SC&A S. COHEN & ASSOCIATES
AN EMPLOYEE-OWNED COMPANY

January 25, 2006

David Staudt
Contracting Officer
Centers for Disease Control & Prevention
Acquisition and Assistance Field Branch
P.O. Box 18070
626 Cochrans Mill Road – B-140
Pittsburgh, PA 15236-0295

Subject: Contract No. 200-2004-03805, Task Order 1: Draft Review of the NIOSH Site Profile for the Idaho National Laboratory, Idaho – Revision 1

Dear Mr. Staudt:

S. Cohen & Associates (SC&A, Inc.) is pleased to submit its revised draft *Review of the NIOSH Site Profile for the Idaho National Laboratory, Idaho*. This report includes Attachment 3, "Summaries of Site Expert Reviews," that was missing from Revision 0 of the document submitted to the Advisory Board and NIOSH on September 23, 2005. Revision 1 also includes an Attachment 5, "Issue Resolution Matrix for Findings and Key Observations." Both of these attachments have undergone review by DOE. In preparing this deliverable, the authors have also made some minor editorial changes that do not affect the content of the original material.

Sincerely,

John Mauro, PhD, CHP
Project Manager

cc:
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Draft

ADVISORY BOARD ON RADIATION AND WORKER HEALTH

National Institute of Occupational Safety and Health

Review of the NIOSH Site Profile for the Idaho National Laboratory, Idaho

Contract No. 200-2004-03805
Task Order No. 1

SCA-TR-TASK1-0005

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January 25, 2006

S. COHEN & ASSOCIATES: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program***	Document No. SCA-TR-TASK1-0005 Effective Date: Draft – January 25, 2006 Revision No. 1 (Draft)
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Project Manager: ________________ Date:_____________ John Mauro	

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AEDE	Annual Effective Dose Equivalent
α	Alpha Particle
AMWTP	Advanced Mixed Waste Treatment Program
ANC	Aerojet Nuclear Corporation
ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory West
ANP	Aircraft Nuclear Propulsion
ANPP	Aircraft Nuclear Propulsion Program
ANSI	American National Standards Institute
APS	Atmospheric Protection System
ARA	Army Reactor Area (later Auxiliary Reactor Area)
AREA	Army Reactor Experimental Area
ARMF-1	Advanced Reactivity Measurements Facility No. 1
ARMF-2	Advanced Reactivity Measurements Facility No. 2
ARVFS	Army Reentry Vehicle Facility Site
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical Facility
β	Beta Particle
BORAX	Boiling Water Reactor Experiment
BORAX I	Boiling Water Reactor Experiment No. 1
BORAX II	Boiling Water Reactor Experiment No. 2
BORAX III	Boiling Water Reactor Experiment No. 3
BORAX IV	Boiling Water Reactor Experiment No. 4
BORAX V	Boiling Water Reactor Experiment No. 5
Bq	Becquerel (1 disintegration per second)
C	Celsius/Centigrade
CA	Contamination Area
CAM	Continuous Air Monitor
CAMU	Corrective Action Management Unit
CAS	Criticality Accident System

CDC	Centers for Disease Control and Prevention
CE	Critical Experiment
CEDE	Committed Effective Dose Equivalent
CEL	Chemical Engineering Laboratory
CERCLA	Comprehensive Environmental Response Compensation and Liability Act
CERT	Controlled Environmental Radioiodine Tests (later Radionuclides)
CET	Critical Experiment Tank
CFA	Central Facilities Area
Ci	Curie
cm	Centimeter
cpm	Counts per Minute
CPP	Chemical Processing Plant
D&D	Decontamination and Decommissioning
DAC	Derived Air Concentration
DE	Dose Equivalent
DL	Detection Limit
DNFSB	Defense Nuclear Facilities Safety Board
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOE-HQ	DOE-Headquarters
DOE-ID	DOE-Idaho Operations Office
DOELAP	DOE Laboratory Accreditation Program
DU	Depleted Uranium
EBOR	Experimental Beryllium Oxide Reactor
EBR	Experimental Breeder Reactor
EBR-I	Experimental Breeder Reactor No. 1
EBR-II	Experimental Breeder Reactor No. 2
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EMDR	Environmental Monitoring Data Report
EMR	Environmental Monitoring Report
EPA	Environmental Protection Agency
ERDA	Energy Research and Development Administration

ESRF	Environmental Science and Research Foundation
ETF	Effluent Treatment Facility
ETR	Engineering Test Reactor
ETRC	Engineering Test Reactor Critical
EXCES	Experimental Cloud Exposure Study
F	Fahrenheit
FARET	Fast Reactor Test
FASB	Fuel Assembly and Storage Building
FAST	Fluorinel Dissolution Process and Fuel Storage Facility
FCF	Fuel Cutting Facility
FDF	Fluorinel Dissolution Facility
FEBT	Fuel Element Burn Test
FECF	Fuel Element Cutting Facility
FFTF	Fast Flux Test Facility
FMF	Fuel Manufacturing Facility
FNCF	Facility Neutron Correction Factor
FP	Fission Product
FPF	Fuel Processing Facility
FPFRT	Fission Product Field Release Test
FPR	Fuel Processing Restoration
FSF	Fuel Storage Facility
γ	Gamma
GCRE	Gas-Cooled Reactor Experiment
GE	General Electric Corporation
GE-ANP	General Electric-Advanced Nuclear Propulsion (Program)
H&S	Health & Safety (management organization)
HCA	High Contamination Area
HEPA	High-Efficiency Particulate Air
HEU	Highly Enriched Uranium
HFEF	Hot Fuel Examination Facility
HLLW	High-Level Liquid Waste
HLW	High-Level Waste

HOTCE	Hot Critical Experiment
HP	Health Physicist/Health Physics
HPIL	Health Physics Instrument Laboratory
HPP	Hot Pilot Plant
HRA	High Radiation Area
HSL	Health Services Laboratory
HTRE	Heat Transfer Reactor Experiment
IBM	International Business Machines
IC	Initial Criticality
ICPP	Idaho Chemical Processing Plant (formerly CPP and now INTEC)
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
ID	Idaho
IDO	Idaho Operations Office
IET	Initial Engine Test
IFR	Integral Fast Reactor
IFSF	Irradiated Fuel Storage Facility
ILTSF	Intermediate-Level Transuranic Storage Facility
IMBA	Internal Modular Bioassay Analysis
in	Inch
INC	Idaho Nuclear Corporation
INEC	Idaho Nuclear Energy Commission
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INELHDE	Idaho National Engineering Laboratory Historical Dose Evaluation
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center (formerly ICPP and CPP)
IRC	INEL Research Center (a facility in Idaho Falls)
IREP	Interactive RadioEpidemiological Program
ISFSI	Independent Spent Fuel Storage Installation
ISU	Idaho State University
IWP	Industrial Waste Pond

KERMA	Kinetic Energy Released in Matter
keV	Kilo Electron Volt, 1,000 Electron Volts
kg	Kilogram
kV	Kilovolt
kVp	Kilovolt Peak
kW	Kilowatt
lat	Lateral
lb	Pound
LET	Linear Energy Transfer
LLD	Lower Limit of Detection
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss-of-Coolant Accident
LOFT	Loss of Fluid Test (Facility)
LPT	Low-Power Test (Facility)
LSC	Liquid Scintillation Counter
M	Moderate (solubility rate)
mA	Milliampere
MDA	Minimum Detectable Activity
MDF	Materials Development Facility
MeV	Mega Electron Volt, 1 Million Electron Volts
mg	Milligram
ML-1	Mobile Low-power Reactor No. 1
mm	Millimeter
MPBB	Maximum Permissible Body Burden
MPLB	Maximum Permissible Lung Burden
MPOB	Maximum Permissible Organ Burden
mR	Milliroentgen
mrad	Millirad
mrem	Millirem
MRL	Minimum Reporting Level
msec	Millisecond
MTR	Materials Test Reactor

MW	Megawatt
NASA	National Aeronautics and Space Administration
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurement
NERP	National Environmental Research Park
NIOSH	National Institute of Occupational Safety and Health
NOAA	National Oceanic and Atmospheric Administration
NRAD	Neutron Radiography Facility
NRC	Nuclear Regulatory Commission
NRF	Naval Reactor Facility
NRTS	National Reactor Testing Station
NTA	Nuclear Track Emulsion-Type A
NVLAP	National Voluntary Laboratory Accreditation Program
NWCF	New Waste Calcining Facility
OCAS	(NIOSH) Office of Compensation Analysis and Support
OMRE	Organic Moderated Reactor Experiment
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
PA	Posterior-Anterior
PAS	Personal Air Sampler
PBF	Power Burst Facility
PIC	Pocket Ionization Chamber (i.e., "Pencil" Dosimeter)
PIF	Process Improvement Facilities
POC	Probability of Causation
PREPP	Process Experimental Pilot Plant
psi	Pounds Per Square Inch
R	Roentgen
RAC	Risk Assessment Corporation
rad	Radiation Absorbed Dose
RAF	Remote Analytical Facility
RAL	Remote Analytical Laboratory
RaLa	Radioactive Lanthanum

RAM	Radiation/Remote Area Monitor
RBE	Relative Biological Effectiveness
RBOF	Receiving Basin for Offsite Fuel
RCRA	Resource Recovery and Conservation Act
RCT	Radiological Control Technician
RDT	Relative Diffusion Test
rem	Roentgen Equivalent Man
rep	Roentgen-equivalent-physical
RESL	Radiological and Environmental Services Laboratory
RHA	Radiation Hazards Analysis
RLWTF	Radioactive Liquid Waste Treatment Facility
RM	Radioactive Material
RMF	Reactivity Measurement Facility
RML	Radiation Measurements Laboratory
RPP	Radiological Protection Program
RPSSA	Radioactive Parts Service and Storage Area
RSAC	Radiological Safety Analysis Computer
RSWF	Radioactive Scrap and Waste Facility
RU	Recycled Uranium
RWMC	Radioactive Waste Management Complex
RWMIS	Radioactive Waste Management Information Service
RWP	Radiological Work Permit
S	Slow (solubility rate)
SC&A	S. Cohen & Associates
SDA	Subsurface Disposal Area
SEC	Special Exposure Cohort
SID	Source to Image Distance
SIS	Special Isotope Separations
SL-1	Stationary Low-Power Reactor No. 1
SM-1	Stationary Medium-Power Reactor
SMC	Specific Manufacturing Capability
SNAP	Systems for Nuclear Auxiliary Power

SNAPTRAN	Systems for Nuclear Auxiliary Power Transient
SPERT	Special Power Excursion Reactor Test
SPF	Sodium Processing Facility
SRS	Savannah River Site
SSD	Source to Skin Distance
SSSTF	Staging, Storage, Sizing, and Treatment Facility
STEP	Safety Test Engineering Program
STM	Stack Tritium Monitor
STP	Sewage Treatment Ponds
STPF	Shield Test Pool Facility
STR	Split-Table Reactor
Sv	Sievert
SWDF	Solid Waste Disposal System
SWEPP	Stored Waste Examination Pilot Plant
SWP	Safe Work Permit
t	Thermal
TAN	Test Area North
TBD	Technical Basis Document
TCE	Trichlorethylene
TEPC	Tissue Equivalent Proportional Counter
TLD	Thermoluminescent Dosimeter
TLND	Thermoluminescent Neutron Dosimeter
TMI	Three Mile Island
TRA	Test Reactor Area
TREAT	Transient Reactor Test Facility
TRIGA	Training Research and Isotope General Atomic
TRU	Transuranics
TRUPACT	Transuranic Packaging Transporter
TSA	Transuranic Storage Area
TSF	Technical Support Facility
UNH	Uranium Nitrate Hexahydrate
U.S.C.	United States Code

USGS	U. S. Geological Survey
WAC	Waste Acceptance Criteria
WBRR	Western Beam Research Reactor
WCA	Waste Characterization Area
WCF	Waste Calcining Facility
WERF	Waste Experimental Reduction Facility
WIPP	Waste Isolation Pilot Plant
WROC	Waste Reduction Operations Complex
WRRTF	Water Reactor Research Test Facility
ZPPR	Zero Power Plutonium Reactor (later Zero Power Physics Reactor)
ZPR III	Zero Power Reactor No. 3
ZPR	Zero Power Reactor

1.0 EXECUTIVE SUMMARY

S. Cohen and Associates (SC&A, Inc.) evaluated the following Technical Basis Documents (TBDs) documents related to historical occupational exposures at the Idaho National Laboratory (INL[1]) Site:

- ORAUT-TKBS-0007-1, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Introduction* (Rohrig 2004i)

- ORAUT-TKBS-0007-2, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Site Description* (Rohrig 2005)

- ORAUT-TKBS-0007-3, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational Medical Dose* (Rohrig 2004m)

- ORAUT-TKBS-0007-4, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational Environmental Dose* (Peterson 2004)

- ORAUT-TKBS-0007-5, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational Internal Dose* (Rich and Wenzel 2004)

- ORAUT-TKBS-0007-6, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational External Dosimetry* (Rohrig 2004e)

These six documents TBDs taken together are often referred to as the INL Site Profile.

In addition, SC&A evaluated and made use of the following Technical Information Bulletins (TIBs), the general National Institute for Occupational Safety and Health (NIOSH) dose evaluation guidelines, and a supplemental document that relate to the INL Site Profile:

- ORAUT-OTIB-0002, *Technical Information Bulletin – Maximizing Internal Dose Estimates for Certain DOE Complex Claims* (Rollins 2004)

- ORAUT-OTIB-0006, *Technical Information Bulletin – Dose Reconstruction from Occupationally Related Diagnostic X-Ray Projections* (Kathren 2003)

- ORAUT-OTIB-0007, *Technical Information Bulletin – Occupational Dose from Elevated Ambient Levels of External Radiation* (Strom 2003)

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[1]The laboratory recently changed names to the Idaho National Laboratory (INL) from its previous name, the Idaho National Engineering and Environmental Laboratory (INEEL). Both names will be used in this report to refer to the same organization depending on the context.

- ORAUT-OTIB-0009, *Technical Information Bulletin in Support of INEEL Technical Basis Document Section 6: Reanalysis of Hankins MTR Bonner Sphere Surveys* (Rohrig 2004h)

- ORAUT-OTIB-0023, *Assignment of Missed Neutron Doses Based on Dosimeter Records* (Merwin 2005)

- OCAS–IG-001, *External Dose Reconstruction Implementation Guideline* (Taulbee 2002)

- OCAS–IG-002, *Internal Dose Reconstruction Implementation Guideline* (Allen 2002)

- *Supplement to Technical Basis Document 4 for the Idaho National Engineering and Environmental Laboratory: INEEL Occupational Environmental Dose* (Peterson 2004s)

The SC&A review also was informed by a number of outside documents and sources, including interviews with groups of former and current site personnel (summarized in Attachments 2 and 3), review of U.S. Department of Energy (DOE) Tiger Team findings, review of Defense Nuclear Facilities Safety Board (DNFSB) assessments, some documents originally provided to NIOSH by INL staff for the development of these six TBDs, some site documents provided to the SC&A team by INL staff during the site interview meetings, and written and teleconference communications with personnel from NIOSH's prime contractor for the dose reconstruction project, Oak Ridge Associated Universities (ORAU), and its subcontractors (summarized in Attachment 1). The review also examined the popular (i.e., non-scholarly) history book written by Susan Stacy for the DOE in recognition of the 50th anniversary of the laboratory: *Proving the Principle – A History of the Idaho National Engineering and Environmental Laboratory 1949–1999* (Stacy 2000).

The TBDs, supplementing individual claimant exposure data provided by DOE and information gathered in interviews with claimants, support the performance of individual dose reconstructions under the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA). The TBDs contain compilations and analyses of data, such as those related to facility operations and processes, radiological source term characterizations, chemical and physical forms of the radionuclides, historic workplace conditions and practices, incidents and accidents involving potential exposures, limits of detection of radiation monitoring methods, and direction for assigning internal and external doses to monitored and unmonitored workers.

As the support contractor to the Advisory Board on Radiation and Worker Health (Advisory Board), SC&A has been charged with independently evaluating the approach taken in NIOSH site profiles to gauge their adequacy, completeness, and validity. These evaluations will be used by the Advisory Board to advise the Secretary of Health and Human Services on the scientific validity, quality, and accuracy of dose reconstruction efforts performed by NIOSH and its contractors.

The INL site occupies 890 square miles (572,000 acres) in southeast Idaho. It consists of nine primary facility areas situated on an expanse of otherwise undeveloped, high-desert terrain. Buildings and structures are clustered within these primary facility areas, which are typically less than a few square miles in size and separated from each other by miles of mostly undeveloped land. The site was established in 1949 as the National Reactor Testing Station (NRTS). The initial missions at INL were the development of civilian and defense nuclear reactor technologies and management of spent nuclear fuel. Fifty-two reactors were built, most of them first-of-a-kind, experimental devices, including the Navy's first prototype nuclear propulsion plant. Of those 52 reactors, 3 remain in operation today.

In 1951, INL achieved one of the most significant scientific and engineering accomplishments of the 20th century when the Experimental Breeder Reactor No. 1 (EBR-1) became the first facility to produce a usable quantity of electricity based on nuclear fission. The EBR-1 is now a Registered National Historic Landmark open to the public. During the 1970s, INL's name was changed from the National Reactor Testing Station to the Idaho National Engineering Laboratory (INEL) to reflect its broadened mission into areas like biotechnology, energy and materials research, and conservation and renewable energy. INL's name changed again in the spring of 1997 to the Idaho National Engineering and Environmental Laboratory (INEEL) to reflect a major refocus of the laboratory toward engineering applications and environmental solutions. On February 1, 2005, the name changed once again to its current name, the Idaho National Laboratory (INL). This latest change reflects a renewed mission to support the nation's expanding nuclear energy initiatives (including advanced Generation IV nuclear energy systems, nuclear energy/hydrogen co-production technology, and advanced nuclear energy fuel cycle technologies) and to support the security needs of the nation's critical infrastructure.

INL's historical role in the Federal nuclear complex included:

- A nuclear reactor research, development, and testing site; INL operated 52 reactors for such applications as electricity generation, naval reactor propulsion, aircraft propulsion, and space missions.
- Nuclear materials production and processing for military and civilian applications.
- Transuranic waste storage and disposal, primarily from the Rocky Flats plutonium foundry in Colorado. INL currently stores the nation's largest inventory of transuranic wastes (about 65,000 m^3 (DOE 2000)).
- Nuclear reactor fuel research, development, fabrication, and testing.
- Spent nuclear reactor fuel handling and storage from U.S. and foreign research reactors, commercial nuclear power plants, and naval propulsion reactors.
- Processing and fabrication of depleted uranium armor for the U.S. Army.
- Environmental restoration, waste management, decontamination, and decommissioning.

The INL site is composed of nine individual facilities, largely isolated from one another by design in the interest of safety. The primary facility areas are (1) the Argonne National Laboratory-West, (2) the Central Facilities Area, (3) the Idaho Nuclear Technology and Engineering Center, (4) the Naval Reactors Facility, (5) the Waste Reduction Operations Complex/Power Burst Facility, (6) the Radioactive Waste Management Complex, (7) the Test Area North, (8) the Test Reactor Area, and (9) the Auxiliary Reactor Area. The remainder of the site land is referred to as the Sitewide Area, which comprises all INL land outside the boundaries of the primary facility areas. In addition, several INL laboratories and administrative offices are located in the city of Idaho Falls, some 25 miles east of the INL site boundary. The INL site was and remains a complex operation involved in numerous nuclear-related missions, many of them experimental, each of which has its own unique exposure hazards. Occupational risks of exposure to ionizing radiation are generally defined by INL's past and current missions. These missions are summarized for eight of the nine primary facility areas, omitting the Naval Reactors Facility, which operated under separate administration:

(1) *Argonne National Laboratory-West* (ANL-W) was established in 1957. The original mission was to test nuclear reactors and reactor safety systems, including the Experimental Breeder Reactor No.1 (EBR-I) and Boiling Water Reactor Experiments (BORAX I-V). The current mission includes stabilization, management, and storage of spent nuclear fuel; storage of transuranic waste; and large-scale advanced reactor development. Facilities at ANL-W include EBR-II, Fuel Conditioning Facility (FCF), Hot Fuel Examination Facility (HFEF), Zero Power Physics Reactor (ZPPR), Transient Reactor Test Facility (TREAT), and Sodium Processing Facility.

(2) *Central Facility Area* (CFA), established in the 1940s, predating the nuclear laboratory, covers a large area, and includes 72 buildings. Its original mission was to lodge U.S. Navy gunnery range personnel during World War II, and provide centralized support for the INL (1950s–present). The current mission includes treatment and disposal of non-hazardous commercial/industrial waste and support for other INL facilities (administrative offices, research laboratories, cafeteria, medical services, construction/ support services, workshops, warehouses, landfills, etc.). Facilities at CFA include the Hot Laundry Facility, DOE Laboratory Accreditation Procedure Irradiation Facility, Health Physics Instrument Laboratory, Handling and Open Storage Area, Remote Service Facilities, Administrative Offices and Support Area, Service Shops Area, Radiological and Environmental Sciences Laboratory (RESL), Fire Station and Fire Fighting Training Facility (FFTF), Light Laboratory, Warehousing and Storage, and INL Sanitary Landfill.

