

<p>ORAU Team NIOSH Dose Reconstruction Project</p> <p>Technical Basis Document for the Idaho National Engineering and Environmental Laboratory – Occupational Internal Dose</p>	<p>Document Number: ORAUT-TKBS-0007-5 Effective Date: 08/11/2004 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 49</p>
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RECORD OF ISSUE/REVISIONS

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Draft	08/11/2004	00-F	Incorporates NIOSH review comments. Initiated by Norman Rohrig.
08/11/2004	08/11/2004	00	First approved issue. Initiated by Norman Rohrig.

ACRONYMS AND ABBREVIATIONS

α	alpha particle
ACGIH	American Conference of Governmental Industrial Hygienists
AEC	Atomic Energy Commission
AECL	Administrative Exposure Control Level
AEDE	Annual Effective Dose Equivalent
AL	Analytical Laboratory
AMAD	activity median aerodynamic diameter
AMWTP	Advanced Mixed Waste Treatment Program
ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory West
ANP	Aircraft Nuclear Propulsion
ANPP	Aircraft Nuclear Propulsion Program
ANSI	American National Standards Institute
ATR	Advanced Test Reactor
β	beta particle
Bq	becquerel
BZ	breathing zone (air monitor)
CAM	continuous air monitor
cc	cubic centimeter
CEDE	committed effective dose equivalent
CERT	Controlled Environmental Radioiodine Tests or Controlled Environmental Radionuclides Tests
CFA	Central Facilities Area
CFR	Code of Federal Regulations
CHTRU	contact handled TRU
Ci	curie
COO	Chicago Operations Office
CPP	Chemical Processing Plant
cpm	counts per minute
d	day
D&D	decontamination and decommissioning
DAC	Derived Air Concentration
DAC-hr	Derived Air Concentration hour
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DU	depleted uranium
DOE-ID	DOE-Idaho Operations Office
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
EBR	Experimental Breeder Reactor
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

FAST	Fluorinel Dissolution Process and Fuel Storage Facility
FCF	Fuel Cutting Facility
Y	gamma
h	hour
HFEF	Hot Fuel Examination Facility
HP	health physicist/health physics
H&S	Health and Safety
HSD	Health and Safety Division
HSL	Health Services Laboratory
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
IDO	Idaho Operations Office
IET	initial engine test
IMBA	Integrated Modules for Bioassay Analysis
INC	Idaho Nuclear Corporation
INEL	Idaho National Engineering Laboratory
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
L	liter
LANL	Los Alamos National Laboratory
LLW	low level waste
keV	kiloelectron volt, 1000 electron volts
μCi/cc	microcurie per cubic centimeter
μg	microgram
μm	micrometer
m	meter
M	moderate absorption type
MAP	mixed activation products
MFP	mixed fission products
mCi	millicurie
MeV	megaelectron volt
mg	milligram
MDA	Minimum Detectable Activity or Amount
MDL	Minimum Detectable Level
min	minute
mL	milliliter
M & O	Management and Operating (Contractor)
MPBB	Maximum Permissible Body Burden
MPLB	Maximum Permissible Lung Burden
MPC	Maximum Permissible Concentration
MPC _a	MPC for airborne activity
mrem	millirem
MRL	Minimum Recording/Reporting Level
MTR	Materials Testing Reactor
NaK	sodium potassium

Nal(Tl)	sodium iodide doped with thallium
NBS	National Bureau of Standards
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH OCAS Claims Tracking System
NRC	Nuclear Regulatory Commission
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
OCAS	(NIOSH) Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
PCS	primary coolant system
PBF	Power Burst Facility
pCi	picocurie
QA	quality assurance
RAF	Remote Analytical Facility
RaLa	radioactive lanthanum
RAM	Radiation/Remote Area Monitor
RBE	Relative Biological Effectiveness
RCIMS	Radiation Control Information Management System
RDR	Radiation Dosimetry and Records
RDL	Research and Development Laboratories
RESL	Radiological Environmental Sciences Laboratory
rep	roentgen equivalent physical
RU	recycled uranium
RWMC	Radioactive Waste Management Complex
S	slow absorption type
SDA	Subsurface Disposal Area
SL-1	Stationary Low-Power Reactor
SMC	Specific Manufacturing Capability
T _{1/2}	Half-life
TAN	Test Area North
TBD	technical basis document
TLV	Threshold Limit Value
TMI	Three Mile Island
TRA	Test Reactor Area
TSA	Transuranic Storage Area
TRU	transuranic
WBC	whole body counter or whole body count
WERF	Waste Experimental Reduction Facility
WIPP	Waste Isolation Pilot Plant
WROC	Waste Reduction Operations Complex
ZPPR	Zero Power Plutonium Reactor (later Zero Power Physics Reactor)

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 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 facility" as defined in the Energy Employee Occupational Illness Compensation Program Act of 2000 (42 U.S.C. § 7384l (5) and (12)).

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

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

- The original mission of the NRTS was (as the name implied) high enriched (>50% and mostly >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 8-10 years 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 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, making three Federal organizations utilizing the NRTS and its infrastructure. Personnel employed in the Naval program and site are not included in this study.
- In addition to the Federal Agencies involved at the Site, there were numerous contractors through the years, who operated the many facilities for the Federal Agencies, and shared some support personnel to varying degrees.
- In order to provide needed consistency of radiation safety programs at the NRTS among a large variety of facilities and constantly changing Contractors, the AEC established a unique feature at the NRTS of a Health and Safety Laboratory (HSL) to provide technical support in the areas of 1) environmental surveillance, 2) external dosimetry (personnel dosimeters of all types), 3) portable radiation detection instrumentation inventories, calibration, and maintenance, 4) internal *in vitro* and *in vivo* bioassay analytical laboratories, 5) technical support in quality

assurance of external and internal radiation dose evaluation, 6) maintenance and documentation of formal/legal personnel dosimetry records, and 7) research and development in these areas of responsibility.

- Though the design and administration of the radiation safety programs in the work place were the responsibility of each facility contractor, oversight of the programs was conducted by the AEC; and technical data, information (particularly in the instances of detectable worker intake), and analytical internal dose calculations/evaluations were exchanged between the AEC HSL and each contractor.

As a consequence, and in spite of the constant changes at the NRTS/INEEL, basic assumptions related to minimum detectable activities (MDAs) or minimum detectable levels (MDLs), missed dose potential, etc. are relatively consistent across the INEEL through the years. There were differences in the available nuclear materials from facility to facility, however, as early as 1955-6 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 identity through gamma spectral and/or chemical separation. If the bioassay records do not include the radionuclide analyses and only record gross beta or alpha results, default assumptions are described in the following text.

There were times when incidents occurred. It was the policy to thoroughly investigate incidents and identify 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 supporting evidence supporting that an individual was involved in the incident, it should be assumed that the individual was not involved in the incident.

The facility descriptions below are provided to aid the internal dose reconstruction through increased general understanding, data interpretation, defaults, etc. Specific information necessary for dose reconstruction is contained in Section 5.5.

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 have been designed, constructed, and operated (including operation-to-destruction tests) on the NRTS. INEEL has experienced a number of reactor episodic events, both planned and accidental [for example, the Stationary Low-Power (SL-1) military reactor 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) project 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 have 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 Chemical Processing Plant (ICPP), currently known as the Idaho Nuclear Technology and Engineering Center (INTEC). The ICPP reprocessed high-enriched reactor fuel (^{235}U enrichments in the expended fuel of

50% to 93%) for 39 years, beginning in 1952 and ending in 1991. Aged mixed fission products (MFPs) were the predominant internal hazard, although enriched uranium isotopes and plutonium isotopes (Pu-238 enhanced) were limiting in specific process locations. 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 also documented in the personnel dosimetry files.

5.1.3 Other Nuclear Facilities and Processes

Other nuclear facilities on the INEEL that resulted in internal dose potential or experience include those described below.

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 have occurred during handling and processing, resulting in releases, this in turn caused intakes of aged MFP, U isotopes, transuranic (TRU) radionuclides, and aged mixed activation products (MAP) (Reilly 2001).

