5	189	1	10	18	6
1971	NA	NA	3	NA	NA	NA	NA
1972	NA	NA	NA	NA	NA	104	29
1973	NA	NA	NA	NA	NA	NA	58
1974	NA	NA	1	12	13	6	35
1975	NA	22	7	5	38	2	8

a. Sources: NIOSH [2017a, p. 181]; ORAUT [2020].
b. NA = not available.
c. Cesium analyses were most frequently for Cs-137, but sometimes also included other isotopes such as Cs-134.
d. Strontium analyses were most frequently for total strontium and Sr-90, but sometimes also included other isotopes such as Sr-89.
e. Uranium analyses before 1971 were for total uranium in urine. After 1975, they were predominately isotopic uranium analyses performed on either urine or fecal samples.
f. Plutonium analyses before 1972 were for total plutonium in urine and feces. For 1972 and later, they were predominately isotopic plutonium analyses performed on either urine or fecal samples.

The in vitro bioassay tallies in Table 5-8 are primarily for urine samples. Before 1963, fecal sampling was relatively rare. Beginning in 1963, fecal sampling became more common, but was still much less common than urine sampling. Initially, fecal samples were analyzed primarily for either gross gamma radioactivity or plutonium. In late 1966, INL also began analyzing fecal samples for uranium [ORAUT 2020].

As indicated in Table 5-8, until 1961, the primary in vitro bioassay method for monitoring workers for internal exposures to MFAPs was gross beta and/or gross beta-gamma analysis of urine samples. Beginning in 1961, the gross gamma analysis of urine samples became the primary in vitro bioassay method for MFAPs, which was often supplemented with a strontium analysis of the same sample. Table 5-8 also shows that in vitro bioassays for uranium and plutonium were relatively uncommon during those years. By 1978, there was a significant increase in the routine bioassays for uranium and plutonium for certain workers at ANL-W, CPP, and RWMC [ORAUT 2020].

In vivo bioassay, specifically whole body counting, started at INL in January 1961. In later years, other types of in vivo bioassays were added such as lung counting, thyroid counting, wound counting and skull counting for Sr/Y-90. Table 5-9 provides a summary of the in vivo bioassay measurements performed for INL and ANL-W workers during the years of 1961 through 1975. When whole body counting began, it was often performed along with in vitro bioassay for MFAPs (i.e., ^{90}Sr and/or gross gamma analysis of urine samples). By the late 1960s, INL began phasing out in vitro bioassay for MFAPs and began to rely solely on whole body counting to monitor workers for potential intakes of MFAPs.

Table 5-9. In vivo bioassay measurements at INL and ANL-W, 1961 through 1975.[a]

Year	Number of results
1961	1,344
1962	2,667
1963	3,839
1964	1,998
1965	2,646
1966	3,072
1967	1,511
1968	1,534
1969	1,158
1970	1,350
1971	1,210
1972	1,396
1973	1,285
1974	794
1975	1,626

a. Source: NIOSH [2017a, p. 181].

Significant intakes of radioactivity were uncommon at INL. This determination is supported by an evaluation of the bioassay data in the Site Research Database (SRDB). Of the more than 89,000 urine samples analyzed for gross beta radioactivity and gross gamma radioactivity, less than 2% of those measurements were above the MDA values in Section 5.2.1 [ORAUT 2010b]. Of more than 69,000 whole body counts (WBCs), which is typically a more sensitive measurement than urinalysis, less than 10% of those measurements were reported as having results above their respective detection levels [ORAUT 2010c]. In addition, over 92% of the WBC results that were above their respective detection levels were still below the 0.1-µCi reporting level [ORAUT 2010c], which indicates that large intakes were uncommon at INL.

5.1.8 Recordkeeping

Required internal dose data were maintained by the DOE HSD in individual hardcopy folders until 1989 when all technical support service functions, including those related to internal dosimetry, were transferred to the INL site's prime contractor. At that time, in vitro analytical functions were transferred to an onsite analytical laboratory. The in vivo counting laboratory provides support directly through the Radiation Dosimetry and Records (RDR) organization, which administers external and internal dosimetry support programs. The current contractor's subject matter expert reviews, validates, and prepares official internal dose assessments. A DOE staff member at RESL is responsible for oversight of INL internal dosimetry program functions and provides quality assurance. The RDR unit functions include documentation and records custodial responsibilities. In 1999, the Radiation Control Information Management System (RCIMS) database was placed in service to support the radiation protection program, including internal dosimetry. RCIMS reports internal doses as committed effective dose equivalent (CEDE) when an individual's dose history is prepared [BBI 2001].

The following information is important to internal dose reconstruction because the worker files from DOE can contain a variety of internal dose information including the calculated internal doses as well as the in vitro and in vivo individual bioassay results. Changing regulations influenced the level of internal dose evaluation and documentation, but they 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 5-10 describes internal dose information that could appear in pre-1989 records. Table 5-11 describes coded information that could appear in records after 1989. Tables 5-12 and 5-13 provide the column descriptions for the sample record sheets that were used for the urine sample data. Column descriptions are provided for the sample record sheets that were used before February 1955 and those for the period of February 1955 and later. Table 5-14 provides INL area codes for various work areas that could be in worker external dosimetry records. The area codes in the external dosimetry records can be used to identify where a worker worked during a given monitoring period.

Table 5-10. Internal dose assessment information before 1989.[a]

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

a. AEDE = annual effective dose equivalent; CDE = committed effective dose.

Table 5-11. Internal dose assessment information after 1989.[a]

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.

a. AEDE = annual effective dose equivalent; CDE = committed effective dose; CEDE = committed effective dose equivalent.

Table 5-12. Column descriptions for urine sample record sheets before February 1955.

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

Analytical information	Description
Corr.	This column is used for the background count rate. A 1σ counting error is sometimes also reported with the background count rate.
Corr. read.	This column is used for the net count rate, which is calculated by subtracting the background count rate from the calculated gross count rate.
Mass equiv.	This column is often used to report the amount of K-40 in dpm that the sample contains. Before February 1955, there is no indication that the sample results were corrected to account for the K-40 activity.
Corrected value	This column is used to report the net counting results in units of activity, usually dpm. When the results are reported in dpm, it should be assumed that they are actually representing units of dpm/sample, unless indicated otherwise.

Table 5-13. Column descriptions for urine sample record sheets from February 1955 and later.[a]

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

Analytical information	Description
Foreign activity, cpm and dpm	Unless stated otherwise, the results in this column are reported in units of cpm and dpm. This is the net count rate, which is sometimes corrected for K-40, and then converted to dpm based on the detector's counting efficiency. Uncertainty also included, which is recorded based on 1σ counting statistics. When the results are reported in dpm, it should be assumed that they are actually representing units of dpm/sample, unless indicated otherwise. At times the dpm units are crossed out and the results are recorded in other units. The other units will then be specified, and are typically units of dpm/mL or µCi/mL. In some instances the units may be in µg/L, which indicates the results for a uranium analysis whether or not a uranium analysis was previously specified.

a. NU = natural uranium.

Table 5-14. Area codes that could be in worker external dosimetry records.

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

a. See the acronyms and abbreviations list. Two-letter descriptions that contain an “X” were used for workers involved with construction activities.

Federal regulations about permissible internal dose and formal reporting requirements to the AEC, the ERDA, and DOE changed periodically during the site’s history. As the internal dose limits were reduced over time, the frequency of monitoring typically increased, new analytical methods were developed for existing types of bioassay to detect lower quantities, and new types of bioassay were developed and implemented (e.g., whole-body counting, chest counting, routine fecal sampling, etc.). As a result, a site’s ability to detect intakes was progressively enhanced over time because of the changes in the permissible levels of internal dose. During the early years internal dose was usually considered separately from external dose in terms of meeting specific exposure limits, and the calculated dose was reported and documented only if specific dose levels were exceeded [Aoki 1979]. AEC and ERDA policies required periodic urinalyses or in vivo counting or evaluation of air concentrations if the whole body dose or committed dose could exceed 300 mrem in a calendar quarter [AEC 1958, 1963, 1968, 1975; ERDA 1975]. Each individual analytical result was documented and placed in individual exposure files regardless of the formal reporting requirements.

The investigation levels (the levels at which positive bioassay results triggered followup sampling to verify that detectable activity had been taken into the body) have also changed little from the early years to the present [32]. Dickson [1977] established official investigation levels (Table 5-15) 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 annual effective dose equivalent (AEDE) or above as the point at which followup and reporting was required. With the DOE *Radiological Control Manual* [DOE 1994], this changed to

100-mrem CEDE. In addition, a calculated dose of 10 mrem or above would be recorded as an internal dose [DOE 1994].

Table 5-15. Derived investigation levels in 1977 for acute exposures.[a]

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

a. Source: Dickson [1977].

5.1.8.1 Radiological Incident Records

When an incident occurred, it was the policy to investigate thoroughly and identify all individuals involved in the incident [33]. Therefore, if no evidence of involvement is in the incident file or the individual's dosimetry file and no other evidence shows involvement, dose reconstructors should assume the individual was not involved.