(3) *Idaho Nuclear Technology and Engineering Center* (INTEC), formerly Idaho Chemical Processing Plant (ICPP or CPP), was established in the early 1950s. In this report, INTEC, ICPP, and CPP designations are used interchangeably. Its original mission was to reprocess spent nuclear fuels by chemically separating out the reusable uranium (until 1992) and calcine high-level waste (until 2000). The current mission includes storage of low-level, mixed low-level, and high-level radioactive waste and spent nuclear fuel, and development of treatment methods for high-level radioactive waste. Facilities at INTEC include Fluorinel Dissolution Process and Fuel Storage Facility (FAST), Remote Analytical Laboratory, Wet and Dry Fuel Storage Facility, TMI-2 Independent Spent Fuel Storage Installation (ISFSI), High-Level Liquid Waste Underground Storage Tank Farm, New Waste Calcining Facility, and INTEC Processing Corridors. The old, entombed, Waste Calcining Facility is still present.

(4) *Waste Reduction Operations Complex/Power Burst Facility* (WROC/PBF) was established in the late 1950s. Its original mission was to perform research on small power reactors, and to investigate and promote reactor safety. The current mission

includes storage of spent nuclear fuel, treatment and storage of mixed low-level and low-level waste, and research to reduce hazardous and mixed waste. This area housed the Special Power Excursion Reactor Tests (SPERT) reactors. Current facilities at WROC/PBF include the Power Burst Facility (PBF, formerly SPERT-I), Waste Engineering Development Facility/WROC Lead Storage Facility (formerly SPERT-II), Waste Experimental Reduction Facility (WERF) (formerly SPERT-III), and Mixed Waste Storage Facility (formerly SPERT-IV).

(5) *Radioactive Waste Management Complex* (RWMC) was established in late 1952. Its original mission was to provide solid, low-level radioactive waste disposal; burial of transuranic waste and hazardous substances, such as organic and inorganic chemicals (until 1970); storage of transuranic waste on pads above ground; and disposal of other waste in 20 pits, 58 trenches, and 21 soil vault rows (1970 to present). The current mission calls for interim storage of transuranic waste and shipment of stored transuranic waste to WIPP for permanent disposal. Facilities at RWMC include Subsurface Disposal Area (SDA), Intermediate Level Transuranic Storage Facility, Transuranic Storage Area (TSA), Stored Waste Examination Pilot Plant, TRUPACT Loading Station, and Advanced Mixed Waste Treatment Project (AMWTP).

(6) *Test Area North* (TAN) was established in 1951, originally for the Aircraft Nuclear Propulsion Program (ANP). Later, its mission included investigating core material from the damaged Three Mile Island-II reactor, test reactors and nuclear fuel, and manufacturing operations. The current mission includes inspection and storage of spent nuclear fuel and manufacturing depleted uranium armor for military vehicles (at the Specific Manufacturing Facility). Facilities at TAN include ANP, Loss of Fluid Test Reactor (LOFT), Three Mile Island Unit 2 (TMI-2) Core Offsite Examination Program, Technical Support Facility, Initial Engine Test Facility (IET), Specific Manufacturing Capability Program (formerly Containment Test Facility), and Water Reactor Research Test Facility. There are other support facilities at TAN, including Hot Shop, Warm Shop, Hot Cells, Storage Pool, Storage Pads, Radioactive Liquid Waste Disposal System, Radioactive Parts Service and Storage Area, and Radiography Facility.

(7) *Test Reactor Area* (TRA) was established in 1982. Its original mission was to study the effect of radiation on materials, fuels, and equipment using seven reactors, especially the Materials Test Reactor (MTR: 1952–1970), the Engineering Test Reactor (ETR: 1957–1982), and the Advanced Test Reactor (ATR: 1967–present). The current mission includes wet storage of spent nuclear fuel, operation of the ATR, research supporting the U.S. Navy and other customers, and production of isotopes for medicine and industry. Other formerly and currently operating facilities at TRA include ATR Critical Facility, TRA Hot Cell Facility, TRA Gamma Facility, Radiation Measurements Laboratory, Radiochemistry Laboratory, Safety and Tritium Applications Research Facility, Liquid Waste Disposal Ponds, and High-Level Liquid Waste Tanks and Transfer Facility.

(8) *Auxiliary Reactor Area* (ARA), originally called the Army Reactor Experimental Area (AREA), was established in 1958 and ceased operations in the 1990s. Its original mission was to test stationary, portable, or mobile reactors of low, medium, or high power. ARA is located 10 miles east of the Central Facility Area. It began with the ARA-1 site, then ARA-II, -III, and -IV at half-mile intervals: ARA-1 housed a hot cell facility; ARA-II was renamed the Stationary Low Power Reactor Number 1 (SL-1), which was shut down due to an excursion accident; ARA-III was the site for the Army Gas Cooled Reactor Experiment (GCRE); and ARA-IV was the site for the Mobile Low Power Reactor (ML-1).

There are other, smaller facilities and experiments located elsewhere on the vast INL site. These facilities include the Organic Moderated Reactor Experiment (OMRE), INL Research Center, Army Reentry Vehicle Facility Station, and Test Grid III. At Test Grid III, INL housed several important experimental facilities, including Fuel Element Burn Tests, Fission Products Field Release Tests, Relative Diffusion Tests, and Experimental Cloud Exposure Study.

It has not been possible within the time and resources available and SC&A's scope for this review to examine in detail all aspects of the site profile, due to the immense complexity and long history of the INL facilities, and the many changes that have occurred over the decades. SC&A has selected only certain issues for detailed consideration and discussion, because they may significantly affect dose reconstruction, and, ultimately, determination of a claimant's petition for compensation.

Based upon the *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004a), approved by the Advisory Board on March 18, 2004, the SC&A review has identified a number of issues. These issues are sorted into the following five categories:

(1) Completeness of data sources
(2) Technical accuracy
(3) Adequacy of data
(4) Consistency among site profiles
(5) Regulatory compliance

After the introduction and a description of the criteria and methods employed to perform the review, this report discusses the strengths of the TBDs, followed by a description of the major "issues" identified during our review. The 35 issues were carefully assessed with respect to the five review criteria listed above. Seventeen of the issues are designated as "findings," because they represent what SC&A believes are deficiencies in the TBDs that need to be corrected, and which have the potential to substantially impact at least some dose reconstructions.[2] The remaining issues are designated as "observations," which represent areas that SC&A feels the TBDs could improve on.

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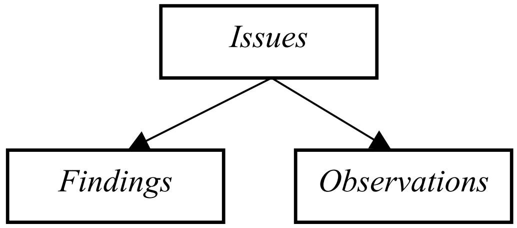
[2] Several findings are combined, resulting in eleven distinct findings.



These issues, and accompanying characterizations and section numbers where they are discussed in the report, are listed in Table 1-1. An issue resolution matrix, Table A-5, for these identified issues (findings and observations) is provided in Attachment 5 of this report. In addition, Section 1.2 summarizes the findings.

Table 1-1: Issue Matrix for the INL Site Profile

Descriptions (a)	Issue Classification	Objective 1: Completeness Of Data	Objective 2: Technical Accuracy	Objective 3: Adequacy of Data	Objective 4: Site Profile Consistency	Objective 5: Regulatory Compliance (b)
Issue 1: (5.1.1.1) Routine Airborne Releases	Finding (5)	X	X	X	X	Category 2
Issue 2: (5.1.1.2) Episodic Airborne Release	Finding (6)	X	X	X	X	Category 2
Issue 3: (5.1.1.3) Direct Gamma Exposures	Finding (7)	X	X	X	X	Category 2
Issue 4: (5.1.2.1) Completeness and Quality of INL Internal Dosimetry Programs	Finding (8)	X	X	X	X	Category 3
Issue 5: (5.1.2.2) High-Risk Jobs (Internal Exposure)	Finding (9)	X				Category 1
Issue 6: (5.1.2.3) Calibration of Internal Dosimetry Analytical and Monitoring Equipment	Observation		X		X	
Issue 7: (5.1.2.4) Changes of Internal Dose Limits	Observation		X			
Issue 8: (5.1.2.5) High Fired Plutonium and Uranium Intakes	Finding (10)	X				Category 1
Issue 9: (5.1.2.6) Skin and Facial Contamination	Observation	X				Category 3
Issue 10: (5.1.2.7) Breathing Rates	Observation	X				
Issue 11: (5.1.2.8) Non-Occupational Worker Elimination of DU Background	Finding (11)	X	X	X		Category 2
Issue 12: (5.1.2.9) Unmonitored Workers	Observation	X				
Issue 13: (5.1.2.10) Naval Reactor Facility Workers	Observation	X				Category 2
Issue 14: (5.1.2.11) Plutonium Monitoring	Observation	X	X	X		Category 1
Issue 15: (5.1.3) SL-1 Accident Dose Reconstructions	Finding (1)	X	X	X		
Issue 16: (5.1.4.1.1) Completeness and Quality of INL Beta/Gamma Dosimetry and Record Keeping Programs	Finding (8)	X		X	X	Category 3
Issue 17: (5.1.4.1.2) Penetrating and Non-Penetrating Doses	Finding (4)	X	X	X	X	Category 3
Issue 18: (5.1.4.1.3) Correction For Beta Doses	Observation		X		X	
Issue 19: (5.1.4.1.4) Angular Dependence Correction Factor for Gamma Dose	Observation	X				

Table 1-1: Issue Matrix for the INL Site Profile

Descriptions (a)	Issue Classification	Objective 1: Completeness Of Data	Objective 2: Technical Accuracy	Objective 3: Adequacy of Data	Objective 4: Site Profile Consistency	Objective 5: Regulatory Compliance (b)
Issue 20: (5.1.4.1.5) Restating Beta Dose As Gamma Dose	Observation		X			
Issue 21: (5.1.4.1.6) Photon Spectrum Split	Observation		X		X	
Issue 22: (5.1.4.1.7) Immersion Dose	Observation	X	X	X		
Issue 23: (5.1.4.1.8) High-Risk Jobs (Beta/Gamma Exposure)	Finding (9)	X				Category 1
Issue 24: (5.1.4.1.9) Extremity Dose	Observation	X				Category 2
Issue 25: (5.1.4.1.10) Discrepancies between PIC and Film Reading	Observation	X				
Issue 26: (5.1.4.1.11) Minimum Detection Limit	Observation			X	X	
Issue 27: (5.1.4.1.12) Minimum Reporting Level (Beta/Gamma)	Finding (3)	X	X	X		Category 2
Issue 28: (5.1.4.2.1) Minimum Reporting Level (Neutron)	Finding (3)	X	X	X		Category 2
Issue 29: (5.1.4.2.2) Failure to Properly Address Neutron Exposures	Finding (2)	X	X	X	X	Category 1
Issue 30: (5.1.4.2.3) Neutron Calibration Deficiencies	Finding (2)	X	X	X		Category 3
Issue 31: (5.1.4.2.4) Completeness and Quality of INL Neutron Dosimetry and Record Keeping Programs	Finding (8)	X	X	X	X	Category 3
Issue 32: (5.1.4.2.5) Uncertainty Estimation for Neutron Doses	Observation	X	X	X		
Issue 33: (5.1.4.2.6) Neutron Organ Dose	Observation	X				
Issue 34: (5.1.4.2.7) High-Risk Jobs (Neutron Exposure)	Finding (9)	X				Category 1
Issue 35: (5.1.4.2.8) Multiplying Factors for Missed Neutron Dose	Observation	X	X	X	X	Category 3

Table Notes:

(a) Report section numbers discussing the issues are given after the issue number.

(b) Category 1: Least challenged by any deficiencies in available dose/monitoring data are dose reconstructions for which even a partial assessment (or minimized dose(s)) corresponds to a probability of causation (POC) value in excess of 50%, and assures compensability to the claimant.

Category 2: The use of upper bound values is limited to those instances where the resultant maximized doses yield POC values below 50%, which are not compensated. For this second category, the dose reconstructor needs only to ensure that all potential internal and external exposure pathways have been considered.

Category 3: The most complex and challenging dose reconstruction represents claims where the case cannot be dealt with under one of the two other categories.

An “X” in the table indicates significant shortfalls in meeting the corresponding review objectives for the indicated topics in the INL Site Profile. These shortfalls have been discussed either within the text of the findings themselves, or, in many cases, in special sections that address one or more of these shortfalls. The first column of the table indicates the primary place within the report that addresses each issue.

There is some redundancy in the report by virtue of the standard format adopted, where a single item may be discussed from different perspectives in several different places. For example, the SC&A site profile review procedure calls for both a "vertical" assessment for purposes of evaluating specific issues of adequacy and completeness, as well as a "horizontal" assessment of how the profile satisfies its intended purpose and scope.

1.1 SUMMARY OF STRENGTHS

For the purpose of reconstructing occupationally related doses (from medical, environmental, internal, and external sources) based on historical operations, NIOSH compiled an enormous amount of data describing the radioactive materials, operations, and processes in the various facilities at INL. The associated TBDs generally provide a wealth of useful information to aid the dose reconstructors, who have the task of determining individual claimant radiation exposures.

The INL Site Description TBD (Rohrig 2005), in addition to providing an overview of the site history, layout, facilities, and present status, and a framework in which to consider occupational exposures, is very thorough and detailed. It describes the past and current operations and missions of 14 areas and 101 facilities and processes. The TBD lists major radionuclides of concern at each facility and identifies potential sources of internal and external exposures. This information is very helpful to the dose reconstructions for the claimants.

The Occupational Medical Dose TBD is thorough and methodical. For the purpose of estimating occupational medical exposure for workers at INL, NIOSH includes a table (Table 3A-1), which presents various organ doses resulting from chest x-ray examination for employees for the entire period of INL operation. This table is conveniently structured and easy to use. The organ doses were also reasonably estimated using available information on site practices and medical equipment.

For airborne emissions from routine facility operations, the Occupational Environmental Dose TBD relies heavily on two previous works: (1) *Idaho National Engineering Laboratory Historical Data Evaluation* (INELHDE, DOE 1991a); and (2) *Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory*, Final Report (RAC 2002). Using data from these two reports, NIOSH developed worker inhalation intake values for eight INL facilities (ANL, ARA, CFA, ICPP, RWMC, SPERT, TAN, TRA) for nine key radionuclides (Ce-144, I-131, Pm-147, Pu-238, Pu-239/240, Ru-106, Sr-89, Sr-90, and Y-91) from 1952 to 2002. These intake values are presented in Table 4-1 to Table 4-8 of the TBD. For direct gamma exposures to environmental releases, NIOSH used facility fence-line TLD measurement data from Environmental Monitoring Data Reports (EMDRs) between 1972 and 1983. The environmental gamma dose values are presented for eleven facilities in Table 4-13. For airborne emissions from episodic events, such as criticality accidents or special tests, NIOSH again used data from INELHDE and developed three tables providing worker inhalation intake values for primary radionuclides for criticality events (Table 4-10), special tests (Table 4-11), and initial engine tests (Table 4-12). These tables are helpful for dose reconstructions for certain claimants who worked at the INL site and, hence, were exposed to airborne emissions.

A prominent component of the INL Occupational Internal Dose TBD is Table 5.7-1, which presents default assumptions that could be used to calculate missed dose. Missed dose default assumptions are based on upper bound operating source terms from INL facilities. The table is easy and convenient for dose reconstructors to use if worker intake records are available and complete and exposures are not too high. On the other hand, this table does not include all potential sources of missed internal doses for workers; these will be discussed in the following sections. The table presents recommended missed dose default assumptions for four periods of operations as follows:

(1) **Beginning to 1960:** The intake defaults are based on gross β activities in urine samples. A different, claimant-favorable weighting factor is applied to each of four radionuclides considered (Sr-90, Cs-137, Pu-238, and Ce-144). Default intake values are presented for personnel who worked at any of the INL facilities in this period, with an additional (in vivo) intake of I-131 added for personnel who worked at the TRA.

(2) **1961 to 1970:** The defaults are based on in-vivo (whole-body counting) measurement of Cs-137. A claimant-favorable weighting factor is applied to each radionuclide (Sr-90 and Pu-238). Default intake values are presented for personnel who worked at any of the INL facilities in this period, with additional intakes of Ce-144, I-131, Ag-110m, and Ta-182 added for personnel who worked at the TRA.

(3) **1971 to 1980:** The defaults are based on in-vivo measurement of Cs-137. A claimant-favorable weighting factor is applied to each radionuclide (Sr-90 and Pu-238) for all facilities in this period. For those personnel who worked at the INTEC (previously ICPP) facility or an unknown INEL facility (e.g., NRF), an intake of Ce-144 is added. For those personnel who worked at ANL-W, intakes of Pu-239 and Ce-144 are added. For those personnel who worked at the TRA, an intake of I-131 is added. For those personnel who worked at other areas (other than INTEC, ANL-W, TRA, and unknown facilities), intakes of Pu-238 and Ce-144 are added.

(4) **1981 to present:** The defaults are based on in-vitro bioassay measurement. Intakes and weighting factors based on radionuclides associated with aluminum-clad fuel intakes are applied to personnel at all INL facilities except ANL-W, INTEC, and SMC in this period. For those who worked at ANL-W, intakes and weighting factors associated with stainless steel-clad fuel are used. For those who worked at INTEC or an unknown INL facility, intakes and weighting factors associated with zirconium-clad fuel are used. For those who worked at SMC, the DU uranium isotopic activities are used.

Table 6B-1 pf the INL site occupational external exposure TBD presents maximum annual missed dose values for gamma radiation exposures for different films (e.g., 552 Dupont film, 558 Dupont film, 508 Dupont film) and for different TLDs (LiF, LiF in Teflon, Harshaw Two-Chip, Panasonic Four-Chip) based on the MRL (maximum recording level) for six different time periods; (1) 1951–1958, (2) 1958–1966, (3) 1966–1974, (4) 1974–1975, (5) 1974–1985, and (6) 1986 to present. The missed dose values are given for different dosimeter exchange frequencies (weekly, biweekly, monthly, quarterly, and semi-annually). For gamma dose reconstruction for a claimant, the equation (N x MRL/2) given in OCAS-IG-001 (Taulbee 2002) should be used, where N is the number of zero dose results in a claimant's file. There is no multiplying factor for gamma missed dose uncertainties.

For missed neutron dose, Table 6B-2 of the external exposure TBD provides maximum annual missed dose values for different neutron dosimeter types (NTA film and TLD) based on MRL for three different time periods; (1) 1951–1958, (2) 1958–1976, and (3) 1976 to present. The table also shows the maximum annual missed neutron dose. For a neutron dose reconstruction for a claimant, the equation (N x MRL/2) given in OCAS-IG-001 (Taulbee 2002) should again be used; however, here N is the number of periods (week, month, or quarter) a worker had missed neutron dose. The TBD provides two multiplying factors when the MRL for NTA film is used in estimating the missed neutron dose. The missed neutron dose should be multiplied by 1.25 for most workers and by 2 for workers on the MTR experimental floor for the period before October 1976. For the period after October 1976, no multiplying factor is needed.

1.2 SUMMARY OF FINDINGS

Table 1-1 presents a list of 35 issues that were identified during the review process. Seventeen of these issues were judged potentially important enough to dose reconstruction to categorize as findings. Those findings are summarized below in the order according to what SC&A believes is their level of significance to the dose reconstructions; references are given to report sections where more detailed discussions may be found. Note that 2 sets of 3 issues have been combined into Findings No. 8 and No. 9, and another 2 sets of 2 issues into Findings No. 2 and No. 3, reducing the final number of findings to 11.

Finding 1 (Issue 15): SL-1 Accident Dose Reconstructions (Section 5.1.3) – The TBDs do not evaluate the potential missed internal and external doses or the associated uncertainties for the over 1,000 rescue and cleanup workers involved with the SL-1 accident that occurred in January 1961. There was a high potential for significant exposures, because the equipment used and the radiological control policies in place in that era were not as advanced and protective as those in current use. The TBDs should develop adjustment factors related to stay time, dose field estimates, internal dose results, external dose readings, and contamination level estimates. NIOSH did not evaluate potential missed neutron doses for the first responders and rescue workers of the accident.

Finding 2 (Issues 29 and 30): Missed Neutron Dose (Sections 5.1.4.2.2 and 5.1.4.2.3) – The TBD presumes that neutron exposures at INL's reactors are not a problem and, therefore, are adequately addressed. But INL had a total of 52 reactors most of which were experimental/ prototype in design, which typically operated with high-power densities and with minimum shielding and neutron moderation. It is, therefore, unjustified to presume that there are no missed neutron doses. In addition, there are deficiencies associated with neutron calibrations. Due to the use of the PoBe source for neutron calibration, dosimeters would significantly under-measure neutron doses from sources with lower energy spectra. NIOSH should re-evaluate the entire approach in the TBD to account for potential missed neutron doses.

Finding 3 (Issues 27 and 28): Minimum Reporting Levels (Sections 5.1.4.1.12 and 5.1.4.2.1) – NIOSH does not provide adequate information supporting the use of chosen detection threshold levels to represent the MRL values for gamma film badges and TLDs. Most significantly, NIOSH's approach for determining the MRL values for NTA emulsion film is not thorough and supported. For example, NIOSH uses 10 neutron readings in one data sheet from March 1958 to determine the MRL values for the period between 1951 and 1957, and 6 neutron readings to represent all neutron measurements between 1959 and 1976. Furthermore, the use of MRL/2 as the missed external dose for dose reconstruction per OCAS-IG-001 is not claimant-favorable for claims where the probability of causation value is close to 50%. In addition, NIOSH's MRL values of 14 mrem and 20 mrem appear low and are inconsistent with generic values given for NTA dosimeters, as well as values cited by other DOE facilities with similar neutron source terms and detectors. NIOSH should re-evaluate the MRL values used and provide more supportable default values.