The Specific Manufacturing Capability Project (SMC) is a depleted uranium specialty-parts production plant that was built in 1985 in the ANP hanger on the Test Area North (TAN) site. The SMC processes metric tons of depleted U metal for the production of military shielding units (Stacy 2000). The processes of cutting, machining, and handling U metal produce environments in which both chronic and accidental intakes of depleted U 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 been related primarily to experimental reactor design and development, spent fuel processing, depleted uranium parts production, and low- and high-level radioactive waste treatment and disposal and are more fully described in the INEEL Site Description TBD (ORAU 2003). Table 5.1.4-1, which lists the internal doses received at the INEEL from 1992 to 2000, demonstrates the radionuclides of more recent concern from an internal dose standpoint are MFP (primarily aged), uranium and decay products, and TRU and decay products. This information was presented in Reilly 2001 and can be useful to the reconstruction effort primarily as an indicator of intake radionuclide identity and quantities for this recent ten-year period. Prior to 1990 higher doses were received from the radionuclides listed above, as well as MAP (primarily longer-lived, i.e., Ag-110m and others). Most internal doses have been identified following an incident rather than as a result of routine bioassay measurements.

From these programs and as documented in *INEEL Management and Operating (M&O) Contractor Technical Basis Document (TBD) for Internal Dosimetry, General Technical Basis and Facility Specific Documents* (Reilly 2001) and from Health Services Laboratory/ Radiological Environmental Sciences Laboratory (HSL/RESL) reports, Table 5.1.4-2 lists radionuclides of concern at INEEL. These radionuclides are those for which internal dose has been determined in the past and/or methods have been developed for their detection. The INEEL program follows the DOE guidance and uses the

ICRP 30 nomenclature for solubility as noted in the table. Also provided is the ICRP 68 (ICRP 95) recommended absorption type.

Table 5.1.4-1. INEEL recorded internal doses for 1992-2000.

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	Am-241, Cs-137, Sr-90, PU-238, Pu-239/240
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 known monitoring "flags" being identified.

Through the years at INEEL, plotting urine and fecal elimination curves has shown that an "effective solubility" of M to S per the ICRP 66 system (ICRP 94) is to be expected for the radionuclides listed in Table 5.1.4-2, except for Sr, Cs, I, Hg, and tritium. Elimination curves for ³H, ⁹⁰Sr, ¹³¹I, ¹³⁷Cs, and ²⁰³Hg show them to be typically of F type absorption. The chemical explanation is that radioactive materials in oxygen atmospheres oxidize rapidly, forming less soluble compounds. The default assumption of M or S would be appropriate, based on the most claimant-favorable result to the organ in question.

Table 5.1.4-2. Primary radionuclides of concern at INEEL.

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)

^b Y 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.

5.2 INEEL 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-year history are listed in Tables 2.1 and 2.2 of the INEEL Site Description TBD of this Site Profile (ORAU 2003) and 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-year 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). A unique AEC function organized as the Health and Safety (H&S) Laboratory provided a variety of health and safety support functions to the entire Site, which included external and internal dosimetry, health physics instrumentation, fire department, medical department, 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 used Site support services, including internal dosimetry support, although the bioassay results will be reported through COO.

The INEEL personnel dosimetry records have been and are documented and permanently maintained by the AEC Health and Safety Laboratory and the subsequent managers of these support functions. Records related to the individual facility/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 constant change in operational responsibility through the years and the movement of workers among facilities, there has been a basic level of site-wide consistency in the internal dosimetry programs applied to INEEL facilities and programs, and particularly the bioassay analytical techniques and calculational processes. Though 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 any needed corrective practices/procedures.

Typically, employees were 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 (HP) technicians] worked at multiple facilities and were exposed to a variety of radioactive materials in a variety of work situations.

The internal dose reconstruction for personnel who have 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 AEC H&S Laboratories had radionuclide identification capabilities from the early 1960 time period. Positive bioassay results (analyses in which the results exceeded 2-sigma counting statistics) were normally followed by a confirmatory analysis, identifying a specific radionuclide. In the cases where only gross beta and/or alpha bioassay results are available, the results are normally less than two-sigma. However if it is necessary to evaluate intakes from the gross beta and/or gross alpha results, the radionuclide defaults should be Sr/Y-90 and Pu-238, respectively.

5.2.2 Bioassay Programs

Routine bioassay of radiation workers has been conducted since the beginning of the site. However, formal documentation of the bioassay programs has not been found for times before 1981. Some of the data sheets on individuals indicate that bioassay sampling was routinely done every 6 months in 1953. Table 5.2.2-1 presents the reconstructed history of routine bioassay frequency.

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/or 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 function includes 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 an electronic database that carries reported internal doses as committed effective dose equivalent when an individual's dose history is prepared.

Table 5.2.2-1 Routine bioassay history summary

Year	Typical Frequency	Type	Groups Analyzed/Sampled	Investigating Level	Comments	Reference
1953 - 1960	Annually	<i>in vitro</i> urine	Radiation workers		Frequency is inferred from individual data sheets.	Individual data sheets. Table 5.3.4-1. AEC 1958 AEC 1961
1961	Annually	<i>in vitro</i> urine <i>in vivo</i>	Radiation workers		Frequency is inferred from individual data sheets.	Table 5.3.4-1. AEC 1962a
1962-1972	Annually	<i>in vitro</i> urine <i>in vivo</i>	Radiation workers		Frequency is inferred from individual data sheets.	AEC 1962a AEC 1963b
1973 - 1981	Annually When internal intake suspected Annually	<i>in vitro</i> urine fecal <i>in vivo</i>	Radiation workers	<u>Reporting</u> Annual DE >10% quarterly standard in ERDA Manual Chapter 0524.	Frequency is inferred from individual data sheets.	AEC 1968 AEC 1975 ERDA 1975
1982-1987	Annual When internal intake suspected Annual termination	<i>in vitro</i> urine fecal <i>in vivo</i>	CPP-603 workers Fuel reprocessing operators Waste reprocessing operators Shift lab workers Health physics technicians Selected radio-chemistry workers Maintenance workers Denitrator operators All radiation workers	<u>Reporting</u> 50-year CDE >10% quarterly standard in ERDA Manual Chapter 0524.	Staggered to monitor group throughout the year	Exxon 1981
1988-1989	1 to 6 months Annually 18 to 24 months termination new hire	<i>in vivo</i> <i>in vitro</i> fecal <i>in vitro</i> urine <i>in vivo</i> <i>in vitro</i> <i>in vivo</i>	All radiation workers When internal exposure suspected. Depending on review of radiation dose history.	<u>Investigating</u> Lung 50-yr CEDE > 0.5 – 1.0 rem Bone Surface 50-yr CEDE > 1.0 – 2.0 rem Other Organs 50-yr CEDE > 0.5 - 1.0 rem	Staggered so that worker receives some sort of analysis or sampling every 3 months.	Tschaeche 1988

Table 5.2.2-1 Routine bioassay history summary (continued)

Year	Typical Frequency	Type	Groups Analyzed/Sampled	Investigating Level	Comments	Reference
1990-1994	annually	<i>in vitro</i>	All radiation workers where exposure to surface or airborne radioactive contamination could give at least 0.1 mrem AEDE from occupational sources, or give an organ or tissue DE > 5 rem annual.	<u>Reporting</u> In accordance with DOE Order 5480.11. Workers that could receive 0.1 rem AEDE or 5 rem AEDE organ or tissue dose. <u>Investigating</u> AEDE ≥ 0.01 rem.	Bioassay requested when workplace monitoring program indicates > 0.02 ALI. 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	<i>in vitro</i> fecal				
	annually	<i>in vitro</i> urine				
	new hire	<i>In vivo</i>	Worked at a facility where gamma emitting radionuclides were handled.			
			<i>in vitro</i> urine fecal			
termination	<i>in vivo</i> <i>in vitro</i>	Any employee suspected of having an internal exposure or that was on a scheduled monitoring program.				
1995	Appropriate to the facility mission, potential uptakes. When work-place monitoring indicates significant potential for intakes.	<i>in vivo</i> <i>in vitro</i> 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 U and TRU source terms. 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 non-radiation workers and a radiation worker population.	<u>Reporting</u> In accordance with DOE 5480.1 and 10 CFR 835. Workers that could receive 0.1 rem CEDE. Declared pregnant workers when embryo/fetus could receive 0.05 rem DE. <u>Investigating</u> Internal doses resulting from all confirmed intakes are to be evaluated.	Each facility has a specific Technical Basis Document for Internal Dosimetry. Follow-up for any suspected intake of radionuclides and to more accurately identify and characterize the amount of intake and excretion pattern.	Andersen 1995