5.2 IN VITRO MINIMUM DETECTABLE AMOUNT, COUNTING METHODS, AND REPORTING PROTOCOLS

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

In compliance with the November 1998 requirement in 10 C.F.R. Part 835 [10 C.F.R. 835, 1999] for the DOE Laboratory Accreditation Program, and based on Health Physics Society (HPS) N13.30, *An American National Standard – Performance Criteria for Radiobioassay* [HPS 1996], both the in vitro and in vivo radiobioassay laboratories at INL received accreditation from DOE in February 1998 [BBI 2001, p 11]. In accordance with this accreditation, MDAs and decision levels at the 95% confidence level are performed. The tables shown in Sections 5.2 through 5.3 list the current MDAs for the various bioassay methods employed at INL along with values from historical documents and the recommended periods of use for the MDA values.

5.2.1 Urine Sample Analyses

The majority of the urine samples taken at the site were single voidings; 24-hour samples were used Lrfor special sampling purposes (i.e., followup samples, primarily to extend the sensitivity). Urine sample results are typically reported in units of activity per sample along with the total sample and/or aliquot size. Table 5-16 provides the known MDAs for this in vitro bioassay method.

Table 5-16. Urine sample MDAs by period.[a]

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

a. KPA = kinetic phosphorescence analysis; PF= photofluorimetry.
b. Based on 1,400-mL daily volume and typical sample size and would need adjusted for different sample volumes.
c. MDA calculated from an inferred 2σ uncertainty.
d. Smallest reported value. Not MDA.

5.2.2 Fecal Sample Analyses

At times, fecal samples were collected from workers to assess potential intakes of certain radionuclides. Table 5-17 provides the known MDAs for this in vitro bioassay method.

Table 5-17. Fecal sample MDAs by period.

Radionuclide	Period	Fecal[a] (pCi/sample)	Sources
Co-60	1963–present	10	Rich 1990
Sr-90	1963–1994	10	Rich 1990
Sr-90	1995–present	1.9	Andersen et al. 1995; BBI 2001
Cs-134	1963–present	10	Rich 1990
Cs-137	1963–1999	0.01	Rich 1990
Cs-137	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 et al. 1995; BBI 2001
U-233/234	2003–present	0.05	Bhatt 2003
U-235	1970–2003	0.038	Andersen et al. 1995; BBI 2001
U-235	2003–present	0.09	Bhatt 2003
U-238	1970–1994	0.5	Rich 1990
U-238	1995–2002	0.03	Andersen et al. 1995; BBI 2000, 2001
U-238	2003–present	0.09	Bhatt 2003
Pu-238	1974–1994	0.03	AEC 1974
Pu-238	1995–2002	0.022	Andersen et al. 1995; BBI 2001
Pu-238	2003–present	0.02	Bhatt 2003
Pu-239/240	1964–1973	0.4[b]	Dodd 1964
Pu-239/240	1974–1994	0.02	AEC 1974
Pu-239/240	1995–present	0.03	Andersen et al. 1995; BBI 2000, 2001; Bhatt 2002
Am-241	1974–1994	0.07	AEC 1974
Am-241	1995–2001	0.023	Andersen et al. 1995; BBI 2001
Am-241	2002–present	0.04	Bhatt 2002
Cm-244	1974–present	0.02	AEC 1974
Cf-252	1974–present	0.02	AEC 1974

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

5.3 IN VIVO MINIMUM DETECTABLE AMOUNT, COUNTING METHODS, AND REPORTING PROTOCOLS

5.3.1 Whole Body Counting

5.3.1.1 General Information on Whole Body Counting

Whole body counting was introduced at INL in 1961; Table 5-18 lists MDAs by radionuclide and period. As early as 1961 one of the fundamental conclusions from experience at INL with in vivo and in vitro internal dosimetry analytical techniques was that a large proportion of the internal exposures were to insoluble materials. Radionuclides (e.g., ^{125}Sb, ^{110m}Ag, ^{65}Zn, and ^{95}Zr/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 urine as to be undetectable even in a 24-hour specimen [Horan 1962; Sill et al. 1964]. Whole body counting was demonstrated to detect activity as low as 0.01 µCi in a 10-minute count [Horan 1962]. This detection level was several orders of magnitude more sensitive than the maximum permissible body burdens for most beta/gamma-emitting fission and activation products.

Table 5-18. WBC MDAs by period.

Radionuclide	Period	In vivo MDA (nCi)	Count time (min)	Sources
Cr-51	1961–2000	5	10	Percival and Anderson, no date
Cr-51	2001–present	32	5	BBI 2001
Mn-54	1962–2000	5	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Mn-54	2001–present	2.6	5	BBI 2001
Mn-54	2001–present	1.3	10	BBI 2001
Fe-59	1962–2001	4.5	5	BBI 2001
Fe-59	2001–present	1.5	10	BBI 2001
Co-58	1961–2000	5	10	Percival and Anderson, no date
Co-58	2001–present	2.5	5	BBI 2001
Co-58	2001–present	1.1	10	BBI 2001
Co-60	1961–2000	5	10	Percival and Anderson, no date
Co-60	1971–1988	5[a]	10	AEC 1972, 1974
Co-60	1989–2000	7	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Co-60	2001–present	2.5	5	BBI 2001
Co-60	2001–present	1.1	10	BBI 2001
Zn-65	1961–2000	5	10	Percival and Anderson, no date
Zn-65	1989–2000	10	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Zn-65	2001–present	4.9	5	BBI 2001
Zn-65	2001–present	2	10	BBI 2001
Zr/Nb-95	1961–2000	5	10	Percival and Anderson, no date
Zr/Nb-95	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Zr/Nb-95	2001–present	2.6	5	BBI 2001
Ru-106	2001–present	27	5	BBI 2001
Ru-106	2001–present	7.6	10	BBI 2001
Ag-110m	1961–2000	5	10	Percival and Anderson, no date
Sb-125	1961–1988	5	10	Percival and Anderson, no date
Sb-125	1989–present	14	10	Martin 1989; Grothaus 1993
I-131	1961–2000	5	10	Percival and Anderson, no date
I-131	2001–present	3.8	5	BBI 2001
Cs-134	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Cs-134	2001–present	3	5	BBI 2001
Cs-134	2001–present	0.96	10	BBI 2001
Cs-137	1961–1970	5	10	Percival and Anderson, no date
Cs-137	1971–1988	2	10	AEC 1972, 1974; Guinn and Mansfield 1988
Cs-137	1989–2000	5	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Cs-137	2001–present	3.1	5	BBI 2001
Cs-137	2001–present	1.9	10	BBI 2001
Ba/La-140	2001–present	12	5	BBI 2001
Ce-141	2001–present	9.9	5	BBI 2001
Ce-141	2001–present	3.2	10	BBI 2001
Ce-144	1989–2000	50	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Ce-144	2001–present	44	5	BBI 2001
Ce-144	2001–present	15	10	BBI 2001
Eu-152	2001–present	4	10	BBI 2001
Eu-154	2001–present	2	10	BBI 2001

Radionuclide	Period	In vivo MDA (nCi)	Count time (min)	Sources
Eu-155	2001–present	1	10	BBI 2001
Ga-153	2001–present	6.5	10	BBI 2001
Hf-181	1989–present	5	10	Martin 1989; Grothaus 1993; Andersen et al. 1995
Ta-182	1961–1962	5	10	Percival and Anderson, no date
Hg-203	1961–1962	5	10	Percival and Anderson, no date

a. MDA calculated from an inferred 2σ uncertainty.

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

5.3.1.2 Attributing Positive Whole Body Counts to Cesium-137 Fallout

Fallout affected everyone in North America, and body burdens of ^{137}Cs measurable in the whole body counters were common in the 1960s and 1970s. Section 2.4.4 in OTIB-0060 provides guidance on how to account for ^{137}Cs fallout that might have been detected in a WBC [ORAUT 2018, pp. 16–17]. In addition to the guidance in Section 2.4.4 of OTIB-0060, if the intakes of other fission or activation product radionuclides were detected by a different bioassay method (e.g., ^{90}Sr in urine or $^{90}Sr/Y$ skull count) for the same monitoring period as the WBC, it should be assumed that the ^{137}Cs detected by the WBC is entirely from occupational sources.

5.3.2 Lung Counts

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

Table 5-19. Lung count MDAs by period.

Radionuclide	Period	In vivo MDA[a] (nCi)	Count time (min)	Sources
Ce-141	2001–present	0.11	60	BBI 2001
Ce-144	2001–present	0.44	60	BBI 2001
Eu-152	2001–present	0.18	60	BBI 2001
Ga-153	2001–present	0.096	60	BBI 2001
Th-234	2001–present	1.4	60	BBI 2001
U-235	1990–2000	0.2	Unknown	Rich 1990
U-235	2001–present	0.11	60	BBI 2001
DU/NU[b]	1989	3	60	Martin 1989; Grothaus 1993
Pu-238	1984–1988	16	Unknown	Guinn and Mansfield 1988
Pu-238	1989–1995	26	60	Martin 1989; Rich 1990; Grothaus 1993; Andersen et al. 1995
Pu-238	1996–present	54	60	BBI 2001
Pu-239/240	1974–1983	74	100	AEC 1974
Pu-239/240	1984–1988	20	Unknown	Guinn and Mansfield 1988
Pu-239/240	1989–1995	80	60	Martin 1989; Rich 1990; Grothaus 1993; Andersen et al. 1995
Pu-239/240	1996–present	140	60	BBI 2001
Am-241	1984–1988	0.15	Unknown	Guinn and Mansfield 1988
Am-241	1989–1995	0.6	60	Martin 1989; Rich 1990; Grothaus 1993; Andersen et al. 1995
Am-241	1996–present	0.14	60	BBI 2001

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

5.3.3 Thyroid Counts

At times, thyroid counts were performed on workers to assess potential intakes of radioactive iodine. Table 5-20 provides the known MDAs for this in vivo bioassay method.

