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06/05/2006	01 PC-1	Page change revision to correct error on page 24 in Table 5-12 in Section 5.5 on sample volume in 1980s. Approved issue of Rev 01 PC-1. No sections were deleted. Incorporates NIOSH formal review comments. This revision results in a reduction in assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Norman D. Rohrig. Approval: Signature on File 05/23/2006 John M. Byrne, TBD Team Leader Signature on File 05/23/2006 John M. Byrne, Task 3 Manager Signature on File 05/22/2006 Edward F. Maher, Task 5 Manager Signature on File 05/24/2006 Kate Kimpan, Project Director Signature on File 06/05/2006 James W. Neton, Associate Director for Science

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AEDE	annual effective dose equivalent
AMAD	activity median aerodynamic diameter
ANL-W	Argonne National Laboratory-West
ANP	aircraft nuclear propulsion
ANSI	American National Standards Institute
ATR	Advanced Test Reactor
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
COO	Chicago Operations Office
cpm	counts per minute
CPP	Chemical Processing Plant
d	day
DAC	derived air concentration
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
DU	depleted uranium
EBR-I	Experimental Breeder Reactor No. 1
EBR-II	Experimental Breeder Reactor No. 2
ERDA	Energy Research and Development Administration
ETR	Engineering Test Reactor
F	fast absorption type
FCF	Fuel Cutting Facility
gal	gallon
H&S	Health and Safety
HFEF	Hot Fuel Examination Facility
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
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
L	liter

m	meter
M	moderate absorption type
MAP	mixed activation product
mCi	millicurie
MDA	minimum detectable activity or amount
MDL	minimum detectable level
MeV	megavolt-electron, 1 million electron volts
MFP	mixed fission product
mg	milligram
min	minute
ml	milliliter
MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MPC_a	MPC for airborne activity
mrem	millirem
mrep	millirep
MTR	Materials Testing Reactor
NaI(Tl)	sodium iodide doped with thallium
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
PBF	Power Burst Facility
pCi	picocurie
PCS	primary coolant system
RaLa	radioactive lanthanum
RAM	radiation area monitor or remote area monitor
RCIMS	Radiation Control Information Management System
RDR	Radiation Dosimetry and Records
RESL	Radiological Environmental Sciences Laboratory
RWMC	Radioactive Waste Management Complex
S	slow absorption type
SDA	Subsurface Disposal Area
SL-1	Stationary Low-Power Reactor
SMC	specific manufacturing capability
TAN	Test Area North
TBD	technical basis document
TLV	threshold limit value
TMI	Three Mile Island
TRA	Test Reactor Area
TRU	transuranic
TSA	Transuranic Storage Area
U.S.C.	United States Code
WBC	whole-body counting
WERF	Waste Experimental Reduction Facility

yr	year
ZPPR	Zero Power Plutonium Reactor (later Zero Power Physics Reactor)
α	alpha particle
β	beta particle
γ	gamma
σ	standard deviation
µCi	microcurie
µg	microgram
µm	micrometer
§	section

5.1 INTRODUCTION AND HISTORICAL OVERVIEW

Technical basis documents and site profile documents are general working documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist the National Institute for Occupational Safety and Health (NIOSH) in the completion of the individual work required for each dose reconstruction.

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

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled "Exposure in the Performance of Duty." That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer "shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer … was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the [probability of causation] guidelines established under subsection (c) …." [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work.

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

This site profile provides specific information concerning documentation of historical practices at the Idaho National Engineering and Environmental Laboratory (INEEL).

In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) and started construction of facilities on a 572,000-acre site approximately 50 miles west of Idaho Falls in southeastern Idaho. NRTS was later the Idaho National Engineering Laboratory and then the INEEL. In 2005, DOE shortened the name to the Idaho National Laboratory.

Each of the original AEC laboratories was unique in both mission and location. Because the early days of the AEC programs represented the beginnings of the nuclear age, significant technical developments were a necessity, not the least of which were developments in radiation safety. Some of the unique characteristics of radiation safety (and internal dosimetry specifically) at the NRTS that had a marked influence on the internal dosimetry programs at each of the facilities were:

The original mission of the NRTS was (as the name implied) high-enriched (over 50% and mostly over 90%) uranium reactor concept development, materials testing through high-flux test reactor operation, and chemical processing of those high-enriched (valuable) uranium fuels. The production of weapons-grade nuclear materials was not a mission.

The NRTS beginning was 8 to 10 yr later than Oak Ridge National Laboratory and Hanford. During those developmental years significant technical progress in professional skills, instrumentation, analyses, procedures, and techniques were accomplished. Radiation safety programs and techniques from Oak Ridge (ACC 1952) were adopted at startup of the NRTS facilities.

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

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

To provide consistency of radiation safety programs at the NRTS among a large variety of facilities and constantly changing contractors, the AEC established a Health and Safety (H&S) Laboratory at NRTS to provide technical support in the areas of 1) environmental surveillance, 2) external dosimetry (personnel dosimeters of all types), 3) portable radiation detection instrumentation inventories, calibration, and maintenance, 4) internal *in vitro* and *in vivo* bioassay analytical laboratories, 5) technical support in quality assurance of external and internal radiation dose evaluation, 6) maintenance and documentation of personnel dosimetry records, and 7) research and development in these areas of responsibility. The name of this organization changed to Health Services Laboratory (HSL), then to Health and Safety Division (HSD), then to Idaho Center for Radiological and Environmental Sciences, and most recently to the Radiological and Environmental Sciences Laboratory (RESL).

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

As a consequence, and in spite of the constant changes at the INEEL, basic assumptions about minimum detectable activities (MDAs) or minimum detectable levels (MDLs), missed dose potential, and so forth are relatively consistent across the years. There were differences in the available nuclear materials from facility to facility, but as early as 1955 or 1956 gamma spectral analysis capabilities at the NRTS allowed the significant bioassay results (those which would result in reportable internal dose) to be defined in terms of the specific radionuclides. The practice in the case of a higher urine sample result was to attempt radionuclide identification through gamma spectral analysis and chemical separation. This document describes default assumptions for use in cases

when the bioassay records for a worker do not include the radionuclide analyses and only record gross beta or gross alpha results.

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

The majority of this document provides background information to aid the internal dose reconstructor through increased general understanding, data interpretation, defaults, and so forth. Sections 5.1.1 to 5.1.3 provide facility descriptions, and Section 5.1.4 details the radionuclides of concern. Section 5.2 describes the INEEL radiological protection program as it evolved over the years, Section 5.3 discusses internal dose control, and Sections 5.4 and 5.5 describe MDAs and whole-body counting (WBC), respectively. Section 5.6 contains specific information necessary for dose reconstruction when the facility or facilities where the worker worked are known. Sections 5.7 and 5.8 describe the treatment of missed dose and unmonitored workers.

5.1.1 Test Reactors

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

The largest internal exposures at INEEL resulted from accidental intakes associated with episodic events or planned major releases, for which the time and characterization of the materials of the intakes were well known. These exposures were documented in each exposed employee's file.

5.1.2 High-Enriched Spent Fuel Chemical Processing

In addition to being the site for experimental test reactors, INEEL is the home of the Idaho Nuclear Technology and Engineering Center (INTEC), formerly known as the Idaho Chemical Processing Plant (ICPP). The ICPP reprocessed high-enriched reactor fuel (^{235}U enrichments of 50% to 93%) for 39 yr from 1952 to 1991. Aged mixed fission products (MFPs) were the predominant internal hazard, although enriched uranium isotopes and plutonium isotopes (^{238}Pu enhanced) were limiting in specific process locations. ICPP experienced not only operational containment barrier failures but also had accidental criticality events in 1959, 1961, and 1978. Because the criticality accidents occurred in process vessels in heavily shielded cells, these events resulted in relatively minor worker intakes (Stacy 2000; AEC 1960; AEC 1962a). These exposures are documented in the personnel dosimetry files.

5.1.3 Other Nuclear Facilities and Processes

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

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

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

At the Naval Reactors Facility (NRF), the U.S. Navy used the INEEL for operating reactors and as a naval reactor training center. Because this is not a U.S. Department of Energy (DOE) program and not under the oversight of DOE, NRF is not part of the dose reconstruction and compensation program. However, through the years NRF has participated in limited coordination of radiological protection programs and site support services. It is possible that some workers' internal dose could have resulted from their support work at the NRF.

5.1.4 Radioactive Nuclides of Concern and Solubility

INEEL facilities and activities have related primarily to experimental reactor design and development, spent fuel processing, DU parts production, and low- and high-level radioactive waste treatment and disposal. The INEEL Site Description TBD (ORAUT 2004) describes these activities in more detail. Table 5-1, which lists the internal doses received at the INEEL from 1992 to 2000, demonstrates that the radionuclides of more recent concern to internal dose are MFPs (primarily aged), uranium and its decay products, and TRU wastes and their decay products. The table lists internal doses as committed effective dose equivalents (CEDEs). This information is from Reilly (2001) and can be useful to the reconstruction effort primarily as an indicator of intake radionuclide identity and quantities

Table 5-1. Recorded internal doses for 1992 to 2000.

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

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

for this recent 10-yr period. Before 1990, workers received higher doses from the radionuclides in Table 5-1, as well as from MAPs (primarily longer-lived, i.e., ^{110m}Ag and others). Most internal doses have been identified following an incident rather than as a result of routine bioassay measurements.