Finding 4 (Issue 17): Penetrating and Non-Penetrating Doses (Section 5.1.4.1.2) – The procedures and algorithms used in the film badge dosimetry service in the early days underestimated the Hp(10) dose, because the low-energy photons reaching the dosimeter were considered beta radiation. Surprisingly, the film service then added this beta dose (to the skin) to the "deep" dose; a practice that is claimant favorable. However, the TBD also correctly requires the dose reconstructor to consider only the "deep dose" as Hp(10), but in doing so, the low-energy photon contribution to Hp(10) is lost. To be claimant favorable, INL calculated the beta dose and the gamma dose, summed these two together, and recorded the result in the worker files as a whole-body dose. The current dose reconstruction process is applying the gamma dose correctly as the effective dose. The problem is that this gamma dose is not claimant favorable, as the information on dose due to low-energy gammas (E < 100 keV) has been lost. NIOSH should re-evaluate the missed gamma dose due to the deficiencies in the procedures and algorithms.

Finding 5 (Issue 1): Routine Airborne Releases (Section 5.1.1.1) – For airborne emissions from routine facility operations, NIOSH relies heavily on two previous works: (1) *Idaho National Engineering Laboratory Historical Data Evaluation* (INELHDE, DOE 1991a); and (2) *Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory*, Final Report (RAC 2002). SC&A found that the source terms provided require improvement for use in determining the worker intake from airborne releases at different INL facilities. The data NIOSH uses do not take into account the deficiencies in the environmental monitoring equipment and their locations, and, in addition, NIOSH does not assess the uncertainties associated with the meteorological dispersion model used for the INL site. Most importantly, the source terms do not account for worker inhalation of resuspended contaminated soils and materials around the INL facilities.

Finding 6 (Issue 2): Episodic Airborne Releases (Section 5.1.1.2) – For airborne releases from episodic events (such as criticality and special tests), NIOSH again relies on the two primary documents, INELHDE and RAC 2002, for determining onsite concentrations of radionuclides. In a previous study (SC&A 2003), SC&A determined that the airborne releases associated with several of the Initial Engine Tests of the Aircraft Nuclear Propulsion (ANP) Program were likely to have been underestimated by factors ranging from 2 to 7. Also, NIOSH did not evaluate the uncertainties associated with the deficiencies in air monitoring equipment.

Finding 7 (Issue 3): Direct Gamma Exposures (Section 5.1.1.3) – For direct gamma exposure from environmental releases, NIOSH used fence-line TLD measurement data from Environmental Monitoring Data Reports (EMDRs) between 1972 and 1983. SC&A believes that these TLD measurements are not adequate for reconstructing direct gamma doses to personnel working outdoors at and around a specific INL facility inside the fence-line boundary. The NIOSH approach assumes all outdoor workers at a facility would receive an average direct gamma dose from a normalized ground concentration of radionuclides. If the assumption were valid, the fence-line TLD results should be multiplied by a weighting factor to account for uncertainties in TLD sensitivity and geometry. NIOSH has not done that in the TBD. However, NIOSH's assumptions are not claimant favorable, because they do not take into account the most bounding scenarios; (1) personnel working outdoors may become immersed in the plume of routine or episodic releases from the facility stack, (2) personnel working outdoors may inhale resuspended cumulative ground radionuclide depositions or windblown contaminated soils from any neighboring dry ponds, and (3) the cumulative ground concentrations inside the fence line are generally higher than those near the fence line. The fence-line TLDs are too far from the bounding source terms to represent the actual direct gamma doses received by the outdoor workers. Therefore, this TBD approach does not appear to be claimant favorable. In addition, NIOSH should re-examine the direct gamma dose values, because some of these values presented in Table 4-13 of the TBD appear to be much lower than the recent INL-published values.

Finding 8 (Issues 4, 16 and 31): Completeness and Quality of INL Radiological Protection, Personnel Dosimetry, and Record Keeping Programs (Sections 5.1.2.1, 5.1.4.1.1, and 5.1.4.2.4) – The tone of the TBDs strongly suggests that NIOSH (i.e., the TBD authors) had full confidence in the radiological protection programs, the internal dosimetry programs, and the dosimetry record keeping systems at the INL site, in the past and at the present. The identification and determination of missed internal and external dose for workers are heavily influenced by this assumption of confidence, but SC&A found this premise to be unsupported after examining several critical DOE-HQ Tiger Team and DNFSB site audit reports. In addition, many site experts interviewed by SC&A indicated that there were significant deficiencies and inconsistencies in radiation work practices throughout the operating history of the INL facilities. These observations jeopardize the validity of the TBD approaches in reconstructing missed worker internal and external doses.

Finding 9 (Issues 5, 23 and 34): High-Risk Jobs (Sections 5.1.2.2, 5.1.4.8, and 5.1.4.2.7) – Site experts interviewed by SC&A classified INL as an "acute dose" site, with a significant number of facilities, operations, experiments, and occurrences providing the possibility of personnel receiving dangerous levels of radiation. However, NIOSH did not evaluate comprehensively the facility and field data to identify and separate out the high-risk or high-dose jobs for worker internal and external exposures. For example, the TBD does not evaluate the potential missed external dose (extremity, bladder, lens, or prostate) received by a pipefitter, who worked on a reactor vessel or process waste tank in a tight space environment where the dosimeter measurements would not be effective and representative due to angular dependence or bodily shielding. This information is essential for dose reconstructors to fill in the data gap when dose records in a claimant's file are not complete.

Finding 10 (Issue 8): High-Fired Plutonium and Uranium Intakes (Section 5.1.2.5) – The Occupational Internal Dose TBD did not evaluate the hazard associated with high-fired plutonium and uranium at the INTEC (ICPP) and RWMC facilities. High fired Pu-238, Pu-239, and uranium are not easily dissolvable, nor do they readily break into very small particles. They also emit some gamma rays and neutrons. Similar to the treatment of recycled uranium, NIOSH should evaluate the lung dose for intake of high-fired uranium and plutonium oxide particulates (alveolar deposition).

Finding 11 (Issue 11): Non-Occupational Worker Elimination of DU Background (Section 5.1.2.8) – The derivation of the background value of 0.16 μg/L used for subtraction from each urinalysis result of uranium prior to assessment of occupational internal dose for SMC radiation workers is not technically sound. The baseline background (population) intake value was determined by a study of urine samples submitted by non-radiation workers at the SMC facility. A better approach would be to use the urine excretion samples by non-INL people in the Idaho Falls areas. This approach would not create a suspected bias due to uranium intake through various pathways (inhalation and ingestion) by non-radiation workers while working at the SMC facility. During the site expert interview, the dosimetry staff indicated that they tried to use residents from the Idaho Falls area, but none were willing to sign a liability waiver form. In a subsequent study, the dosimetry staff used sixteen non-radiation workers from the INL CFA facility. In addition, this background value is much higher than the national survey data. NIOSH should consider this subtraction from urinalysis results as a missed internal dose.

1.3 OPPORTUNITIES FOR IMPROVEMENT

Not withstanding the many positive, helpful features of the TBDs in providing guidance to the dose reconstructors, SC&A identified a number of areas that represent potential opportunities for improvement. These are summarized below:

Incomplete Dose Records in Worker Files: NIOSH should look into the possibility of many missing dose records in worker files. From interviews of retired, past, and current workers, and also from audit reports (DOE-HQ and DNFSB), there appear to be many incident reports, occurrence reports, contamination reports, and worker uptake reports that were not included in the worker records. Most of these reports were kept, historically, at the INL facilities. NIOSH holds the position that the dosimetry records reflect accurately all the doses workers received while working at INL. The suspected incompleteness of the worker records is a serious issue, since it may lead to significant underestimation of workers' radiation dose.

Use of Untrained Workers: NIOSH should look into the worker utilization practices at INL in the early years (from 1950s to 1980s). From interviews of retired, past, and current workers, it appears that INL often sent workers, even untrained ones (e.g., secretaries, yardmen, etc.), from one INL facility to perform emergency or high-dose radiation work in another INL facility. After these workers reached their weekly or bi-weekly dose limit, in most cases 200 mrem, they would be sent back to their original workplace. For example, INTEC (ICPP) is one of the facilities that required frequent utilization of workers from other INL facilities to augment the work force, due to its high contamination and high radiation level (acute dose) environment. Several site experts interviewed indicated that they personally had the experience of being sent to

INTEC to be "burned" to their dose limit. In addition, some workers interviewed indicated that they had experienced that the doses they received after being "burned" at the other facilities were frequently not recorded in their worker exposure files.

Angular Dependence Correction Factor for External Dosimeter: NIOSH should provide angular dependence (anatomic geometry) correction factors for external gamma doses, particularly for low-photon energies, where the angular dependence of the sensitivity of the dosimeter is most pronounced. These correction factors are used to account for, for example, the bias introduced by a dosimeter worn at the neck level and the higher doses received by tissues/organs below the waist.

Breathing Rate: In the INL occupational environmental dose evaluation (TBD Section 4.2.1), NIOSH uses an at-rest breathing rate, which is not claimant favorable.[3]

Plutonium Monitoring: NIOSH should provide historical information on the plutonium analysis methods used at the INL site. It is entirely possible that selective plutonium monitoring of workers was used at INL until 1980. Without this information, the dose reconstructors would not be able to assign missed internal dose, due to plutonium intakes in the time period before 1980.

Dose Calculation Example: NIOSH should provide an example (or examples) in the TBD text of a hypothetical dose reconstruction using recorded records, missed dose assignment, and dose assignments when dosimeters read zero dose.

Consistent Air Dispersion Model: NIOSH should use a consistent methodology for calculation of occupational dose that is appropriate for application to onsite workers. In addition, the components of environmental dose should be consistent among the DOE facilities evaluated. For routine releases, the INL Occupational Environmental Dose TBD uses the MESODIF model, employing an objective regional trajectory computational scheme combined with the Gaussian diffusion equation for a continuous point source. It is a forward time-marching Gaussian plume model in which successive, small plume elements (or puffs) are advected throughout the computational area. For episodic releases, the TBD used the RSAC program primarily for calculating onsite meteorological dispersion parameters for the various airborne release incidents. RSAC provides the option to use various types of meteorological diffusion models and parameters applicable at INL, all of which are based on the Gaussian plume model. In the case of Hanford Site and SRS, for example, a RATCHET puff model and the Gaussian model were used, respectively.

Recycled Uranium and Plutonium: The dose contribution from trace radionuclides in recycled uranium and plutonium should be evaluated in terms of dose to particular organs of concern and the relative impact on internal dose reconstruction. NIOSH should evaluate the lack of formal policies for considering trace radionuclides in recycled uranium and plutonium, and develop

[3] See Attachment 1, Occupational Environmental Dose, Question/Response No. 7, of this report for further discussion of the at-rest breathing rate assumption.

bounding conditions that can be applied at all applicable DOE facilities, including INTEC (formerly ICPP) at the INL site.

Beta/Gamma Dosimeter Adjustments and Uncertainties: A method to consistently account for laboratory, radiological, and environmental uncertainties in dosimeter readings should be developed and appropriately applied to recorded dosimeter results, so that it is clear what sigma value should be entered into Interactive RadioEpidemiological Program (IREP) Parameter 2.

Tank Farm Worker: NIOSH should complete an evaluation of the relative hazards associated with work at the Tank Farms and the completeness of monitoring related to Tank Farm workers, including subcontractor and construction workers.

Use of Site Expert Input: NIOSH should make a greater effort to take into account site expert information and to investigate worker accounts. First-hand experience and association with the INL facilities enable site experts and workers to provide original perspectives and information concerning site practices and exposure histories that may not appear in the official records.

Missed Dose and Off-normal Practices: NIOSH should evaluate the significance of off-normal practices for missed dose by analysis of film badge data and site expert interviews. This is essential to determine if there were areas or periods where badges may not have been consistently worn when the actual dose was near the administrative control limit.

Data Uncertainty: The statutory requirement of a claimant-favorable dose reconstruction process is achieved by (1) giving the benefit of doubt when there are unknowns, and (2) defining uncertainties for measured data and selecting the 99th percentile values of a Monte Carlo distribution. In the site profile TBDs, it is often found that lower percentile values are used instead of the 99th percentile values as default assumptions for missed dose calculations. NIOSH should re-examine all the data uncertainty values used in the TBDs to ensure that this statutory requirement of claimant-favorable dose reconstruction is met.

2.0 SCOPE AND INTRODUCTION

> *INL is an 890-square-mile reservation encompassing almost 572,000 acres with a maximum distance of about 39 miles from north to south and 36 miles from east to west. It is 30 to 60 miles west of Idaho Falls, Idaho. Major site-related research facilities and offices are in Idaho Falls. The site, situated on the Snake River Plain of southeastern Idaho at an elevation of about 5,000 ft., is above the Snake River Plain Aquifer. In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) in Idaho as a Federal reservation to build, test, and operate nuclear reactors. The site utilized a variety of support facilities and equipment. In 1974, the NRTS became the Idaho National Engineering Laboratory (INEL) and, in 1997, the Idaho National Engineering and Environmental Laboratory (INEEL). On February 1, 2005, the site became the Idaho National Laboratory (INL) combining the research side of the INEEL and ANL-W and the Idaho Cleanup Project (ICP) working on closure of inactive portions of the site. INL is unique among U.S. Department of Energy (DOE) facilities in its size and complexity and diversity of it facilities and operations, with many independent technical areas, contractors, goals, and missions.* (Rohrig 2005, pg. 13)

Under the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) and Federal regulations defined in Title 42, Part 82, *Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program,* of the *Code of Federal Regulations*, of the *Code of Federal Regulations* (42 CFR Part 82), the Advisory Board on Radiation and Worker Health (Advisory Board) is mandated to conduct an independent review of the methods and procedures used by the National Institute for Occupational Safety and Health (NIOSH) and its contractors for dose reconstruction. As a contractor to the Advisory Board, S. Cohen and Associates (SC&A) has been charged to support the Advisory Board in this effort by independently evaluating a select number of site profiles that correspond to specific facilities at which energy employees worked and were exposed to ionizing radiation.

This report provides a review of the following Technical Basis Documents (TBDs) related to historical occupational exposures at INL and which, together, constitute the site profile:

- ORAUT-TKBS-0007-1, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Introduction* (Rohrig 2004i)

- ORAUT-TKBS-0007-2, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Site Description* (Rohrig 2005)

- ORAUT-TKBS-0007-3, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational Medical Dose* (Rohrig 2004m)

- ORAUT-TKBS-0007-4, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) –Occupational Environmental Dose* (Peterson 2004)

- ORAUT-TKBS-0007-5, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational Internal Dose* (Rich and Wenzel 2004)

- ORAUT-TKBS-0007-6, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational External Dosimetry* (Rohrig 2004e)

SC&A has critically evaluated the INL TBDs and other applicable documents in order to accomplish the following:

- Determine the completeness of the information gathered by NIOSH in behalf of the site profile, with a view to assessing the profile's adequacy and accuracy in supporting individual dose reconstructions
- Assess the technical merit of the data/information
- Assess NIOSH's use of the data in dose reconstructions

SC&A's review of the six TBDs focuses on the quality and completeness of the data characterizing the facility and its operations, and the methods prescribed by NIOSH for the use of these data in the dose reconstruction process. The review was conducted in accordance with the objectives stated in SC&A's *Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004a). The criteria for evaluation include whether the TBDs provide a basis for scientifically supportable dose reconstruction in a manner that is adequate, complete, efficient, and claimant favorable. Specifically, these criteria were viewed from the prospective of whether dose reconstructions based on the TBD would support robust compensation decisions.

The review is directed at "sampling" the site profile analyses and data for validation purposes; the review does not provide a rigorous quality control process, whereby actual analyses and calculations are duplicated or verified. The scope and depth of the review are focused on aspects or parameters of the site profile that would be particularly important in deriving dose reconstructions, bridging uncertainties, or correcting technical inaccuracies.

The basic goal of dose reconstruction is to characterize the radiation environments to which workers were exposed and, supplementing the workers' individual dose records, determine claimant exposures through different pathways over time. The hierarchy of data used for developing dose reconstruction methodologies is (1) dosimeter readings and bioassay data, (2) co-worker data and workplace monitoring data, and (3) process description information or source term data.

In accordance with directions provided by the Advisory Board and the SC&A site profile review procedure, this report is organized into the following sections:

(1) Executive Summary
(2) Scope and Introduction
(3) Assessment Criteria and Method
(4) Site Profile Strengths
(5) Vertical Issues
(6) Overall Adequacy of the INL Site Profile as a Basis for Dose Reconstruction

Based on the issues identified in each of these sections, SC&A prepared a list (Tables 1-1 and 6-1) of "issues" briefly described in the Executive Summary and later in the report; this list functions as a convenient roadmap to the issues discussed throughout the report. Issues are designated as "findings" if SC&A believes that they represent deficiencies in the TBD that need to be corrected and which have the potential to have a substantial impact on at least some dose reconstructions. Issues are designated as "observations" if they simply raise questions, which, if addressed, would further improve the TBDs and may possibly reveal deficiencies that will need to be addressed in future revisions of the TBDs. In this review, SC&A has identified 35 issues, categorized into 11 findings (some issues are combined), and 18 observations. The TBDs, in many ways, have done a successful job in addressing a series of technical challenges. In other areas, the TBDs exhibit shortcomings that may influence some dose reconstructions in a substantial manner.

Since many of the issues that surfaced in the report correspond to more than one of the major objectives (i.e., strengths, completeness of data, technical accuracy, consistency among site profiles, and regulatory compliance), there is a degree of redundancy in the report, where different sections may address the same issue, but from different perspectives. For example, the SC&A site profile review procedure calls for both a "vertical" assessment for purposes of evaluating specific issues of adequacy and completeness, as well as a "horizontal" assessment of how the profile satisfies its intended purpose and scope.

3.0 ASSESSMENT CRITERIA AND METHOD

SC&A is charged with evaluating the approach set forth in the site profile used in the individual dose reconstruction process. SC&A reviewed the site profile documents with respect to the degree to which technically sound judgments or assumptions are employed, and assessed the degree to which they fulfill the objectives delineated in SC&A's review procedure.

3.1 OBJECTIVES

Documents are reviewed for their completeness, technical accuracy, adequacy of data, consistency with other site profiles, and compliance with the stated objectives, as defined in the *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004a). These objectives are discussed in the following sections:

3.1.1 Objective 1: Completeness of Data Sources

SC&A reviewed the site profile with respect to Objective 1, which requires SC&A to identify principal sources of data and information that are essential to the development of the site profile. The two elements examined under this objective are (1) determining if the site profile made proper use of available data considered relevant and significant to the dose reconstruction, and (2) investigating whether other relevant/significant sources are available, but were not used in the development of the site profile. For example, if relevant data are available in site technical reports or other site documents for particular processes, and if the TBDs have not taken these data into consideration, this would constitute a completeness of data issue. The Oak Ridge Associated Universities (ORAU) site profile document database, including the referenced sources in the TBDs, was evaluated to determine the relevance and use of the data collected by NIOSH to the development of the site profile. Additionally, SC&A evaluated selected records publicly available relating to the INL site and records provided by site experts. SC&A requested a number of documents during the site expert interview visit, which have all been provided by INL.

3.1.2 Objective 2: Technical Accuracy

SC&A reviewed the site profile with respect to Objective 2, which requires SC&A to perform a critical assessment of the methods used in the site profile to develop technically defensible guidance or instruction, including evaluating field characterization data, source term data, technical reports, standards and guidance documents, and literature related to processes that occurred at INL. The goal of this objective is to first analyze the data according to sound scientific principles, and then evaluate this information in the context of compensation. For example, if SC&A found that the technical approach used by NIOSH was not scientifically sound or claimant favorable, this would constitute a technical accuracy issue.

3.1.3 Objective 3: Adequacy of Data

SC&A reviewed the site profile with respect to Objective 3, which requires SC&A to determine whether the data and guidance presented in the site profile are sufficiently detailed and complete to conduct dose reconstruction, and whether a defensible approach has been developed in the absence of data. In addition, this objective requires SC&A to assess the credibility of the data used for dose reconstruction. The adequacy of the data identifies gaps in the facility data that may influence the outcome of the dose reconstruction process. For example, if a site did not monitor all workers exposed to neutrons who should have been monitored, this would be considered a gap and, thus, an inadequacy in the data.

3.1.4 Objective 4: Consistency Among Site Profiles

SC&A reviewed the site profile with respect to Objective 4, which requires SC&A to identify common elements within site profiles completed or reviewed to date, as appropriate. In order to accomplish this objective, the INL TBDs were compared to the Hanford and Savannah River Site (SRS) TBDs. Both the Hanford and SRS site profiles are appropriate for comparison as the sites, all large DOE laboratories, had similar missions. Attachment 4 consists of a table comparing key assumptions and approaches taken by site profiles for the three sites. This assessment was conducted to identify areas of inconsistencies and determine the potential significance of any inconsistencies with regard to the dose reconstruction process.

3.1.5 Objective 5: Regulatory Compliance

SC&A reviewed the site profile with respect to Objective 5, which requires evaluation of the degree to which the site profile complies with stated policy and directives contained in 42 CFR Part 82. In addition, SC&A evaluated the TBDs for adherence to general quality assurance policies and procedures utilized for the performance of dose reconstructions. In order to place the above objectives into the proper context as they pertain to the site profile, it is important to briefly review key elements of the dose reconstruction process, as specified in 42 CFR Part 82. Federal regulations specify that a dose reconstruction can be broadly placed into one of three discrete categories. These three categories differ greatly in terms of their dependence on the availability, completeness, and accuracy/uncertainty of available dose data. The first two categories represent "extreme cases," where, in the first (Category 1), exposures are so obviously high as to lead quickly to a probability of causation (POC) of at least 50% and, in the second (Category 2), exposures are so obviously low, as to lead quickly to determination of a POC of less than 50%. The third category (Category 3) is the most difficult one to assess, as the claimant's exposure falls between the two extremes of the first two categories.