Table 5-20. Thyroid count MDAs by period.

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

5.3.4 Skull Counts

At times skull counts were performed to assess potential intakes of ^{90}Sr/Y. The skull counts measured bremsstrahlung radiation that is emitted from the skull after the bone has had an uptake of radioactive strontium. Table 5-21 provides the MDAs for this in vivo bioassay method.

Table 5-21. Skull count MDAs by period.

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

a. MDA calculated from an inferred 2σ uncertainty.

5.3.5 Wound Counts

Wound counts have been performed at INL to assess potential intakes from contaminated wounds. Table 5-22 provides the known MDAs for this in vivo bioassay method.

Table 5-22. Wound count MDAs by period.[a]

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

a. Source: Grothaus [1993].

5.4 INTERFERENCES AND UNCERTAINTIES

5.4.1 Interferences

5.4.1.1 Contamination of Samples

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

5.4.1.2 Naturally Occurring Uranium in Uranium Bioassay Measurements

Uranium is present in secular equilibrium throughout the Earth. Consequently, there is a continuous source of intake of naturally occurring uranium via ingestion of drinking water and food and inhalation of particulate matter, which can interfere with bioassay measurements being used to detect intakes of uranium from occupational exposures. Because the uranium bioassay measurements performed for the CPP workers who were exposed to HEU were isotopic analyses, and because the mixture of uranium isotopes in recycled HEU is significantly different from natural uranium (NU), positive sample results attributable to naturally occurring uranium can be readily distinguished from results attributable to HEU. However, the kinetic phosphorescence analysis (KPA) that is performed on SMC Project urine samples cannot distinguish between different types of uranium (naturally occurring, depleted, or enriched). Therefore, the uranium in urine sample results for SMC Project workers have been adjusted to discount the interference from naturally occurring uranium. Consequently, the reported bioassay results in the worker files reflect the subtraction of 0.16 µg U/L from the value determined in the laboratory bioassay result [King 2001].

The 0.16 µg U/L adjustment value is based on measured background levels of uranium in urine samples submitted in 1987, 1994, and 1998 by SMC Project workers who were not radiological workers. Those measurements were assumed to represent the nonoccupational excretion levels of the SMC Project worker population because performing background measurements using members

of the public was not feasible. The background measurement results ranged from 0.04 to 0.33 µg U/L with wide fluctuations in individual measurements, some as high as 1.0 µg U/L [King 2001]. The average reported uranium concentration was 0.157 ±0.109 µg U/L at 1σ uncertainty. Therefore, 0.16 µg U/L is used as the nonoccupational component of uranium excretion for SMC Project workers and is subtracted from each urine result before assessment of occupational internal dose. ICRP Publication 23 lists the daily intake of naturally occurring uranium as 1.9 µg U/d and a daily urine excretion rate of 1.4 L for Reference Man [ICRP 1975]. Using a uranium intake of 1.9 µg U/d and a daily urine excretion rate of 1.4 L, the typical urinary concentration ranges between 0.4 and 0.5 µg U/L for naturally occurring uranium [King 2001]. Therefore, the adjustment value used at INL is consistent with the ICRP reference values for naturally occurring uranium.

5.4.1.3 Cesium-137 from Fallout

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

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

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

5.4.1.4 Intakes of Radionuclides for Medical Reasons

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

5.4.2 Uncertainties

The uncertainty values for all types of bioassay measurements are typically included in the bioassay records that are provided by the DOE. When measurement-specific uncertainty values are available, those values are preferred for the data analysis over generic values. When the uncertainty values are not included with the bioassay records, the uncertainty values for the data analysis should be determined in accordance with the recommendations in OTIB-0060.

5.5 INTAKE AND INTERNAL DOSE ASSESSMENT FOR MONITORED WORKERS

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

When bioassay data are used to assess the intakes and internal doses for monitored INL workers, the recommendations in OTIB-0060 should be followed. In addition, when workers have monitoring data for specific radionuclides, the radionuclide-specific data are the preferred data for estimating intakes and internal doses for those radionuclides. Because the majority of INL workers were only monitored for beta- and/or gamma-emitting radionuclides, and because the reported bioassay results typically do not provide any indication of the specific radionuclides involved with the intake or potential intake, intakes of specific radionuclides will typically need to be assigned based on ratios that are applied to the estimated ^{90}Sr and/or ^{137}Cs intakes. The details on when this needs to be done and on how to do it are provided in the following subsections.

5.5.1 Activation and Fission Product Intakes

Because the majority of the bioassay measurements performed for INL workers did not determine the potential mixtures of the activation and fission products that the workers were exposed to, and because those potential mixtures have a significant impact on the worker doses, a method for estimating the mixtures of the activation and fission products needed to be selected for the INL site. As a result, the method described in OTIB-0054 was determined to be appropriate and applicable for the INL site, because some INL reactor data were used to develop that method.

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

Section 3.0 of OTIB-0054 identifies two exclusions that were potentially applicable to CPP. The following paragraphs describe why those exclusions do not apply to CPP.

Even though RaLa work was performed at CPP between 1956 and 1963, the exclusion in OTIB-0054 does not apply to CPP because the irradiated nuclear fuel that was reprocessed at CPP as part of RaLa operations accounted for less than 1% of the total amounts of irradiated reactor fuel reprocessed at CPP during each of those years. Therefore, the radioactive materials associated with the RaLa operations only account for an insignificant fraction of the total amount of irradiated reactor fuels that were reprocessed at CPP during each of the years between 1956 and 1963, and those materials are not considered to be representative of the radioactive materials that the majority of workers were exposed to. However, there were a limited number of CPP workers who did receive intakes of radioactive materials due to the RaLa operations that were performed. Based on the bioassay results for those exposure incidents, the gamma spectrometry results indicated that the intakes from the RaLa operations were usually limited to radioactive isotopes of iodine versus a mixture of all the activation and fission products present in the source term for the RaLa operations. When the dosimetry records for a CPP worker indicate that they received an intake from the RaLa operations or when there are positive bioassay results during the years of 1956–1963 that include radioactive isotopes of iodine, those potential intakes should be assessed in accordance with OTIB-0060 and Section 5.5.4 below.

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

5.5.1.1 Assessment of Missed Intakes

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

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

5.5.1.2 Assessment of Fitted Intakes

Fitted intakes are assessed when there are positive bioassay measurement results. A bioassay measurement is considered to be positive when its result is greater than the reporting level (this could be the MDA, detection level, or some other value that the site used), and is considered to be negative when its result is less than or equal to the reporting level [ORAUT 2018].

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

Unfortunately, the radionuclides responsible for many positive urine sample results cannot be determined when only a gross radioactivity analysis was performed on them. In those instances, a ^{90}Sr or ^{137}Cs intake should be calculated in accordance with the recommendations provided in OTIB-0054 and OTIB-0060. Even though no information was provided in the dosimetry records to indicate that the positive urine sample results were attributable to either ^{24}Na or radioactive iodine, it may be possible to determine if a radioactive iodine or ^{24}Na intake occurred. Sections 5.5.4 and 5.5.6 provide guidance on how to determine if an intake was likely attributable to ^{24}Na or an isotope of radioactive iodine.

5.5.1.3 Assessment of Unmonitored Intakes

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

5.5.1.4 Decay Period Selections

The decay periods to be used for the selection of the appropriate urine activity fractions and associated radionuclide ratios for the intake estimates are to be based on a combination of the recommendations in Table 5-3 of OTIB-0054 and the available site-specific information.

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

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

For CPP, intake estimates up through 1992 should be assessed based on the 40-day, 180-day, and 1-year decay periods, and the decay period resulting in the highest internal dose for a given organ should be used for the dose assignment. Because CPP reprocessed irradiated fuels with varying decay periods, this approach is considered reasonable for all types of claims regardless of the compensability decision. At CPP some irradiated fuels were reprocessed with less than a 40-day decay period because of RaLa operations. However, a decay period shorter than 40 days is not justified because of the relatively small amount of reactor fuel involved with the RaLa operations. The RaLa operations performed at the CPP between 1957 and 1963 involved processing irradiated fuel with only a 2-day decay period; however, as indicated above, this fuel represented less than 1% of the fuel that was reprocessed during each of those years. Irradiated fuels were also stored at the CPP Fuel Storage Facility (Building CPP-603). Table 5-3 of OTIB-0054 recommends a 10-day decay period for spent fuel storage facilities when a more appropriate decay period is unknown. However, site-specific information indicates that, with the exception of the RaLa operations, the irradiated reactor fuels had a minimum of 90 days of cooling before being received at the CPP Fuel Storage Facility [Allied Chemical Corporation, no date]. Therefore, the decay period recommendations for CPP also apply to the Fuel Storage Facility. After CPP was shut down in 1992, only a decay period of 1-year needs to be assessed.