Table 5.2.2-1 Routine bioassay history summary (continued)

Year	Typical Frequency	Type	Groups Analyzed/Sampled	Investigating Level	Comments	Reference
1995-2000	new hire		Based on screening to determine internal conditions from previous uptakes or to establish baseline for those continuing to work s radiation workers.			
	termination	<i>in vivo</i> <i>in vitro</i>	Any employee that was on a scheduled monitoring program.			
2001	As developed by individual facilities based on analysis tables developed for each radionuclide.	<i>in vivo</i> <i>in vitro</i> urine fecal	All radiation workers	<u>Reporting</u> In accordance with DEOE 5480.1 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	<i>in vivo</i> <i>in vitro</i>	Any employee that was on a scheduled monitoring program.	<u>Investigating</u> uranium > 1.0 µg/l <i>in vitro</i> activity detected > 2σ. <i>in vivo</i> > 2.33 σ. Default trigger levels exceeded.		

5.2.4 Internal Dose Regulations, Investigation Levels, and Data Codes

The following information is important to internal dose reconstruction, since the claimant files from DOE can contain a variety of internal dose information, ranging from the calculated internal doses as well as the *in vitro* and *in vivo* individual bioassay results. The 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 5.2.4-1 contains the description of coded information that may be encountered post 1989. Table 5.2.4-2 contains internal dose information that may be in pre-1989 records. Table 5.2.4-3 contains analytical nomenclature that may be encountered. Table 5.2.4-4 contains INEEL area codes.

Federal regulations related to permissible internal dose and formal reporting requirements to the AEC/ERDA/DOE changed periodically through the years of the history of the NRTS/INEEL. While the regulations influenced the level of calculated internal dose, which would be reported and/or documented, 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 an investigation of any potential internal intake as indicated by positive air sampling, personnel contamination, etc., by *in vitro* and/or *in vivo* bioassay analyses. 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 (AEC/ERDA 0524, 1968-

1977, required periodic urinalyses and/or in vivo counting and/or evaluation of air concentrations if the whole body dose or dose commitment could exceed 300 mrem in a calendar quarter). Changes in the reporting levels did not generally result in changes to the air monitoring and bioassay sampling programs. Each individual analytical result was documented and placed in individual exposure files, regardless of the formal reporting requirements.

Table 5.2.4-1 Internal dose assessment information post 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)- No. represents 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 30 solubility class symbol D, W, or Y. Amount in either μCi or Bq
CEDE rem	Calculated Committed Effective Dose Equivalent in rem
Organ (Max.)	Organ that received the maximum dose from the specified intake
Organ CDE rem	Committed Dose Equivalent calculated for the listed organ in rem
Employer and Exp. Location	Abbreviation of DOE Site Contractor and the plant site of exposure (may 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 5.2.4-2 Internal dose assessment information pre 1989.

Dose Information	Description
Name, Soc. Sec. No.	Employee name, social security number, and (Contractor Abbreviation/Plant or Facility)
Nuclide	Nuclide symbol followed by ICRP solubility class (D, W, or Y)
Intake Period	Month and Year for single exposure or period of time by month and year in which exposure occurred
CEDE rem	Calculated Committed Effective Dose Equivalent in rem
AEDE rem	Calculated Annual Effective Dose Equivalent in rem
Year	Year for which the AEDE was calculated

The “investigation levels” have also changed little from the early times to the present, i.e. positive bioassay results triggered follow-up sampling to verify that detectable activity had been taken into the body. Dickson (1977) established official investigation levels, which are shown in Table 5.2.4-5, for acute uptakes of radionuclides corresponding to one-tenth of the quarterly radiation standard. Later procedures (DOE 1988) have set specific limits on those positive bioassay results which could result in 100 mrem annual EDE or above as the point at which followup and reporting was required. With the DOE Radiological Control Manual, this changed to 100 mrem CEDE. Also a calculated dose of 10 mrem or above would be recorded as an internal dose. These procedural limits did not materially affect the bioassay sampling frequency and the recording of even non-detectable radioactivity in bioassay samples, although the request for and number of followup samples and/or analyses may have been different as a function of the formal regulations in effect.

Table 5.2.4-3 Analytical information that may be in claimant dose files.

Analytical Information	Description
Sample No.	Sample log number
Date and Time	Generally clear interpretation
Sample Description	Name of the employee, numerical sample number frequently included, additional special analyses that were performed, i.e. Sr-90, Y Separation, etc.
* Anal. For	Generally gross beta and/or gross gamma. Sample aliquots evaporated for gross beta and/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 K and/or U
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 – c/m	Counts/min. determined by dividing total counts by time of count
Bkgd. c/m	Background counts/minute recorded
Net count-c/m	Gross c/m minus background c/m
K ⁴⁰ corr. – c/m	Additional background from K-40 identified. ⁴⁰ K is not a facility occupational product – ignore from an internal dose reconstruction
Foreign Activity – c/m and d/m	Net counts corrected for ⁴⁰ K and then converted to d/m based upon counter calibration. Uncertainty also included, which is recorded as 1 standard deviation 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 5.2.4-4 Area codes that may be in claimant dose files.

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

Table 5.2.4-5 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	

5.3 INTERNAL DOSE CONTROL

The radiological protection program was established to provide detection of barrier or ventilation failure in a timely manner and 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 (respirators, etc.), 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. Generally, radiological materials handled at the Site were of relatively low volume and mass and of higher activity concentration, as opposed to 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 generating airborne contamination was thought to exist, regardless of the actual measured air or surface contamination. These practices influence the assumptions related to dose evaluation in internal dose reconstruction.

As a related note, 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 ALARA, some reduction in acceptable release levels was also 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/or follow-up evaluations and actions. Beta/gamma MDAs typically were a factor of 5 below the limits. Table 5.3-1 is a summary of control limits primarily from the CPP Health Physics Manual, April 1952 (ACC 1952) and current operating procedures.

Table 5.3-1 Surface Contamination Control and MDAs.

Time Period	Surface Location	Detection Technique	Control Levels	MDA - Typical
1952 – 1960s	Plant/Equipment	Smears	500 d/m β & 20 d/m α per 100 cm ²	150 d/m β & 10 d/m α per 100 cm ²
	Personal Clothing	Portable Survey Instruments	1500 d/m β & 500 d/m α per 100 cm ²	1000 d/m β & 500 d/m α per 100 cm ²
	Personal Skin	Portable Survey Instruments	Any detectable reported, e.g. 1000 d/m β & 500 d/m α per 100 cm ²	1000 d/m β & 500 d/m α per 100 cm ²
	Shipments	Smears/Portable Survey Instrument	500 d/m β & 20 d/m α per 100 cm ² – smears 0.1 mrep/hr β & 500 d/m α per 100 cm ²	150 d/m β & 10 d/m α per 100 cm ² smears. 0.01 mrep/hr β & 500 d/m α per 100 cm ²
1970s - present	Plant/equipment Surfaces	Smears	300 d/m β & 20 d/m α per 100 cm ²	30 d/m β & 10 d/m α per 100 cm ²
	Personnel	Portable Survey Instruments	Any detectable reported, e.g. 300d/m β & 200 d/m α per 80-100 cm ²	300d/m β & 200 d/m 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 SMC (1985) 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) in the $> 10^{-9}$ $\mu\text{Ci/cc}$ range. TRU and U 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.

A 1952 CPP Health Physics Manual describes a CAM and three other air sampling systems. The 1952 manual required use of a “filter type respirator” when airborne activity exceeded 10^{-8} $\mu\text{Ci/cc}$ for beta-gamma activity or 10^{-11} $\mu\text{Ci/cc}$ for alpha activity. An army assault-type mask was invoked when levels exceeded this by a factor of 10. Positive air-supply masks were invoked if levels larger by a factor of 1000 were experienced (ACC 1952, p IX: 4-1).