Category 1: Least challenged by any deficiencies in available dose/monitoring data are dose reconstructions for which even a partial assessment (or minimized dose(s)) corresponds to a POC value in excess of 50%, and assures compensability to the claimant. Such partial/incomplete dose reconstructions with a POC greater than 50% may, in some cases, involve only a limited amount of external or internal data. In extreme cases, even a total absence of a positive measurement may suffice for an assigned organ dose that results in a POC greater than 50%. For this reason, dose reconstructions in behalf of this category may only be marginally affected by incomplete/missing data or uncertainty of the measurements. In fact, regulatory guidelines recommend the use of a partial/incomplete dose reconstruction, the minimization of dose, and the exclusion of uncertainty for reasons of process efficiency, as long as this limited effort produces a POC of greater than or equal to 50%.

Category 2: A second category of dose reconstruction is defined by Federal guidance, which recommends the use of "worst-case" assumptions. The purpose of worst-case assumptions in dose reconstruction is to derive maximal or highly improbable dose assignments. For example, a worst-case assumption may place a worker at a given work location 24 hours per day and 365 days per year; while not logical, the assumption is certainly upper bounding on exposure. The use of such maximized (or upper bound) values, however, is limited to those instances where the resultant maximized doses yield POC values below 50%, which are not compensated. For this second category, the dose reconstructor needs only to ensure that all potential internal and external exposure pathways have been considered.

The obvious benefit of worst-case assumptions and the use of maximized doses in dose reconstruction is efficiency. Efficiency is achieved by the fact that maximized doses avoid the need for precise data and eliminates consideration for the uncertainty of the dose. Lastly, the use of bounding values in dose reconstruction minimizes any controversy regarding the decision not to compensate a claim.

Although simplistic in design, to satisfy this type of a dose reconstruction, the TBD must, at a minimum, provide information and data that clearly identify (1) all potential radionuclides, (2) all potential modes of exposure, and (3) upper limits for each contaminant and mode of exposure. Thus, for external exposures, for example, maximum dose rates must be identified in time and space that correspond to a worker's employment period, work locations, and job assignment. Similarly, in order to maximize internal exposures, the highest possible air concentrations and surface contaminations must be identified.

Category 3: The most complex and challenging dose reconstruction represents claims where the case cannot be dealt with under one of the two categories above. For instance, when a minimum dose estimate does not result in compensation, a next step is required to make a more complete estimate. Or, when a worst-case dose estimate that has assumptions that may be physically implausible results in a POC greater than 50%, denial is not possible. A more refined estimate may be required to support a recommendation either to deny or to compensate the claimant. In such dose reconstructions, which may be represented as "reasonable," NIOSH has committed to resolve uncertainties in favor of the claimant. According to 42 CFR Part 82, NIOSH interprets "reasonable estimates" of radiation dose to mean the following:

> *. . . estimates calculated using a substantial basis of fact and the application of science-based, logical assumptions to supplement or interpret the factual basis. Claimants will in no case be harmed by any level of uncertainty involved in their claims, since assumptions applied by NIOSH will consistently give the benefit of the doubt to claimants.*

3.2 ASSESSMENT METHODOLOGY

In order to assess the degree of compliance with the five objectives described above, SC&A reviewed each of the six TBDs, their supplemental attachments, TIBs, and other relevant documents, giving due consideration to the three categories of dose reconstructions that the site profile is intended to support (Section 3.1.5). The TBDs generally provide well-organized and somewhat or partially user-friendly information for the dose reconstructor when adequate data are available to do that comprehensively. During the course of its review, SC&A was cognizant of the fact that the site profile is not required by the EEOICPA or by 42 CFR Part 82, which implements the statute. Site profiles were developed by NIOSH as a resource to the dose reconstructors for identifying site-specific practices, parameter values, and factors that are relevant to dose reconstruction, and which may be used to supplement a claimant's own employment and exposure record. Based on information provided by NIOSH personnel, SC&A understands that site profiles are living documents, which are revised, refined, and supplemented with TIBs as required, to help dose reconstructors. Site profiles are not intended to be prescriptive nor necessarily complete in terms of addressing every possible issue that may be relevant to a given dose reconstruction. In addition, they are not intended for the "layman," but for the health physics personnel immersed in the review process. The principal documents and data sources SC&A examined in the course of its review, and which were most influential in informing SC&A's assessment, are the following:

- ORAUT-TKBS-0007-1, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Introduction* (Rohrig 2004i), explains the purpose and the scope of the site profile. SC&A was attentive to this section, because it provides a useful overview and explains the role of each TBD in support of the dose reconstruction process. Hence, the introduction helps in framing the scope of the site profile. As will be discussed later in this report, NIOSH may want to include additional qualifying information in the introduction to this and other site profiles describing the dose reconstruction issues that are not explicitly addressed by a given site profile. NIOSH may also want to include a roadmap to the dose reconstructor expounding on the entire dose reconstruction process and showing how the TBDs fit into that process.

- ORAUT-TKBS-0007-2, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Site Description* (Rohrig 2005), is an extremely important document, because it provides a description of the facilities and processes, as well as historical information that serve as the underpinning for subsequent INL TBDs. Specifically, this document describes the history and current status of 14 areas, 101 facilities and processes, and their associated source terms that are relevant to dose reconstruction. SC&A's review of this section addresses whether all the potentially important site activities and processes are described, and whether characterization of source terms is complete and sufficient to support dose reconstruction.

- ORAUT-TKBS-0007-3, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL)– Occupational Medical Dose* (Rohrig 2004m), provides a set of procedures and default dose values for reconstructing the radiation exposures of workers from medical radiographic procedures that were required of employees at INL. SC&A reviewed this section for completeness and technical adequacy.

- ORAUT-TKBS-0007-4, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL)– Occupational Environmental Dose* (Peterson 2004), provides background information and guidance to dose reconstructors for reconstructing the doses to unmonitored workers outside of the facilities at INL who may have been exposed to routine and episodic airborne emissions from these facilities. SC&A reviewed this section from the perspective of the source terms and the atmospheric transport, deposition, and resuspension models used to derive the external and internal doses to these workers. In addition, SC&A also reviewed the *Supplement to Technical Basis Document 4 for the Idaho National Engineering and Environmental Laboratory: INEEL Occupational Environmental Dose* (Peterson 2004s).

- ORAUT-TKBS-0007-5, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational Internal Dose* (Rich and Wenzel 2004), presents background information and guidance to dose reconstructors for deriving occupational internal doses to workers. This section was reviewed with respect to information and guidance regarding the types, mixes, and chemical forms of the radionuclides that may have been inhaled or ingested by the workers, the recommended assumptions for use in reconstructing internal doses based on whole-body counts and bioassay data, the methods recommended for use in the reconstruction of missed internal doses, and the methods recommended for characterizing uncertainty in the reconstructed internal doses.

- ORAUT-TKBS-0007-6, *Technical Basis Document for the Idaho National Engineering and Environmental Laboratory (INEEL) – Occupational External Dose* (Rohrig 2004e), presents background information and guidance to dose reconstructors for deriving occupational external doses to workers. This section was reviewed with respect to information and guidance regarding the different types of external radiation (i.e., gamma, beta, and neutron) and the energy distribution of this radiation to which the workers may have been exposed. SC&A also reviewed the recommendations for converting external dosimetry data to organ-specific doses, the methods recommended for reconstructing missed external doses, and the methods recommended for characterizing uncertainty in the reconstructed external doses.

- ORAUT-OTIB-0009, *Technical Information Bulletin in Support of INEEL Technical Basis Document Section 6: Reanalysis of Hankins MTR Bonner Sphere* (Rohrig 2004h), presents re-analyzed neutron dose equivalents of the original Hankins measured values for three IREP energy groups at the MTR floor (Hankins 1961). This TIB also provides a distribution of the neutron dose equivalent among the energy groups and their corresponding fractional NTA film responses.

- *Supplement to Technical Basis Document 4 for the INEEL: INEEL Occupational Environmental Dose* (Peterson 2004s) presents supportive material for occupational environmental dose information and data. In particular, Attachment 4-A of this supplement provides 18 tables listing 14 total organ doses (thyroid, skin, lungs, bone, breast, stomach, ULI, LLI, red marrow, gonads, SI, spleen, liver, kidneys) for the inhalation and air-immersion pathways at different INL facilities, due to either operational releases or episodic events from other facilities. The reason why these organ dose values are not included in the TBD is not specified; but the TBD indicates that the air-immersion dose calculations are not necessary, because NIOSH believes that they would be recorded in the fence-line TLD doses (pg. 24).

In accordance with SC&A's site profile review procedure, the reviewers performed an initial assessment of the six TBDs, their supporting documentation, and several TIBs. SC&A then submitted questions to NIOSH with regard to assumptions and methodologies used in the site profile. A teleconference was conducted with staff members of NIOSH, ORAU, an ORAU subcontractor (Intrepid), and the SC&A team to discuss SC&A's questions, and to allow NIOSH to provide explanations and clarifications. NIOSH, ORAU, and ORAU subcontractor personnel subsequently were given the opportunity to comment on a draft summary of the teleconference. A summary of the questions, conference call, and some subsequent clarifications is provided in Attachment 1 of this report.

SC&A conducted site expert interviews to help obtain a comprehensive understanding of the radiation protection program, site operations, and environmental contamination that might be present in some areas. While it is recognized that peoples' memories may not be wholly reliable, especially when trying to recall information from decades ago, the interviews, especially taken in the aggregate, provided much useful insight from the perspective of the INL workers themselves. Attachment 3 provides summaries of the interviews conducted by SC&A in the Idaho Falls area during the course of this review. The site experts interviewed include current, former, and retired staff from dosimetry, radiation control, operations, environmental monitoring, maintenance, instrumentation, electrical, mechanical, security, engineering, management, and other support organizations. Each summary is an edited paraphrase of conversations held with a number of site experts, rather than a verbatim transcript. Personnel statements have been grouped into categories to provide a linkage with various portions of the INL Site Profile. References that may identify specific site experts have been omitted for privacy reasons. Interviewed individuals were given the opportunity to review the interview summary for accuracy. This is an important safeguard against missing key issues or misinterpreting some vital piece of information. Most, but not all, of the individuals interviewed by SC&A provided comments on the summaries. The DOE-ID Classification Office also reviewed the summaries.

Information provided in the teleconference with NIOSH mentioned above was evaluated against the preliminary findings to finalize the vertical issues addressed in the report (Section 5). There are three levels of review for this report. First, SC&A team members review the report internally. Second, SC&A engages an outside consultant, who has not participated in the preparation of this document, to review all aspects of this report. The third level, referred to as the expanded review cycle, will consist of a review of this draft by the Advisory Board and NIOSH. The first two review levels have been completed.

After the Advisory Board and NIOSH (and its contractors and subcontractors) have an opportunity to review this draft, SC&A plans to request a meeting with Advisory Board members and NIOSH representatives to discuss the report. Following this meeting, SC&A will revise the draft and deliver the final version to the Advisory Board and to NIOSH. We anticipate that, in accordance with the procedures followed during previous site profile reviews, the report will then be published on the NIOSH web site and discussed at an Advisory Board meeting. This last step in the review cycle completes SC&A's role in the review process, unless the Advisory Board requests SC&A to participate in additional discussions regarding the closeout of issues, or if NIOSH issues revisions to the TBDs or additional TIBs, and the Advisory Board requests SC&A to review these documents.

Finally, it is important to note that SC&A's review of the TBDs and their supporting TIBs is not exhaustive. These are large, complex documents, and SC&A used its judgment in selecting those issues that we believe would be important with respect to dose reconstruction.

4.0 SITE PROFILE STRENGTHS

In developing a TBD, the assumptions used must be fair, consistent, and scientifically robust, and uncertainties and inadequacies in source data must be explicitly addressed. The development of the TBD must also consider efficiency in the process of analyzing individual exposure histories, so that claims can be processed in a timely manner; this is clearly in the best interest of the claimants. With this perspective in mind, SC&A identified a number of strengths in the INL TBDs. These strengths are described in the following sections.

4.1 COMPLETENESS OF DATA SOURCES

The INL TBDs exhibit the following strengths in terms of the completeness of their data sources:

(1) In an effort to be comprehensive in addressing the range of facilities and processes at INL, NIOSH effectively compiled facility-specific information from a number of descriptions and historical records, the most complete and expansive of which is Susan Stacy's book, commissioned by DOE to commemorate 50 years of INL operations, *Proving the Principle, A History of the Idaho National Engineering and Environmental Laboratory, 1949–1999* (Stacy 2000). In developing the site profile, NIOSH drew upon information contained in 287 reports and documents cited in the reference sections of five of the six TBDs. Facilities are grouped into 14 areas and 101 facilities and processes. A concerted effort was made to characterize the principal types and relative importance of the various radionuclides that may have contributed to internal and external exposures at the various facilities and associated processes over the life of the facilities. SC&A considers this compilation to be an important strength of the report.

(2) NIOSH made an admirable effort to compile historical data on medical x-ray equipment and techniques used at INL. From these data, NIOSH was able to develop a comprehensive, user-friendly table listing organ doses from occupational medical x-ray exposure to workers. For missing data prior to 1954, NIOSH has been able to use estimates from a document prepared by Ron Kathren (Kathren 2003) to fill the gap. Notwithstanding SC&A's opinion that there are opportunities for improvement in treating the uncertainties associated with the derivation of the organ doses, SC&A considers the medical exposure section to be one of the strengths of the site profile.

(3) In compiling the atmospheric source terms for deriving outdoor occupational exposures to unmonitored and monitored workers, NIOSH depended heavily on two reports; INELHDE (DOE 1991a) and RAC 2002. In addition, a supporting document, *Supplement to Technical Basis Document 4 for the Idaho National Engineering and Environmental Laboratory: INEEL Occupational Environmental Dose* (Peterson 2004), was developed to provide more detailed information on the data and methods used in deriving the potential worker intake values at 8 primary INL facilities for routine releases, potential worker intake values for identified episodic events, and direct gamma exposure values at 11 INL facilities. Though these reports are comprehensive and provide much useful information and guidance, SC&A believes that there are opportunities for

improvement in identifying other (missing) source terms and the methods used to reconstruct the missed environmental gamma doses.

(4) For the purpose of developing data needed to reconstruct missed internal doses based on historical operations, NIOSH compiled a significant amount of data identifying the radioactive materials at the various INL facilities, and describing the relevant operations and their associated processes. Notwithstanding this achievement, there are opportunities for improvement in the data sets and guidance for the dose reconstructors for reconstructing missed internal exposures, and also for identifying missed internal doses due to deficient work practices and inadequate instrumentation.

(5) In compiling historical data needed to reconstruct missed external (gamma, beta, and neutron) doses, NIOSH compiled a significant amount of data identifying potential external radiation sources at various INL facilities, and describing the relevant operations and their associated processes. NIOSH also compiled data related to external dosimetry used at INL over its entire operating history. From these gathered data, NIOSH developed the missed gamma and neutron dose values for different time periods. Opportunities for improvement remain, however, in the areas of adding the missed beta dose values and identifying high-risk jobs.

4.2 TECHNICAL ACCURACY/CLAIMANT-FAVORABILITY

The INL site TBDs exhibit the following strengths, in terms of their technical accuracy and claimant-favorability:

(1) Medical x-ray procedures have been investigated thoroughly to determine the radiographic techniques used at INL. The frequencies of medical examinations performed on new hires and on workers over the operating years have been exhaustively reviewed and determined. The INL Site Occupational Medical Dose TBD (Rohrig 2004m) provides methods for calculating doses from both posterior-anterior (PA; back to front) and lateral chest x-rays in Section 3.4 (pg. 6). For dose calculations, NIOSH provides default skin entrance air kerma values for PA and lateral chest x-ray examinations. It was determined that no photofluoroscopic examination has ever been performed at the INL site. Table 3A-1 of the TBD presents the recommended default organ doses for six time periods of INL operations. An assigned uncertainty of ±30% at 1 sigma (84% confidence) was recommended for the calculated dose results.

(2) NIOSH made a concerted effort to determine the airborne release source terms due to routine operations from major INL facilities and episodic events, such as unplanned criticalities and planned special fuel tests. NIOSH also thoroughly evaluated the environmental monitoring data over the operating history to determine the fence-line dose results. The Occupational Environmental Dose TBD provides a set of tables listing the worker intake quantities by year due to routine operations (Tables 4-1 to 4-8) and episodic events (Table 4-10 to 4-11), and the direct gamma dose to workers (Table 4-12) at primary INL facilities. The overall accuracy of the information provided is judged acceptable, however, NIOSH did not include other potential sources of missed dose to

workers, such as resuspended contaminated soils and accumulated ground depositions. In addition, NIOSH did not evaluate the validity and quality of the monitoring data to determine uncertainties associated with the source terms.

(3) NIOSH identified the primary contributing facilities to worker internal exposures by extensively analyzing the operations and processes at different INL facilities. NIOSH also evaluated the radiologically significant radionuclides for these operations and processes. The most limiting source terms for worker internal exposure were then determined to arise from the processed nuclear fuels at ICPP (Table 5.6.2.5-1, pg. 32). The TBD selected the most limiting radionuclides in the fuel inventory as the default worker intakes to calculate the missed internal doses at different INL facilities by multiplying by a set of weighting factors (Table 5.7-1, pg. 38). However, NIOSH did not evaluate other potential sources of missed internal dose to workers, which may have been due to inconsistent and deficient field work practices. In addition, NIOSH did not identify potential high-risk or high-internal dose jobs leading to missed worker internal doses.

(4) NIOSH made a concerted effort to determine the minimum reporting levels (MRLs) that were associated with the various types of dosimeters used over the life of the lab, for the different types of operations, and for the different types of external exposures (Rohrig 2004e, Tables 6B-1 and 6B-2, pg. 47). In addition, NIOSH determined the potential gamma, beta, and neutron radiation fields at different INL facilities (Section 6.3.4). The assessment was not as complete and as accurate as it could have been, however, as NIOSH did not evaluate other potential sources of missed external dose to workers due to inconsistent and deficient field work practices. NIOSH also did not identify potential high-risk or high-external dose jobs, which may have led to significant missed worker external doses.

(5) Similar to the Hanford site profile, the development of separate TBDs for the six primary areas, i.e., introduction, site description, Occupational Medical Dose, occupational environmental dose, occupational internal dose, and occupational external dosimetry, forms a model for later site TBDs. The format of the INL TBDs is more user-friendly than some of the previous TBDs, such as for Savannah River. The use of examples is beneficial to the dose reconstructor, as they summarize and clarify the suggested procedures to be followed. NIOSH should expand the presentation of examples in revised or new site profile TBDs; for instance, there is only one example provided in the Occupational External Dose TBD.

4.3 ADEQUACY OF DATA

The developers of the TBDs benefited from having access to information and data that were compiled as a part of several INL site programs. The following summarizes these programs and how the TBDs benefit from them, and also points out areas where the programs may have been deficient or where the TBDs could have made better use of the available data.

(1) While INL's radiological control program has undergone continuous improvement since the inception of site operations, there are significant concerns about deficiencies in the program and its implementation in the early years of the site history. Radiological control personnel have implemented upgraded procedures and technology over time to reduce radiation dose to workers, and have enhanced personnel monitoring programs as more advanced equipment and instrumentation were added.

(2) The personnel monitoring program utilized timekeeping, film badges (later TLDs), pocket ionization chambers (PICs), and air sampling to monitor and control worker exposure. The two-element film badge was used as early as August 1951 (Rohrig 2004e), but INL continued to use timekeeping and PICs to monitor real-time personnel dose after the implementation of the film badge. However, these PIC data were not provided in the TBD for dose reconstructors to use, although there were significant concerns about the validity of the film badge results among many site experts interviewed.

(3) The INL dosimetry program initially used neutron film badges with Kodak nuclear track emulsion Type A (NTA) for monitoring worker exposure to neutron fields. Because of the missed dose from any neutrons below the NTA energy threshold of 0.5 to 0.8 MeV, INL switched to Hankins albedo dosimeters (Rohrig 2004e).

(4) The external dosimetry staff characterized the workplace radiation fields at INL in response to a Tiger Team finding, making field measurements with a NaI(Tl) spectrometer and TLDs mounted on a phantom. The measurement results were compared to determine relative bias. These data, however, were not used in determining the missed external gamma and neutron doses.

(5) In 1996, the INL external dosimetry staff performed neutron radiation field characterization to determine a facility neutron correction factor (FNCF) from the 9-inch to 3-inch ratio in the worker location. This FNCF is used to adjust the measurement result to dose equivalent.

(6) INL has identified the MTR experimental floor as the largest contributor of neutron exposures to workers. In 1961, INL used 2-, 3-, and 8-inch polyethylene Bonner balls in a cadmium shield to characterize the intermediate and fast neutron at 21 locations around the MTR floor. Measurements of thermal neutron components were also made at six other locations. These 1961 data are reanalyzed in the TBD (Rohrig 2004h) using more recent Bonner response curves. Correction factors for missed neutron dose are then estimated.

(7) Routine bioassay involving in-vitro and urine sampling were implemented in the beginning of the site history (probably from 1953). Whole-body counting became routine in the 1960s. However, thyroid counting used for high-risk workers was not mentioned in the TBD. The frequencies of the bioassay program evolved from an annual routine program to an event-based program in the later years, but there are significant gaps in these data, which need to be addressed. NIOSH gives incomplete instructions on how to use bioassay data to calculate doses.

(8) INL implemented environmental monitoring from the beginning of its operational history. All of the airborne releases and their impact to the populations around INL boundaries were reviewed and analyzed. With the help of NOAA, hourly meteorological data were available for the analyses. All the releases and analyzed results up to 1991 are documented in the INLEHDE (DOE 1991a).