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

For exposures at the Central Facilities Area (CFA), intake estimates should be assessed based on the 40-day, 180-day, and 1-year decay periods, and the decay period resulting in the highest internal dose for a given organ should be used for the dose assignment. This is based on the available information for the CFA, which indicates that irradiated fuels with varying decay periods may have been present. Because there is no indication of irradiated fuels with a 10-day decay period being present at the CFA, this decay period does not need to be assessed for CFA intake estimates.

5.5.2 Unmonitored Actinide Intakes when MFPs were Present

In most instances at INL and ANL-W, the only actinides that the workers were exposed to were the actinides that were present in irradiated nuclear fuels (i.e., when the MFPs were also present). Because the much more readily detectable fission products were present along with the actinides in those instances, the potential intakes and doses attributable to those actinides can be estimated by applying ratios to the ^{90}Sr and/or ^{137}Cs intakes assessed for that worker.

As indicated in Section 5.1.3.3, internal doses can only be reconstructed during the ANL-W SEC period for which sufficient internal monitoring data are available to reconstruct those specific doses. Because this unmonitored actinide approach utilizes radionuclide ratios applied to ^{90}Sr and/or ^{137}Cs intakes, this approach can be used for partial dose reconstructions during the ANL-W SEC period when there are sufficient internal monitoring data to reconstruct a worker's internal doses to exposures to ^{90}Sr and/or ^{137}Cs.

As indicated in Section 5.1.3.3, internal doses from exposures to alpha-emitting radionuclides can only be reconstructed during the INL SEC periods for which internal monitoring data are available to reconstruct those specific doses. Because the SEC-00219 and SEC-00238 evaluations and subsequent DHHS designations said the CPP internal doses from exposures to alpha-emitting radionuclides cannot be bound, this unmonitored actinide approach should not be used for the workers with CPP external dosimetry during the CPP-only SEC periods or for the workers with INL external dosimetry during the all-INL SEC period. Refer to Section 5.1.3.3 for the applicable information for a given SEC period. Section 5.1.3.3 also indicates that environmental doses are feasible and can be assigned in a partial dose reconstruction. Therefore, for partial dose reconstructions, the environmental actinide intakes can be assigned during all three of the INL SEC periods.

Workers bioassayed for potential actinide intakes likely had a potential to be internally exposed to an actinide when the MFPs were not present. This also indicates a worker had a higher potential for internal exposure to an actinide. For the periods when a worker was monitored for actinide intakes, the actual bioassay results should be used to estimate internal doses for those actinide exposures instead of using this approach.

Actinide to ^{90}Sr and actinide to ^{137}Cs activity ratios are provided in Tables 5-23 and 5-24 for the three major irradiated nuclear fuel types that were present at INL. Tables 5-23 and 5-24 also identify the maximum ratio for a given actinide, which is needed when exposures to multiple fuel types were possible or when the fuel type is unknown. Table 5-25 identifies the actinide isotopes to use for the dose calculations. The actinide isotopes listed in Table 5-25 are the predominant alpha-emitting actinides in the source term for a given fuel type.

Table 5-23. Activity ratios of actinides to ^{90}Sr by reactor fuel type.[a]

Actinide	Aluminum	Zirconium	Stainless steel	Maximum
Ac	8.0E−12	1.3E−11	2.3E−10	2.3E−10
Th	2.4E−08	6.4E−08	2.3E−07	2.3E−07
Pa	1.2E−10	1.1E−10	3.8E−09	3.8E−09
U	5.6E−05	6.2E−06	1.4E−03	1.4E−03
Np	3.4E−06	3.7E−06	6.8E−07	3.7E−06
Pu	8.7E−03	1.5E−02	3.7E−03	1.5E−02
Am	1.4E−04	3.9E−06	9.0E−08	1.4E−04
Cm	4.9E−05	1.8E−06	1.1E−10	4.9E−05

a. Source: The values in this table are from the spreadsheet *INEL – Actinide Ratios* [ORAUT 2009a].

Table 5-24. Activity ratios of actinides to ^{137}Cs ratios by reactor fuel type.[a]

Actinide	Aluminum	Zirconium	Stainless steel	Maximum
Ac	7.6E−12	1.3E−11	2.1E−10	2.1E−10
Th	2.3E−08	6.2E−08	2.1E−07	2.1E−07
Pa	1.2E−10	1.1E−10	3.5E−09	3.5E−09
U	5.3E−05	6.0E−06	1.3E−03	1.3E−03
Np	3.2E−06	3.5E−06	6.2E−07	3.5E−06
Pu	8.3E−03	1.4E−02	3.4E−03	1.4E−02
Am	1.3E−04	3.7E−06	8.3E−08	1.3E−04
Cm	4.7E−05	1.7E−06	1.0E−10	4.7E−05

a. Source: The values in this table are from the spreadsheet *INEL – Actinide Ratios* [ORAUT 2009a].

Table 5-25. Actinide isotope assumptions for each reactor fuel type.[a]

Actinide	Aluminum	Zirconium	Stainless steel	Maximum
Ac	Ac-227	Ac-227	Ac-227	Ac-227
Th	Th-228	Th-228	Th-228	Th-228
Pa	Pa-231	Pa-231	Pa-231	Pa-231
U	U-234	U-236	U-234	U-234
Np	Np-237	Np-237	Np-237	Np-237
Pu	Pu-238	Pu-238	Pu-239	Pu-238
Am	Am-241	Am-241	Am-241	Am-241
Cm	Cm-244	Cm-244	Cm-242	Cm-244

a. Source: The appropriate actinide isotope assumptions are from the spreadsheet *INEL – Actinide Ratios* [ORAUT 2009a].

For the INL reactor facilities, the actinide ratios for aluminum fuels should be used. Even though some INL reactors did not use aluminum fuel types, the aluminum fuel actinide ratios for those reactors likely provide a reasonable overestimate of the actinides in the other fuel types, because the fuels from non-aluminum-fueled reactors typically had much lower burnups than what was assumed for the aluminum fuel calculations.

For ANL-W reactor facilities, the actinide ratios for stainless-steel fuels should be used.

For CPP, the actinide ratios for the aluminum fuels likely provide a reasonable overestimate of the actinides present before 1971, because the nonaluminum fuels that were reprocessed at CPP before 1971 had much lower burnups than what was assumed for the aluminum fuels. Because CPP was reprocessing a combination of different reactor fuels that had higher burnups by 1971, and because CPP workers might have been simultaneously exposed to an unknown mixture of those fuel types, the maximum ratio between the three major fuel types (i.e., aluminum, zirconium, and stainless steel) for a given actinide should be used for CPP workers after 1970. As discussed above and in Section 5.1.3.3, during the INL SEC periods, no unmonitored actinide intakes from this approach should be assigned to account for potential internal actinide exposures received at the CPP.

For INL waste management facilities and CFA, the maximum actinide ratios should be used for the intake estimates.

The radioactive source terms used to generate the values in Tables 5-23 and 5-24 were based on information provided in *Determination of the Normalized Mass of Individual Radionuclides in the Dissolver Product for Aluminum, Zirconium and Stainless Steel Fuels Previously Processed at INTEC* [Wenzel 2000]. This document indicated that the source term values were generated using the ORIGEN2 computer program and that calculations were only performed for the three main types of irradiated reactor fuel that were reprocessed at CPP (also known as INTEC). In addition, only a single source term calculation was performed for each of the reactor fuel types using the typical composition, configuration, nuclear fuel burnup, and postirradiation decay periods that were applicable to those reactor fuels. Because no uncertainty information was provided for the ORIGEN2 calculations, the source term information and the actinide ratios that were calculated for this TBD will be treated as constants.

Because the number of favorable to claimant simplifying approaches or assumptions used to develop the ratios for the actinides was minimal, the distribution(s) associated with the indicator radionuclide intakes will need to be accounted for. Therefore, the distribution associated with the unmonitored actinide intakes and internal doses should be indicative of the distributions associated with the indicator radionuclide intakes that they were based on. For a fitted dose calculation, the distribution should be lognormal with a geometric standard deviation of 3.0 [ORAUT 2018]. For a missed dose calculation, the distribution should be triangular [ORAUT 2018].

5.5.3 Contaminant Intakes for Chemical Processing Plant Workers Who Handled Recycled Highly Enriched Uranium

If a CPP worker has bioassay data for uranium, the worker likely handled recycled HEU that was separated from the irradiated reactor fuels. Because nearly all of the HEU at the CPP was recycled uranium, the HEU intakes also need to account for the intakes of the dosimetrically significant radioactive contaminants that were also likely present and not directly monitored for. This section provides the basis and guidance for estimating the intakes of those radioactive contaminants.

As indicated in Section 5.1.3.3, internal doses from exposures to alpha-emitting radionuclides can only be reconstructed during the INL SEC periods for which internal monitoring data are available to reconstruct those specific doses. Because this section applies only to CPP workers with bioassay data for uranium, the uranium doses based on that internal monitoring data are not affected by the INL SECs. Additionally, the dose infeasibility for CPP does not apply to the following approach that is being used to account for the unmonitored intakes of radioactive contaminants in the HEU. For the periods when a worker was monitored for intakes of any of the radioactive contaminants while also being monitored for HEU intakes (e.g., periods when the worker has both uranium and plutonium bioassay data), the actual bioassay results should be used to estimate intakes for those radioactive contaminants instead of the following approach.