Table 5-2 lists radionuclides of concern at INEEL from these programs and as documented in *INEEL M & O Contractor Technical Basis for Internal Dosimetry, General Technical Basis and Facility Specific Documents* (Reilly 2001) and from HSL/RESL reports. These radionuclides are those for which internal doses were determined in the past and/or for which detection methods were developed. The INEEL program follows the DOE guidance and uses the International Commission on Radiological Protection (ICRP) Publication 30 nomenclature for solubility (ICRP 1979) as noted in the table. In addition, Table 5-2 provides the ICRP Publication 68 (ICRP 1995) recommended absorption type.

Through the years at INEEL, plotting urine and fecal elimination curves has shown that an effective solubility of moderate (M) to slow (S) per the ICRP Publication 66 system (ICRP 1994) is to be expected for the radionuclides listed in Table 5-2, with the exceptions of strontium, cesium, iodine, mercury, and tritium. Elimination curves for ^{3}H, ^{90}Sr, ^{131}I, ^{137}Cs, and ^{203}Hg show they are typically of a fast absorption type (F). The chemical explanation is that radioactive materials in oxygen

Table 5-2. Primary radionuclides of concern.

Element	Radionuclides	ICRP 30[a] retention class	ICRP 68 absorption type	Preferred analysis technique
Hydrogen	H-3 (assume HTO)	D	F	Urine
Chromium	Cr-51		F,M,S	WBC
Manganese	Mn-54		F,M	WBC
Iron	Fe-59		F,M	WBC
Cobalt	Co-58, Co-60	W,Y	M,S	WBC
Zinc	Zn-65		S	WBC
Strontium	Sr-89, Sr-90/Y-90[b]	D	F	Urine
Zirconium/niobium	Zr/Nb-95	W,Y	F,M,S	WBC
Molybdenum	Mo-99		F,S	Urine
Technetium	Tc-99		F,M	Urine
Ruthenium	Ru-103, Ru-106		F,M,S	Urine
Silver	Ag-110m		F,M,S	WBC
Antimony	Sb-122, Sb-125		F,M	WBC
Tellurium	Te-132		F,M	Urine, WBC
Iodine	I-131, I-133 (assume elemental I_2)		F	Urine, thyroid counter
Cesium	Cs-134, Cs-137	D	F	WBC
Barium/lanthanum	Ba/La-140		F/F,M	WBC
Cerium	Ce-141, Ce-144	W,Y	M,S	WBC
Europium	Eu-152, Eu-154, Eu-155	W,Y	M	WBC
Gadolinium	Gd-153		F,M	WBC
Tantalum	Ta-182		M,S	WBC
Mercury	Hg-203 (assume inorganic)		F	Urine, WBC
Protactinium	Pa-233		M, S	WBC
Uranium	U-233, U-234, U-235, U-236, U-238	D,W,Y	F,M, S	Urine, fecal
Neptunium	Np-237		M	Urine, fecal
Plutonium	Pu-238, Pu-239/240	W,Y	M, S	Urine, fecal
Americium	Am-241	W	M	Urine, fecal, lung counter

a. Assigned in the INEEL internal dosimetry technical basis document (Reilly 2001); ICRP 30 = ICRP 1979; ICRP 68 = ICRP 1995.

b. Yttrium has an absorption type of M or S; however, in the case of short-lived Y-90, it tracks its parent Sr-90, which is normally a type F at the INEEL.

atmospheres oxidize rapidly, which forms less soluble compounds. The default assumption of M or S would be appropriate when based on the more claimant-favorable result to the organ in question.

5.2 RADIOLOGICAL PROTECTION PROGRAM MANAGEMENT AND SUPPORT

5.2.1 Internal Dosimetry Issues Related to Contractor Changes

The changes in contractors at INEEL during its 54-yr history [listed in Tables 2-1 and 2-2 of the Site Description TBD of this Site Profile (ORAUT 2004)] resulted in relatively frequent management changes at most of the facilities. (The contract with the University of Chicago to operate the ANL-W facilities has not changed in this 54-yr period.)

The primary AEC/DOE oversight for NRTS/INEEL, which included most projects on the site and all support functions, was assigned to the AEC Idaho Operations Office (IDO). The AEC created the H&S Laboratory to provide a variety of health and safety support functions to the entire site, which included external and internal dosimetry, health physics instrumentation, fire protection, medical services, and environmental surveillance. The Chicago Operations Office (COO) provided oversight for ANL-W (contracted by the University of Chicago) programs and facilities, while the U.S. Navy provided oversight for the NRF program. ANL-W, which is a DOE program, is included in this TBD. The ANL-W program uses site support services, including internal dosimetry support, but the bioassay results are reported through COO.

The INEEL personnel dosimetry records have been and are documented and permanently maintained by the various organizations over time. Records about individual facility or contractor field monitoring programs (air-monitoring data, personnel contamination records, etc.) were maintained by individual contractors and/or site areas and are not maintained in a single recordkeeping system. The field monitoring data were not available for use in this report.

In spite of the frequent changes in operational responsibility through the years and the movement of workers among facilities, there has been a basic level of consistency in the internal dosimetry programs at INEEL, particularly the bioassay analytical techniques and calculation processes. Although the field programs (which monitored the workplace and identified work groups to be included in the routine bioassay programs and workers who needed special bioassays) 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.

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.

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

5.2.2 Bioassay Programs

Routine bioassay of radiation workers has been conducted 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. Table 5-3 lists the reconstructed history of routine bioassay frequency.

Table 5-3. Routine bioassay history summary.[a]

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1953-1960	Annually	*In vitro* urine	Radiation workers		Frequency is inferred from individual data sheets.	Individual data sheets Table 5-10 AEC 1959 AEC 1961
1961	Annually	*In vitro* urine *In vivo*	Radiation workers		Frequency is inferred from individual data sheets.	Table 5-10 AEC 1962a
1962-1972	Annually	*In vitro* urine *In vivo*	Radiation workers		Frequency is inferred from individual data sheets.	AEC 1962a AEC 1963a
1973-1981	Annually	*In vitro* urine	Radiation workers	Reporting Annual DE >10% quarterly standard in ERDA Manual Chapter 0524 (ERDA 1975)	Frequency is inferred from individual data sheets.	AEC 1968 AEC 1975 ERDA 1975
	When internal intake suspected	Fecal				
	Annually	*In vivo*				
1982-1987	Annual	*In vitro* urine	CPP-603 workers Fuel reprocessing operators	Reporting 50-yr CDE >10% quarterly standard in ERDA Manual Chapter 0524 (ERDA 1975)	Staggered to monitor group throughout the year	Exxon 1981
	When internal intake suspected	Fecal	Waste reprocessing operators Shift lab workers Health physics technicians			
	Annual	In vivo	Selected radio-chemistry workers Maintenance workers Denitrator operators			
	Termination	*In vivo*	All radiation workers			
1988-1989	1 to 6 months	*In vivo*	All radiation workers	Investigating Lung, 50-yr CDE >0.5–1.0 rem Bone Surface, 50-yr CDE >1.0–2.0 rem Other Organs, 50-yr CDE >0.5-1.0 rem	Staggered so that worker receives some sort of analysis or sampling every 3 months.	Tschaeche 1988
	Annually	*In vitro* fecal				
	18 to 24 months	*In vitro* urine				
	Termination	*In vivo*	When internal exposure suspected.			
		In vitro				
	New hire	*In vitro*	Depending on review of radiation dose history.			
		In vivo				
1990-1994	Annually	*In vivo*	All radiation workers where exposure to surface or airborne radioactive contamination could give at least 0.1 mrem AEDE from occupational sources, or give an organ or tissue DE >5 rem annual.	Reporting In accordance with DOE Order 5480.11 (DOE 1988). Workers that could receive 0.1 rem AEDE or 5 rem ADE organ or tissue dose.	Bioassay requested when workplace monitoring program indicates >0.02 annual limit of intake. Follow-up triggered by positive results from the workplace monitoring program, positive routine bioassay sample, or in response to incidents involving suspected intakes.	King 1990 Rich 1990
	6 months	*In vitro* fecal				
	Annually	*In vitro* urine				
	New hire	*In vivo*	Worked at a facility where gamma-emitting radionuclides were handled.	Investigating AEDE ≥0.01 rem.		
		In vitro urine fecal	Worked in U manufacturing or recovery facilities. Worked with transuranic materials			
	Termination	*In vivo* *In vitro*	Any employee suspected of having an internal exposure or on a scheduled monitoring program.			

Table 5-3 (Continued). Routine bioassay history summary.