4.4 CONSISTENCY AMONG THE SITE PROFILES

Although INL, Hanford, and Savannah River Site (SRS) missions overlapped to a significant extent, there are many differences in the facility designs and processes, and radiological practices. Attachment 4 compares many parameters and assumptions of these three sites. In some cases, these differences require site-specific assumptions in dose determinations. For example, due to many different nuclear reactor experiments and fuel reprocessing operations at INL, there is a long list of radionuclides of concern, including Cs-137, Co-60, Sr-90, Pu-239, U-235, and I-131. INL, however, did not have to consider the substantial particulate releases of ruthenium from the REDOX facility at Hanford; nor did SRS, whose facilities were built later. NIOSH has made a concerted effort to recognize and address site-specific issues in the TBDs. With respect to the Interactive RadioEpidemiologic Program (IREP) input parameters, the INL, Hanford, and SRS TBDs are consistent in many cases. This consistency was especially apparent in the occupational medical exposure sections, as seen in Table A.4-1 in Attachment 4.

4.5 REGULATORY COMPLIANCE

The TBDs' reliance on environmental monitoring data, but only limited personnel data, to determine dose is not entirely consistent with the requirements outlined in 42 CFR Part 82, which specifies the hierarchy of data that is to be used in dose reconstruction. NIOSH has not effectively complied with the hierarchy of data required in §82.2 and its implementation guides for monitored workers. The two examples that follow show that NIOSH did not compile pertinent data, like bioassays and external doses, and dose rates.

(1) Where in-vivo and in-vitro analyses are available, this information is provided for use in the determination of internal dose. However, the Occupational Internal Dose TBD relies entirely on reprocessed fuel information as the bounding source terms for potential worker inhalation intake.

(2) The Occupational External Dose TBD does not use beta/gamma and neutron dosimetry data in the determination of external exposure. NIOSH uses the MRL values for different types of film badges and dosimeters to calculate the potential maximum external (gamma and neutron) dose to workers. Where environmental measurements are available, these data are used as the basis for environmental dose.

5.0 VERTICAL ISSUES

SC&A has developed a list of issues regarding the INL site profile, which relate to the five objectives defined in the SC&A site profile review procedure (SC&A 2004a). Some issues pertain to a particular objective, while others to several objectives. A matrix relating the objectives and the relative importance of each issue is provided in Section 6.0. The issues identified as findings map into the four broad categories discussed in Section 5.1, and the issues identified as observations map into the two broad categories discussed in Section 5.2. Many of the issues raised are applicable to other DOE and Atomic Weapons Employer sites, and should be considered in the preparation and revision of other site profiles.

5.1 DISCUSSION OF ISSUES

5.1.1 Occupational Environmental Dose Issues

The Occupational Environmental Dose TBD divides potential occupational environmental exposure to workers at the INL facilities into two pathways; (1) airborne releases from INL facilities, and (2) direct gamma doses resulting from facility operations. The airborne releases are further subdivided into routine operational releases and episodic releases. NIOSH used the source terms in the document, *Idaho National Engineering Laboratory Historical Dose Evaluation Report* (INELHDE), to develop a set of annual inhaled quantities (Bq/yr) for routine operational releases for 8 INL facilities for the dose reconstructors to choose from. NIOSH also developed intakes (Bq/event) for all identified episodic releases in the history of INL (from 1952 to 2002). For direct radiation to workers, NIOSH used exclusively fence-line environmental TLD records to develop direct gamma values for 11 INL facilities from 1952 to 2002.

5.1.1.1 Routine Airborne Releases

For airborne emissions from routine facility operations, the Occupational Environmental Dose TBD relies heavily on two previous work; the INELHDE Report (DOE 1991a) and *Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory, Final Report* (RAC 2002). The INELHDE project was chartered in December 1988 in response to inquiries concerning possible radiological consequences to the public from past projects and facility operations at INL. The purpose of the report is to provide estimates of potential offsite radiation doses encompassing the entire operating history of INL using a consistent methodology for all years. While potential ground water, surface water, and biotic pathways exist, they are not significant contributors to the offsite radiation dose, because the principal route by which radioactivity released on the INL site can reach offsite locations is through the airborne pathway. Consequently, this report concentrates on estimates of doses resulting from radioactivity releases to the atmosphere that are subsequently transported offsite through airborne pathways.

Atmospheric releases are categorized as either operational or episodic. Operational releases are continuous and fairly uniform releases, occurring over a year or over a portion of a year, that span a variety of meteorological conditions; the report uses annual average meteorological conditions. Episodic releases took place over a short period of time, typically a few hours, and may have been associated with a particular planned or unplanned event, such as the SL-1 accident. Some tests as long as several months were modeled as episodic releases. Episodic releases are treated as distinct events using the actual meteorological conditions measured at the time of each release. The primary source of information for estimating the operational releases of radioactivity to the atmosphere is the measurement of stack effluents, as reported in the Radioactive Waste Management Information System (RWMIS) database. The report performed supplemental calculations using information from other technical reports to determine the radionuclide composition of the effluent. For episodic releases, estimates are taken from technical reports on individual INL facilities and test programs.

The total amount of radioactivity associated with annual operational releases at INL was greatest in the late 1950s and early 1960s, and has decreased since then. The peak release of radioactivity in the 1980s was approximately one-tenth of that released in 1961. A total of 108 individual episodic releases are considered in the INELHDE, and detailed dose calculations were done for the 54 most important releases. Most of the important episodic releases were associated with reactor fuel development tests conducted before 1961 (DOE 1991a, pp. v–vi).

In the INELHDE document, calculation of atmospheric transport to locations off the INL site was done using the MESODIF computer code. The name MESODIF is derived from the terms mesocale (MESO) and diffusion (DIF) to indicate that the dispersion predictions of the model are valid for distances up to about 150 km (95 miles) from the release point (DOE 1991a, pg. 15). Data for dispersion calculations were taken from the records of the INL network of meteorological stations. Even though the INELHDE was developed only for evaluating the offsite population dose impact due to INL releases, the exposure pathways are similar for the onsite workers. Radiological assessment calculations were made for four exposure pathways; (1) external exposure from immersion in contaminated air, (2) external exposure from radioactivity deposited on the ground, (3) internal exposure from inhalation of contaminated air, and (4) internal exposure from ingestion of contaminated agricultural products. However, the INL Environmental TBD focuses on radiological assessment calculations for pathway (3). It does not evaluate pathway (1), and inadequately evaluates pathway (2). As for pathway (4), the TBD does not consider any ingestion scenarios.

The TBD indicates that potential offsite doses from INL activities have been small. The largest radiation doses were calculated for an infant in 1956, when the effective dose equivalent (EDE) from operational and episodic releases was estimated to be 61 mrem and the thyroid dose equivalent was estimated to be 1,350 mrem. The trend in later years has been toward smaller doses as total INL site releases have declined. Episodic releases have made a substantial contribution to the total potential radiation dose only for a few years during the test program of the 1950s and 1960s. Radiation doses from airborne releases over much of the operating history of INL were small, compared to doses from natural background radiation (DOE 1991a, pg. v).

To determine onsite concentrations of radionuclides from operational releases at INL facilities, NIOSH used the same methodology employed in the INELHDE. The release source term for each year of operation is the same as that documented in the INELHDE, with some minor modifications (see discussion below). With one exception, NIOSH performed an analysis to reduce the number of radionuclides from 56 to 9 for operational releases that contributed less than 95% of the inhalation doses. For operational releases, the TBD used annual average meteorological conditions (mesoscale dispersion isopleths of ground-level air concentrations) from Bowman 1984 for the years prior to 1993. For 1993 to 2002, NIOSH used average meteorological conditions from annual environmental reports (ESRF 1994, 1995, 1996, 1997, 1998, 2000; Stoller 2002a, 2002b, and 2002c) (TBD, pg. 10). For routine releases, NIOSH developed worker inhalation-intake values for 8 INL facilities (ANL-W, ARA, CFA, ICPP, RWMC, SPERT, TAN, TRA) for 9 key radionuclides (Ce-144, I-131, Pm-147, Pu-238, Pu-239/240, Ru-106, Sr-89, Sr-90, and Y-91) from 1952 to 2002. These intake values (Bq/yr) are presented in Table 4-1 to Table 4-8 of the Occupational Environmental Dose TBD (Peterson 2004). The following quotations are from that TBD and set the stage for the discussions that follow in subsequent sections.

> *Other than the early releases from the GE-ANP IET operations, which are treated as episodic releases, by far the greater majority of the airborne radiological releases were contributed by the ICPP, the TRA, and EBR-I reactor, in that order. The ICPP airborne releases peaked in 1959 at about 1.3 million curies, dropped to about 800,000 curies the year after, rose to about 1.2 million the next year and then declined steadily to less than 200,000 curies in 1964 and remained at that level until 1975 when it rose for a year to about 250,000 curies. Since 1976 the annual releases from ICPP have been less than 200,000 curies. The TRA annual releases peaked at slightly less than one half of the ICPP releases (about 600,000 curies) in 1963 and have declined to less than 200,000 curies in 1967 and have remained less than 100,000 curies since 1969. The ANL-W airborne releases, generally associated with EBR-II, peaked at 19,000 curies in 1965 and dropped steadily to less than 1000 curies in 1969 where it has remained since.*

> *Annual site boundary doses for the maximally exposed individual reported in the Annual Environmental Monitoring Reports assume the total INEEL airborne effluent is released from a 250 ft. (76 m) stack midpoint between TRA and ICPP. Meteorological dispersion for the year is calculated for ground-level elevations by NOAA for the release height as presented in the average mesoscale dispersion isopleths. Originally, in the INEEL 4 TBD, INEEL facility personnel intakes for operational releases (Bq/yr) were calculated using INEEL total annual release radionuclides and respective quantities documented in RAC 2002. These release radionuclides and quantities are the result of a critical review by RAC of the releases documented in the INELHDE, performed in 1990 and 1991. The INELHDE took its operational release source term from the Radioactive Waste Management Information Service (RWMIS), the official documentation for gaseous, liquid, and solid radioactive wastes produced by facilities at the INEEL. However, when the INELHDE evaluation was performed, some minor problems were identified with respect to the annual source terms attributed to the TRA and ICPP releases as documented in the RWMIS. Identification and correction of these problems are discussed in length in pages A-12 through A-44 of the INELHDE. In the late 1990s, the State of Idaho requested the Centers for Disease Control (CDC) to critically review the INELHDE, which was contracted out to the RAC. The RAC review of the INELHDE "improved" the source terms for the*

annual operational releases as described in pages 10 through 23 of RAC 2002. In summary, the changes to the RWMIS annual releases made for the INELHDE were in some cases substantial, but annual curie amounts remained unchanged. The changes in the INELHDE source terms made by the RAC were more for completeness and were not substantial. All changes preserved the annual total curies that are documented in RWMIS. In the final RAC assessment, the list for annual releases included 56 radionuclides, which was preserved when the first version of the INEEL 4 TBD was written.

The formal review of INEEL 4 TBD requested that the number of radionuclides be reduced from 56 such that 95% of the original dose was preserved. The reduction was accomplished by using an option provided by the RSAC-6 code that allowed the total CEDE to be listed by radionuclide in ascending dose order. The reanalysis using this option allowed the number of radionuclides to be reduced from 56 to 16. After that reduction was accomplished, a further review suggested that the number of radionuclides be reduced to 10 and still preserve 95% of the original dose. The list was reduced to 11 radionuclides using the ICRP 26/30 methodology with the help of the RSAC-6 code. Calculations at that time showed that eliminating one more radionuclide would have reduced the original total CEDE to 92%–93% of the original dose. The final review suggested that the final list include only those nuclides contained in a list treated by the Integrated Modules for Bioassay Analysis (IMBA), which meant that the two radionuclides, ^{132}I and ^{89}Rb, which contributed the least dose of the 11 be omitted. The final reduction, completed with a table of doses, created by the ICRP 60/66 methodology for the INEEL operational releases, indicated that ^{132}I and ^{89}Rb each contributed only about 0.7% of the semi-final dose. Therefore, those two radionuclides were deleted leaving the total list of radionuclides to a total of nine, all of which are treated by the IMBA code used by the dose reconstructors. (Peterson 2004, pp. 9–10)

5.1.1.1.1 <u>Completeness and Quality of Release Data Used</u>

(1) The INELHDE report uses effluent release data primarily from stack monitoring and air sampling systems installed across the INL site. There were a total of 23 air samplers; 12 within the site, and 11 outside the INL boundaries. The INELHDE claims:

> *The best and most comprehensive available data on releases of radioactivity to the atmosphere from INEL facilities have been used. The primary source of information on operational releases was the measurement of stack effluents as reported in the Radioactive Waste Management Information System Information System (RWMIS) data base. The data in RWMIS are generally most complete and reliable for the years since this data base was established in 1971. Data for the years from 1962 to 1970 are less complete and reflect less sophisticated monitoring instrumentation and record keeping systems. Data from 1952 to 1961 do not identify the specific mixture of radionuclides released; therefore*

> *additional assumptions and calculations were required to determine the radionuclide composition of the effluent and perform the dose assessment* (DOE 1991a, pg. 11).

The ambient air monitors at various INL site locations had been found to be deficient in meeting the siting requirements specified in 40 CFR 58 (near obstructions) and meeting minimum flow rate for particulates specified in 40 CFR 50 (e.g., 2.5 cfm versus 39 cfm), as cited in a DOE report, Tiger Team Assessment of the Idaho National Engineering Laboratory, DOE/EH-0178, 1991 (DOE-HQ 1991). NIOSH should evaluate the adequacy of the stack release data for use in the TBD. The Tiger Team report states the following:

> *Deficiencies exist within data residing in the RWMIS data base in that the data is not subject to formal validation and verification procedures by the generator before the annual certification is made. RESL does not believe that verification/validation is part of their responsibility; however, no other entity at INEL is performing that function. In one instance, data has been omitted from this data and not identified through any verification/validation process. The example noted by the Tiger Team involved the omission of Krypton-85m, Krypton-87, and Krypton-88 from the emission inventory reports for the ZPPR at ANL-W between 1980 and 1991.* (Vol. 1, pg. 3-265)

> *There is one particulate radiation monitor in the immediate vicinity of ICPP. This monitor is located near the ICPP fence line but is not downwind of ICPP radiation sources during predominant wind directions, and therefore is not sited properly to observe most low-level radiation releases from ICPP.* (Vol.1, pg. 3-48)

> *The ambient air monitors observed did not meet the siting requirements specified in 40 CFR 58, Appendix E, Paragraph 8.2, with respect to locating monitors near obstructions (i.e., buildings, parking lots) and maintaining an unrestricted airflow 270 degrees around the air sampler. For example, the ambient air monitor located at RESL is within 10 feet of the corner of CF-676 behind CF-690, and the ambient air monitors located at TAN, and ANL-W are located in parking lots adjacent to the fenceline.* (Vol. 1, pg. 3-50)

> *The type of ambient air monitors being used by RESL and Rockwell are of the low volume variety. The flow rate of the air samplers vary between 1.5 and 2.5 cubic feet per minute. 40 CFR 50, Appendix B, states that the minimum and maximum flow rate for particulate monitors is 39 to 60 cubic feet per minute, respectively.* (Vol. 1, pg. 3-50)

> *The height of the ambient air monitors maintained by RESL and Rockwell is approximately 1 meter. 40 CFR 58, Appendix E, states that the required*

> *height of the air intake for large scale samplers is 2–15 meters.* (Vol. 1, pg. 3-50)

Radiological effluent sampling and monitoring systems throughout INL facilities had not been evaluated to ensure that they would detect, quantify, and respond adequately to unplanned releases (DOE-HQ 1991). Also, airborne effluent particulates from INL operations had not been adequately characterized, and measurements and sampling techniques did not ensure a representative sample for effluent monitoring systems (DOE-HQ 1991). Several quotations from the Tiger Team report illustrate these deficiencies (and point to areas that NIOSH should investigate further):

> *A review of the radiological effluent sampling and monitoring systems at contactor facilities throughout the INEL has indicated that the majority of the stacks with the potential for release of radioactive particulates in the effluent have not been evaluated to ensure that particulate sampling is representative or that it will detect, quantify, and adequately respond to unplanned releases of radioactive material to the environment as required in DOE 5400.5. ANSI N13.1-1969 provides data which shows that the longer the sample delivery line and the greater the number of 90 degree turns, the greater the potential for particulate losses (depending on particulate sizes, sampling flow rate, particulate density, etc.). Examples of facilities in question are described below.* (Vol. 1, pg. 3-59)

> *The MTR Stack (TRA-710), the Hot Cell (TRA-632), Science & Technology Building (TRA-604), alpha Wing Addition (TRA-661), and the ATR stack (TRA-771) have not been evaluated for particulate losses in the sample delivery lines, sampling probe integrity, and the proper placement of the sampling probe. Sample delivery lines are typically too long and possess 90 degree bends. A study has not been conducted to determine whether or not the sample collection flow is isokinetic. An EG&G internal review of the ATR stack stated that 'there is no evidence that representative samples are taken.'* (Vol. 1, pg. 3-59)

> *Flow rate measurements are not taken for the TRA-604 stack sampling system as required by DOE 5400.1, Chapter IV, Section 5.a.(2). The Sampling system flow rate is determined by the power of the motor providing the suction. This method also fails to correct for flow rate changes due to filter loading or equipment malfunctions.* (Vol. 1, pg. 3-59)

> *The CFA laundry fails to take flow rate measurements at the beginning and end of each sampling period to account for dust loading (I-R-204). Additionally, during air sample filter analysis, the wrong alpha self-absorption factor was being used per EG&G Idaho procedures (R-223). These factors will cause an underestimation of the effluent release concentrations.* (Vol. 1, pg. 3-59)

The air sample extraction systems on the stacks at PBF-WERF Incinerator and TAN Hot Cell/Hot Shop have not been evaluated for particulate losses resulting in the sample delivery line (I-R-257, I-R-274). (Vol. 1, pg. 3-59)

The TAN Decontamination Shop Stack was found to have no stack monitoring system after the high efficiency particulate air (HEPA) filtration. This facility is not presently utilized. (Vol. 1, pg. 3-59)

The RWMC Drum Vent System stack monitoring system has not been evaluated for particulate losses in the sample delivery line. The sample delivery line possesses three 90 degree bends, and is approximately 20 feet long. This facility is currently not in use and a larger facility is expected to house the operation, however, without any planned upgrades to the stack monitoring system other than extending the stack height. EG&G Idaho has determined that this facility will be required to perform continuous monitoring under NESHAP when operational and , hence, the stack monitoring system will be required to conform to 40 CFR 61.93 and ANSI N13.1-1969 (I-R-427). It currently does not meet these standards. (Vol. 1, pp. 3-59 and 3-60)

The Main Stack and FAST stack continuous, on-line radiological monitoring systems for airborne effluents collect samples of the particulate in the stack gas through the use of a slip-stream. The slip-stream is drawn at a constant flow from the primary sample delivery line, and particulates are collected on a bulk filter. The slip-stream sample delivery line extends away from the primary sample delivery line at a 45 degree angle incident to the direction of the flow. This primary sample delivery line has been characterized for particulate losses during routine operations and unplanned release events (R-477). However, the particulate losses and the adequacy of the slip-stream design to respond in a timely manner during an unplanned release event has not been determined. (Vol. 1, pg. 3-61)

The stack monitoring delivery systems at the NWCF have not been evaluated for particulate losses in the sample delivery lines from the calciner and the HVAC system and have not been evaluated to determine whether the sample flow rate is isokinetic. The sample delivery lines exceed 30 feet in length and have at least three 90 degree bends. (Vol. 1, pg. 3–61)

ICPP identified 49 sources that were evaluated for radioactive air source emissions as part of the document prepared for the NESHAP waiver. This document presents the release rates prior to abatement for sources of radioactive air emissions in worksheet form. Some of the abated release rates provided by ICPP are not consistent with what is reported in previous Environmental Monitoring Reports. For example, the NESHAP

> *emission report presents the FAST stack unabated release for tritium as 2.31 curies, which does not reflect operating conditions. The 1987 and 1988 Annual Monitoring Reports present the measured releases for tritium as 480 and 183 curies, respectively. .. Approximately 150 air radioactive emission sources were presented in INEL's Air Emission Inventory. As a result of INEL's review of source terms, only three sources of radioactive emissions were only slightly below the limit requiring continuous monitoring (e.g., 0.067 mrem/yr from Vent 008 at CPP-627) and were rejected as requiring continuous monitoring (continuous monitoring required at 0.1 mrem/yr). Considering the above deficiencies in source term characterization, the potential exists that other sources of radioactive air emissions at INEL may require continuous monitoring as prescribed in 40 CFR 61.93 in addition to those already identified.* (Vol. 1, pg. 3-67)

Given these deficiencies in the INL stack monitoring and air sampling systems identified by the DOE Tiger Team audit, it is unlikely that the INELHDE results would be complete and representative of the actual effluent releases from different INL facilities. This would further impact the quality and the validity of the dose assessments. NIOSH should evaluate the uncertainties associated with these issues, so that the recommended worker intake values from environmental releases would be truly claimant favorable. In 2003, SC&A performed a study of radioactive release source terms for two major INL programs for the CDC (SC&A 2003). This study reviewed stack monitoring data and uncertainties associated with the data. SC&A provided this report to NIOSH for use in the preparation of the INL Site Profile.

(2) During the site expert interview conducted by SC&A, INL environmental staff indicated that other unplanned (episodic) releases occurred at different facilities that are not included in the INELHDE. Therefore, they are also not included in the NIOSH TBD. For instance, there was an incident at the INTEC (ICPP) in the early 1990s where particulate releases were observed as a result of a new steam cleaning process of the CPP stack. The airborne material released was believed to be Cs-137 attached to white insulation material. Measurable radioactivity was associated with these releases. This information is not included in the TBD for the use of dose reconstructions.