Because the majority of the CPP HEU product was shipped to the Y-12 Plant, the activity ratios for CPP HEU are based on information provided for recycled uranium in ORAUT-TKBS-0014-5, *Y-12 Plant – Occupational Internal Dose* [ORAUT 2023]. The default ratios to estimate the unmonitored radioactive contaminant intakes are provided in Table 5-26. The unmonitored radioactive contaminant intakes are calculated by multiplying the default ratios by the total uranium intake that was calculated based on an assessment of the bioassay data.

Table 5-26. Default ratios for recycled HEU intakes.[a]

Radionuclide	Default ratio (pCi/µg total U)
Tc-99	1.9
Th-228	0.05
Np-237	0.06
Pu-238	0.02

a. Source: The values in this table are based on Table 5-8 of the Y-12 Plant TBD [ORAUT 2023]. The values have also been adjusted in accordance with that TBD for use on all types of cases, including best estimate cases.

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

5.5.4 Radioactive Iodine Intakes

5.5.4.1 General Guidance on Identified Radioactive Iodine Intakes

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

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

Isotope-specific information provided with positive bioassay results should be used to assess the intakes that are attributable to radioactive iodine. If multiple iodine isotopes are identified and no information regarding the specific mixture is provided, the bioassay results should be assessed assuming that the intake was attributable solely to each iodine isotope and the isotope generating the highest dose estimate should be used for the assigned dose. When there is no isotopic information provided for a radioactive iodine intake, the dose reconstructor should assume that the iodine isotope is solely ^{131}I, unless there is information to indicate that the worker's intake occurred during RaLa operations at CPP.

If a worker's radioactive iodine intake occurred during RaLa operations at CPP, the isotopic mixture provided in Table 5-27 should be used to assess the radioactive iodine intakes and doses. The isotopic fractions for the iodine isotopes provided in Table 5-27 are based on an ORIGEN computer program run for the MTR fuel with a 2-day decay period, which was the fuel type and typical decay period used for the RaLa operations. The isotopes listed in Table 5-27 account for more than 99% of the iodine radioactivity.

Table 5-27. RaLa iodine.

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

5.5.4.2 Special Considerations for Thyroid Cancer Cases

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

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

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

Because OTIB-0054 does not provide ^{129}I-to-indicator nuclide ratios and because the aged iodine is likely in a less volatile form, an approach similar to the approach being used for assessing unmonitored actinide intakes in Section 5.5.2 should be used to assess unmonitored ^{129}I intakes. Table 5-28 provides ^{129}I ratios for indicator radionuclides that can be used to estimate the unmonitored ^{129}I intakes when the applicable decay period exceeds 40 days. Because elemental iodine is very reactive and has had over 40 days to react with other materials, ^{129}I should be assessed as an inhalation intake and a material having Type F lung absorption properties.

Table 5-28. ^{129}I-to-indicator nuclide ratios.[a]

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

a. Source: ORAUT [2009b].

The radioactive source term that was used to generate the values in Table 5-28 was based on the information in Wenzel [2000]. This document indicated that the source term values were generated using the ORIGEN2 computer program and that calculations were only performed for the three main types of irradiated reactor fuel that were reprocessed at CPP. In addition, only a single source term calculation was performed for each of the reactor fuel types using the typical composition, configuration, nuclear fuel burnup, and postirradiation decay periods that were applicable to those reactor fuels. Because no uncertainty information was provided for the ORIGEN2 calculations, the source term information and the actinide ratios that were calculated for this TBD will be treated as constants.

Because the number of favorable to claimant simplifying approaches or assumptions used to develop the ratios for ^{129}I was minimal, the distribution(s) associated with the indicator radionuclide intakes will need to be accounted for. Therefore, the distribution associated with the unmonitored ^{129}I intakes and internal doses should be indicative of the distributions associated with the indicator radionuclide intakes they were based on. For a fitted dose calculation, the distribution should be lognormal with a geometric standard deviation of 3.0 [ORAUT 2018]. For a missed dose calculation, the distribution should be triangular [ORAUT 2018].

5.5.5 Uranium Intakes for Specific Manufacturing Capability Project Workers

The SMC Project was located at TAN at the old ANP Program facilities [ORAUT 2010a]. In 1985, the SMC Project started manufacturing DU armor for the U.S. Army [ORAUT 2010a]. As of June 2000, records show that the SMC Project has received 10,129,000 lb of DU for processing [Lewis et al. 2000]. Of this, 4,726,000 lb were received from the DOE Feed Materials Production Center in Fernald, Ohio, and 5,403,000 lb were received from RFP [Lewis et al. 2000]. The records also indicate that the

DU being used by the SMC Project was recycled DU and contained other radioactive impurities [Lewis et al. 2000].

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

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

The uranium values provided in Table 5-29 are representative of the typical composition of recycled DU. Based on the Table 5-29 information, the specific activity of the DU at the SMC Project is 0.402 pCi/µg DU. Average values for the radiologically significant impurities in the SMC Project DU are provided in Table 5-30.

Table 5-29. Uranium mass and activity ratios for SMC Project DU.[a]

Uranium isotope	Mass %	Activity %
U-234	0.0010	15.46
U-235	0.1991	1.07
U-236	0.0003	0.05
U-238	99.7996	83.42

a. Source: The values in this table are the default values for DU that are provided in the IMBA computer program because the site-specific information [King 2001] did not account for the U-236 in recycled DU.

Table 5-30. SMC Project DU average impurity levels.[a]

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

a. Source: Lewis et al. [2000].

Inhalation Absorption Type

Respirable particulates associated with SMC Project operations are probably a mixture of metal and metal oxides. The actual exposures are undoubtedly due to mixtures of absorption types. During the 18 years of operation, much bioassay data have been collected on a large number of individuals. The overall elimination patterns are consistent with Type M but probably are a mixture of all types. It could be too simplistic to assume a pure absorption type when the chemical form is not known for certain. The dose reconstructors should assume either Type M or Type S uranium to maximize the dose to the organ of concern. Exposure to significant quantities of Type F uranium at the SMC Project is not considered credible [34].

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 [BBI 2001]. This default particle size is applicable to both the DU intakes and the intakes for the impurity radionuclides.

5.5.6 Intakes for Radioactive Waste Management Center Workers

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

The comprehensive radiation protection program for RWMC includes extensive air monitoring, personnel monitoring, and surface contamination surveillance. Although infrequent, there have been instances of inadvertent intakes. Therefore, bioassay is conducted randomly at the present time [BBI 2001, p. 88-89].

Because of the variety of radiological materials that were present at RWMC, more than one mixture of radioactive materials may need to be considered for the intake and internal dose calculations for RWMC workers. For RWMC workers who were likely exposed to materials bearing irradiated fuels or for instances when the material that the worker was exposed to was unknown, the guidance in Sections 5.5.1 and 5.5.2 should be followed.

Tables 5-31 and 5-32 summarize the major radionuclides in the RWMC waste inventory for the SDA and TSA, respectively. TRU radionuclides are the primary contaminants in the TSA waste; all but ^{241}Pu are alpha emitters. Because the materials are not homogeneous, it should not be assumed that failing to detect one radionuclide in the inventory invalidates detection of other radionuclides.

Table 5-31. Radioactive low-level waste inventory in the active pits in the SDA.[a]

Radionuclide	Concentration (Ci/m^3)	Percentage
Co-60	4.1	92.00%
Ni-63	3.3E−01	7.40%
Sr-90	9.7E−03	0.22%
Cs-137	9.7E−03	0.22%
H-3	5.8E−03	0.13%
C-14	8.9E−04	0.02%

a. Source: BBI [2001] which indicated a total of 75,600 m^3 with total activity of 3.35E+05 Ci.

Table 5-32. Stored contact-handled TRU waste inventory in the TSA.[a]

Radionuclide	Concentration (Ci/m^3)	Percentage
Pu-241	2.5	44.1%
Am-241	1.4	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%

a. Source: BBI [2001] which indicated a total of 65,000 m^3 with total activity of 4.06E+05 Ci.

5.5.7 Intakes for Waste Reduction Operations Complex Workers

The Waste Reduction Operations Complex (WROC) includes several reactor facilities that operated from the 1950s to the late 1960s along with 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, radiation area monitors, surface and personnel contamination surveillance, and effluent monitors.

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

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

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

Sodium metal would not have been in areas routinely accessed by personnel until most of the ^{24}Na had decayed due to the intense gamma radiation from the ^{24}Na. Given that the half-life of ^{24}Na is only 15 hours, a decay period of only about 6.25 days would be required to eliminate nearly all of the ^{24}Na in the sodium metal. Significant intakes of ^{24}Na from sodium metal in the solid or liquid forms are also unlikely due to its reduced dispersion potential in those forms. It is also unlikely that unknown accidental exposures to ^{24}Na would have occurred, because sodium metal is highly reactive and reacts violently with water. Because of the significant hazards associated with sodium metal, any exposures to the metal would have been minimized and would have also been known when they occurred. When an exposure was known to have occurred, the radiological protection practices being

performed throughout the INL would have likely required that bioassay measurements capable of detecting ^{24}Na intakes be performed on the exposed workers. Therefore, no special assessments need be performed for workers who were not monitored for ^{24}Na exposures at the fast reactors.