The CAM monitoring systems provided “live” or “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 system 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 HPs were established for interpreting CAM responses for various situations, radionuclides, response times, filter accumulation, etc. 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.

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

However, in the absence of bioassay data for a known radiation worker in one of the functioning facilities as a default, claimant favorable, unmonitored dose 100 DAC/MPC-hours for Sr/Y-90 or Pu-238 could be assumed per year. This is based upon a consistent and standing policy of taking confirmatory bioassay for workers exposed to known levels of air activity. However for an MDA for chronic activity detection would be < 0.01 MPC (in the range of 10^{-11} $\mu\text{Ci/cc}$ for beta) $\times 10$ ratio of general area to breathing zone concentration $\times 1000$ hours of undetected exposure = approx 100 DAC-hrs or approximately 2×10^5 pCi intake.

5.3.2 Early Technical and Analytical Capabilities at NRTS

DOE HSL technical reports and annual reports, coupled with facility memos and reports, document the analytical detection capability dating from the early 1950s and 1960s. For example, during ANP-IET runs 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 (strontium, iodine, etc.) 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 and/or gamma count directly on a liquid sample. Any detectable activity triggered a specific chemical separation analysis – generally for Sr. Early analyses for Pu generally were gross alpha counts on a Pu separation – later alpha spectroscopy was used to count and better characterize the results.

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

AEC (1961) outlined a basic philosophy related to gamma counting of bioassay samples. Gamma counting would be effective in all situations except for exposure to pure strontium isotopes. In order to guard against this unlikely possibility, the procedure was established of performing a strontium analysis for individual “workers at risk” (radiation workers) every two years and/or at termination. Because of the improbability of finding detectable activity, all activities were to be precipitated by oxalic acid in a weak acid solution, gross beta counted, and the strontium analysis not completed unless a detectable count was obtained on the precipitate. A 100-mL sample of urine permitted the detection of approximately 8×10^{-8} $\mu\text{Ci/mL}$ of Sr-90.

Table 5.3.2-1 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 (1962). 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 urinalysis in 1959 (11,066) when 3524 people had radiation badges with 715 receiving external doses above 500 mrem demonstrates that people provided urine samples multiple times during the year.

Table 5.3.2-2 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/substances for the purpose of bioassay (AEC 1964).

Table 5.3.2-1. Urinalysis results in 1959, 1960, and 1961

Nuclide/ Element of Interest	Type Activity	Total Number Performed			Statistically Significant						Highest Result ^a		
					Number			Percent					
		1959	1960	1961	1959	1960	1961	1959	1960	1961	1959	1960	1961
	Gross β^b	8,546	8,546	30	65	15	5	0.76	0.18	17	18,820 ± 632 d/m/5 mL	992± 40 d/m/5 mL	172±16 d/m/mL
	Gross γ^c	2,433	2,712	9,120	174	129	-- ^d	7.15	4.76	4	35,972 ± 310 d/m/5mL	19,817±105 d/m/5 mL	1,900,235±876 d/m/75 mL
Co-60	β	--	--	1	--	--	0	--	--	0	--	--	300±75 d/m/450 mL
Sr-90	β	3	105	3,248	3	0	2	100	0	2	4.12 x 10 ⁻² d/m/mL	Insignificant	183±8 d/m/75 mL
Sr-91	β	20	37	2	19	0	--	95	0	0	388 ± 1.6	Insignificant	4±8 d/m/mL
I^e	β	--	9	--	--	2	--	--	22	--	0	9992±80 d/m/mL	--
Cs-137	β	--	--	40	--	--	0	--	--	0	--	--	1460±10% d/m/1700 mL
Ba-139	α	20	--	--	16	0	--	80	0	--	120 ± 0.8 d/m/mL	0	--
Th^f	α	7	0	--	0	0	--	0	0	--	Insignificant	0	Insignificant
U^g	α	--	--	4	--	--	--	--	--	0	--	--	10 µg/L
U-233	α	17	3	--	1	0	--	.06	0	--	180 ± 4.0 d/m/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			

1

^a All except two ¹³¹I 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 I isotope(s) not identified in reference. Assume I-131.

^f Th isotope(s) not identified in reference. Assume Th-228.

^g U isotope(s) not identified in reference. Assume U-235.

Table 5.3.2-2. Detection limits applicable to environmental sample analyses (1953 to 1965)

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

^aReduced in October 1962 from 1.5E-7 μ Ci/mL

^bReduced in September 1962 from 50 pCi/liter.

The special and routine bioassay sample analyses were performed and documented by the DOE analytical laboratory. The individual analytical data were recorded, as were the specific doses for which formal evaluations were required. Prior to 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 DOE Manual Chapter 0524, the controlling regulation at the time (Aoki 1979). In 1977, the policy was changed to assign internal doses when the committed dose equivalent (CDE) to an organ exceeded one-tenth of the quarterly radiation protection standard. Replies to requests for radiation exposure history prior to 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 on NOCTS.

5.4 MINIMUM DETECTABLE ACTIVITIES

In compliance with the November 1998 Code of Federal Regulations requirement (10 CFR 835) for DOE Laboratory Accreditation Program (DOELAP), and based on American National Standards Institute (ANSI) N 13.30, *Performance Criteria for Radiobioassay* (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 (2- sigma) are performed. Current MDAs for urine and fecal sample analysis are listed respectively in Table 5.4-1 and Table 5.4.2 along with values gleaned from historical documents. By far the majority of the urine samples taken at the Site were single voidings, with 24-hr samples used for special sampling purposes, i.e. followup samples, primarily to extend the sensitivity. The MDAs listed are those for the primary samples. The recommended time periods for the MDA values have been included in the table.

The current *in vivo* MDAs are listed in Table 5.4-3 along with values gleaned from historical documents. The recommended time periods for the MDA values have been included in the table.

5.5 DEVELOPMENT AND IMPACT OF WHOLE BODY COUNTING

Whole body counting 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., ¹²⁵Sb, ^{110m}Ag, ⁶⁵Zn, and ⁹⁵Zr/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-hour specimen (AEC 1962a; Sill et al. 1964). Whole-body counting was demonstrated to detect activity as low as 0.01 μ Ci in a 10 minute count. (AEC 1962a) This detection level was several orders of magnitude lower than the maximum permissible body burdens for most beta/gamma fission and activation products.

As a consequence, the *in vivo* counting program was used to count a) all terminating employees, who required physicals exams, b) employees who were suspected of having a possible internal intake, and c) selected groups of individuals scheduled for semi-routine analyses by Health Physics Supervisors (Sommers 1961). In CY 1963 approximately 1650 counts were performed, with only those activities >0.1 μCi being quantified further. This level was determined to be less than one-tenth of the maximum permissible body burden for most of the gamma-emitting isotopes. Table 5.4-1 contains MDLs for internal dosimetry analytical counting by date.

The 1963 Annual Summary Report (AEC 1964) describes the year's follow-up for the whole-body counting program as presented in Table 5.5-1. 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 might be found in someone in earlier years before the whole body counter was operational.

Table 5.4-1. Internal dosimetry *in vitro* MDAs for urine samples by date.

Radiation/ Radionuclide	Year	Urine $\mu\text{Ci}/\text{mL}$	Reference
Gross β	Start date - 1953	8.4	Data Sheet
	1954-1959	9.3	Ebersole & Flygare 1957
	1960-70	3.5	AEC 1961 ^a
	1971-present	1	AEC 1972 ^a
Gross γ	1960-1964	3.5	AEC 1961 ^a
	1965-present	2.3	Data Sheet
H-3	Start date - 1994	35	AEC 1972 ^a , ACB 1974
	1995-present	1.4	Andersen et al. 1995, Rielly 2001
Co-60	1963-present	0.01	Rich 1990
Sr-90	1953-1959	9.2	Ebersole 1957
	1960-1970	0.2	AEC 1961
	1971-1989	0.02	AEC 1972 ^a , ACB 1974
	1990-present	0.01	Rich 1990
I-131	1963-present	0.01	Rich 1990
Cs-134	1963-present	0.01	Rich 1990
Cs-137	1963-present	0.01	Rich 1990
Th-230	1974-present	5.E-5	ACB 1974
Np-237	1974-present	0.00005	ACB 1974
U (FP)	1954-1985	14 $\mu\text{g}/\text{L}$	
U (KPA)	1985-2001	0.17 $\mu\text{g}/\text{L}$	
U-233/234	1970-present	1.E-4	Rich 1990
U-235	1970-present	1.E-4	Rich 1990
U-238	1970-present	1.E-4	Rich 1990
Pu-238	1974-present	2. E-5	ACB 1974
Pu-239/240	1964-1970	4.E-4	AEC 1964 ^a
	1971-1973	5.E-4	AEC 1972 ^a
	1974-1989	2 E-5	ACB 1974
	1990-present	6.E-5	Rich 1990
Am-241	1974-1989	7 E-5	ACB 1974
	1990-present	2. E-4	Rich 1990
Cm-244	1974-present	1. E-5	ACB 1974
Cf-252	1974-present	1. E-5	ACB 1974

^a MDA calculated from inferred 2σ uncertainty.