Year	Typical frequency	Type	Groups analyzed/sampled	Investigating level	Comments	Reference
1995	Appropriate to the facility mission, potential uptakes.	*In vivo* *In vitro* urine fecal	All radiation workers that enter radiological buffer areas or areas of higher radiological controls and are likely to receive intakes resulting in a CEDE of 0.1 rem or more. Type of bioassay based on source term. Urine requested when pure beta, uranium, or TRU was of interest. Feces requested primarily for uranium and TRU source terms.	Reporting In accordance with DOE 5480.11 (DOE 1988) and 10 C. F. R. 835. Workers that could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE.	Each facility has a specific Technical Basis Document for Internal Dosimetry.	Andersen, Perry, and Ruhter 1995
	When work place monitoring indicates significant potential for intakes.			Investigating Internal doses resulting from all confirmed intakes are to be evaluated.	Follow-up for any suspected intake of radionuclides and to more accurately identify and characterize the amount of intake and excretion pattern.	
1995-2000	New hire		Random sampling is performed to demonstrate the adequacy of the radiological controls in limiting the internal intake of radionuclides. Employees are selected at random from both nonradiation workers and a radiation worker population.			
	Termination	*In vivo* *In vitro*	Based on screening to determine internal conditions from previous uptakes or to establish baseline for those continuing to work as radiation workers. Any employee that was on a scheduled monitoring program.			
2001	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 CFR 835. Workers that could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE.	Bioassay is mandatory when an employee or visitor is involved in an event where the internal uptake of radionuclides was likely to have occurred.	Rielly 2001
	Termination	*In vivo* *In vitro*	Any employee that was on a scheduled monitoring program.	Investigating Uranium, >1.0 µg/L *In vitro* activity detected >2σ. *In vivo* >2.33 σ. Default trigger levels exceeded.		

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

5.2.3 Internal Dose Records

Formal or *legal* internal dose data were maintained by the DOE HSD in individual hard-copy folders until 1989, when all technical support service functions, including those related to internal dosimetry, were transferred to the INEEL 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 subject matter expert reviews, validates,

and prepares official internal dose assessments. A DOE staff member at RESL is responsible for oversight of INEEL 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) was placed in service to support the Radiation Protection program, including internal dosimetry. RCIMS is a database that lists reported internal doses as CEDE when an individual's dose history is prepared.

5.2.4 Internal Dose Regulations, Investigation Levels, and Data Codes

The following information is important to internal dose reconstruction because the worker files from DOE can contain a variety of internal dose information including the calculated internal doses as well as the *in vitro* and *in vivo* individual bioassay results. The changing regulations influenced the level of internal dose evaluation and documentation, but did not change the fact that all (negative as well as positive) bioassay data were recorded in the individual dosimetry files.

The information used in internal dose assessments and analytical data sheets has varied through the years. Table 5-4 describes coded information that could appear in records after 1989. Table 5-5 describes internal dose information that could appear in pre-1989 records. Table 5-6 contains analytical nomenclature. Table 5-7 contains INEEL codes for various site areas.

Table 5-4. Internal dose assessment information after 1989.

Coded information	Description
Name & SS No.	Exposed employee by name and social security number.
Asmt. Nos.	This assessment number is the calendar year, i.e., (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 bequerels.
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 (TL) exposures are summed for the year of intake for each employee.

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

Dose information	Description
Name, Soc. Sec. No.	Employee name, social security number, and (Contractor Abbreviation/Plant or Facility).
Nuclide	Radionuclide symbol followed by ICRP solubility class (D, W, or Y) (ICRP 1979).
Intake Period	Month and Year for single exposure or period of time by month and year in which exposure occurred.
Organ (Max.)	Organ that received the maximum dose from the specified intake.
Organ CDE rem	CDE calculated for the listed organ in rem.
CEDE rem	Calculated CEDE in rem.
AEDE rem	Calculated AEDE in rem.
Year	Year for which the AEDE was calculated.

Federal regulations about permissible internal dose and formal reporting requirements to the AEC, the Energy Research and Development Administration (ERDA), and the DOE changed periodically through the site's history. While the regulations influenced the level of calculated internal dose that would be reported, these changes did not significantly change the analytical programs for the detection of internal intake. For example, the monitoring and analytical programs were designed to initiate, through *in vitro* and/or *in vivo* bioassay analysis, an investigation of any potential internal

intake as indicated by positive air sampling, personnel contamination, etc. Most of these recorded analyses did not result in detectable radionuclides. 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 only reported and documented if specific dose levels were exceeded. Regulations required periodic urinalyses or *in vivo* counting or evaluation of air concentrations if the whole-body dose or dose commitment could exceed 300 mrem in a calendar quarter (AEC 1958, 1963b, 1968, 1975; ERDA 1975). Changes in the reporting levels did not generally result in changes to the air-monitoring or bioassay programs. Each individual analytical result was documented and placed in individual exposure files regardless of the formal reporting requirements.

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

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

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

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

The investigation levels (the levels at which positive bioassay results triggered follow-up sampling to verify that detectable activity had been taken into the body) have also changed little from the early years to the present. Dickson (1977) established official investigation levels (Table 5-8) for acute

uptakes of radionuclides corresponding to one-tenth of the quarterly radiation standard. Later procedures (DOE 1988) set specific limits on those positive bioassay results that could result in 100 mrem AEDE or above as the point at which follow-up and reporting was required. With the DOE *Radiological Control Manual* (e.g., DOE 1994), this changed to 100 mrem CEDE. In addition, a calculated dose of 10 mrem or above would be recorded as an internal dose. These procedural limits did not materially affect the bioassay sampling frequency or the recording of even non-detectable radioactivity in bioassay samples, although the request for and number of follow-up samples and analyses could have been different as a function of the formal regulations in effect.

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

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

5.3 INTERNAL DOSE CONTROL

The radiological protection program was established to provide detection of barrier or ventilation failure in a timely manner. The program consisted of continuous and retrospective air and effluent monitoring combined with personnel and surface contamination monitoring. Detection of barrier failure provided the information for making decisions on evacuating personnel, increasing personnel protection equipment (e.g., respirators), and requesting bioassay analyses to identify internal intake. As a consequence of a consistent AEC/DOE policy to avoid detectable internal exposures, coupled with the time and technical complexity of an internal dose evaluation, the general policy at INEEL for internal exposures has been preventive in nature. 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 NRTS/INEEL policy and practice was to require respiratory protection on jobs when the possibility of airborne contamination was thought to exist regardless of the actual measured air or surface contamination. These practices influenced the assumptions for dose evaluation in internal dose reconstruction.

In a related matter, the contamination control limits for the detection and control of released activity beyond the control boundaries were related to instrumentation capabilities and the basic philosophy of acceptance of detectable contamination. As a result of increased emphasis on exposures that were as low as reasonably achievable, some reduction in acceptable release levels was implemented. The contamination control limits for alpha on plant surfaces and particularly personnel were always set close to the MDA, such that *any detectable* contamination was a signal for preventative and follow-up evaluations and actions. Beta/gamma MDAs typically were a factor of 5 below the limits. Table 5-9 is a summary of control limits primarily from the *CPP* [Chemical Processing Plant] *Health Physics Manual* (ACC 1952) and current operating procedures.

Table 5-9. Surface contamination control and MDAs.

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

5.3.1 Air Monitoring

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

ACC (1952) describes a CAM and three other air-sampling systems. The manual required use of a filter-type respirator when airborne activity exceeded 1×10^{-8} µCi/cm^3 for beta/gamma activity or 1×10^{-11} µCi/cm^3 for alpha activity. An army assault-type mask was invoked when levels exceeded this by a factor of 10. Positive-pressure air masks were invoked if levels larger by a factor of 1,000 occurred (ACC 1952, p IX: 4-1).

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. 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. 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. In the event 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.

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

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

DOE HSL technical reports and annual reports, coupled with facility memoranda and reports, documented the analytical detection capability of NRTS in the early 1950s and 1960s. Internal monitoring programs were in place when facility operations began in late 1951. For example, during ANP Program-IET activity in 1956, particulate and liquid caustic filter samples of effluent were analyzed with gamma spectroscopy and specific chemical separations of the identified radionuclides (Ebersole 1956). This analytical capability to identify radionuclides by their energy spectra was available and used for urine and other bioassay samples. Specific separations (e.g., strontium, iodine) were available to quantify the radioactive components of a variety of samples of interest.

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

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

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

As part of the EEOICPA coworker data program for the INEEL, images of all bioassay results prior to 1986 were provided to the ORAU Team and the results were entered into an Access database. This database has over 150,000 bioassay results. The nearly 135,000 urine bioassay results fall into over 80 categories of analytes, a few of which are the result of minor typographical differences. Figure 5-1 shows the time course of the gross beta, gross gamma, and strontium analyses which were performed the most. The group beta-gamma in the early years seems to have followed the beta protocol and after 1964 follows the gamma protocol. Early on the gross beta assay using a 5 ml volume was used, but it was replaced with the gross gamma assay supplemented with strontium analyses typically using a 75 ml volume. In the early 1970s, the *in vitro* bioassay was largely replaced by whole body counting. In the 50s and 60s, several thousand assays were performed each year, well over one per monitored worker per year. The early results were reported in d/m/sample except for certain assays like Uranium reported in g U/l and tritium in d/m/ml. About September 1973 the activity units changed to µCi with a large negative exponent so that pCi would be a more natural unit. There is some sloppiness in the database record with the units between dpm and µCi and /sample or /ml so if the results or uncertainties look anomalous, that is a likely explanation. The sample size for alpha spectrometry went from the urine sample size (about a liter) to 1 ml in mid 1979 with no change in the sample uncertainty.