5.1.1.1.2 Dispersion Model

The Environmental TBD uses a mesoscale model (MESODIF), which employs an objective regional trajectory computational scheme, combined with the Gaussian diffusion equation for a continuous point source, to estimate dispersion for transport of releases. It is a forward time-marching Gaussian plume model in which successive, small plume elements (or puffs) are advected throughout the computational area. The following quotations, taken from the INELHDE (DOE 1991a), describe the use of the mesoscale dispersion data from MESODIF model for the calculations of average ground level air concentrations at INL facilities:

> *Since 1973, meteorological information gathered from the 26-station INEL telemetry network in the Upper Snake River Plain has been evaluated annually using the MESODIF computer code. The results were illustrated in dispersion isopleth maps contained in annual INEL Site environmental monitoring reports. Adequate meteorological information was not available before 1973 or in 1978, when the telemetry system was being upgraded, to produce annual dispersion isopleth maps. For these years, the MESODIF computer code was run on a 9-year (1974–1983, but excluding 1978) set of meteorological data from the telemetry system. The resulting "average dispersion conditions" were used in computing the annual impact from operational releases before 1973 and in 1978. The annual dispersion isopleth maps prepared from the MESODIF computer code output were used from 1973 through 1989, except for 1978. The annual average wind speed toward the inhabited location of highest annual average concentration from a point midway between the Idaho Chemical Processing Plant (ICPP) and Test Reactor Area (TRA) was estimated based on the joint frequency distribution of wind speeds. The wind speed distribution for 9 years (1974–1983) was used to estimate the annual average wind speed before 1984. The annual distributions were used for this purpose beginning in 1984. The detailed data from a grid of 176 points, on which the dispersion isopleth map is based, were used to calculate the ground-level dispersion coefficient at Atomic City, the location of the maximally exposed individual for the 9-year average. This was done with a log-log interpolation from the four grid points surrounding Atomic City.* (DOE 1991a, Appendix B, pg. B-9)

> *MESODIF was also used to calculate dispersion coefficients for episodic events occurring at the INEL. The data flow for discrete episodic events is shown in Figure B-i. For these individual events MESODIF was run on a finer scale grid, grid point spacing of 2 mi as opposed to 5 1/3 mi for the annual operations. Available meteorological data for the years before 1967 were limited to wind sensor tower sites located at the Central Facilities Area (CFA) and Initial Engine Test (JET). The accuracy of the plume trajectories calculated for the past events depends on how well these two wind stations were able to represent the actual winds affecting the plume. It is reasonable to assume that the further the plume traveled from the wind tower locations, the less reliable were the calculations. Most of the events were initiated within a few miles of one of these sites, so the important initial portions of these calculated trajectories are considered reasonable representations.* (DOE 1991a, Appendix B, pg. B-11)

MESODIF is a trajectory model that specifically requires spatial information describing upper boundary layer meteorological conditions. INL, however, does not have a real-time database that could be used in defining model trajectories. This caused significant uncertainties in accurately estimating the dispersion of released materials. This deficiency was noted in the DOE-HQ 1991 Tiger Team report (DOE-HQ 1991), a contemporaneous report of the INELHDE:

> *The site has an adequate surface meteorological observation network, but does not conduct regular monitoring to describe meteorological conditions above the*

> *heights of the surface meteorological towers, and does not have a complete upper air database. The INEL site encompasses 890 square miles and has only three meteorological towers with observation levels above 50 feet, two of which have observation levels at 200 feet and one at 250 feet. The remainder of the boundary layer, above 250 feel to approximately 5,000 feet, is not characterized. The INEL has numerous sources which utilize stacks equal to or greater than 200 feet to vent air emissions. Plume rise from these sources was frequently observed during this assessment to exceed 500 feet above the ground, and the air emissions are transported in an atmospheric region that that site is currently not characterizing…. Upper air data is available from the National Weather Service offices in Boise, ID and Salt Lake City, UT, but these data are not representative of the INEL site conditions given the close proximity and orientation of the mountains to the site.* (DOE-HQ 1991, Vol. 1, pp. 3-56 and 3-57)
>
> *RESL performs an analysis of all onsite monitor locations using the results generated by the mesoscale diffusion (MESODIF) model. The MESODIF model is designed for transport distances of greater than 20 km. This model is not appropriate for evaluating monitor locations within 20 km, and this type of model is not recommended by EPA-540/4-87-007 for evaluating appropriateness of monitor locations within these transport distances.* (DOE-HQ 1991, Vol. 1, pg. 3-47)
>
> *RESL has performed an evaluation of offsite monitors using the annual average results of the annual MESODIF and CAP-88 National Emission Standards of Hazardous Air Pollutants evaluation. As discussed in Finding A/CF-6, the lack of a representative upper air meteorological data for the INEL limits the accuracy of the MESODIF model because it requires upper air data to define representative trajectories for plumes. Without this data, it cannot be verified that offsite (>20 km) monitors are located to be representative of maximum INEL related pollutant concentrations.* (DOE-HQ 1991, Vol. 1, pg. 3-47)

First, the mesoscale model used by INL in the INELHDE is only appropriate for evaluating dispersion coefficients for locations at greater than 20 km distance, and is not appropriate for those facilities that are within 20 km of each other. It is definitely not suitable for determining dispersion of airborne releases within several hundred feet from a facility building or a stack. Many INL facilities, however, are within 20 km from each other. Second, even for facilities located more than 20 km from each other, the dispersion coefficients calculated by this model are deficient and not representative. Third, and most important, the workers considered in this site profile are more impacted by the release plumes at the facility where they worked than those from more distant facilities; this mesoscale model is not capable of addressing such short distance dispersion coefficient factors. NIOSH should re-examine the validity of the mesoscale model data used in the occupational environmental TBD.

5.1.1.1.3 Other Observations

NIOSH should list the routine airborne release activities and associated uncertainties for each INL facility in the Occupational Environmental Dose TBD. This data would be helpful for the dose reconstructors to assess whether the worker intakes are applicable to the claim they are considering. An example would be useful, showing how the worker exposure could be calculated using the release activities, uncertainty values, and weighting factors.

5.1.1.2 Episodic Airborne Releases

To determine onsite concentrations of radionuclides from episodic releases at INL facilities, NIOSH used the Radiological Safety Analysis Computer code, RSAC-6 (Wenzel and Schrader 2001; Peterson 2004s) for the pathway calculations. NIOSH employed the total list of radionuclides that are applicable to the respective episodic release from INELHDE. In the Occupational Environmental Dose TBD, NIOSH developed three tables providing worker inhalation intake values (Bq/event) for important radionuclides for (1) seven criticality events (Table 4-10), (2) special tests (Table 4-11), and (3) initial engine tests (Table 4-12). Peterson, in the environmental TBD supplement, wrote about the use of RSAC:

> *For the INEEL technical basis document (TBD), the RSAC program was used primarily for calculating on-site meteorological dispersion parameters for the various airborne release incidents; radiological doses due to these airborne releases was a minor additional calculation and was originally to be incorporated into the TBD only as an indication of the magnitude of the dose since they were relatively small and easy to calculate. These doses were originally incorporated into the TBD to be an indicator to the dose reconstructor only of the magnitude of the relative doses. They were removed as a result of reviewer comments.* (Peterson 2004s, pg. 7)

> *The RSAC program has the ability to calculate a fission product inventory for a given reactor operation (or a criticality) that is comparable to an ORIGEN fission product inventory calculation, the industry standard. The program can then tailor modeled releases to emulate actual releases from the standpoint of material released, attenuate the material by filters or by holdup within a volume containment, and for the rate of release, either instantaneously or over a period of time. The program provides the option to use various types of meteorological diffusion and contains the meteorological parameters applicable at the INEEL, all of which are based on the Gaussian plume model. From the meteorologically dispersed material, the program calculates, according to the ICRP 26 and ICRP 30 methodology, radiological doses from the ingestion, inhalation, air-immersion, cloud-gamma, and deposition pathways to 19 different organs of the body and a corresponding external effective dose equivalent (EDE) or a committed effective dose equivalent (CEDE) to the whole body, depending on the pathway for dose calculation. All of the doses discussed here were calculated with default parameters (plume centerline, particle size, solubility, etc.) such that radiological doses are maximized.* (Peterson 2004s, pg. 7)

> *The code has been extensively used for safety analysis work at the INEEL since 1965 when it was used for dose calculations for the Safety Analysis Report for the Advanced Test Reactor. After the publication of the INELHDE (DOE 1991*[a]*), the Centers for Disease Control (CDC) and the State of Idaho instigated a technical review of the methodology for the analyses for this report. As a result of this technical review, minor changes were recommended in some of the airborne source terms. These minor changes did not change the total curies released, but added small amounts of other radionuclides and respective quantities to earlier years where detection had not been as 'low level' as in the more recent years. In the course of the review, the Radiological Assessment Corporation (RAC) examined the RSAC-4 program that defined the radiological doses for each of the releases.* (Peterson 2004s, pg. 8)

SC&A considers use of the RSAC code and underlying model adequate for calculating dispersion factors and the resulting dose to workers.

5.1.1.2.1 Completeness and Quality of Episodic Releases Data

The airborne releases associated with several of the Initial Engine Tests (IETs 3, 4, and 10) of the Aircraft Nuclear Propulsion (ANP) Program, as estimated by the INELHDE, were likely to have been underestimated as follows:

- IET 3 – underestimate of total radionuclide release by up to a factor of about 3
- IET 4 – underestimate of noble gases by up to a factor of about 16, halogens by up to a factor of about 7, and solids by a factor of up to about 2
- IET 10 – underestimate of total radionuclide releases by up to a factor of about 7

These concerns were cited in the SC&A report, *A Critical Review of Source Term for Select Initial Engine Tests Associated with the Aircraft Nuclear Propulsion Program at INEL* (SC&A 2003, -g. 2-24), which states the following:

> *The HDE Task Group acknowledged the absence of available raw effluent data as well as the deficiencies/limitations of summary data contained in the report by Thornton et al. (1962b). The HDE, therefore, modeled release estimates that were principally based on historical operating records and photographic evidence, which characterized the extent of fuel damage to the HTRE No. 1 reactor core. … Embedded in the HDE model of radioactive releases are several assumptions that potentially may have underestimated the true release quantities of fission products. Identified below are four key model parameters whose values may have differed significantly from those assumed by the HDE.*

5.1.1.2.2 Other Observations

Similar to the problem with routine releases, NIOSH should list the episodic airborne release activities for each INL facility used in the Occupational Environmental Dose TBD. Uncertainties associated with the release activities should also be provided. This data would be helpful for the dose reconstructors to assess whether the worker intakes are applicable to their claim. An example may be given showing how the worker exposure could be calculated using the release activities, uncertainty values, and weighting factors.

5.1.1.3 Direct Gamma Exposure

For direct gamma exposures to environmental releases, NIOSH used facility fence-line TLD measurement data from Environmental Monitoring Data Reports (EMDRs) between 1972 and 1983. The environmental gamma dose values (in the unit of mR) are presented for 11 facilities in environmental TBD Table 4-13:

> *Monitoring locations, established for the 1950s and 1960s, are shown on Figure 4. As shown on this map, the 118 area film badge locations completely surround the IET testing area and effectively monitor gamma/beta releases from other facilities at the INEEL. Results of film badge monitoring for the 4th quarter of 1960 from 340 badges, pulled on a monthly basis, showed the average gamma reading to be <20 mrem/month and the beta to be <10 mrem/month. The maximum gamma was 40 mrem/month and the maximum beta was 20 mrem/month, but locations for the maximum readings are unknown. Film numbers and environmental readings are provided in Table 7-1.* (Peterson 2004s, pg. 30)
>
> *It should be mentioned that the above table gives data principally for off-site locations. However, especially during the early years, the data do include on-site locations. These locations were primarily along the highways; at the north end of the Site, along highways 22, 28, and 33, and at the south end of the Site, along 20, 26, and along Lincoln Boulevard, which runs north-south through the Site. To illustrate, the H&S Division 1959 Annual Report (AEC 1960) briefly summarizes the onsite gamma monitoring program (pg. 120):* (Peterson 2004s, pp. 30 and 32).
>
>> *Film badges were located at 30 stations throughout the NRTS as a means of area monitoring for external radiation. Fig. LIII shows the average monthly dose in millirem due to gamma radiation. Estimates of external doses downwind of ICPP as the result of the criticality incident of October 16, 1959 were determined by this program.*
>
> *A 1970 environmental report (AEC 1970, pg. 3) states:* (Peterson 2004s, pg. 33).
>
>> *Only the TLD's located on the highways passing through the NRTS boundaries received exposures statistically above the background level. The highest exposure was 26 mrem for the six month period from*

> *November 1969 through April 1970. Trucks carrying radioactive shipments on the NRTS highways are the likely cause of this exposure, which is 15% of the standard.*
>
> *For the period described, 69 TLDs at 69 locations were placed, retrieved, and read; 35 along highway 20 at the south end of the Site, 18 along highway 88 and 5 along highway 28 which are both at the north end of the Site, and 11 along highway 22 which runs north-south along the western lower half of the site. All TLDs, except those along highway 20, averaged 20.9 mrem. Those along highway 20, the highway between CFA and Idaho Falls, Idaho, averaged 26 mrem. Since most radioactive shipments would come through Idaho Falls to CFA, the quoted statement about radioactive shipments is reasonable.*

In addition to the exposure readings described above, TLDs were placed around the major facilities at the NRTS and were used to measure the radiation exposure between 5/1 and 11/1/70. For example, 8 TLDs were placed around the EBOR-LPT facility with the highest reading of 19 mrem above background and the average being 12.9 mrem. Around the burial ground there were 18 TLDs with a highest reading of 760 mrem above background and an average of 257 mrem. At CFA, 4 TLDs averaged 16.8 mrem above background with the highest being 20 mrem. At the TRA in 1970 there were 15 locations shown on Figure 6. At badge location #8 the badge reading was lost, but the other 14 readings averaged 74.9 mrem. The highest reading was for location #10 (300 mrem), which is on the perimeter of the "north storage area" where contaminated or mildly activated items were stored. TLD locations #1 through #5 are along Monroe Blvd. where radioactive shipments come into and leave TRA and average 87.8 mrem. Of these 5 TLDs, #3 and #4 are in one of two of the predominant wind directions and are most probably influenced by the effluent of the ETR and MTR stacks; they read 110 and 130 mrem, respectively. Location #13 is practically adjacent to the ATR stack and reads 31 mrem above background. Badge #15, located on the fence southeast of the ATR reactor building, is at a location that would 'see' radioactive shipments coming into and leaving the TRA. Also, as radioactive shipment drivers are processing in and out of TRA, their shipments, sitting outside the guard gate, are influencing the exposure of badge #15. Also, the gamma irradiation facility is located just south of the TRA Guard Post #1, where the TRA badge rack is located. Badges #12 and #14 have the lowest readings, 16 and 18 mrem, respectively, and are probably most representative of the higher exposures for personnel within the TRA not performing "radiation worker" work.

Between the latter part of 1970 and the latter part of 1972, facility fence monitoring and facility locations had been established. Beginning in 1972 facility fence TLD measurements, made on a 6-month basis, are available in the Environmental Monitoring Data Reports (EMDRs) that were not made "public" but were sent to the contractor H&S managers. The data retrieved from these reports from 1970 to 1983 are tabulated in Table 7-2. For the years 1984

> *through 1992 facility fence TLD measurements could not be located, but for 1993 and beyond, facility fence TLD measurements are again included in the INEL/INEEL EMRs. Background TLD measurements, corresponding with the facility fence TLD measuring periods, were also recorded in the respective EMR. All of the facility fence TLD data located is presented in Table 7-2.* (Peterson 2004s, pg. 33)
>
> *As shown in the table, all data for 1984 through 1992, is highlighted … to indicate data that had to be added by extrapolation. For all facilities except TRA and ICPP, extrapolated data is the average of TLD data for that facility from 1993 through 1999, "0s" excluded. For example, the extrapolated values for TAN-TSF is the average of four TLD-Bkgd values, 18, 4, 14, and 3, i.e., 39/4=10. For TRA and ICPP there is a downward trend in the readings with time. Therefore, for these two facilities, the bottom four values (1989 through 1992) are the averages of the 7 values below and the top five values (1984 through 1988) are the averages of the seven numbers directly above. Background values are provided in the last column of the table.* (Peterson 2004s, pg. 34)

First, NIOSH believes the fence-line TLDs adequately measured worker doses from direct beta/gamma radiation from the facility and also gaseous effluents released from the facility or from adjacent facilities (TBD, pg. 22). This is only valid, however, for the portion concerning "gaseous effluents released from adjacent facilities." For "direct beta/gamma radiation from the facility" and for "gaseous effluents released from the facility," the TLD measurements are not representative of what a worker would receive while working at or around the facility within the fence-line boundary. In fact, the TLD measurements would be too low to be representative since the fence-line TLDs were situated too far away from the locations of the workers and the source terms. Second, the TBD indicates that there was general contamination of the surrounding facility ground due to facility operations and release depositions. However, the TBD does not provide any data or instruction for dose reconstructions to account for this missed dose that would not be picked up by the fence-line TLDs. Third, NIOSH believes that it is unnecessary for the dose reconstructors to calculate missed dose due to facility air immersion from noble gas and halogen releases because they would have been measured by the fence-line TLDs. This assumption is not supported by any data. In addition, there are not enough fence-line TLDs to cover the facility ground to measure all potential air immersion releases.

5.1.1.3.1 <u>Validity of the Fence-Line TLD Direct Gamma Values</u>

Table 6-1 of the Occupational External Dose TBD presents the fence-line TLD direct gamma values for eleven facilities from 1952 to 2002. The values are calculated as mean values of several TLD measurements. Each of the TLD measurements has an uncertainty of one standard deviation. NIOSH should evaluate the resultant uncertainty associated with these values.

Most important, a more recent INL Environmental Monitoring Fact Sheet, *Direct Radiation 2004*, shows different direct gamma values. In particular, for TRA, it presents much higher values from 2000 to 2002: 126 mrem for 2000, 172 mrem for 2001, 133 mrem for 2002, as compared to 40, 27, and 34, respectively, from Table 6-1. These are significant differences. NIOSH should re-examine the direct gamma values, as well as the background values used in the TBD.

5.1.1.3.2 Weakness of Fence-Line Monitoring Systems

INL uses fence-line environmental TLD systems to monitor environmental releases from routine operational releases and episodic releases. These TLD systems are commonly used in all other DOE facilities and commercial plants, and, in general, are accurate and easy to use. However, they are intended for providing data for evaluating potential exposures to offsite populations around the INL area; they are not located or intended for providing exposure data for onsite workers. Therefore, the issue is whether these fence-line TLDs are measuring the environmental doses to personnel working at or around the facility where the release is originating.

According to Table A1 of RAC 2002, the stack heights of various facilities at INL range from 9.1 m (ARA) to 76.2 m (ATR and ICPP, respectively). With lower stacks, the possibility of onsite plumes during routine or episodic releases is higher. The worker exposures would be increased depending on the type and quantity of releases. NIOSH should evaluate this issue to determine the worker exposure to onsite plumes, especially since the fence-line TLDs would not be capable of identifying the doses to personnel who were working in the path of these ground-level plumes within or around the facility. Mostly likely, the radioactive particulates of these ground level plumes would settle partially on the ground of the originating facility. Therefore, it is also necessary to evaluate the worker exposures to the resuspension of these ground depositions of the ground-level plumes and their accumulation over the operating years of the facility. For example:

(1) For a ground-level radioactive halogen plume emitted at the facility where a worker was located, the fence-line TLD data would not be representative of the worker’s immersion doses.

(2) As a facility ages, radioactive sources tend to accumulate around it, causing the general facility background to increase with time. The fence-line TLD data would not be representative of the actual direct gamma doses to workers working within the fence-line boundary because the workers were closer to the sources than the TLDs.

(3) During the site interviews, INL management personnel indicated that they have soil sampling data, site survey reports, and facility baseline studies. NIOSH could develop a facility background dose rate within the fence-line boundary for each facility.

(4) The INL environmental monitoring and surveillance program did not adequately monitor, evaluate, and report airborne effluents and ambient environmental conditions at INL (DOE-HQ 1991). The DOE-HQ Tiger Team observed deficiencies in the field sample controls, calibration and use of environmental monitoring equipment, and the use and storage of calibration standards. These deficiencies jeopardize the long-term reliability and integrity of the measurement process, and the validity of the monitoring data.

NIOSH should evaluate the representativeness of the fence-line TLD data for the onsite workers by comparing them to doses calculated from facility-specific on-ground radiation survey data, soil sampling data, and air sampling data.

5.1.1.4 Other Observations

5.1.1.4.1 Soil Sampling Data

NIOSH did not evaluate soil-sampling data at different facilities to determine potential worker intakes from resuspension of radioactive materials deposited on facility grounds and fugitive emissions from radiologically contaminated soil piles.

(1) As INL facilities age, cumulative depositions from effluent releases (both operational and episodic) on the grounds of each facility increase. Personnel working outside the facility within the fence-line boundary would have intakes from inhaling resuspended radioactive particulates. During the site interviews, INL management indicated that there are soil sampling data, aerial survey reports, site survey reports, and CERCLA reports that would provide such information.

(2) Radiologically contaminated soil piles were observed at the ICPP, PBF, and ARA facilities by the DOE-HQ inspection team. These piles were enclosed by a rope barrier to prevent personnel from entering the area, but no mechanism was used to prevent windblown soil, as cited in a 1991 DOE Tiger Team Report (Vol. 1, pg. 3-69).