Table 5.4-2. Internal dosimetry *in vitro* MDAs for fecal samples by date

Radiation/ Radionuclide	Year	Fecal ^a pCi/ Sample	Reference
Co-60	1963-present	10	Rich 1990
Sr-90	1963-1994	10	Rich 1990
	1995-present	1.9	Andersen et al. 1995, Rielly 2001
Cs-134	1963-present	10	Rich 1990
Cs-137	1963-1999	0.01	Rich 1990
	2000-present	0.3	PLN-153 2000
Th-230	1974-present	0.0003	ACB 1974
Np-237	1974-present	0.0003	ACB 1974
U-233/234	1970-2002	0.041	Andersen et al. 1995, Rielly 2001
	2003-present	0.05	Bhatt 2003
U-235	1970-2003	0.038	Andersen et al. 1995, Rielly 2001
	2003-present	0.09	Bhatt 2003
U-238	1970-1994	0.5	Rich 1990
	1995-2002	0.03	Andersen et al. 1995, PLN-153 2000, Rielly 2001
	2003-present	0.09	Bhatt 2003
Pu-238	1974-1994	0.03	ACB 1974
	1995-2002	0.022	Andersen et al. 1995, Rielly 2001
	2003-present	0.02	Bhatt 2003
Pu-239/240	1964-1973	0.4	AEC 1964 ^b
	1974-1994	0.02	ACB 1974
	1995-present	0.03	Andersen et al. 1995, PLN-153 2000, Rielly 2001, Bhatt 2002
Am-241	1974-1994	0.07	ACB 1974
	1995-2001	0.023	Andersen et al. 1995, Rielly 2001
	2002-present	0.04	Bhatt 2002
Cm-244	1974-present	0.02	ACB 1974
Cf-252	1974-present	0.02	ACB 1974

^a When sample size is not identified in individual's records, assume the activity is that excreted per day.

^b MDA calculated from inferred 2σ uncertainty.

Table 5.4-3. Internal dosimetry *in vivo* MDAs by date.

Radiation/ Radionuclide	Year	<i>In vivo</i> nCi	<i>In vivo</i> Count Time (min)	Reference
γ Spectrometer	1974	2.E-4 (thyroid)	10	ACB 1974
Cr-51	1962-2000	12	10	Percival 1962 ^a
	2001-present	32	5	Rielly 2001
Mn-54	1962-2000	5	10	Martin 1989, Grothaus 1993, Andersen et al. 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 1962 ^a
	2001-present	2.5	5	Rielly 2001
	2001-present	1.1	10	Rielly 2001
Co-60	1962-1970	12	10	Percival 1962 ^a
	1971-1988	5	10	AEC 1972 ^a , ACB 1974
	1989	7	10	Martin 1989
	1990-1992	2 (lung)		Rich 1990
	1993-2000	7	10	Grothaus 1993, Andersen et al. 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 et al. 1995
	2001-present	4.9	5	Rielly 2001
	2001-present	2	10	Rielly 2001
Sr-90	1968-1977	70 (skull)	10	Voelz 1969 ^a , AEC 1972 ^a , ACB 1974
	1978-present	34 (skull)	10	Martin 1989, Grothaus 1993
Zr/Nb-95	1962-1988	12	10	Percival 1962 ^a
	1989-2000	5	10	Martin 1989, Grothaus 1993, Andersen et al. 1995
	2001-present	2.6	5	Rielly 2001
Ru-106	2001-present	27	5	Rielly 2001
	2001-present	7.6	10	Rielly 2001
Ag-110 ^m	1962-present	12	10	Percival 1962 ^a
Sb-125	1962-present	14	10	Martin 1989, Grothaus 1993
I-131	1962-1989	12	10	Percival 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
I-133	1989-1998	5	10	Martin 1989, Grothaus 1993, Andersen et al. 1995
	1999-2000	2 (lung)		Rich 1990
	2001-present	3.	5	Rielly 2001
	2001-present	0.96	10	Rielly 2001

^a MDA calculated from inferred 2σ uncertainty.

Table 5.4-3. Internal dosimetry *in vivo* MDAs by date (continued).

Radiation/ Radionuclide	Year	<i>In vivo</i> nCi	<i>In vivo</i> Count Time (min)	Reference
Cs-137	1962-1970	12	10	Percival 1962 ^a
	1971-1998	5	10	AEC 1972 ^a , ACB 1974, Martin 1989, Grothaus 1993, Andersen et al. 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 et al. 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 et al. 1995
Ta-182	1962-present	12	10	Percival 1962 ^a
Hg-203	1962-present	12	10	Percival 1962 ^a
Th-230	1974-present		1000	ACB 1974
Th-234	2001-present	1.4 (lung)	60	Rielly 2001
Np-237	1974-present			ACB 1974
Uranium	1974	30 mg (lung)	40	ACB 1974
U-233/234	1990-present	0.2 (lung)		Rich 1990
U-235	1993	0.2 (wound)	20	Grothaus 1993
	1962-present	0.2 (lung)		Rich 1990
	2001	0.11 (lung)	60	Rielly 2001
U-238	1990-present	0.2 (lung)		Rich 1990
U-dep/nat	1989	3 (lung)	60	Martin 1989, Grothaus 1993
	1989-1998	26 (lung)	60	Martin 1989, Grothaus 1993, Andersen et al. 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 ^b
	1974-1988	74 (lung)	100	ACB 1974
	1989-1988	80 (lung)	60	Martin 1989, Grothaus 1993, Andersen et al. 1995
	1993	2 (wound)	20	Grothaus 1993
	1990-2000	30 (lung)		Rich 1990
	2001-present	140 (lung)	60	Rielly 2001

Table 5.4-3. Internal dosimetry *in vivo* MDAs by date (continued).

Radiation/ Radionuclide	Year	<i>In vivo</i> nCi	<i>In vivo</i> Count Time (min)	Reference
Am-241	1989-1999	0.6 (lung)	60	Martin 1989, Grothaus 1993, Andersen et al. 1995
	1993	0.1 (wound)	20	Grothaus 1993
	1990-2000	0.2 (lung)		Rich 1990
	2001-present	0.14 (lung)	60	Rielly 2001
Cm-244	1974-present		1000	ACB 1974
Cf-252	1974-present		1000	ACB 1974

Table 5.5-1. Summary statistics from the 1963 whole-body counting program.

Radionuclide	Times Reported	Number of Individuals	Maximum Activity (μ Ci)
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	427	232	1.66
Ru-103-106	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

5.6 SPECIFIC FACILITIES

Each of the facilities were responsible for conducting internal dosimetry monitoring programs, which were designed both to prevent and/or mitigate internal exposure and to evaluate and document internal dose above the detectable limits. The descriptions below provide insight and default instructions for dose reconstruction if the specific operational facility is known.

5.6.1 Test Area North

GE 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. 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. Large hot cells, maintenance shops, water storage pools, waste management areas, etc. are featured (Stacy 2000).

The Initial Engine Tests (IET) in the 1950s were conducted under area controls and radiological monitoring surveillance. The reactors operated in the open environment and each test involved the release of large quantities of short-lived radioactive fission product gasses and volatiles. However, workers were protected by enclosures (control point buildings, etc.) and constant monitoring for identification of unanticipated exposures, etc.