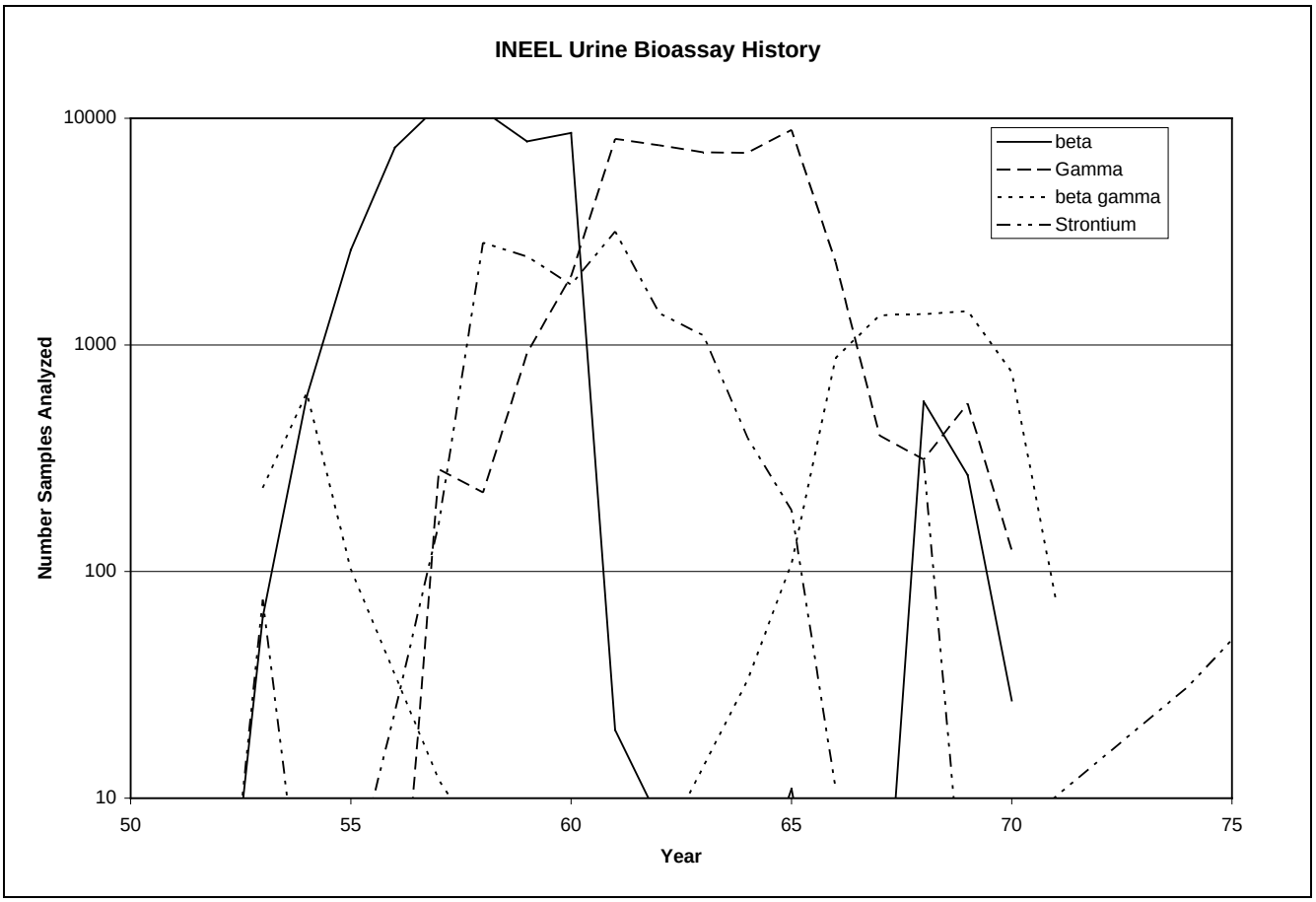

Figure 5-1:History of Urine Bioassay at the INEEL.

Table 5-10 reproduces reports of the urinalysis results for 1959, 1960, and 1961 as obtained from Table 4 of AEC (1960), Table IV of AEC (1961), and Table XIII of AEC (1962a). These results are not identical but quite similar to those from the new database. The practice was to perform a gross beta or gamma analysis and identify specific radionuclides if the gross counts indicated activity above background levels. The total number of urinalyses in 1959 was 11,066; 3,524 people had radiation badges; and 715 received external doses above 500 mrem. These numbers demonstrate that people provided urine samples multiple times during the year.

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

The special and routine bioassay sample analyses were performed and documented by the DOE analytical laboratory. Puphal (1994) reported on the procedures used for bioassay in the DOE Analytical Chemistry Branch beginning in 1960. These procedures were collected into a procedures manual in 1982 to be revised periodically (RESL 1982). Another version of the procedures after the analytical work was transferred to WINCO is documented (DOE 2002).

The individual analytical data were recorded, as were the specific doses for which formal evaluations were required. Before 1977, internal doses were assigned if the internal dose or dose commitment was greater than 50% of the yearly allowable body or organ doses by Manual Chapter 0524, the controlling regulation at the time (Aoki 1979). In 1977, the policy was changed to assign internal

Table 5-10. Urinalysis results in 1959, 1960, and 1961.

Nuclide/ element of interest	Type activity	Total number performed			Statistically significant Number			Statistically significant Percent			Highest result[a]		
		1959	1960	1961	1959	1960	1961	1959	1960	1961	1959	1960	1961
	Gross β[b]	8,546	8,546	30	65	15	5	0.76	0.18	17	18,820 ±632 dpm/5 ml	992 ±40 dpm/5 ml	172 ±16 dpm/ml
	Gross γ[c]	2,433	2,712	9,120	174	129	--[d]	7.15	4.76	4	35,972 ±310 dpm/5 ml	19,817 ±105 dpm/5 ml	1,900,235 ±876 dpm/75 ml
Co-60	β	--	--	1	--	--	0	--	--	0	--	--	300 ±75 dpm/450 ml
Sr-90	β	3	105	3,248	3	0	2	100	0	2	4.12×10^{-2} dpm/ml	Insignificant	183 ±8 dpm/75 ml
Sr-91	β	20	37	2	19	0	--	95	0	0	388 ± 1.6	Insignificant	4 ±8 dpm/ml
I[e]	β	--	9	--	--	2		--	22		0	9992 ±80 dpm/ml	--
Cs-137	β	--	--	40	--	--	0	--	--	0	--	--	1,460 ±10% dpm/1700 ml
Ba-139	α	20			16	0	--	80	0	--	120 ±0.8 dpm/ml	0	--
Th[f]	α	7	0	--	0	0	--	0	0	--	Insignificant	0	Insignificant
U	α	--	--	4	--	--	--	--	--	0	--	--	10 µg/L
U-233	α	17	3	--	1	0	--	.06	0	--	180 ±4.0 dpm/ml	Insignificant	--
Pu-239	α	18	0	29	0	0		0	0	0	Insignificant	Insignificant	2E-9 µCi/ml
Am-241	α	2	0	--	0	0	--	0	0	--	Insignificant	0	--
Totals		11,066	11,352	12,494	278	146	--	2.51	1.29	4			

a. All except two I-131 exposures in 1961 listed under gross gamma activity are less than 10% of the permissible body burden for the radionuclide of interest.
b. If only gross β analyses are available, the default should be Sr-90.
c. If only gross γ analyses are available, the default should be Cs-137.
d. No data reported.
e. Iodine isotope(s) not identified in reference. Assume I-131.
f. Thorium isotope(s) not identified in reference. Assume Th-228.

Table 5-11. Detection limits applicable to environmental sample analyses (1953 to 1965).

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

a. Reduced in October 1962 from 1.5E-7 µCi/ml.
b. Reduced in September 1962 from 50 pCi/L.

doses when the CDE to an organ exceeded one-tenth of the quarterly radiation protection standard. Replies to requests for radiation exposure history before 1979 stated that there was "no positive exposure reported" when the dose was below the reporting levels noted above. Because individuals remembered receiving some internal exposure, the statement, "no positive exposure reported," was determined to be misleading and was changed in 1977 to "no reportable levels recorded." In all cases, copies of the bioassay results were placed in the individual's radiation exposure file and should be in the individual's dose file in the NIOSH Office of Compensation and Support Claims Tracking System database.

5.4 MINIMUM DETECTABLE ACTIVITIES

In compliance with the November 1998 Code of Federal Regulations requirement (10 C. F. R. pt. 835) for the DOE Laboratory Accreditation Program (DOELAP), and based on American National Standards Institute (ANSI) N 13.30, *Performance Criteria for Radiobioassay* (ANSI 1996), both the *in vitro* and *in vivo* radiobioassay laboratories at INEEL received DOELAP accreditation in February 1998. In accordance with this accreditation, MDAs and decision levels at the 95% confidence level (2σ) are performed. Tables 5-12 and 5-13 list current MDAs for urine and fecal sample analysis, respectively, along with values gleaned from historical documents. A large majority of the urine samples taken at the site were single voidings; 24-hr samples were used for special sampling purposes (i.e., follow-up samples, primarily to extend the sensitivity). The MDAs listed are those for the primary samples; the tables include the recommended periods for the MDA values. Table 5-14 lists the current *in vivo* MDAs along with values gleaned from historical documents and the recommended periods for use with the MDA values.

5.5 DEVELOPMENT AND IMPACT OF WHOLE-BODY COUNTING

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

As a consequence, the *in vivo* counting program was used to count 1) all terminating employees who required physical exams, 2) employees suspected of having a possible internal intake, and 3) selected groups of individuals scheduled for semi-routine analyses by health physics supervisors

(Sommers 1961). In 1963 approximately 1,650 counts were performed; only those activities greater than 0.1 µCi were further quantified. This level was determined to be less than one-tenth of the MPBB for most of the gamma-emitting isotopes.

Table 5-12. MDAs for urine samples by period.