(3) Dry evaporation, warm waste, and percolation ponds with radiologically contaminated soil could release fugitive emission, as cited in a 1991 DOE Tiger Team Report:

> *The evaporation pond at the PBF was found to be dry during the inspection. It was noted that low-level radiologically-contaminated cooling tower water is released to the pond approximately twice a year. The pond would be dry for much of the year which may allow for wind blown contaminated soil being emitted from the pond. Other locations of dry ponds containing contaminated soil include the three warm ponds at the TRA, the percolation pond at ICPP, and the ponds at ARA. .*
> (Vol. 1, pg. 3-70)

According to Table A1 of RAC 2002, there are other ponds that may be major release points to the environment, including the following:

a. ANL-W – 3 seepage ponds, leaching pits, and sanitary lagoon
b. EBR-II – 1 industrial pond
c. ARA – 2 surface depressions (1/3 acre and ½ acre)
d. CFA – sewage plant tile drain field
e. ICPP – 1 percolation pond and sewage plant to tile field
f. LOFT – 1 pond
g. NRF – 1 pond, 2 sewage ponds, and waste ditch
h. PBF – 1 evaporation pond

i. TAN – 1 pond
j. TRA – 2 ponds, 1 chemical waste pond and 1 leaching pond

(4) Resuspension of radiologically contaminated soils in identified surface soil contamination at ICPP and other facilities due to wind blowing, personnel tracking, or vehicular activities, may cause significant worker intakes (Vol. 1, pg. 268).

(5) Adequate protection for workers and the environment against the spread of radioactive contamination was not provided for known and potentially radiologically contaminated materials (or equipment) stored outdoors at INL facilities. Outdoor radioactive waste storage areas existed at TRA, PBF, RWMC, WERF, and CFA (Vol. 1, pp. 3-269, 3-271 and 3-273).

(6) The major contributor to dose at INL is radioactive gaseous releases. The dose fraction resulting from radioactive particulate releases during routine operations may not be accurate. Therefore, the calculated dose to workers may not be as accurate as could be achieved. Dose calculated as a result of unplanned releases may not be accurate also. This is cited as a finding in the 1991 DOE Tiger Team Report (Vol. 1, pg. 236).

(7) Besides the contaminated soil piles, dry contaminated ponds, and contaminated equipment stored outdoors, several types of spills occurred at INL facilities that could be sources of resuspended radioactive material for worker inhalation exposure. During the interviews, site experts indicated that they had been personally involved in many spills of radioactive materials and liquids during transportation, transfer of materials, and maintenance works.

5.1.1.4.2 Multiplying Factors

The suggested multiplying factors for intakes given in the Occupational Environmental Dose TBD may not be claimant favorable for the following reasons:

(1) The multiplying factor of two for intake values due to operational releases is based on a suggestion in the INELHDE Report. The authors of the INELHDE, however, further suggested that the uncertainty could be higher when the annual normalized ground-level concentration values are applied to the operational releases.[4]

(2) The multiplying factor of 3 for intake values due to episodic releases was based on suggestion of the authors of the INELHDE Report. SC&A's review of the INELHDE in 2003 found the airborne releases associated with several of the Initial Engine Tests (IETs 3, 4, and 10) of the Aircraft Nuclear Propulsion (ANP) Program were likely to have been underestimated as follows; (1) IET 3 – underestimate of total radionuclide release by up to a factor of about 3, (2) IET 4 – underestimate of noble gases by up to a factor of about

[4] It should be noted that two of the authors of the INELHDE Report were also authors of the NIOSH INL Site Profile Report.

16, halogens by up to a factor of about 7, and solids by a factor of up to about 2, and (3) IET 10 – underestimate of total radionuclide releases by up to a factor of about 7.

.

(3) NIOSH performed an analysis to reduce the number of radionuclides from 56 to 9 for the operational releases. The 9 radionuclides retained contribute about 95% of the inhalation doses to the workers. However, it is not explained why this reduction is necessary, since a spreadsheet can readily calculate the inhalation doses for 56 radionuclides almost as fast as for 9 radionuclides. There seems to be no attempt in the TBDs to make the final inhalation doses claimant favorable by adding a multiplying factor of 1.05 to compensate for the missing 5% doses. While this 5% may not be significant in most cases, there are many small percentage take-aways in the TBDs that could add up to a significant amount when the claimant dose is approaching the 50% POC threshold.

(4) The uncertainty of measurements made with film badges and TLDs could be as high as 100%. It is not clear whether the dose reconstructor should use a multiplying factor of 2 for direct gamma doses. In addition, NIOSH should address angular dependence of the gamma doses due to sensitivity of dosimeters used, as the procedures provided in OCAS-IG-001 Appendix B were found to be in error (SC&A 2005, pp. 143–145). It is essential for NIOSH to provide correction factors for dealing with angular dependence in the dose conversion factors used.

(5) It is indicated in the Environmental TBD that the 1952–1962 values for direct gamma doses at TAN-TSF, TAN-LOFT, and TAN-LPT could be low by a factor of 3 (TBD, pg. 25). It is not clear whether the dose reconstructor should use a multiplying factor of 3.

(6) For direct gamma doses after 1967, the uncertainty could be as high as 20%. It is not clear whether the dose reconstructor should use a multiplying factor of 1.2.

5.1.1.4.3 Breathing Rate

The NIOSH TBD (Section 4.2.1, pg. 10) uses an annual breathing rate of 2.4×10^3 m^3/yr rather than the 2.88×10^3 m^3/yr (1.6 m^3/hr for 1800 hr/yr) of ICRP 68 or the 8.0×10^3 m^3/yr (RAC2002, pg. 38) of the NCRP. The TBD assumption appears less claimant favorable than the ICRP or NCRP assumptions. However, NIOSH believes the chosen breathing rare is applicable to a worker exposed to the environment, who is not breathing hard from strenuous labor. NIOSH also believes that this choice of breathing rate was made uniformly for consistency for all the evaluated sites. This consistency choice is not claimant favorable for INL claimants. There is also no data supporting the claim that INL workers exposed to outside facility environments are not breathing hard from strenuous labor.[5]

[5] At the time of the preparation of this report, SC&A was informed that NIOSH has addressed the issue as part of the latest revision of the Bethlehem Site Profile Report.

5.1.1.4.4 Naval Reactor Facility Data

Although exposures to workers at the Naval Reactor Facility (NRF) located on the INL site are outside the scope of the dose reconstruction project, operations at the NRF have the potential of exposing personnel located outside the facility. Release data from the Naval Reactor Facility was deemed classified until recently. The INELHDE Report assumed values for environmental release from this facility (Attachment 3) and identified four NRF releases (1955, 1975, 1976, 1977) (DOE 1991a, pp. A-52 and A-55). In 2004, NRF provided environmental release and emission information to the INL site. NIOSH did request NRF environmental release documents from NRF more than 1 year ago. These documents were redacted and ready to be picked up by Intrepid, a subcontractor to ORAU, the NIOSH dose reconstruction contractor. NIOSH has since indicated no further interest in this data. SC&A believes that NIOSH should review this newly released data to improve estimation of the intake values and improve understanding of the origin of personnel exposure from NRF.

5.1.1.4.5 Background Dose Subtraction

It is confusing to compare INL fence-line direct gamma values provided in Table 4-13 of the environmental TBD with values given in Table 7-2 of the supplement to the TBD (Peterson 2004s). The background values in Table 4-13, which are supposed to be subtracted out from the TLD values, are higher from 1970 to 1983; NIOSH does not explain of the differences. To achieve claimant-favorable conditions, NIOSH should use the lower background values. In addition, a more recent INL Environmental Monitoring Fact Sheet, *Direct Radiation 2004*, shows some different direct gamma background values for 2001, 2002, and 2003.

5.1.1.4.6 Releases of Uranium and Its Daughters

The Occupational Environmental Dose TBD indicates that an analysis was performed to reduce the number of radionuclides from 56 to 9 for the operational releases. Actually, the TBD reduced the number of radionuclides from a total of 71 to 9 listed in the INELHDE for the operational releases. It also indicates that the dose contributions from these 9 radionuclides would represent 95% of the inhalation dose. Similarly, the TBD reduces the number of radionuclides from a total of 52 to 15 for criticality events, 9 for special tests, and 7 for IETs. However, NIOSH does not provide any compensations or adjustments to the lost 5% corresponding to operational releases or whatever amount corresponding to the episodic releases.

As a result of these reductions in the radionuclides considered, uranium and its decay daughters are not included in the worker intake value tables. The concern may be less significant if we consider only the dose impact to workers at a facility distant from the (release) source facility. However, as already pointed out in the above discussions, the workers would receive considerably higher doses from inhaling released radioactive material in a ground-level moving plume or resuspended contaminated material from ground deposition at the facility where the workers were located. Inhalation, ingestion, or oro-nasal intake of uranium and its daughters might then be a significant issue. NIOSH should re-examine the potential missed worker dose due to intake of uranium and its daughters.

5.1.2 Occupational Internal Dosimetry Issues

The NIOSH INL occupational internal dosimetry TBD reflects an extensive and thorough review of selected existing documents related to INL internal dosimetry programs, bioassay records, and reports. NIOSH interviewed INL Dosimetry Department staff and requested individual dosimetry records and documentation. NIOSH also interviewed some workers and retirees as part of an outreach program after the TBD was done. However, NIOSH (apparently) did not look at or use facility-specific field logs, RWPs, PEQs, facility survey reports, incident reports, occurrence reports, contamination reports, and ALARA records.

The authors of the Occupational Internal Dose TBD display confidence in the past and current INL radiological protection programs and its implementation, the accuracy of the internal dosimetry programs including whole-body counting and bioassay program (both routine and event specific), and the dosimetry record-keeping systems. During the site interviews, the INL Dosimetry staff showed similar confidence:

> *Except for several criticalities, there has been no significant radiological control incident for more than 55 years of operations. The site records anomalies, personnel contamination, skin exposure, and over-exposure incidents on a Personnel Exposure Questionnaire (PEQ) form. For example, a lost dosimeter by a worker would trigger a PEQ process. Anyone onsite can issue a PEQ, but they are usually generated from the Radiological Control Organization. In the case of skin exposure evaluations, the PEQ will be placed in the dosimetry record. Other PEQs are stored together and are not placed in the dosimetry file. If there was a dose adjustment as a result of a PEQ evaluation, this would be noted in the electronic dosimetry database. The PEQ itself is not provided to NIOSH as a part of the claimant packet.*
>
> *Personnel contamination reports are maintained as a field record. Only the exposure assessment is maintained in the dosimetry record. This includes primarily dosimetry information such as organ doses, radioisotopes of concern, total doses for the events, and bioassay results.*

It is important to point out here that many records and data pertinent to the worker claims are not included historically in the workers' files. These potentially missing records and data include personnel contamination reports, PEQs, RWPs, incident reports, and occurrence reports. For example, in the DOE record of some workers, the only indication of an incident is a brief comment on the bioassay report. No PEQ, incident report, or personnel contamination record are available to ascertain the details of the incident that may affect the dose reconstructions; this is apparent, for example, in two case reports that SC&A examined.[6,7]

Based on historical internal exposure source terms and known exposure incidents (both in-vivo and in-vitro records) at 13 specific INL facilities, the TBD identifies radionuclides of concern for

[6] Case No. 11654 from Idaho National Laboratory, Dose Reconstruction in progress.
[7] Case No. 10689 from Idaho National Laboratory, Dose Reconstruction Approval Date February 26, 2004.

key INL facilities and develops potential inhalation intakes for workers. Among these inhalation intakes, NIOSH recommends a set of defaults for specific radionuclides (such as Cs-137, Sr-90, Ce-144, Pu-238, or Pu-239) with weighting factors for the missed dose calculations based on urine, gross beta, in-vivo, and other bioassay data. The dose weighting factors are intended to account for missed inhalation dose from the other radionuclides. However, these weighting factors may not be claimant favorable, because there are no uncertainty evaluations for the missed doses and their weighting factors.

NIOSH's missed dose defaults are developed for four time periods; (1) start up through 1960, (2) 1961–1970, (3) 1971–1980, and (4) 1981 to present. For the first two time periods, NIOSH recommends defaults for two components: (1) workers at all INL facilities; and (2) workers at TRA. For the third time period, NIOSH recommends defaults for five components; (1) workers at all INL facilities, (2) workers at INTEC and unknown locations, (3) workers at ANL-W, (4) workers at other locations, and (5) workers at TRA. For the last time period, NIOSH recommends defaults for four components; (1) workers at all INL facilities except ANL-W, INTEC, and SMC, (2) workers at ANL-W, (3) workers at INTEC and unknown locations, and (4) workers at SMC. These missed dose defaults are inhalation doses only. There were no considerations given for potential ingestion or oro-nasal doses for uptakes due to unplanned events or specific exposure incidents.

The missed internal dose defaults are presented in Table 5.7-1, and are to be applied to dose reconstructions according to the following condition:

> *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* (pg. 8).

NIOSH's key assumption for dose reconstruction of missed dose is as follows:

> *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.* (TBD, pg. 37)

Another key assumption that NIOSH makes in the TBD is that claimant files are all complete with all documented incidents and internal doses included. NIOSH believes that the inclusion of these outlined missed doses would fill any gap in the information for undocumented missed worker internal doses.

In addition, the TBD specifies a guideline for dose reconstruction:

> *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 accident, it should be assumed that the individual was not involved in the incident.* (TBD, pg. 8)

It is ironic that the above guideline dictates how the completeness or the incompleteness of the incident file or the individual's dosimetry file would become conclusive evidence supporting that an individual was not involved in the accident. But, at the end of the TBD, NIOSH amends its guideline somewhat to read the following:

> *Thus the probability that a worker received 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 personal 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.* (TBD, pg. 39)

The following enumerates and comments on some of the pertinent information NIOSH has included in the internal dose TBD:

(1) NIOSH evaluates recorded internal doses for 1992–2000 in Table 5.1.4-1. This table also provides radionuclides of concern at the major INL facilities in recent years, but it is quite inadequate for dose reconstructions, with almost 42 years of missing internal dose data from 1949 to 1991. It also does not provide radionuclides of concern for all INL facilities. Furthermore, there are no instructions for dose reconstructors on how to use these internal dose data.

(2) NIOSH summarizes the primary radionuclides of concern at INL in Table 5.1.4-2. This table also lists the default solubility class and identifies the preferred bioassay techniques for the radionuclides. The TBD indicates that the default assumption of M or S should be used, based on the most claimant-favorable result to the organ in question.

(3) NIOSH summarizes the history of internal dosimetry efforts at INL and concludes the following:

> *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.* (TBD, pg. 8)
>
> *Most internal doses have been identified following an incident rather than as a result of routine bioassay measurements.* (TBD, pg. 9)
>
> *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.* (TBD, pg. 12)

These conclusions lead to NIOSH's missed internal dose approach. However, they are consistent with the findings of the DOE-HQ Tiger Team and DNFSB audits, as well as the information provided by the site experts interviewed.

(4) NIOSH believes that 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" (TBD, pg. 12). However, this conclusion is not consistent with the information provided by the site experts interviewed.

(5) NIOSH believes that the internal dose records at INL are adequate and effective to support the dose reconstruction process. This is again not consistent with the findings of the DOE-HQ Tiger Team and DNFSB audits, as well as the information provided by the site experts interviewed.

(6) NIOSH summarizes the routine bioassay history at the INL site in Table 5.2.2-1, which provides the type of bioassay (in-vitro, in-vivo, urine, or fecal), frequency, group of worker sampled, investigating level, and sources of information. It covers the operating years of INL from 1953 to 2001. The table shows that individual data sheets were reviewed to determine the bioassay programs between 1953 and 1981. After 1981, several technical papers were used as the sources for bioassay program information, and it is explained in the TBD why individual data sheets were not reviewed. It is also interesting to point out that the RCIMS database was not referenced or used in any way in this table.

(7) NIOSH believes the INL internal dose information, including the calculated internal doses as well as the in-vitro and in-vivo individual bioassay results, is in full compliance with Federal regulations. NIOSH believes "all (negative as well as positive) bioassay data were recorded in the individual dosimetry files" (pg. 15). This presumption leads to NIOSH's approach in this TBD, but is inconsistent with the findings of the DOE-HQ Tiger Team and DNFSB audits, as well as the information provided by the site experts interviewed.

(8) The TBD indicates that, "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 level 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" (TBD, pp. 15–16). NIOSH believes and concludes here that the air monitoring programs, bioassay sampling programs, and worker exposure record-keeping programs at INL are consistently effective and complete. This conclusion again is inconsistent with the findings of the DOE-HQ Tiger Team and DNFSB audits, as well as the information provided by the site experts interviewed.

(9) NIOSH lists the possible internal dose assessment information that may be in claimant files after 1989 in Table 5.2.4-1, and before 1989 in Table 5.2.4-2. There are many differences in the details of the information provided between the two periods. The TBD also provides other analytical information in Table 5.2.4-3 and area coding used in the claimant dose files in Table 5.2.4-4. In Table 5.2.4-5, NIOSH lists the derived investigation levels (µCi) in 1977 for 15 primary radionuclides of concern at INL for acute exposures, which represent one-tenth of the then quarterly radiation standard. However, the TBD does not provide guidance on how to fill the data gap if any information essential to the dose reconstruction is missing from the claimant's files.

(10) NIOSH presents a history of surface contamination control and MDAs used at INL, including an effective air monitoring program with the use of CAMs, gamma counting of bioassay samples, urinalysis, fecal sample analysis, and whole-body counting. Table 5.3-1 of the TBD lists the control levels and detection techniques for surface contamination and corresponding MDAs for INL between 1952 to present. However, NIOSH states the following:

> *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.* (TBD, pg. 18)

NIOSH should perform uncertainty analyses on the contamination control equipment and analytical capabilities at INL. The uncertainties due to the sensitivities of the equipment and practices are essential for dose reconstruction, especially for the early days of the lab when the equipment was not as accurate and the program practices were less rigorous.

(11) NIOSH believes "workers were asked to submit to bioassay whenever they were in an area in which a CAM alarmed" (TBD, pg. 19). NIOSH should review site audit and inspection reports by DOE-HQ (Tiger Team), DOE-ID, and DNFSB. A 1991 Tiger Team report found the set point for CAM alarms was faulty. Site experts interviewed indicated that the practices at some INL facilities frequently ignored CAM alarms.

(12) NIOSH presents some urinalysis results for 1959, 1960, and 1961 in Table 5.3.2-1 of the TBD, and summarizes the numbers of urinalyses performed and the highest results for 11 primary radionuclides of concern, as well as gross beta and gamma exposures. These urinalyses were limited to data from 1959, 1960 and 1961. NIOSH should provide more information for post-1961 years that constitutes almost 45 years of site operations.

(13) Instead of evaluating the analytical sensitivities of the INL bioassay equipment and adequacy of procedures, NIOSH applies environmental sample analysis detection limits and sensitivities for water (alpha, beta, and tritium) and milk (I-131, Sr-90) to INL special and routine bioassay sample analyses. The TBD lists the detection limits applicable to environmental sample analyses from 1953 to 1965 in Table 5.3.2-2. It is not clear how dose reconstructors should use these data; should they also apply them to

post-1965 analyses? NIOSH should provide equipment-specific and method-specific information. The broad application approach taken lacks technical validity, as well as data support. Not only is it limited to the period of 1953 to 1965, but also equipment and practices changed significantly after 1965. NIOSH should perform uncertainty analyses on the bioassay equipment and analytical capabilities at INL. The uncertainties due to the sensitivities of the equipment and analytical methods are essential for dose reconstructions, especially for the early days of the site when the equipment was not as accurate and the analytical methods were less rigorous than they became in later years.

(14) In TBD Table 5.5-1, NIOSH presents a single set of whole-body counting summary statistics for INL workers in 1963, showing the number of times reported, number of individuals, and maximum activity (µCi) for 24 radionuclides of concern at INL. The TBD states that, "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" (TBD, pg. 23). Since the table only shows data from 1 year of operations, NIOSH should compile data for all post-1963 whole-body counting statistics. In addition, NIOSH has stated repeatedly in the TBD that the worker exposures were comparatively worse in the early years, due to the nature of the operations and the improvement in instrumentation, program standards, and methodologies. Therefore, the maximum activity detected in 1963 may not represent an upper bound for exposures occurring in earlier years.

(15) NIOSH evaluated 13 specific facilities to identify potential sources of internal exposures based on radionuclide activities from facility operations.

- Test Area North (TAN)
- Specific Manufacturing Capability Project (SMC)
- Idaho Chemical Processing Plant (ICPP, now INTEC)
- High-Enriched Spent Fuel Storage (CPP-603)
- High-Level Waste Storage Tank Farm
- High-Level Waste Calcination
- Process Analytical Facilities
- Spent Fuel Processing
- Argonne National Laboratories – West (ANL-W)
- Radioactive Waste Management Complex (RWMC)
- Waste Reduction Operations Complex (WROC)
- Test Reactor Area (TRA) Reactors
- Test Reactor Area Laboratories

Similar to the Occupational External Dose TBD, NIOSH should provide information on airborne concentrations and contamination levels in these facilities. It would be even better if NIOSH could develop a list of high-risk (dose) jobs in each of these facilities.

(16) NIOSH indicates that detailed particle size analyses were performed for SMC operations. The internal dose TBD states, "an AMAD of 2.4 µm is appropriate for typical SMC operations" (pg. 29), but NIOSH does not provide support information for this value in

the TBD. In addition, particle size analyses are not provided for the other 12 facilities. This omission could significantly impact the missed dose calculations for dose reconstructions. It is, otherwise, interesting to note that the Occupational Environmental Dose TBD indicates that, since there was no particle size analysis performed at the INL facilities, default values should be used for worker dose reconstructions. The site experts interviewed indicated that a particle size study was completed for the calciner with a cascade impactor at ICPP by the field organization. Particle sizes determined range from 0.3 to 1 µm. There is no other particle size study recently. The default particle size for internal dose calculations is 1 µm AMAD.