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 ¹³⁷Cs and ⁹⁰Sr/Y) with periods when ⁹⁵Zr/Nb and ¹⁴⁴Ce were present. Activation products were also encountered – primarily ⁶⁰Co. Alpha emitters (U isotopes and TRU radionuclides) were present, but in ratios of 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 operation-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 site in the large hangar facility. The program uses depleted uranium (DU) to produce armor packages for the U.S. Army M1-A1 and M1-A2 main battle tanks (Stacy 2000). During DU parts fabrication, small quantities of finely divided U 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 > 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 isotopes of depleted U as listed in Table 5.6.1.-1. The mass percentages, relative activities in pCi/μg, and the total pCi/μg are based upon IMBA NIOSH default values.

Table 5.6.1.-1. Mass and activity ratios of SMC DU uranium isotopes^a

Isotope	Mass %	Activity %	Relative activity (pCi μg ⁻¹)	Total pCi μg ⁻¹
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 U isotopes, two beta-emitting radionuclides are found in DU. Due to short half-lives, ²³⁴Th and ^{234m}Pa reach equilibrium with the ²³⁸U 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 operations are probably a mixture of metal and metal oxides. The actual exposures are undoubtedly to mixtures of absorption types. During the 18 years 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 might 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 SMC is not considered credible.

Particle Size: Detailed particle sizing analyses of representative samples from the various operations indicate that an AMAD of 2.4 μm is appropriate for typical SMC operations. This site-specific value of 2.4 μm AMAD is used for assessments of intakes at the SMC and is the default SMC 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 (ACGIH) is 0.2 mg/m^3 . The TLV is defined by the ACGIH as "...the time-weighted average concentration for a normal 8-hr workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect." Converting 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} $\mu\text{Ci}/\text{cc}$. The DAC for Type M U is 3×10^{-10} $\mu\text{Ci}/\text{cc}$, which is a factor of 4 more than the chemical TLV. The SMC staff has always been aware of the need to consider the chemical toxicity of SMC DU exposure in addition to the radiological limit.

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

5.6.2 Idaho Chemical Processing Plant (ICPP) – currently INTEC

The ICPP consists of a complex of high-enriched spent fuel storage basins, fuel dissolution and U extraction processing facilities, 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 considerations. Rather than being a plant with large canyon construction and complete remote control and maintenance, the ICPP processes were remotely controlled, but contact maintenance (i.e., maintenance personnel entered process cells and repaired equipment using hands-on maintenance) was required. The process equipment in the cells (5 feet of high-density concrete) were decontaminated by flushing/rinsing with concentrated acids and complexing agents prior to entry by health physics and maintenance personnel. These occasional operations were well planned, but they carried high potential for internal exposures.

The primary internal dose experience at the ICPP resulted from accidental releases. Table 5.6.2-1 lists unusual and episodic events that have occurred at the ICPP.

Table 5.6.2-1: Notable Airborne Incidents at ICPP.

Date	Incident	Radionuclide(s) 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 HP technicians and operators in the 40- μ Ci 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-meter 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-year 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-meter 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 et al. 1989

5.6.2.1 High-Enriched Spent Fuel Storage

The original spent fuel storage facility (CPP-603) was a 1.5-million-gallon, 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 years 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 and > 0.5 μ Ci/mL 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 in the few tenths to 1 MPC_a levels 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 MFP – primarily ⁹⁰Sr/Y and ¹³⁷Cs. For claimants in which it can be established that they worked in the Building 603 storage facility for extended time periods, and specific bioassay analyses are either not available or insufficient, a claimant favorable default intake of 1000 DAC-hours per year should be assumed, i.e. 1000 hours/year at 2 x 10⁻⁹ μ Ci/cc:

$$20,000 \text{ cc/min} \times 60 \text{ min/h} \times 1000 \text{ h/y} \times 2 \times 10^{-9} \text{ } \mu\text{Ci/cc} \times 10^6 \text{ pCi}/\mu\text{Ci} = 2.4 \times 10^6 \text{ pCi/y.}$$

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

5.6.2.2 High-Level Wastes

The high-level waste storage tank farm consists of a series of 500,000-gallon, 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-mild steel connections. These spills resulted in extremely high-levels of contaminated soil that were removed with remote equipment and transport. 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, leaving the radionuclides high-fired coated on calcine granules. The potential for release of high-level MFP and MAP and TRU particulate activity was significant, but 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 (RAF) 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, etc. These incidents frequently involved loss of control of TRU materials during analytical procedures in hoods or benchtop 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 PUREX process), producing a uranyl nitrate product. The first-, second-, and third-cycle raffinates contained high levels of fission and activation products along with TRU. 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 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 fuels reprocessed were highly enriched with ²³⁵U enrichments in the expended fuels 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 program. In addition, some fuels that had no burnup or very little burnup were reprocessed in a hands-on operation known as Custom Processing.

In order to minimize the radiological safety hazard due to the relatively volatile halogens, the fuel was normally decayed a minimum of 120 days 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 decay 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 120 days. Stainless steel fuel, based on fuel from EBR-II, was decayed 3 years. Zirconium fuel, based on various reactors, was decayed 5 years. These decay times represent the minimum decay times that occurred prior to processing of the fuels. Table 5.6.2.5-1 lists the radiologically significant nuclides and the percent of the inhalation dose from inhaling activity from these three types of fuel. The inhalation dose percentages were calculated using ICRP 68 5 µm AMAD dose conversion factors.

Table 5.6.2.5-1. Radiologically significant radionuclides for ICPP processed fuels^a.

Nuclide	Half-life	Absorption Type	Aluminum-Clad Fuel (decayed 120 d)		Stainless Steel-Clad Fuel (decayed 3 y)		Zirconium-Clad Fuel (decayed 5 y)	
			Relative Activity	Percent Inhalation Dose	Relative Activity	Percent Inhalation Dose	Relative Activity	Percent Inhalation Dose
Sr-89	50.5 d	F	7.8E-2	1.7	1.0E-6	0.0	2.1E-11	0.0
Sr-90	28.78 y	F	5.7E-3	2.7	8.6E-2	13.3	2.0E-1	6.0
Y-90	64.00 h	S	5.7E-3	0.2	8.6E-2	0.8	2.0E-1	0.3
Y-91	58.51 d	S	1.1E-1	10.9	9.7E-6	0.0	8.0E-10	0.0
Zr-95	63.98 d	S	1.4E-1	8.9	3.3E-5	0.0	5.6E-9	0.0
Nb-95	35.15 d	M/S	2.4E-1	4.9	7.3E-5	0.0	1.3E-8	0.0
Ru-103	39.26 d	S	6.10	1.09	0.00	0.00	0.00	0.00
Ru-106	368.2 d	S	8.9E-3	4.8	3.1E-2	5.7	4.5E-3	0.2
Cs-134	2.07 y	F	9.2E-09	0.7	3.1E-12	0.1	5.0E-09	0.4
Cs-137	30.07 y	F	6.3E-3	0.7	9.3E-2	3.2	2.1E-1	1.4
Ce-144	284.3 d	M	1.3E-1	46.7	1.6E-1	18.4	2.2E-2	0.5
Pm-147	2.623 y	M	1.6E-2	0.9	1.8E-1	3.2	8.4E-2	0.3
U-234	245500 y	S	6.3E-10	0.0	1.1E-4	4.0	7.2E-8	0.0
Pu-238	87.71 y	M	3.1E-5	14.4	7.6E-6	1.2	3.0E-3	90.0
Pu-239	24110 y	M	1.1E-7	0.1	3.0E-4	50.0	1.6E-6	0.1
Total ^b				98.6		99.8		99.2
Total of highlighted select nuclides				64.5		86.1		97.9
Weighting factor for the 4 selected radionuclides				1.6		1.2		1.0
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.

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-

lived 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 5.6.2.5-1 contains too many radionuclides for efficient dose reconstruction. Rather than include all of the radionuclides in the default summary table for missed dose (Table 5.7-1), 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 *in vivo* results rather than for its dose contribution. The potential missed inhalation dose from the other radionuclides is accounted for by weighting the dose from these selected radionuclides by the weighting factors at the bottom of Table 5.7-1. This gives an equivalent to 100% of the dose from the radionuclide distributions for the three types of fuels.