Radiation/ radionuclide	Period	Typical Volume (ml)	Typical MDA (dpm/sample)	Typical Daily[c] MDA (dpm/d)	Reference
Gross β	1951-1953	5	86	24000	Data Sheet
	1954-1960	5	93	26000	Ebersole & Flygare 1957[a]
Gross γ	1957-1964	75	580	10800	AEC 1961[a]
	1965-1971	75	205	3800	Data Sheet
H-3	1972-1994	3	0.5 dpm/ml	700	AEC 1972[a], AEC 1974
	1995-present		3 dpm/ml	4200	Andersen, Perry, and Ruhter 1995, Rielly 2001
Co-60	1957-1958	50	51	1400	Database
Sr-90	1953-6/14/62	75	37	700	Database
	6/15/62-1970	75	20	370	Database
	1971-1989	75	1.7	32	AEC 1972[a], AEC 1974[a]
	1990-present	500 min	1.9	5	Andersen, Perry, and Ruhter 1995
I-131	1957-1970	75	370	6900	Database
Cs-134	1974-present	400	2	7	Database
Cs-137	1961-present	400	410	1435	Database
Th-230	1974-present	1000	0.1	0.14	AEC 1974[a]
Np-237	1974-present	1000	0.1	0.14	AEC 1974[a]
U (FP)	1954-1961	0.1	1E-5 g U/L[b]	14 µg U[b]	Database
	1962-1971	0.1	5E-6 g U/L[b]	7 µg U[b]	Database
U (KPA)	1985-present		0.2 µg/L	0.28 µg U	Rich 1990
U-233/234	1979-1986	700	0.52	1.0	Database
	1995-present	500 min	0.091	0.25	Andersen, Perry, and Ruhter 1995
U-235	1970-1979	1000	0.22	0.31	Rich 1990
	1980-1994	700	0.13	0.26	Database
	1995-present	500 min	0.084	0.24	Andersen, Perry, and Ruhter 1995
U-238	1970-1979	1000	0.22	0.31	Rich 1990
	1980-1994	700	0.21	0.42	Database
	1995-present	500 min	0.067	0.19	Andersen, Perry, and Ruhter 1995
Pu-238	1981-1984	700	0.072	0.14	Database
	1990-1994	1000	0.13	0.18	Rich 1990
	1995-present	500 min	0.049	0.14	Andersen, Perry, and Ruhter 1995
Pu-239/240	1964-1970	1000	0.93	1.3	AEC 1964[a]
	1971-1973	1000	1.03	1.4	AEC 1972[a]
	1974-1979	1000	0.47	0.66	AEC 1974[a]
	1980-1989	700	0.073	0.14	Database
	1990-1994	1000	0.060	0.084	Rich 1990
	1995-present	500 min	0.060	0.17	Andersen, Perry, and Ruhter 1995
Am-241	1977-1979	1000	0.16	0.22	AEC 1974[a]
	1980-1989	700	0.29	0.6	Database
	1990-1994	1000	0.2	0.28	Rich 1990
	1995-present	500 min	0.051	0.14	Andersen, Perry, and Ruhter 1995
Cm-244	1974-present	1000	0.155	0.22	AEC 1974[a]

a. MDA calculated from inferred 2σ uncertainty.
b. Smallest reported value. Not MDA.
c. Based on 1400 ml daily volume and typical sample size. May need adjustment for larger or smaller sample..

Table 5-13. MDAs for fecal samples by period.

Radiation/ radionuclide	Period	Fecal[a] (pCi/sample)	Reference
Co-60	1963-present	10	Rich 1990
Sr-90	1963-1994	10	Rich 1990
	1995-present	1.9	Andersen, Perry, and Ruhter 1995, Rielly 2001
Cs-134	1963-present	10	Rich 1990
Cs-137	1963-1999	0.01	Rich 1990
	2000-present	0.3	Bechtel BWXT 2000
Th-230	1974-present	0.0003	AEC 1974
Np-237	1974-present	0.0003	AEC 1974
U-233/234	1970-2002	0.041	Andersen, Perry, and Ruhter 1995, Rielly 2001
	2003-present	0.05	Bhatt 2003
U-235	1970-2003	0.038	Andersen, Perry, and Ruhter 1995, Rielly 2001
	2003-present	0.09	Bhatt 2003
U-238	1970-1994	0.5	Rich 1990
	1995-2002	0.03	Andersen, Perry, and Ruhter 1995, Bechtel BWXT 2000, Rielly 2001
	2003-present	0.09	Bhatt 2003
Pu-238	1974-1994	0.03	AEC 1974
	1995-2002	0.022	Andersen, Perry, and Ruhter 1995, Rielly 2001
	2003-present	0.02	Bhatt 2003
Pu-239/240	1964-1973	0.4	AEC 1964[b]
	1974-1994	0.02	AEC 1974
	1995-present	0.03	Andersen, Perry, and Ruhter 1995, Bechtel BWXT 2000, Rielly 2001, Bhatt 2002
Am-241	1974-1994	0.07	AEC 1974
	1995-2001	0.023	Andersen, Perry, and Ruhter 1995, Rielly 2001
	2002-present	0.04	Bhatt 2002
Cm-244	1974-present	0.02	AEC 1974
Cf-252	1974-present	0.02	AEC 1974

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

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

Radiation/ radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
Cr-51	1962-2000	12	10	Percival and Anderson 1962[a]
	2001-present	32	5	Rielly 2001
Mn-54	1962-2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, Ruhter 1995
	2001-present	2.6	5	Rielly 2001
	2001-present	1.3	10	Rielly 2001
Fe-59	1962-2001	4.5	5	Rielly 2001
	2001-present	1.5	10	Rielly 2001
Co-58	1962-2000	12	10	Percival and Anderson 1962[a]
	2001-present	2.5	5	Rielly 2001
	2001-present	1.1	10	Rielly 2001
Co-60	1962-1970	12	10	Percival and Anderson 1962[a]
	1971-1988	5	10	AEC 1972[a]; AEC 1974
	1989	7	10	Martin 1989
	1990-1992	2 (lung)		Rich 1990
	1993-2000	7	10	Grothaus 1993; Andersen, Perry and Ruhter 1995
	2001-present	2.5	5	Rielly 2001
	2001-present	1.1	10	Rielly 2001
Zn-65	1962-1988	12	10	Percival 1962[a]
	1989-2000	10	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001-present	4.9	5	Rielly 2001
	2001-present	2	10	Rielly 2001
Sr/Y-90	1968-1977	70 (skull)	10	AEC 1969[a]; AEC 1972[a]; AEC 1974
	1978-present	34 (skull)	10	Martin 1989; Grothaus 1993
Zr/Nb-95	1962-1988	12	10	Percival and Anderson 1962[a]
	1989-2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995

Radiation/ radionuclide	Period	*In vivo* MDA (nCi)	Count time (min)	Reference
	2001-present	2.6	5	Rielly 2001
Ru-106	2001-present	27	5	Rielly 2001
	2001-present	7.6	10	Rielly 2001
Ag-110^{m}	1962-present	12	10	Percival and Anderson 1962[a]
Sb-125	1962-present	14	10	Martin 1989; Grothaus 1993
I-131	1962-1989	12	10	Percival and Anderson 1962
	1990-1992	2 (thyroid)	10	Rich 1990
	1993-2000	0.3 (thyroid)	10	Grothaus 1993
	2001-present	3.8	5	Rielly 2001
	2001-present	0.13 (thyroid)		Rielly 2001
Cs-134	1989-2000	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1990-present	2(lung)		Rich 1990
	2001-present	3	5	Rielly 2001
	2001-present	0.96	10	Rielly 2001
Cs-137	1962-1970	12	10	Percival and Anderson 1962[a]
	1971-1998	5	10	AEC 1972[a]; AEC 1974; Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1999-2000	2. (lung)	10	Rich 1990
	2001-present	3.1	5	Rielly 2001
	2001-present	1.9	10	Rielly 2001
Ba/La-140	1962-present	12	5	Rielly 2001
Ce-141	1962-present	9.9	5	Rielly 2001
	2001-present	3.2	10	Rielly 2001
	2001-present	0.11 (lung)	60	Reilly 2001
Ce-144	1962-2000	50	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	2001-present	44	5	Rielly 2001
	2001-present	15	10	Rielly 2001
	2001-present	0.44 (lung)	60	Rielly 2001
Eu-152	1962-present	4	10	Rielly 2001
	2001-present	0.18 (lung)	60	Rielly 2001
Eu-154	1962-present	2	10	Rielly 2001
Eu-155	1962-present	1	10	Rielly 2001
Ga-153	1962-present	6.5	10	Rielly 2001
	2001-present	0.096 (lung)	60	Rielly 2001
Hf-181	1962-present	5	10	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
Ta-182	1962-present	12	10	Percival and Anderson 1962[a]
Hg-203	1962-present	12	10	Percival and Anderson 1962[a]
Th-230	1974-present		1,000	AEC 1974
Th-234	2001-present	1.4 (lung)	60	Rielly 2001
Np-237	1974-present			AEC 1974
U-235	1993	0.2 (wound)	20	Grothaus 1993
	1962-present	0.2 (lung)		Rich 1990
	2001	0.11 (lung)	60	Rielly 2001
U-dep/nat	1989	3 (lung)	60	Martin 1989; Grothaus 1993
Pu-238	1989-1998	26 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	1 (wound)	20	Grothaus 1993
	1999-2000	30 (lung)		Rich 1990
	2001	54 (lung)	60	Rielly 2001
Pu-239/240	1971-1993	30	100	AEC 1972[a]
	1974-1988	74 (lung)	100	AEC 1974
	1989-1988	80 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	2 (wound)	20	Grothaus 1993
	1990-2000	30 (lung)		Rich 1990
	2001-present	140 (lung)	60	Rielly 2001
Am-241	1989-1999	0.6 (lung)	60	Martin 1989; Grothaus 1993; Andersen, Perry, and Ruhter 1995
	1993	0.1 (wound)	20	Grothaus 1993
	1990-2000	0.2 (lung)		Rich 1990
	2001-present	0.14 (lung)	60	Rielly 2001

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

Table 5-15. Summary statistics from the 1963 WBC program.