If it can be determined that a worker received an intake of ^{24}Na or was monitored for a ^{24}Na exposure, fitted and/or missed ^{24}Na intakes and doses might need to be assessed. When bioassay measurements were used to monitor a worker for ^{24}Na exposures, the potential ^{24}Na intakes and doses should be assessed in accordance with OTIB-0060.

5.6 INTAKE AND INTERNAL DOSE ASSESSMENT FOR UNMONITORED WORKERS

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

For most of the history of INL, personnel dosimeters were issued to all workers who entered the security access control points at each facility regardless of their work assignment. For example, administrative and clerical personnel were required to wear dosimeters even though they were not exposed to external or internal doses that were above the onsite ambient levels. Dose reconstructors should determine the appropriate unmonitored dose assignment in accordance with the following recommendations:

- If the worker's file includes positive external dosimeter readings during a given calendar year that the worker was not monitored for internal dose, the worker should be treated as a radiation worker and a default missed internal dose (i.e., internal missed doses based on hypothetical bioassay data) should be assessed. The default missed internal dose is assessed by assuming that the most likely type of bioassay measurement was performed on the last day of the calendar year with a positive external dose and that bioassay measurement was negative. For the years from INL's startup through 1960 the most likely type of bioassay method was a urine sample analyzed for gross beta radioactivity; after 1960 the most likely type of bioassay method was a WBC. The missed intake and dose calculations are the same as described in Sections 5.5.1 and 5.5.2 above. For readily noncompensable claims, a simpler but favorable to claimant alternative to assigning a default missed internal dose is to assign ORAUT-OTIB-0018, *Internal Dose Overestimates for Facilities with Air Sampling Programs* [ORAUT 2022], based internal doses.

  This approach can be used for the INL SEC periods, because no infeasibilities were identified for the internal doses attributable to MFAPs. However, as indicated in Sections 5.1.3.3 and 5.5.2, the unmonitored actinide portion of this approach should not be used during the INL SEC periods for the specified worker categories. Based on the guidance in OTIB-0018 for SEC periods, the simplifying OTIB-0018 approach can still be used during the INL SEC periods for the radionuclides that were not designated as infeasible [ORAUT 2022, p. 17]. Because of the infeasibility for the internal doses from alpha-emitting radionuclides, the OTIB-0018 exposure type options for alpha-emitting radionuclides should not be used during the INL SEC periods [ORAUT 2022, p. 17; NIOSH 2017a, p. 248, 251; NIOSH 2017b, p. 41]. Again, this is only applicable when the worker falls within the worker category for a given SEC period. Refer to Section 5.1.3.3 for the applicable information for a given SEC period.

  As indicated in Section 5.1.3.3, internal doses can only be reconstructed during the ANL-W SEC period for which sufficient records are available to reconstruct those specific doses. Because this approach is an unmonitored dose approach that utilizes hypothetical bioassay data to estimate the intakes of MFAPs, it should not be used during the ANL-W SEC period. The OTIB-0018 approach should not be used during the ANL-W SEC period, because the

infeasibility for internal dose included all radionuclides [ORAUT 2022, p. 17; NIOSH 2016, p. 215].

- If a worker was not monitored for internal dose and had no detectable external doses reported for a given calendar year, only the environmental internal doses need to be assessed for that calendar year. The environmental internal doses should be assessed in accordance with the guidance in ORAUT-TKBS-0007-4, *Idaho National Laboratory and Argonne National Laboratory - West – Occupational Environmental Dose* [ORAUT 2010d]. Because no infeasibilities were identified for the environmental internal doses, those doses can still be assessed during the SEC periods, as indicated in Section 5.1.3.3.

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

5.7 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.

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 TBD, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source.

Bryce Rich served as one of the initial Subject Experts for this document. Mr. Rich was previously employed at 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. Rich, those materials are fully attributed to the source.

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. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports.

[2] Rich, Bryce. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports.

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

[4] Rich, Bryce. ORAU Team. Consulting Health Physicist. 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.

[5] Peterson, Henry. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on experience as a radiological engineer at TRA.

[6] Peterson, Henry. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on experience as a radiological engineer at TRA.

[7] Peterson, Henry. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on experience as a radiological engineer at TRA.

[8] Jenkins, JoAnn M. ORAU Team. Senior Health Physicist. 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.

[9] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement 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.

[10] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement 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.

[11] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
The fuel at CPP 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 5-19 that demonstrate that most of the isotopes were beta and gamma emitters.

[12] Rich, Bryce. ORAU Team. Consulting Health Physicist. 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.

[13] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on Hayden [1958].

[14] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on knowledge of NRF from working at INL. Because NRF had operating reactors, the potential for internal exposure existed and, therefore, individuals who worked at the facility could have received internal doses.

[15] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on 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.

[16] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on 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.

[17] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports. It is further supported by Stacy [2000].

[18] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports. It is further supported by Stacy [2000].

[19] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports. It is further supported by Stacy [2000].

[20] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement 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].

[21] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This conclusion is based on a review of Andersen et al. [1995], BBI [2001], King [1990] and NIOSH [1994].

[22] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on experience as a member of the H&S management team that supported the INL internal dosimetry program and is supported by NIOSH [1994].

[23] Rich, Bryce. ORAU Team. Consulting Health Physicist. 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 INL, 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.

[24] Rich, Bryce, and Jenkins, JoAnn M. ORAU Team. Consulting Health Physicist and Senior Health Physicist. 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 internal dose. The statement about the equivalency of the bioassay programs at the different facilities is based on 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.

[25] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement 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.

[26] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports and internal dosimetry TBDs.

[27] Rich, Bryce. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports.

[28] Rich, Bryce. ORAU Team. Principal Health Physicist. November 2004.
This statement is based on a review of the referenced AEC annual reports.

[29] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of facility bioassay data and is supported by the referenced AEC annual reports and internal dosimetry TBDs.

[30] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of facility bioassay data and is supported by NIOSH [1994].

[31] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on information from an extensive review of INL bioassay records.

[32] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of facility bioassay data and is supported by the referenced AEC annual reports and internal dosimetry TBDs.

[33] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of dosimetry records. It is also supported by INL incident reports. It is common industry practice to conduct incident investigations in a timely manner.

[34] Rich, Bryce. ORAU Team. Consulting Health Physicist. November 2004.
This statement is based on a review of the *INEEL M&O Contractor Technical Basis for Internal Dosimetry* [BBI 2001].

REFERENCES

10 C.F.R. 835, [1999]. Occupational radiation protection. [SRDB Ref ID: 165859]

42 C.F.R. 83, [2023]. Procedures for designating classes of employees as members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000. [SRDB Ref ID: 201527]

ACC [1952]. CPP health physics manual. National Reactor Testing Station, Idaho Falls, ID: American Cyanamid Company. IDO 14055, April. [SRDB Ref ID: 8095]

AEC [1954–1960]. Maps of INEL areas. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. [SRDB Ref ID: 117427]

AEC [1957–1975]. National Reactor Testing Station plot plans. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. [SRDB Ref ID: 86891]

AEC [1958]. Permissible levels of radiation exposure. Chapter 0524. In: AEC manual. Washington, DC: U.S. Atomic Energy Commission. February 1. [SRDB Ref ID: 10678]

AEC [1960]. Annual report of the Health and Safety Division 1959. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12014, October. [SRDB Ref ID: 8080]

AEC [1961]. Annual report of the Health and Safety Division 1960. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12019, September. [SRDB Ref ID: 8078]

AEC [1963]. Standards for radiation protection. Chapter 0524. In: AEC manual. Washington, DC: U.S. Atomic Energy Commission. August 12. [SRDB Ref ID: 13037]

AEC [1968]. Standards for radiation protection. Chapter 0524. In: AEC manual. Washington, DC: U.S. Atomic Energy Commission. November 8. [SRDB Ref ID: 10668]

AEC [1972]. 1971 annual report of the Health Services Laboratory. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12076. [SRDB Ref ID: 1431]

AEC [1974]. Detection limits. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. December. [SRDB Ref ID: 13448]

AEC [1975]. Occupational radiation exposure information. Chapter 0525. In: AEC manual. Washington, DC: U.S. Atomic Energy Commission. January 7. [SRDB Ref ID: 11594]

Allied Chemical Corporation [no date]. Idaho Chemical Processing Plant. Idaho National Engineering Laboratory, Idaho Falls, ID. [SRDB Ref ID: 29247]

Andersen BL, Perry OR, Ruhter PE [1995]. Technical basis for internal dosimetry at LITCO. Idaho National Engineering Laboratory, Idaho Falls, ID: EG&G Idaho. RDRU-95-3.0, January 31. [SRDB Ref ID: 13453]

Aoki I [1979]. Reporting of dose or dose commitment from internal sources. Memorandum to Dosimetry File. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. July 9. [SRDB Ref ID: 13455]

Azar AM II [2019]. U.S. Department of Health and Human Services designation of additional members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation

Program Act of 2000 designating a class of employees from Idaho National Laboratory (INL) Scoville, Idaho. Washington, DC: U.S. Department of Health and Human Services, Office of the Secretary. June 21. [SRDB Ref ID: 192249]

BBI [2000]. Quality assurance project plan for the Analytical Laboratories Department Radioanalytical Section. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. PLN-153 Rev. 1, August 17. [SRDB Ref ID: 8540]