One exception to this planned fuel aging was the RaLa process, which operated in "L" cell of the 601/602 process building beginning in February 1957 to 1963. This process was designed for the ICPP to extract radioactive lanthanum (RaLa) from "green" fuel from the MTR reactor with as little decay as manageable (less than 2 days decay). Fuel was removed from the MTR reactor, transported about 2 miles to the ICPP in a heavy shielded transport container by a "straddle carrier," immediately dissolved, and the barium element was extracted. The $^{140}\text{Ba/La}$ product was shipped immediately to Los Alamos National Laboratory (LANL). 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 the 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 field office. However, over the 52 years of ANL-W experience, informal program coordination has occurred, such that technical consistency with the other INEEL facility internal dosimetry programs exists.

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, U-fueled, Pu-fueled, breeder reactor designs; and self-destruct experiments. As listed in Table 5.6.3-1, the radionuclides of concern ranged the spectrum of MFP, MAP, U, and TRU.

5.6.4 Radioactive Waste Management Complex

The RWMC has supported NRTS/INEEL operations as a waste management complex since 1952 and has received major 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 to open trenches, were relatively vulnerable to airborne release compared 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 be exhumed and repackaged; the 58-acre Transuranic Storage Area (TSA) for temporary storage, examination, and certification prior to shipment to the Waste Isolation Pilot Plant (WIPP); operations area; and an administrative area, where no radioactive waste is permitted.

Table 5.6.3-1. Radionuclides of concern, ANL-W locations, and MDAs.

Radionuclides	Absorption Type	Sources/characteristics	Urine pCi/mL	<i>In vivo</i> ^a 5 min count nCi	<i>In vivo</i> ^b 10 min count nCi
H-3 (HTO)	F	EBR-II Reactor Facility & Sodium Components Maintenance Shop	2.		
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, gloveboxes, 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.	
Pu-239			0.02 ^c	161.	
Am-241	M	FCF, HFEF, ZPPR, Analytical Laboratory		0.1	

^aUsing a Canberra WBC with a 5-minute count.

^bUsing a large NaI WBC Detector with a 10-minute count.

^cAssumes a 100 mL sample.

A comprehensive radiation protection program is practiced, which includes extensive air monitoring, personnel contamination, and surface contamination surveillance. Though infrequent, there have been instances of inadvertent intakes (recently there was one in 1996 and two in 1992). Therefore, current routine bioassay is random.

Tables 5.6.4-1 and 5.6.4-2 summarize the major radionuclides in the RWMC waste inventory. TRU radionuclides are the primary contaminants in the TSA waste. All but ²⁴¹Pu are alpha emitters. Because the materials are not homogenous, it should not be assumed that no detection of one radionuclide in the inventory invalidates detection of other radionuclides.

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 68.

Table 5.6.4-1. Radioactive waste inventory in the TSA.

Waste type	Volume (m ³)	Total curies	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 5.6.4-2. Radioactive low-level waste inventory in the active pits in the SDA.

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

5.6.5 Waste Reduction Operations Complex

The Waste Reduction Operations Complex (WROC) includes several reactor facilities that operated from the 1950s to the late 1960s and the Power Burst Facility (PBF), a reactor that 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 (D&D), was a low-level waste incinerator, including some mixed-waste treatment (Stacy 2000).

The waste at WERF is in the form of burnable containers and resulting high-fired and solidified ash. The radioactive waste at the mixed waste storage facility and the reactors were the sources of the radioactivity inventory. The ashes were removed remotely to a glove box and solidified in 55-gallon drums.

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

Since the type of radioactive materials processed at WERF varied, depending on the area shipping the waste to WERF, the claimant favorable assumption is that radioactive materials came from Zr fuel as processed at the ICPP (see Table 5.6.2.5-1, columns 7 and 8).

5.6.6 Test Reactor Area

5.6.6.1 TRA Reactors

The MTR was the second operating reactor at the NRTS (from March 1953 to 1970); the ETR (1957-1981) and the ATR (1967-present) along with the six reactor critical facilities supporting the test reactors were also at the TRA. 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, etc. They were not involved in the production of Pu or any other weapons materials as were some DOE reactors.

The uranium in the TRA reactors is enriched to 93% with ²³⁵U. The fuel cladding for the TRA reactors is Al. The predominant activation product in the cladding of Al is ²⁴Na, formed by activation of Na present in the Al. Sodium-24 has a half-life of 15 hours and emits a high-energy gamma ray (2.75 MeV). The inhalation dose to personnel from ²⁴Na is insignificant compared to the fission products in the fuel. There are minor levels of activation products of stainless steel, ⁵⁸Co, ⁶⁰Co, ⁵¹Cr, ⁵⁶Mn, 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 claimant files directly. If applicable data is absent, the *in vivo* MDLs may be assumed, consistent with information in Tables 5.1.1-1 and 5.4-3.

Several factors contributed to unusual amounts of fission products in the PCS of the MTR and ETR during early operations. With fuel element 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 that was a contaminant on the outside of the cladding. During reactor operation, fission products created in tramp U were released directly to the PCS. Reactor operators and the fuel manufacturer resolved these deficiencies over time such that 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 AI fuel processed at the ICPP in Table 5.6.2.5-1 are also 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 particulate. The principle dose to personnel from releases of noble gases was direct radiation, not inhalation. The direct radiation caused instruments to alarm, resulting 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 ORIGEN-2 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 an inhalation dose. Table 5.6.6.1-1 contains the relative amounts of halogens and particulate radionuclides from ATR fuel that contribute significantly to the inhalation dose. Table 5.6.6.1-1 contains too many radionuclides for efficient dose reconstruction. As shown in Table 5.6.6.1-1, the iodine isotopes contribute by far the majority of the dose from a test reactor gaseous release, 94.5 percent. Therefore, I-131 has been selected to be the representative radionuclide for missed dose in Table 5.7-1, and the dose from ¹³¹I is weighted by a factor of 1.6 to account for the total iodine dose. When there is measured I-131 for any facility, a dose reconstructor may choose to apply this factor of 1.6 to the I-131 dose to account for shorter lived iodines and may choose to ratio Ce-144, Sr-89, Y-91, and Zr-95 activities to the I-131 activity using the values in Table 5.6.6.1-1

Table 5.6.6.1-1. 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 h	F	0.1298	
I-133	20.8 h	F	0.2040	
I-134	0.876 h	F	0.2225	
I-135	6.61 h	F	0.1855	
Ce-144	284.3 d	S	0.0012	1.5E-02
Total			0.834	

Another potential source of inhalation dose was from the pressurized loop experiments in the TRA reactors. These loop experiments sometimes contained Ag and Ta, which became activated, producing ^{110m}Ag and ¹⁸²Ta. 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 5.6.6.1-2 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 5.4-3) for $^{110\text{m}}\text{Ag}$ and ^{182}Ta .

Reactor components, such as fuel elements, reactor loop components, etc., 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 for listed in Table 5.6.2.5-1 for AI fuel processed at the ICPP, are applicable to this TRA contamination also.

Table 5.6.6.1-2. Major airborne incidents at TRA.

Date	Incident	Radionuclide(s) released	Reference
3/28/1954	GE-ANP-1 depressurization	Noble gas	Sommers, 1954
12/17/1958	GEH-4 rupture	Noble gas + I	Sommers 1958
6/13/1967	GA-18-1 depressurization	Ta-182 and Ta-183	Nertney et al. 1967
1/6/1967	Noble gas release at ATR	Noble gas	Sommers 1977

5.6.6.2 TRA Laboratories

The wing buildings of the MTR at TRA house chemistry and other laboratories. Over the years, experiments in the laboratories resulted in contamination and airborne activity incidents. These experiments involved various isotopes of Pu, U, and other radionuclides. The radiologically significant radionuclides listed in Table 5.6.2.5-1 for AI fuel processed at the ICPP are applicable for the experiments done in the TRA laboratories.

Beginning in 1980, many of the laboratories were involved in various assays of TMI reactor fuel from the accident in March 1979. Because the TMI core had only the equivalent of 100 effective full-power days 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 Zr fuel processed at the ICPP in Table 5.6.2.5-1 is claimant favorable for evaluating inhalation dose from exposure to contamination from the TMI fuel.