Radionuclide	Times reported	Number of individuals	Maximum activity (µCi)
Cr-51	15	10	1.2
Co-60 / Fe-59	848	387	1.5
Mn-54	98	51	0.16
Co-58	62	50	0.03
Zn-65	505	171	1.20
Zr-/Nb-95[a]	427	232	1.66
Ru-103-106[b]	93	75	0.22
Ag-110m	583	186	0.93
Sb-122	2	2	0.08
I-131	110	82	5.0
Cs-134	361	168	0.14
Cs-137	2332	573	1.32
Ba-/La-140	90	51	0.07
Ce-141-144	59	49	0.16
Ta-182	50	36	0.02
Hg-203	28	6	0.16
Pa-233	13	10	0.48
Np-239	1	1	1.68
Sb-125	3	3	0.1
Mo-Tc-99	8	5	0.72
I-132	8	7	<0.1
I-133	3	3	<0.1
Te-132	6	6	<0.1
Hg-197	7	3	0.7

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

5.6 SPECIFIC FACILITIES

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

5.6.1 Test Area North

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

Following termination of the IET programs, the TAN facilities were used to handle, inspect, store, and prepare for disposal the materials from unplanned reactor excursions. Major reactor components of the damaged SL-1 reactor were examined in the large hot cell and prepared for ultimate disposal. Fuel from the damaged Three Mile Island (TMI) reactor was brought to and stored in the large water pool facility, where it was examined and prepared for permanent dry storage.

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

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

5.6.1.1 Specific Manufacturing Capability Project

The SMC Project began late in the site history (1985) at the ANP Program site in the large hangar facility. The program uses DU to produce armor packages for the U.S. Army M1-A1 and M1-A2 tanks (Stacy 2000). During DU parts fabrication, small quantities of finely divided uranium metal and oxides present inhalation and ingestion potential, as indicated by routine positive personnel bioassays.

Air monitoring is the primary method used at the SMC Project for evaluating 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 indicate potential internal intake. Exposures to concentrations above 0.1 DAC generally will indicate the use of respiratory protection and require bioassay follow-up (King 2001).

The radionuclides of concern at SMC are the isotopes of DU listed in Table 5-16. The mass percentages, relative activities in picocuries per microgram, and the total picocuries per microgram are based upon Integrated Modules for Bioassay Analysis NIOSH default values.

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

Isotope	Mass %	Activity %	Relative activity (pCi μg^{-1})	Total (pCi/µg)
U-238	99.8	83.42	0.3354	0.402
U-236	0.0031	0.05	0.0002	
U-235	0.20	1.07	0.004	
U-234	0.001	15.46	0.062	

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

Inhalation Absorption Type: Respirable particulates associated with SMC Project operations are probably a mixture of metal and metal oxides. The actual exposures are undoubtedly to mixtures of absorption types. During the 18 yr of operation, much bioassay data has been collected on a large number of individuals. The overall elimination patterns are consistent with type M, but probably are a mixture of all types. It could be too simplistic to assume a pure absorption type when the chemical form is not known for certain. The dose reconstructor should assume either type M or type S 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.

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 (Rielly 2001).

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

Natural Background Uranium Excretion: Urine samples submitted by SMC Project nonradiation workers in 1987, 1994, and 1998 were assumed to represent nonoccupational elimination of the SMC Project worker population. The results ranged from 0.04 to 0.33 µg/L with wide fluctuations in individual measurement, some as high as 1.0 µg/L (King 2001). The average reported uranium concentration was 0.157 ±0.109 µg/L at 1σ uncertainty. Therefore, 0.16 µg/L is used as the nonoccupational component of uranium excretion for SMC Project workers, and is subtracted from each urine result before assessment of occupational internal dose. The bioassay results in the worker files reflect the subtraction of 0.16 µg/L from the value determined in the laboratory bioassay result.

5.6.2 Idaho Chemical Processing Plant

The ICPP, now known as INTEC, consists of a complex of high-enriched spent fuel storage basins, fuel dissolution and uranium extraction processing facilities, a high-level liquid waste storage tank farm, high-level waste calcining processes, and associated analytical and support capabilities. The ICPP was a process facility for the recovery of high-enriched uranium from spent fuels from a variety of national and a few foreign reactors. Because high-enriched uranium was the product, the process vessels had to be small for criticality control. Rather than being a plant with large canyon construction and complete remote control and maintenance, the ICPP processes were remotely controlled but contact maintenance was required; that is, maintenance personnel entered process cells and repaired equipment by hand. The process equipment in the cells (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.

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

5.6.2.1 High-Enriched Spent Fuel Storage

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

The basin was used for the storage of aluminum-clad, stainless-steel-clad, zirconium-clad, and sodium-bonded stainless-steel fuels. During 34 yr of operation, cases of cladding leaks resulted in MFP contamination of the basin water. Breaks in the stainless-steel cladding of sodium-bonded elements allowed a sodium–water reaction that increased the dispersal of fission products to the pool water. Pool water contamination up to 0.15 µCi/ml (Rich et al. 1974) existed for periods during which cleanup methods were developed. As a result, during the late 1950s through the 1970s, air activity above the pool area was routinely measured at 10 to 25% (Rich et al. 1974) of the RCG_a [1×10^{-9}

Table 5-17. Notable airborne incidents at ICPP.

Date	Incident	Radionuclides released	Internal dose discussion	Reference
05/15/57	Iodine release to Y-Cell	I-131	Y-Cell modifications resulted in 8 personnel receiving minor thyroid doses in the range of 600 mrem.	Vance 1957
03/20/58	Iodine release at ICPP	I-131	Radioactive iodine spread through makeup area to operating corridor. Thyroid intake to several health physics technicians and operators in the 40-µCi range.	Rich 1958 Hayden 1958
10/16/59	Criticality accident - in shielded process system	Short-lived noble gases and I-131, -132, -133, etc.	Short-lived radioactive gases released to plant areas; internal doses reported as minimal.	Ginkel et al. 1960
01/25/61	Criticality accident - in shielded process system	Short-lived noble gases & I-131, -132, -133, etc.	Short-lived radioactive gases released through process off-gas system to 76-m stack. Internal doses reported as minimal.	Paulus et al. 1961
01/72	Release of ~1.0 Ci Ru-106 from ICPP main stack	Ru-106	No internal doses detected.	ERDA 1977
11/17/72	ICPP mass-spectrometry Pu contamination Incident	Pu-238, -239	An exposure incident involving about a dozen personnel resulted in 50-yr exposure lung doses ranging up to about 4 rem.	Wenzel 1973 Wenzel 1974
10/17/78	Criticality accident in shielded process system	Short-lived noble gases	Short-lived lived radioactive gases released through process off-gas system to 76-m stack. Internal doses reported as minimal.	Casto 1980
11/85	N-Cell Pu Uptake	Pu-238	Internal exposures were far below DOE exposure limits, but showed a weakness in the radiological control program.	Henry and Slagle 1985
10/30/88	Release of ~0.2 Ci of Ru-106 from ICPP main stack	Ru-106	No internal doses detected.	Hoff, Mitchell, and Moore 1989

µCi/ml for soluble ^{90}Sr (AEC 1968)] and was one of the very few operations at INEEL in which operators were allowed to work in fractional MPC_a levels without respiratory protection for several hours a day. Routine bioassay sampling was increased for those personnel. The primary contaminants were aged MFPs, primarily ^{90}Sr/^{90}Y and ^{137}Cs. For workers with established work histories in the Building 603 storage facility for extended periods, and if specific bioassay analyses are either not available or insufficient, a claimant-favorable default intake of 250 RCG_a-hr/yr should be assumed (i.e., 1,000 hr/yr at 0.25 RCG_a):

$$20{,}000\ cm^3/min \times 60\ min/hr \times 1{,}000\ hr/yr \times 0.25 \times 1\times10^{-9}\ \mu Ci/cm^3 \times 1\times10^{6}\ pCi/\mu Ci = 3\times10^{5}\ pCi/yr.$$

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

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

Table 5-18. Radiologically significant radionuclides for ICPP processed fuels[a].