BBI [2001]. INEEL M&O contractor technical basis for internal dosimetry. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. INEEL-2001-025, May 30. [SRDB Ref ID: 8479]

Bhatt AR [2002]. Bioassay sample analysis technical basis document. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. EDF-In-Vitro-1 Rev. 0, August 21. [SRDB Ref ID: 7981]

Bhatt AR [2003]. In vitro analysis measurement techniques. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. INEEL/INT-03-00702 Rev. 0, September. [SRDB Ref ID: 8414]

Bodnar LZ, Percival DR, eds. [1982]. RESL Analytical Chemistry Branch procedures manual. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. IDO-12096. [SRDB Ref ID: 16103]

Burwell SM [2016a]. U.S. Department of Health and Human Services designation of additional members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000 designating a class of employees Idaho National Laboratory (INL) Scoville, Idaho. Washington, DC: U.S. Department of Health and Human Services, Office of the Secretary. June 3. [SRDB Ref ID: 192250]

Burwell SM [2016b]. U.S. Department of Health and Human Services designation of additional members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000 designating a class of employees Argonne National Laboratory-West Scoville, Idaho. Washington, DC: U.S. Department of Health and Human Services, Office of the Secretary. June 3. [SRDB Ref ID: 192248]

Casto WR, ed. [1980]. ICPP criticality event of October 17, 1978. Nuclear Safety *21*(5):648–653. [SRDB Ref ID: 1218]

Cipperley FV [1958]. Personnel metering operating techniques self-service system effective August 20, 1951 through March 29, 1958. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12007. [SRDB Ref ID: 8112]

Cipperley FV [1968]. Personnel dosimetry standard operating procedure effective March 30, 1958. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12070, December. [SRDB Ref ID: 8086]

Dickson RL [1977]. Reporting levels for whole-body counting. Memorandum to L.Z. Bodnar and D.G. Olson. Idaho Falls, ID: U.S. Energy Research and Development Administration, Idaho Operations Office. July 15. [SRDB Ref ID: 7976]

Dodd AO, ed. [1963]. Health and Safety Division annual progress report 1962. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12033. [SRDB Ref ID: 8075]

Dodd AO, ed. [1964]. Health and Safety Division annual progress report 1963. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12037. August. [SRDB Ref ID: 13449]

DOE [1988]. Radiation protection for occupational workers. Washington, DC: U.S. Department of Energy, Office of Environment, Safety and Health. DOE 5480.11, December 21. [SRDB Ref ID: 8073]

DOE [1991]. Idaho National Engineering Laboratory historical dose evaluation. 2 vols. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. DOE/ID-12119, August. [SRDB Ref ID: 1107]

DOE [1994]. Radiological control manual. Washington, DC: U.S. Department of Energy, Office of Assistant Secretary for Environment, Safety and Health. DOE/EH-0256T Rev. 1, April. [SRDB Ref ID: 14856]

DOE [1997]. Linking legacies connecting the Cold War nuclear weapons production processes to their environmental consequences. Washington, DC: U.S. Department of Energy, Office of Environmental Management. DOE/EM-0319, January. [SRDB Ref ID: 11930]

Ebersole ER [1956]. Analysis of stack and field samples during IET operations June 1956. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12009, July. [SRDB Ref ID: 8393]

Ebersole ER, Flygare JK Jr. [1957]. Determination of gross beta activity in urine with K-40 correction. In: Proceedings of the Health Physics Society first annual meeting June 25-27, 1956 University of Michigan. Ann Arbor, Michigan, pp. 72–78. [SRDB Ref ID: 193026]

ERDA [1975]. Standards for radiation protection. Chapter 0524. In: ERDA manual. Washington, DC: U.S. Energy Research and Development Administration. April 8. [SRDB Ref ID: 8083]

ERDA [1977]. Final environmental impact statement waste management operations Idaho National Engineering Laboratory Idaho. Washington, DC: U.S. Energy Research and Development Administration. ERDA-1536, September. [SRDB Ref ID: 1143]

Gammill WP [1955]. Phillips personnel metering procedure at the NRTS. Memorandum to J.W. McCaslin. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. November 9. [SRDB Ref ID: 141721]

Ginkel WL, Bills CW, Dodd AO, Kennedy KK, Tingey FH [1960]. Nuclear incident at the Idaho Chemical Processing Plant on October 16, 1959 report of the investigating committee. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-10035, February 15. [SRDB Ref ID: 12784]

Grothaus GE [1993]. MDAs for in-vivo counting. Memorandum to C. Bloom, EG&G. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. February 19. [SRDB Ref ID: 7978]

Guinn L, Mansfield WG [1988]. Internal dosimetry procedures manual. Idaho National Engineering Laboratory, Idaho Falls, ID: Westinghouse Idaho Nuclear Company. July 14. [SRDB Ref ID: 13435]

Hargon ED [2017]. U.S. Department of Health and Human Services designation of additional members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000 designating a class of employees Idaho National Laboratory Scoville, Idaho. Washington, DC: U.S. Department of Health and Human Services, Office of the Secretary. November 22. [SRDB Ref ID: 192253]

Hayden RE [1958]. I-131 incident report Ha-46-58A March 20, 1958. Memorandum to J.P. Lyon. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. March 26. [SRDB Ref ID: 8010]

Henry RN, Slagle WL [1985]. Report of MORT investigation of N-Cell incident. Memorandum to W.C. Mallory. Idaho National Engineering Laboratory, Idaho Falls, ID: Westinghouse Idaho Nuclear Company. April 30. [SRDB Ref ID: 8424]

Hoff DL, Mitchell RG, Moore R [1989]. The Idaho National Engineering Laboratory site environmental report for CY1988. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. DOE/ ID-12082(88), June. [SRDB Ref ID: 193027]

Horan JR, ed. [1958]. Annual report of the Health and Safety Division 1958. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12012. [SRDB Ref ID: 8079]

Horan JR, ed. [1962]. Health and Safety Division annual report, 1961. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12021, May 21. [SRDB Ref ID: 8076]

Horan JR, Braun JB [1993]. Occupational radiation exposure history of Idaho Field Office operations at the INEL. Idaho National Engineering Laboratory, Idaho Falls, ID: EG&G Idaho. EGG-CS-11143, October. [SRDB Ref ID: 1023]

HPS [1996]. HPS N13.30-1996 An American national standard – performance criteria for radiobioassay. McLean, VA: Health Physics Society. May. [SRDB Ref ID: 167246]

ICPP internal dose assessment manual [1981]. Idaho National Engineering Laboratory, Idaho Falls, ID. March. [SRDB Ref ID: 1223]

ICRP [1975]. ICRP 23 - Report of the task group on reference man. Oxford, England: Pergamon Press. ICRP Publication 23. [SRDB Ref ID: 22735]

ICRP [1979]. ICRP 30 Part 1 - Limits for intakes of radionuclides by workers. ICRP Publication 30. Ann ICRP *2*(3/4). [SRDB Ref ID: 29220]

ICRP [1995]. ICRP 68 - Dose coefficients for intakes of radionuclides by workers. ICRP Publication 68. Ann ICRP *24*(4). [SRDB Ref ID: 22731]

INEEL [2002]. Analytical chemistry methods manual, Vol. VI (ACMM-VI). Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID. TOC-ACMM-VI Rev. 13, September 11. [SRDB Ref ID: 13456]

King EA [1990]. WINCO internal dose technical basis document. Idaho National Engineering Laboratory, Idaho Falls, ID: Westinghouse Idaho Nuclear Company. December. [SRDB Ref ID: 26111]

King VA [2001]. Technical basis for internal dosimetry at the Specific Manufacturing Capability Project (SMC). Idaho National Environmental and Engineering Laboratory, Idaho Falls, ID. SMC-2001-02 Rev. 03, May 31. [SRDB Ref ID: 193029]

Lewis LC, Barg DC, Bendixsen CL, Henscheid JP, Wenzel DR, Denning BL [2000]. Idaho National Engineering and Environmental Laboratory site report on the production and use of recycled uranium. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. INEEL/EXT-2000-00959, September. [SRDB Ref ID: 16495]

Martin DB [1989]. MDAs for in vivo counting. Memorandum to C. Bloom, EG&G Idaho. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. November 16. [SRDB Ref ID: 7977]

NCRP [1987]. NCRP 94 - Exposure of the population in the United States and Canada from natural background radiation. Bethesda, MD: National Council on Radiation Protection and Measurements. NCRP Report No. 94, December 30. [SRDB Ref ID: 22736]

Nertney RJ, Brown ES, Buckham JA, Weber LJ, Jenson DK [1967]. The GA-18-1 contamination incident at the TRA National Reactor Testing Station June 13, 1967. National Reactor Testing Station, Idaho Falls, ID: Idaho Nuclear Corporation. Ne-312-67, July 10. [SRDB Ref ID: 8381]

NIOSH [1994]. History of the INEL bioassay programs. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. [SRDB Ref ID: 14664]

NIOSH [2002]. Internal dose reconstruction implementation guideline. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. OCAS-IG-002 Rev. 0, August. [SRDB Ref ID: 22402]

NIOSH [2010]. Radiation exposures covered for dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. DCAS-IG-003 Rev. 1, October 5. [SRDB Ref ID: 88929]