5.7 DEFAULT FOR MISSED DOSE

Based upon the INEEL characteristics and circumstances, a number of missed dose default assumptions have been derived. Table 5.7-1 is a summary of these recommended defaults. The bases for these defaults are discussed in more detail in Table 5.7-1 below:

For most of the history of the INEEL personnel dosimeters were issued to all workers who entered the security access control points at each facility, regardless of the work assignments. For example, administrative and clerical personnel were required to wear these radiation monitoring dosimeters though they were not exposed to elevated backgrounds or internal dose potential. The default categories listed in the Table 5.7-1 are determined by the following: If claimant file includes positive external dosimeter readings, they should be treated as radiation workers and the default internal missed dose is applied as outlined in the table. If no detectable external or internal dose information is recorded, only the environmental dose should be included.

Table 5.7-1. Default table for missed dose

Time Period	Based On	Recommendation	Site Area	Basis
Start up date through 1960	Urine gross β	Calculate chronic Sr-90 intake that results in a urine activity of 0.4 x gross β	All	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 = 0.008 x Sr-90 intake		Pu:Sr-90 ratio of 0.005 in AI fuel weighted by a factor of 1.6. See Table 5.6.2.5-1, column 5.
		Ce-144 intake = 35 x Sr-90 intake		Ce-144:Sr-90 ratio of 22 in AI fuel weighted by a factor of 1.6. See Table 5.6.2.5-1.
		I-131 = 5 μ Ci <i>in vivo</i>	TRA ^a	Assumes I-131 <i>in vivo</i> is equal to the maximum <i>in vivo</i> measured in 1963.
1961 – 1970	<i>in vivo</i> Cs-137	Calculate chronic Cs-137 intake that results in the <i>in vivo</i> measurement	All	Half-lives and fission yields are approximately equal.
		Sr-90 intake = Cs-137 intake when no <i>in vitro</i> measurement		
		Pu-238 intake = 0.008 x Cs-137 intake when no <i>in vitro</i> measurement		
		Ce-144 intake = 35 x Cs-137 intake when no measurement	TRA	Ce-144:Sr-90 ratio of 22 in AI fuel weighted by a factor of 1.6. See Table 5.6.2.5-1, column 5.
		I-131 = 19 nCi <i>in vivo</i> when no measurement.		MDA for I-131 of 12 nCi weighted by a factor of 1.6. See Table 5.4-3.
		Acute intake of Ag-110m = 12 nCi <i>in vivo</i> when no measurement. Assume 3 days post intake.		MDA for Ag-110m, see Table 5.4-3. Known incident involving Ag-110m.
		Acute intake of Ta-182 = 12 nCi <i>in vivo</i> when no measurement. Assume 3 days post intake.		MDA for Ta-182, see Table 5.4-3. Known incident involving Ta-182
1971-1980	<i>in vivo</i> Cs-137	Sr-90 intake = Cs-137 intake when no <i>in vitro</i> measurement	All	Half-lives and fission yields of Cs-137 & Sr-90 are approximately equal.
		Pu-238 intake = 0.02 x Cs-137 intake when no <i>in vitro</i> measurement		Pu:Sr-90 ratio of 0.02 in Zr fuel. See Table 5.6.2.5-1, column 9.
		Ce-144 intake = 35 x Cs-137 intake when no measurement	INTEC ^b & unknown ^c	Ce-144:Sr-90 ratio of 22 in AI fuel weighted by a factor of 1.6. See Table 5.6.2.5-1, column 5.
		Pu-239 intake = 0.004 x Cs-137 intake when no <i>in vitro</i> measurement	ANL	Pu:Cs-137 ratio of 0.003 in stainless steel fuel weighted by a factor of 1.2. See Table 5.6.2.5-1, column 7

^a The dose recommendations specific areas, i.e., TRA, INTEC, ANL, are to be added to recommendations for "All".

^b Formerly known as ICPP.

^c When it is not known what site area a worker worked at, or if they worked at many different areas, use the dose recommendations for INTEC.

Table 5.7-1. Default table for missed dose (continued)

Time Period	Based On	Recommendation	Site Areas	Basis
1971-1980 (continued)	<i>in vivo</i> Cs-137 (continued)	Ce-144 intake = 2.4 x Cs-137 assigned intake when no measurement	ANL (continued)	Ce-144:Cs-137 ratio of 2 in stainless steel fuel weighted by a factor of 1.2. See Table 5.6.2.5-1, column 5.
		Pu-238 intake = 0.008 x assigned Cs-137 intake when no <i>in vitro</i> measurement	other area ^a	Pu:Cs-137 ratio of 0.005 in Al fuel weighted by a factor of 1.6. See Table 5.6.2.5-1, column 5.
		Ce-144 intake = 35 x Cs-137 assigned intake when no measurement		Ce-144:Sr-90 ratio of 22 in Al fuel weighted by a factor of 1.6. See Table 5.6.2.5-1, column 5.
		I-131 = 19 nCi when not reported	TRA	MDA for I-131 of 12 nCi weighted by a factor of 1.6. See Table 5.4-3.
1981 - present	Bioassay	Calculate associated radionuclide intakes based on Al fuel distribution	all except ANL, INTEC & SMC	Distribution calculated for Al fuel. See Table 5.6.2.5-1, column 3.
		Calculate associated radionuclide intakes based on stainless steel fuel distribution	ANL	Distribution calculated for stainless steel fuel. See Table 5.6.2.5-1, column 4.
		Calculate associated radionuclide intakes based on Zr fuel distribution	INTEC & unknown	Distribution calculated for Zr fuel. See Table 5.6.2.5-1, column 5.
		Calculate associated intakes based on known isotope distribution	SMC	Isotope distribution, see Table 5.6.1-1.

^a "Other area" or areas are known, just not one of the specific areas with special recommendations.

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 has had a rigorous radiation protection program to detect the presence of and the spread of radioactive contamination. Areas with potential airborne contamination were monitored with alarming continuous air monitors. 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 whole body counting for internal contamination many (often 4 or more) times a year. Most of the activity encountered was well tagged with β/γ activity which would have produced measurable direct radiation doses on a personnel dosimeter. Due to the rigor of the radiation monitoring program, the probability that a worker receiving a significant unmonitored internal intake of radioactive material is very low. It is recommended that workers who have no recorded internal dose and wore a personnel dosimeter be treated the same as a worker who was monitored but had no bioassay results exceeding reporting levels. It is further recommended that individuals that 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 the same as a radiation worker. Construction workers that were issued personnel dosimeters should be treated the same as facility employees that were issued personnel dosimeters. Construction workers that were not issued a personnel dosimeter should be assigned the environmental dose for the facility.

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GLOSSARY

absorption

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

activation

The process of inducing radioactivity by irradiation.

Atomic Energy Commission

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

Becquerel

A unit of radioactivity equal to one-disintegration per second.

beta radiation

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

Breeder reactor

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

calcine

The dry solid (grainy or granular) product of a chemical process of removing liquids from a solution; also, 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 that result from the fissioning of the fuel.

contamination, radioactive

Radioactive material where it doesn't belong.

core

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

criticality

An event which 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 (dps).

decontaminate

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

depleted uranium

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 the isotope ^{235}U . Most processing methods increase the relative abundance of ^{234}U , also.

fission

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

fission product

Radionuclides resulting from fission.

fuel reprocessing

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

gamma rays

Short wave length electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture) in an energy range of 10 keV to 9 MeV.

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 may 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 a differing number of neutrons (different mass number).

millirem

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

microcurie

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

mixed waste

Waste that contains hazardous and radioactive materials.

natural uranium

Uranium that has not been through an enrichment process.

neutron

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

nuclear waste

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

nucleus

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

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, Q

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

radiation

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

radioactivity

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

radionuclide

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

RaLa

Radioactive lanthanum, one of the fission products of a nuclear reaction; a lanthanum recovery process at the INTEC for development weapons.

rem

A unit of dose equivalent, equal to the product of the absorbed dose and the quality factor.

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 years and concentration greater than or equal to 100 nCi/g.

X-ray

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

zirconium

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