Nuclide	Half-life	Absorption type	Aluminum-clad fuel (decayed 1 yr)		Stainless steel-clad fuel (decayed 3 yr)		Zirconium-clad fuel (decayed 5 yr)	
			Relative activity	Percent inhalation dose	Relative activity	Percent inhalation dose	Relative activity	Percent inhalation dose
Sr-90	28.78 yr	F	2.4E-02	13.1	8.6E-2	14.1	2.0E-1	6.3
Y-91	58.51 d	S	2.6E-02	1.1	9.7E-6	0.0	8.0E-10	0.0
Zr-95	63.98 d	S	4.0E-02	1.2	3.3E-5	0.0	5.6E-9	0.0
Ru-106	368.2 d	S	2.3E-02	5.5	3.1E-2	5.7	4.5E-3	0.2
Cs-137	30.07 yr	F	2.5E-02	1.2	9.3E-2	3.2	2.1E-1	1.4
Ce-144	284.3 d	M	3.0E-01	47.4	1.6E-1	18.4	2.2E-2	0.5
Pm-147	2.623 yr	M	5.6E-02	1.3	1.8E-1	3.2	8.4E-2	0.3
U-234	245500 yr	S	2.9E-09	0.0	1.1E-4	4.0	7.2E-8	0.0
Pu-238	87.71 yr	M	1.3E-04	26.1	7.6E-6	1.2	3.0E-3	90.0
Pu-239	24110 yr	M	4.1E-07	0.1	3.0E-4	50.0	1.6E-6	0.1
Total[b]				97.0		99.0		98.4
Mass fraction U-234 to total U				3.4E-5		5.3E-5		1.4E-4

a. Assume exposure to aluminum-clad fuel activity from the beginning through 1970, zirconium-clad fuel activity from 1971 to present, and stainless-steel-clad fuel activity when there is indication that the Pu-239 activity exceeds the Pu-238 activity.

b. The total percent of the inhalation dose is less than 100% because other radionuclides not included in the table contribute small amounts of dose.

5.6.2.2 High-Level Wastes

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

5.6.2.3 High-Level Waste Calcination

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

5.6.2.4 Process Analytical Facilities

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

5.6.2.5 Spent Fuel Processing and ICPP Most-Limiting Radionuclides

High-enriched fuel processing, which began in 1953, consisted of nitric acid dissolution of aluminum-clad elements, electrolytic nitric acid dissolution of stainless-steel-clad elements, hydrofluoric acid dissolution of zirconium-clad elements, and nitric acid dissolution of graphite fuel

following graphite burning. Uranium was extracted from the dissolved elements (using variations of the plutonium–uranium extraction process, which produced a uranyl nitrate product). The first-, second-, and third-cycle raffinates contained high levels of MAPs and MFPs along with TRU radionuclides. These products, in highly corrosive matrices, were difficult to contain in the confinement barriers of the process piping, process off-gas, etc., and resulted in routine leaks and spills. Most of the leaks were confined to process cells, but they occasionally occurred in occupied process control and equipment areas. For this reason, the extensive network of CAMs and radiation area monitors (RAMs) was essential for timely detection of loss of confinement.

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

To minimize the radiological safety hazard due to the relatively volatile halogens, the fuel was normally decayed a minimum of 120 d before shipment to ICPP for processing. Processing of fuels often did not occur until years later. Because of this relatively long decay time, many of the short half-life radionuclides experienced considerable decay, leaving the actinides to make up a larger percentage of the total radionuclide inventory of the processed fuel. To give an indication of the radionuclide inventory in fuels that were processed at the ICPP, the ORIGEN2 code (Croff 1980) was used to determine the decay of previously calculated fuel inventories (Wenzel 2000) to typical decay times before processing. Aluminum fuel, based on fuel from the Advanced Test Reactor (ATR), was decayed 1 yr. Stainless-steel fuel, based on fuel from EBR-II, was decayed 3 yr. Zirconium fuel, based on various reactors, was decayed 5 yr. These decay times represented the minimum decay times that occurred before processing of the fuels. Table 5-18 lists the radiologically significant nuclides and the percent of the inhalation dose from inhaling activity from these three types of fuel. Several short-lived daughters (Y-90, Rh-106, Ba-137m, and Pr-144) have over 1/3 of the activity which will be followed in IMBA calculations and are not shown. The inhalation dose percentages were calculated using ICRP Publication 68 dose conversion factors for AMADs of 5 µm (ICRP 1995).

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

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

One exception to this planned fuel aging was the radioactive lanthanum (RaLa) process, which operated in L cell of the 601/602 process building from February 1957 to 1963. This process was designed to extract RaLa from green fuel from the Material Test Reactor (MTR) with as little decay as manageable (less than 2 d). Fuel was removed from the MTR, transported about 2 miles to the ICPP in a heavily shielded transport container by a straddle carrier, immediately dissolved, and the barium element was extracted. The ^{140}Ba/La product was shipped immediately to Los Alamos National Laboratory. This process released large quantities of volatile radioactive iodines, which have a higher potential for escaping confinement systems. Several significant internal exposure incidents occurred in which ^{131}I, ^{132}I, and ^{133}I thyroid intakes occurred before personnel could respond to CAM alarms and take protective or corrective actions.

5.6.3 Argonne National Laboratories-West

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

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

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

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

a. Using a Canberra WBC with a 5-min count.
b. Using a large NaI WBC Detector with a 10-min count.
c. Assumes a 100-ml sample.

5.6.4 Radioactive Waste Management Complex

The RWMC has supported NRTS/INEEL operations as a waste management complex since 1952 and has received large quantities of TRU waste from Rocky Flats and other DOE facilities. Improved operations have resulted in a decrease in internal dose potential. The original disposal techniques (dumping waste in open trenches) were relatively vulnerable to airborne release in comparison to

current total-containment practices. The four major areas in the RWMC facility are the Subsurface Disposal Area (SDA) for permanent disposal of low-level waste and some early TRU waste (which will eventually be exhumed and repackaged); the 58-acre Transuranic Storage Area (TSA) for temporary storage, examination, and certification before shipment to the Waste Isolation Pilot Plant; the operations area; and an administrative area where no radioactive waste is permitted.

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

Tables 5-20 and 5-21 summarize the major radionuclides in the RWMC waste inventory. TRU radionuclides are the primary contaminants in the TSA waste; all but ^{241}Pu are alpha emitters. Because the materials are not homogenous, it should not be assumed that no detection of one radionuclide in the inventory invalidates detection of other radionuclides.

Table 5-20. Radioactive waste inventory in the TSA.

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

Table 5-21. Radioactive low-level waste inventory in the active pits in the SDA.

Waste type	Volume (m^3)	Total Ci	Radionuclide	Concentration (Ci/m^3)	Percentage
Low-level waste	75,600	3.35E+05	Co-60	4.1E+0	92.
			Ni-63	3.3E-1	7.4
			Sr-90	9.7E-3	0.22
			Cs-137	9.7E-3	0.22
			H-3	5.8E-3	0.13
			C-14	8.9E-4	0.02

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

5.6.5 Waste Reduction Operations Complex

The Waste Reduction Operations Complex includes several reactor facilities that operated from the 1950s to the late 1960s and the Power Burst Facility (PBF) reactor, which operated from 1972 to 1985. These currently inactive facilities are in a common control area. In addition, a low-level waste incinerator called the Waste Experimental Reduction Facility (WERF) burned waste from all INEEL 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 is in the form of burnable containers and the resultant high-fired and solidified ash. The radioactive wastes at the mixed waste storage facility and the reactors were the sources of the radioactivity inventory. The ashes were removed remotely to a glovebox and solidified in 55-gal drums.

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

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

5.6.6 Test Reactor Area

5.6.6.1 Test Reactor Area Reactors

The MTR was the second operating reactor at the NRTS (from March 1953 to 1970). The TRA also hosted the Engineering Test Reactor (ETR; 1957 to 1981) and the ATR (1967 to present) along with six reactor-critical facilities that supported the test reactors. The TRA complex includes hot cells, a gamma irradiation pool facility, research laboratories, and analytical laboratories. The reactors at TRA, as well as the others at the INEEL, were used for testing materials, experiments, neutron irradiation facilities, and so forth. They were not involved in the production of plutonium or any other weapons materials, as were some DOE reactors.

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

Several factors contributed to unusual amounts of fission products in the PCS of the MTR and ETR during early operations. With cladding technology in its infancy, the quality of the cladding was not the best and fission products leaked through the cladding. Another factor was *tramp* fuel, which was a contaminant on the outside of the cladding. During reactor operation, fission products created in tramp uranium were released directly to the PCS. Reactor operators and the fuel manufacturer resolved these deficiencies over time; by the time the ATR became operational, the PCS of the ATR was considerably less contaminated than that of the MTR or the ETR during their early years of operation. The radiologically significant radionuclides for aluminum fuel processed at the ICPP in Table 5-18 are applicable to this PCS contamination.

The majority of radioactivity releases from the TRA reactors to areas potentially occupied by workers consisted of noble gases that promptly decayed to short-lived particulates. The principle dose to personnel from releases of noble gases was direct radiation rather than inhalation. The direct radiation caused instruments to alarm, which resulted in immediate evacuation of the affected areas. The claimant-favorable position is that there could have been some halogens and particulate radionuclides released along with the noble gases. The ORIGEN2 code (Croff 1980) was used to calculate a radionuclide inventory for ATR fuel at the time of reactor shutdown. The radionuclide inventory was then fractionated according to the release percentages often used to assess

hypothetical reactor accidents (AEC 1962b): 100% noble gases, 50% halogens, and 1% solids. A halogen-to-solid ratio of 50:1 was therefore used to determine the radionuclides that could significantly contribute to inhalation dose. Table 5-22 contains the relative amounts of halogens and particulate radionuclides from ATR fuel that contribute significantly to inhalation dose, but there are too many for efficient dose reconstruction. The iodine isotopes contribute the large majority (94.5%) of the dose from a test reactor gaseous release. Therefore, ^{131}I was selected to be the representative radionuclide for missed dose (in Table 5-24), and the dose from ^{131}I was weighted by a factor of 1.6 to account for the total iodine dose. When there is measured ^{131}I for any facility, a dose reconstructor may choose to apply this factor of 1.6 to the ^{131}I dose to account for short half-life iodines and may choose to ratio ^{144}Ce, ^{89}Sr, ^{91}Y, and ^{95}Zr activities to the ^{131}I activity using the values in Table 5-22.