NIOSH [2016]. Special Exposure Cohort petition evaluation report petition SEC-00224 Rev. 0 Argonne National Laboratory-West qualified March 13, 2015. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. Petition SEC-00224 Rev. 0, February 18. [SRDB Ref ID: 156483]

NIOSH [2017a]. Special Exposure Cohort petition evaluation report petition SEC-00219 Rev. 2 Idaho National Laboratory qualified September 16, 2014. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. Petition SEC-00219 Rev. 2, February 22. [SRDB Ref ID: 166679]

NIOSH [2017b]. Special Exposure Cohort petition evaluation report petition SEC-00238 Rev. 0 Idaho National Laboratory qualified March 22, 2017. Cincinnati, OH: U.S. Department of Health and Human Services, Centers for Disease Control and Prevention, National Institute for Occupational Safety and Health. Petition SEC-00238 Rev. 0, July 20. [SRDB Ref ID: 192254]

ORAUT [2009a]. INEL – actinide ratios. Spreadsheet. Oak Ridge, TN: Oak Ridge Associated Universities Team. Rev. 0, December 18. [SRDB Ref ID: 77440]

ORAUT [2009b]. INEL – I-129 ratios for internal TBD. Spreadsheet. Oak Ridge, TN: Oak Ridge Associated Universities Team. Rev. 0, December 18. [SRDB Ref ID: 77441]

ORAUT [2010a]. Idaho National Laboratory and Argonne National Laboratory-West – site description. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0007-2 Rev. 04, August 2. [SRDB Ref ID: 84106]

ORAUT [2010b]. Compiled urine data. Spreadsheet. Oak Ridge, TN: Oak Ridge Associated Universities Team. Rev. 0, January 3. [SRDB Ref ID: 77692]

ORAUT [2010c]. Compiled whole body count. Spreadsheet. Oak Ridge, TN: Oak Ridge Associated Universities Team. Rev. 0, January 3. [SRDB Ref ID: 77693]

ORAUT [2010d]. Idaho National Laboratory and Argonne National Laboratory-West – occupational environmental dose. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0007-4 Rev. 02, January 8. [SRDB Ref ID: 78635]

ORAUT [2011]. Idaho National Laboratory and Argonne National Laboratory-West - occupational external dosimetry. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0007-6 Rev. 03, April 19. [SRDB Ref ID: 94104]

ORAUT [2015]. Fission and activation product assignment for internal dose-related gross beta and gross gamma analyses. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0054 Rev. 04, August 27. [SRDB Ref ID: 146884]

ORAUT [2018]. Internal dose reconstruction. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0060 Rev. 02, April 20. [SRDB Ref ID: 171554]

ORAUT [2020]. Compiled INL and ANL-W in vitro bioassay records for the period of 1953-1998. Spreadsheet. Oak Ridge, TN: Oak Ridge Associated Universities Team. February 20. [SRDB Ref ID: 203715]

ORAUT [2022]. Internal dose overestimates for facilities with air sampling programs. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-OTIB-0018 Rev. 02, July 1. [SRDB Ref ID: 193095]

ORAUT [2023]. Y-12 Plant – occupational internal dose. Oak Ridge, TN: Oak Ridge Associated Universities Team. ORAUT-TKBS-0014-5 Rev 04, August 3. [SRDB Ref ID: 198500]

Paulus RC, Dodd AO, Kennedy KK, Tingey FH, Warzel FM [1961]. Nuclear incident at the Idaho Chemical Processing Plant on January 25, 1961 report of the investigating committee. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-10036, June 1. [SRDB Ref ID: 12735]

Percival DR, Anderson JI [no date]. Whole-body counting at the National Reactor Testing Station. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. [SRDB Ref ID: 8399]

PPC [1959]. Health and safety instruction bulletin 4.01 personnel metering. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. July 7. [SRDB Ref ID: 207120]

Rich BL [1958]. Thyroid exposures from March 1958 incident at ICPP. Memorandum to J.W. McCaslin. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. July 21. [SRDB Ref ID: 8013]

Rich BL [1990]. Internal dosimetry and control. Idaho National Engineering Laboratory, Idaho Falls, ID: EG&G Idaho. EGG-ESQ-8808, May. [SRDB Ref ID: 8412]

Sill CW, Anderson JI, Percival DR [1964]. Comparison of excretion analysis with whole-body counting for assessment of internal radioactive contaminants. In: Proceedings of Symposium on the assessment of radioactive body burdens in man held in Heidelberg, Germany, May 11–16 1964. Vol. 1, pp. 217–229. [SRDB Ref ID: 8003]

Sommers JF [1954]. Radiation incident involving GE-ANP-1 on 3/28/54. Memorandum to J.P. Lyon. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. April 1. [SRDB Ref ID: 8386]

Sommers JF [1958]. Incident report on GEH-4 rupture on 12-17-58. Memorandum to J.P. Lyon. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. December 23. [SRDB Ref ID: 12711]

Sommers JF [1961]. Whole body counting procedures for MTR-ETR HP's. Memorandum to MTR-ETR Health and Safety Personnel. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. September 27. [SRDB Ref ID: 8545]

Sommers JF [1977]. Xenon dose determinations - Som-20-77. Memorandum to D.W. Marshall. Idaho National Engineering Laboratory, Idaho Falls, ID: EG&G Idaho. April 26. [SRDB Ref ID: 8411]

Stacy SM [2000]. Proving the principle a history of the Idaho National Engineering and Environmental Laboratory 1949-1999. Idaho Falls, ID: U.S. Department of Energy, Idaho Operations Office. DOE/ID-10799. [SRDB Ref ID: 1152]

Staiger MD [2003]. Calcined waste storage at the Idaho Nuclear Technology and Engineering Center. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. INEEL/EXT-98-00455 Rev. 1, June. [SRDB Ref ID: 13458]

Till JE, Meyer KR, Mohler HJ, Aanenson JW [2002]. Identification and prioritization of radionuclide releases from the Idaho National Engineering and Environmental Laboratory. Neeses, SC: Risk Assessment Corporation. RAC Report No. 3-CDCTask Order 5-2000-FINAL, October 8. [SRDB Ref ID: 1141]

Vance FP [1957]. Report of incident concerning overexposure to Iodine-131, May 15, 1957. National Reactor Testing Station, Idaho Falls, ID: Phillips Petroleum Company. May 17. [SRDB Ref ID: 8539]

Voelz GL, ed. [1969]. 1968 annual report of the Health Services Laboratory. Idaho Falls, ID: U.S. Atomic Energy Commission, Idaho Operations Office. IDO-12071, April. [SRDB Ref ID: 1359]

Wenzel DR [1973]. Internal doses resulting from ICPP Analytical Chemistry Laboratory plutonium contamination occurrence. National Reactor Testing Station, Idaho Falls, ID: Allied Chemical Corporation. July 24. [SRDB Ref ID: 8014]

Wenzel DR [1974]. Lung doses for Mass Spect. Lab Pu incident November 1972. Letter to F.W. Cipperley, U.S. Atomic Energy Commission, Idaho Operations Office. Idaho National Engineering Laboratory, Idaho Falls, ID: Allied Chemical Corporation. DRW-2-74, July 3. [SRDB Ref ID: 7997]

Wenzel DR [2000]. Determination of the normalized mass of individual radionuclides in the dissolver product for aluminum, zirconium and stainless steel fuels previously processed at INTEC. Idaho National Engineering and Environmental Laboratory, Idaho Falls, ID: Bechtel BWXT Idaho. INTEC-2000-006, February 8. [SRDB Ref ID: 7973]

GLOSSARY

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

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

activation
Creation of a different isotope by bombarding the parent isotope with neutrons, protons, or other types of radiation that results in the formation of a different isotope or element.

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.

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 in 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 fission of the fuel.

contamination
Radioactive material in undesired locations 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.

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.

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 (EU)
Uranium in which processing has increased the proportion of ^{235}U to ^{238}U to above the natural level of 0.7% by mass. 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 (FP)
(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) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons; the difference is that X-rays do not originate in the nucleus.

half-life (T, T1/2, T½)
Time in which half of a given quantity of a particular radionuclide disintegrates (decays) into another nuclear form. During one half-life, the number of atoms of a particular radionuclide decreases by one half. Each radionuclide has a unique half-life ranging from millionths of a second to billions of years.

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

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

in vivo bioassay
Measurements of radioactive material in the human body using instrumentation that detects radiation emitted from the radioactive material in the body.

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

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

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

mixed waste
Unwanted material containing both radioactive and hazardous components.

natural uranium (NU)
Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by mass. The specific activity of this mixture is 2.6×10^7 becquerels per kilogram (0.7 microcuries 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
Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

proton

Basic nuclear 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) with kinetic energy that interact with matter through various mechanisms that involve energy transfer. See *ionizing radiation*.

radioactive lanthanum (RaLa)

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

radioactive waste

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

radioactivity

Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei. See *radionuclide*.

radionuclide

Radioactive nuclide. See *nuclide*.

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 sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from *r*oentgen *e*quivalent in *m*an; 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 fuel storage basin

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

spent nuclear fuel (SNF)

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. See *irradiated fuel*.

transuranic (TRU) elements

Elements with atomic numbers above 92 (uranium). Examples include neptunium, plutonium and americium.

transuranic 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 commission in 1974. The DOE succeeded the U.S. Energy Research and Development Administration in 1979.

X-ray radiation

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

zirconium (Zr)

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