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

Nuclide	Half-life	Absorption type	Fraction activity	Ratio activity to I-131 activity
Sr-89	50.5 d	S	0.0028	3.5E-02
Y-91	58.51 d	M/S	0.0033	4.1E-02
Zr-95	63.98 d	S	0.0036	4.4E-02
I-131	8.041 d	F	0.0816	
I-132	2.3 hr	F	0.1298	
I-133	20.8 hr	F	0.2040	
I-134	0.876 hr	F	0.2225	
I-135	6.61 hr	F	0.1855	
Ce-144	284.3 d	S	0.0012	1.5E-02
Total			0.834	Not applicable

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

Table 5-23. Major airborne incidents at TRA.

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

Reactor components, such as fuel elements, reactor loop components, and so forth that are removed from the reactor and placed in the canal are a source of contamination in the canal water. If these components are not cleaned adequately before they are removed from the canal, the activity on them can become airborne. The radiologically significant radionuclides listed in Table 5-18 for aluminum fuel processed at the ICPP are applicable to this TRA contamination.

5.6.6.2 Test Reactor Area Laboratories

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

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

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

5.7 DEFAULT FOR MISSED DOSE

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

For most of the history of the INEEL, personnel dosimeters were issued to all workers, regardless of work assignment, who entered the security access control points at each facility. For example, administrative and clerical personnel were required to wear these radiation-monitoring dosimeters even though they were not exposed to elevated backgrounds or internal dose potential. Dose reconstructors should determine the appropriate default categories from Table 5-24 as follows: If the worker's file includes positive external dosimeter readings, those readings should be treated as radiation workers and the default internal missed dose should be applied as outlined in the table. If no detectable external or internal dose information is recorded, only the environmental dose should be included. For example, a worker at TRA in 1975 should be assigned the intakes for "All", for "All but ANL-W", and for "TRA".

5.8 UNMONITORED WORKERS

As noted above, INEEL personnel dosimeters were issued to all workers at facilities handling radioactive material. Many of these workers, due to the nature of their work, would not have been exposed to internal activity and would not have been subjected to routine bioassay.

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

Areas with potential airborne contamination were monitored with alarming CAMs.

The spread of contamination was monitored through the use of smears and monitoring instrumentation.

It was the policy to screen radiation workers through urine samples and WBC for internal contamination many (often 4 or more) times a year.

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

Construction workers often worked at the INEEL. Each construction job was evaluated to determine if radiation exposure or internal dose could be received. When construction work was done in an area with potential radiation exposure, including internal dose exposure, construction workers were

monitored in the same fashion as a radiation worker. Construction workers who were issued personnel dosimeters should be treated the same as facility employees who were issued personnel dosimeters. Construction workers who were not issued a personnel dosimeter should be assigned the environmental dose for the facility.

Table 5-24. Default table for missed dose.

Period	Based on	Site area	Recommendation	Basis
Startup–1960	Urine gross β	All	Calculate chronic Sr-90 intake that results in a urine activity of 0.4 × gross β	Typical β activity is 0.33 Sr-90, 0.33 Y-90 & 0.33 Cs-137. Use of 0.4 is claimant-favorable
			Cs-137 intake = Sr-90 intake	Half-lives and fission yields of Cs-137 & Sr-90 are approximately equal.
			Pu-238 intake = 0.005 × Sr-90 intake	Pu:Sr-90 ratio of 0.005 in Al fuel.
			Ce-144 intake = 12.4 × Sr-90 intake	Ce-144:Sr-90 ratio of 12.4 in Al fuel.
			Y-91 intake = 1.1 × Sr-90 intake	Y-91:Sr-90 ratio of 1.1 in Al fuel
			Zr-95 intake = 1.2 × Sr-90 intake	Zr-95:Sr-90 ratio of 1.2 in Al fuel
		TRA[a]	Annual I-131 intake= 25 μCi	Assumes I-131 in thyroid at 3 days is 5 μCi Vapor Type F, the maximum *in vivo* measured in 1963.
1961-1970	*In vivo* Cs-137	All	Calculate chronic Cs-137 intake that results in the *in vivo* measurement	
			Sr-90 intake = Cs-137 intake when no *in vitro* measurement	Half-lives and fission yields are approximately equal.
			Pu-238 intake = 0.005 × Cs-137 intake when no *in vitro* measurement	Pu:Cs-137 ratio of 0.005 in Al fuel.
			Ce-144 intake = 12.4 × Cs-137 intake when no measurement	Ce-144:Sr-90 ratio of 12.4 in Al fuel.
		TRA	Annual I-131 intake= 60 nCi when no measurement.	Based on MDA for I-131 of 12 nCi at 3 days in thyroid Vapor Type F.
			One acute intake of Ag-110m = 71 nCi when no measurement.	Based on MDA for Ag-110m of 12 nCi at 3 days type M. Known incident involving Ag-110m.
			One acute intake of Ta-182 = 80 nCi when no measurement.	Based on MDA for Ta-182 of 12 nCi at 3 days type M. Known incident involving Ta-182.
1971-1980	*In vivo* Cs-137	All	Sr-90 intake = Cs-137 intake when no *in vitro* measurement	Half-lives and fission yields of Cs-137 & Sr-90 are approximately equal.
		All but ANL-W	Pu-238 intake = 0.02 × Cs-137 intake when no *in vitro* measurement	Pu:Sr-90 ratio of 0.02 in Zr fuel. See Table 5-18, column 9.
			Ce-144 intake = 1.05 × Cs-137 intake when no measurement	Ce-144:Sr-90 ratio of 1.05 in Zr fuel.
		ANL-W	Pu-239 intake = 0.003 × Cs-137 intake when no *in vitro* measurement	Pu:Cs-137 ratio of 0.003 in stainless steel fuel.
			Ce-144 intake = 2 × Cs-137 assigned intake when no measurement	Ce-144:Cs-137 ratio of 2 in stainless steel fuel.
		TRA	Annual I-131 intake= 94 nCi when no measurement.	Based on MDA for I-131 of 12 nCi at 3 days thyroid Vapor Type F weighted by a factor of 1.6.
1981-present	Bioassay	All except ANL-W, INTEC & SMC	Calculate associated radionuclide intakes based on Al fuel distribution	Distribution calculated for Al fuel. See Table 5-18, column 3.
		ANL-W	Calculate associated radionuclide intakes based on stainless-steel fuel distribution	Distribution calculated for stainless steel fuel. See Table 5-18, column 4.
		INTEC & unknown	Calculate associated radionuclide intakes based on Zr fuel distribution	Distribution calculated for Zr fuel. See Table 5-18, column 5.
		SMC	Calculate associated intakes based on known isotope distribution	Isotope distribution, see Table 5-16.

a. The dose recommendations for specific areas, i.e., TRA, INTEC, ANL-W, are to be added to recommendations for "All".
b. Formerly known as ICPP.
c. When the worker's area is not known, or if the employee worked in many different areas, use the dose recommendations for INTEC.
d. "Other area" or areas are known, just not one of the specific areas with special recommendations.

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GLOSSARY

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

activation
The process of inducing radioactivity by irradiation.

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

becquerel
A unit of radioactivity equal to 1 disintegration per second.

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

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

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

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

contamination, radioactive
Radioactive material where it does not belong.

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

criticality
An event that occurs as a result of uncontrolled assembly of fissile material capable of sustaining a nuclear chain reaction.

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

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

depleted uranium (DU)
Uranium that has undergone a process to remove one or more isotopes from the element. Depleted uranium is usually depleted in ^{235}U and ^{234}U, leaving primarily ^{238}U.

dosimeter
A device used to measure accumulated radiation exposure.

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

enriched uranium
Uranium processed to increase the relative abundance of ^{235}U. Most processing methods also increase the relative abundance of ^{234}U.

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

fission product
Radionuclides that result from fission.

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

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

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

hot cell
A specialized shielded laboratory in which radioactive materials can be handled with the aid of remotely operated manipulators. The walls and windows of the laboratory are made of materials designed to protect workers from radiation.

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

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

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

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

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

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

mixed waste
Waste that contains hazardous and radioactive materials.

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

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

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

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

photon
A quantum of electromagnetic energy often referred to as X-rays or gamma rays, but also including light and radiant heat.

proton
An elementary atomic particle with a positive electrical charge equal numerically to the charge of the electron and a mass slightly greater than 1 mass unit.

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

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

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

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

radioactive lanthanum (RaLa)
One of the fission products of a nuclear reaction; a lanthanum recovery process at the INTEC for development of weapons.

rem

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

shielding

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

spent nuclear fuel

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

spent fuel storage basin

A pool or pit made of reinforced concrete containing water and used to store spent nuclear fuel. The water acts as shielding and as a coolant.

type

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

transuranic (TRU) waste

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

X-ray

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

zirconium

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