(17) NIOSH lists the depleted uranium (DU) mass and activity ratios for SMC in Table 5.6.1-1. The TBD also indicates that a baseline uranium background concentration study was performed to represent "nonoccupational elimination of the SMC worker population" (TBD, pg. 29). This study used urine samples from SMC non-radiation workers in 1987, 1994, and 1998 to determine the nonoccupational component of uranium excretion for SMC radiation workers. The value of 0.16 µg/L was calculated and subtracted from each worker urine sample result prior to assessment of the occupational internal dose. This approach is not statistically valid, due to the nature of the selected nonoccupational worker group, collocated with the radiation workers. There was an obvious possibility that these non-radiation workers may have intakes of uranium from the surrounding contaminated environment. NIOSH should evaluate the validity of the value of 0.16 µg/L used as nonoccupational subtraction from urine results. More importantly, the background concentration should not be subtracted from the urine sample results. (See Section 5.1.2.8 for further discussion).

(18) NIOSH lists nine notable airborne incidents at ICPP in Table 5.6.2-1 of the TBD. The table provides the date of the incident, radionuclides released, internal dose results, and references. However, the TBD concludes that, "The primary internal dose experience at the ICPP resulted from accidental releases" (pg. 29). This conclusion seems to disagree with audit findings, and past and present worker experiences. NIOSH should re-evaluate the potentials of internal dose exposures at ICPP by examining high-risk (dose) jobs and field records.

(19) NIOSH indicates that for personnel who worked for extended time periods in the Building 603 storage facility at ICPP if, "specific bioassay analyses are either not available or insufficient, a claimant favorable default intake of 1000 DAC-hours per year should be assumed" (TBD, pg. 30). There is a tendency for the TBD authors to use average values for determining default values. In almost all (valid) cases of the dose reconstruction, the missed doses are in the high end of the dose spectrum. Therefore, NIOSH should provide both average and upper bound values for defaults to assist the dose reconstructors. For example, NIOSH could use the 95th percentile values for those workers in high-risk (dose) jobs.

In this case, the 1,000 DAC-hour is assumed in the TBD to be the average intake by a worker in Building 603, with Cs-137 and Sr-90/Y as the primary radionuclides of concern. This assumption is based on a 1 MPC level and 1,000 hours per year, according to NIOSH experts (see Attachment 1). This assumption would short-change some workers who may have worked in the building for more than 1,000 hours or in a very highly contaminated area, receiving the 1,000 DAC-hour in a short period of time. In addition, the default airborne activity of 2×10^{-9} µCi/cc is not justified by supporting data and uncertainty analyses. For example, a pipefitter who worked in Building 603 on a valve or piping replacement job could be working in a contaminated air zone with much higher concentration levels than this default airborne activity value. Therefore, this assumption does not appear to be claimant favorable.

(20) There were no potential internal exposure source term data identified for the High-Level Wastes area, the High-Level Waste Calcination area, and the Process Analytical Facilities at the ICPP. As the TBD and site experts interviewed indicate that the waste calciners (both WCF and NWCF) were among the worst contaminated facilities at INL, NIOSH should have identified the high-risk (dose) jobs and potential missed doses to personnel who worked in these facilities.

(21) NIOSH identifies three types of spent nuclear fuels (aluminum-clad, stainless steel-clad, and zirconium-clad) as the primary source terms for worker internal exposures at ICPP. The TBD further determines that the radionuclides of concern are the "most limiting radionuclides" for all the facilities at the INL site. (Eventually, the TBD uses these radionuclides as the default limiting radionuclides to calculate missed internal doses for INL workers in Table 5.7-1). The radiologically significant radionuclides associated with these fuels are listed in Table 5.6.2.5-1. The inhalation doses from inhaling radioactivity from these three types of fuel were calculated using ICRP 68 5 µm AMAD dose conversion factors. NIOSH selected four key radionuclides as the main dose contributors for potential internal exposures; Sr-90, Cs-137, Ce-144, and Pu-238/Pu-239 (interchanged among fuels). Three weighting factors were developed by ratios and used to account for the dose contributions from unused radionuclides:

> *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 Sr-90, Cs-137, Ce-144, and Pu-238 are included for aluminum and zirconium fuels. For stainless fuels, the Pu-238 is replaced by Pu-239. 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 5.6.2.5-1. This gives an equivalent to 100% of the dose from the radionuclides distribution for the three types of fuels.* (TBD, pg. 33)

Again, NIOSH's approach used here is based on the "average" percent inhalation doses a worker would receive due to the "average" relative activities of the identified significant radionuclides of concern for the ICPP processed fuels. It does not take into account of the upper bound inhalation doses.

(22) NIOSH indicates that there is one exception to the above spent fuel approach; the RaLa (Radioactive Lanthanum) process, in which large quantities of volatile radioactive iodines (i.e., I-131, I-132, and I-133) were released. However, the TBD does not provide instructions to dose reconstructors on how to evaluate missed doses for personnel working in the RaLa process or how to scale from the spent fuel data.

(23) NIOSH identified and evaluated radionuclides of concern at ANL-W, RWMC, and WROC. Analytically determined MDAs, bioassay data, and radioactivity concentrations are also provided in the TBD (Tables 5.6.3-1, 5.6.4-1, and 5.6.4-2). The TBD also states that, "a comprehensive radiation protection program is practiced, which includes extensive air monitoring, personnel contamination, and surface contamination surveillance" (TBD, pg. 34). Once again, the TBD shows a degree of confidence in the INL radiation protection programs that is not consistent with the findings of the DOE-HQ Tiger Team and DNFSB audits, as well as the information provided by the site experts interviewed.

(24) NIOSH concludes that, "For most of the history of the INEEL personnel dosimeters were issued to all workers at facilities handling radioactive material" (TBD, pg. 37). The TBD continues to show the sentiment that the authors have full confidence in the radiological protection policy and programs and internal exposure monitoring systems at INL. NIOSH believes, "the probability that a worker received a significant unmonitored internal intake of radioactive material is very low" (TBD, pg. 37).

5.1.2.1 Completeness and Quality of INL Internal Dosimetry Programs

The authors of the Occupational Internal Dose TBD display full confidence with the radiological protection programs, the internal dosimetry programs, and the dosimetry record-keeping systems at INL. For instance, NIOSH states the following in the TBD:

> *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 exposure has been preventive in nature.* (TBD, pg. 18)
>
> *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.* (TBD, pg. 18)

NIOSH also believes that all worker exposures were documented in each exposed workers' file. As a result, a dose reconstructor could perform the claimant dose reconstruction based on information kept in the claimant file. These conclusions are evident throughout the TBD and can be illustrated by a few excerpts:

Most internal doses have been identified following an incident rather than as a result of routine bioassay measurements. (TBD, pg. 9)

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. (TBD, pg. 8)

Each individual analytical result was documented and placed in individual exposure files regardless of the formal reporting requirements. (TBD, pg. 16)

However, information provided by past and present workers during SC&A's site interviews, as well as INL site inspection findings by the DOE-HQ Tiger Team in 1991 and by the Defense Nuclear Facilities Safety Board in 1994 and 1995, indicates that the INL site programs were deficient in many ways, despite what was claimed. These program deficiencies are serious and likely jeopardize the validity of data NIOSH used and its conclusions in the TBD. There are many examples of potentially important deficiencies, which could adversely affect the dose reconstruction process, cited by the DOE Tiger Team, as follows (DOE-HQ 1991):

(1) The Radiological, Environmental and Safety Laboratory (RESL) was in charge of all air sampling and bioassay analysis and internal dose assessment functions at INL, but RESL did not have an effective safety and health management system. The Tiger Team found 75 safety deficiencies in policy, policy implementation, and procedures. These deficiencies put all the analytical results RESL produced in question. For example (Vol. 2, pg. 4-26):

  a. The calibration of whole-body counter and teletherapy sources were not adequate (Vol. 2, pg. 4-467).

  b. RESL was found to be deficient in monitoring airborne radioactivity in laboratory spaces, in performing pre-employment baseline bioassay for new employees, in performing bioassay surveillance of individuals potentially exposed to airborne radioactivity, in performing fume hood testing, in performing quality assurance sample checks for routine urinalysis, and in performing analysis of urine samples for total uranium from INL radiation workers (Vol.2, pp. 4-613 and 4-614).

  c. RESL was deficient in ensuring accurate documentation of the individual radiation exposure, including routine survey information (instrument type and number and reading), wipe sample, counting, and air sample counting (Vol.2, pp. 4-623 and 4-624).

(2) The INL Internal Dosimetry Program was found to be deficient because compliance with DOE 5480.11 (radiation protection) could not be demonstrated (Vol. 2, pg. 4-181).

  a. For sample implementation of in-vivo and in-vitro bioassay programs, logs were not maintained to provide information such as purpose of bioassay, schedule of bioassay, date when bioassay was completed, data when the analytical results were received, analytical results, and date a dose assessment was completed.

b. For workers, required in-vivo and in-vitro bioassay monitoring was not completed as scheduled. This was primarily due to the cumbersome process of dose assessments involving four different entities, namely RESL, INL Laboratories, RSRS, and ODU. In fact, RESL was not in compliance with DOE Orders.

(3) INL was deficient in providing adequate calibration equipment or facilities to perform tests for insuring reliability of existing radiation protection instrument response in 1991 (Vol. 2, pg. 4-183).

a. There were no defined criteria for acceptance testing of fixed radiation protection instruments.

b. Trending analysis of radiation protection instrument performance was not performed.

c. There was no program to identify problems and improve the response of radiation protection instrumentation to meet the needs of specific applications.

(4) Several deficiencies of the CAMs were identified at TAN, ATR, and other facilities. Their locations were not based on an evaluation of airflow patterns (Vol.2, pg. 4-185).

a. Some of the CAMs detecting beta-gamma radiation had a set point of approximately 8 DAC-hour instead of 1 DAC-hour (i.e., MDL). 8 DAC-hour was the minimum set point for alpha emitters but not for beta-gamma.

b. Some CAMs (e.g., TAN Warm Shop) were not in service. There were no two level alarm systems established for the CAMs.

(5) Maintenance of the INL occupational exposure record was transferred from the RESL in 1988. The quality of the historical occupational exposure records ranges from handwritten in pencil on plain paper to records legibly recorded on standard forms in ink or using computer-generated data. The records of individual occupational exposure histories are readily retrievable. The histories can be obtained from ODU (i.e., manual retrieval from record storage sites). Information prior to 1986 must be retrieved from a combination of microfiche, notebooks, and individual files. Information from 1986 forward is readily retrievable from computer record systems. Documented procedures for retrieval of occupational exposure histories are maintained. The results of in-vivo and in-vitro bioassay are maintained. However, there are certain limitations in the exposure records and the process of incorporating records in the dose reconstruction files. These limitations may hinder dose reconstructions and result in missed doses. Overall, the Tiger Team found the maintenance and retention of occupational exposure record was deficient at INL. For example (Vol. 2, pp. 4-190 to 4-192):

a. The results of bioassay are not included in an individual's file unless a dose assessment was made from a positive result.

b. The individual's file does not indicate that negative bioassay results are available.

c. Records of dose estimates from special studies and unusual exposures are not maintained in an individuals' occupational exposure history.

d. There was no audit mechanism to verify that previous occupational exposure histories, exposures of employees who travel to other sites and internal dose assessments are included in the files.

e. Reporting of occupational exposure to radiation workers was deficient in the areas of submitting annual report of radiation exposures, using the effective dose equivalents, and providing summary of annual, cumulative and committed effective dose equivalents to each radiation worker on an annual basis.

(6) INL was deficient in routine and accident personnel radiation dosimetry program at SMC (Vol. 2, pg. 4-611).

(7) The procedure for personnel entry into potentially airborne radioactivity areas when wearing respirators was deficient at INL because no air sample measurements were taken to assure that respirator protection factors are adequate (Vol. 2, pg. 4-612).

(8) The INL site was found to be deficient in testing and calibration of radiation measurement instruments including gas proportional counters, air sample counters, portable radiation survey and frisking instruments, radiation counters for wipes, special tool survey box instruments, and portal monitors (Vol.2, pp. 4-615 and 4-616).

(9) The INL radiological air sampling program was deficient in ensuring timely and representative airborne activity measurements in the workplace (Vol.2, pp. 4-617 and 4-618).

In addition to the many critical findings by the DOE-HQ Tiger Team, the DNFSB also generated a long list of deficiency findings on the INL radiological protection programs and internal dosimetry programs during their audits. Several examples of the DNFSB findings, which may impact dose reconstruction, follow:

(1) 7/29/1994 DNFSB finding: The staff noted an inconsistency among contractors in the requirements for respiratory protection, as well as some errors in procedures.

(2) 7/29/1994 DNFSB finding: An error was noted in EG&G Procedure 10.3, Airborne Radioactivity Monitoring, dated January 28, 1994, and Document Revision Request (DRR) dated February 14, 1994. These documents gave equations for computing High Alarm set points that are not correct.

(3) 7/29/1994 DNFSB finding: Air monitoring for alpha radioactivity in the building is accomplished by one continuous air monitor.

(4) 7/29/1994 DNFSB finding: Although several individuals are involved in operations in the high-level waste pit area where dose rates can reach several rem per hour, standard actions are not taken to control personnel exposure and preclude unnecessary exposure.

(5) 9/15/94 DNFSB finding: ICPP radiological control practices also remain deficient, including a general lack of concern with personnel and equipment crossing radiological control boundaries, inadequate frisk/swipe survey practices, etc.

(6) 1/17/95 DNFSB finding: Radiological control postings and field documentation on the tank farm were inconsistent and contained mistakes that were not identified and/or corrected by internal reviews.

(7) 1/17/95 DNFSB finding: Several protective clothing doffing areas do not have posted doffing instructions. In one location, a Radiation Area posting was changed by hand to read "High Radiation Area." No radiation levels were listed. Radiation levels were entered inconsistently on many postings throughout the project.

Given these deficiencies noted in the INL radiological protection and internal dosimetry programs, it is unlikely that the information and internal exposure records provided in the worker files are complete. It is also likely that many worker internal exposures associated with high-dose jobs were not monitored or documented. NIOSH's missed dose approach and the default values should be complete and representative of all missed dose scenarios for personnel working at different INL facilities. At a minimum, NIOSH should evaluate the programmatic uncertainties associated with the missed dose values presented in Table 5.7-1, so that the recommended missed dose values would be truly claimant favorable.

5.1.2.2 High-Risk (Dose) Jobs

The DOE Tiger Team and DNFSB findings, made in the 1991 to 1994 time frame, suggest that there may be potentially significant missed doses due to deficient radiological protection work practices at different INL facilities. It is likely that the work practices at INL before this period were even worse, due to less sensitive equipment, less protective policy, programs, and procedures, and also less frequent and rigorous worker training. Instead of merely using inhalation dose defaults for worker missed dose from generic facility operational source terms, NIOSH should develop a list of high-risk jobs for different categories of workers (such as pipefitter, operator, RCT, machinist, mechanic, electrician, maintenance, yardman, etc.) at each INL facility-based on bioassay data, air sampling data, area survey data, and RWP data. High risk, in this context, means the potential for high dose exposure. Some of these high-risk (dose) jobs have been identified by past and present workers during the SC&A site interviews (see Attachment 3 of this report).

It is generally agreed that the INTEC (previously ICPP) and its facilities are the most contaminated facilities at INL, with high potential for significant worker intake due to the fuel processing and reprocessing operations and practices. NISOH should evaluate separately the missed dose for workers who had worked at ICPP facilities.

5.1.2.3 Calibration of Internal Dosimetry Analytical and Monitoring Equipment

The TBD does not provide any information on the calibration procedures, sensitivities, and standards of the internal dosimetry analytical equipment and monitoring instrumentation, including thyroid counters, whole-body counters, laboratory instruments, and continuous air monitoring systems (CAMs). The 1991 DOE Tiger Team findings show the deficiencies in these areas. NIOSH should evaluate the uncertainties and impacts on the internal dose assessment results associated with the deficient calibration programs at INL.

5.1.2.4 Changes of Internal Dose Limits

The TBD describes the historical changes of internal exposure control and monitoring programs at INL and touches on some of the internal dose limit changes. Currently, the INL policy is that all individuals who have the potential to receive a dose exceeding 100 mrem shall require monitoring. In the past, the INL policy was less restrictive and protective. For example, during the site expert interviews, a worker indicated that he often had positive nose swipes of 100mrem/hr, but a bioassay or PEQ was not triggered. This type of inconsistent work practice was prevalent in the early years of the INL operation and may have led to significant missed dose to workers.

NIOSH should evaluate the impacts of these dose limit changes over the operating history of INL to see whether there were missed doses in the early years when the radiation protection policy was less protective and inconsistently implemented.

5.1.2.5 High-Fired Plutonium and Uranium Intakes

Some INL facilities contained high-fired uranium or plutonium oxide produced by heating the material to approximately 1,000ºC. High-fired oxide is more chemically stable than low-fired oxide, because the higher heat removes moisture and other impurities. The Rover facility at ICPP was used to reprocess graphite space reactor fuel, resulting in the formation of high-fired uranium oxide. It was identified as one of the 10 most significant highly enriched uranium safety concerns in the DOE complex. In addition, some high-fired plutonium oxides may have been shipped from Rocky Flats Plant and disposed of at INL. These oxides are commingled with radioactive wastes stored in drums and buried underground or above ground at the RWMC facility.

Plutomium-238 is primarily an emitter of alpha particles, which are easily stopped by the materials encasing it, although it also emits some gamma rays and neutrons. The high-fired Pu-238 ceramic does not easily dissolve nor does it easily break into very small particles. In the case of plutonium, a urine sample would not show small intakes within two days, so INL staff developed a procedure for detecting fired plutonium oxide in fecal samples. Similar to recycled uranium, NIOSH should evaluate the lung dose for intake of high-fired uranium and plutonium oxide particulates (alveolar deposition).

5.1.2.6 Skin and Facial Contamination

This TBD does not consider incidents with workers having skin contamination, facial contamination, and positive nasal swipes in the INL facilities. These kinds of problems would be compounded by the deficiencies in air sampling systems and ineffective respiratory protection programs, some of which have been discussed previously. Consequently, a bioassay may not be triggered for the workers. Guidance should be provided to a dose reconstructor to account for the missed dose due to the unaccounted uptake.

5.1.2.7 Breathing Rate

The NIOSH TBD (Section 4.2.1, pg. 10) uses an annual breathing rate of 2.4 x 10^3 m^3/yr rather than the 2.88 x 10^3 m^3/yr (1.6 m^3/hr for 1800 hr/yr) of ICRP 68 or the 8.0 x 10^3 m^3/yr (RAC2002, pg. 38) of the NCRP. The TBD assumption appears less claimant favorable than the ICRP or NCRP assumptions.

5.1.2.8 Non-Occupational Worker Elimination of DU Background at SMC

The last paragraph of Section 5.6.1.1 of the internal dose TBD (pg. 29) on the SMC project to produce DU armor for tanks discusses natural background uranium baseline for radiation personnel working at the SMC facility. This section states, "Urine samples submitted by SMC nonradiation worker in 1987, 1994, and 1998 were assumed to represent nonoccupational elimination of the SMC worker population." This study used urine sample results from nonradiation personnel working onsite at SMC in 1987, 1994, and 1998 to determine the nonoccupational component of uranium excretion for SMC radiation workers. The results ranged from 0.04 to 0.33 µg/L, with the average uranium concentration as 0.157 ± 0.109 µg/L at 1 sigma uncertainty. The value of 0.16 µg/L was used and subtracted from each worker urine sample result prior to assessment of occupational internal dose.

It is not sound to use a group of co-workers at the same facility as a reference, even though they are non-radiation workers. NIOSH should consider using non-INL individuals from the surrounding populations for the non-occupational elimination approach. The INL approach is not statistically conservative or technically valid due to the nature of the nonoccupational worker group used. The selected group of workers consisted of personnel collocated with the radiation workers. There is a high probability that these non-radiation workers had also been exposed to depleted uranium dusts or contaminated soils from the SMC grounds. NIOSH should, therefore, evaluate the validity of the value of 0.16 µg/L used as nonoccupational subtraction from urine results.

In the teleconference, the NIOSH technical experts explained, "The SMC program was run independently from the other INL programs and considered only its own people for establishing background. NIOSH considered that, but using only the SMC population to determine background is consistent with the approach taken elsewhere on the site and at other sites (e.g., Fernald)" (see Attachment 2). The idea of consistency with the approach taken elsewhere on the site and at other sites (e.g., Fernald) is understandable and should be encouraged. However, in this case, data validity is more important than consistency.

In the site interview, the INL Dosimetry Department staff indicated that they had attempted to use non-INL workers from the surrounding populations, but the individuals they had contacted did not agree to sign a liability waiver form. That caused the change of plan to use onsite non-radiation workers. NIOSH should consider using uranium background data from other studies performed for the Idaho Falls area to compare with this value.

A survey by the National Center for Health Statistics (NCHS) of CDC, *National Health and Nutrition Examination Survey 1999–2000*, shows the uranium concentration in urine for the U.S. population aged 20 and older ranging from 0.006 µg/L to 0.054 µg/L, with a geometric mean of 0.007 µg/L. This is much lower than the INL background worker elimination value of 0.16 µg/L. In addition, NIOSH may consider not subtracting contributions from environmental background from the urine monitoring results. This approach is also used at Y-12 (Eckerman 1999).

5.1.2.9 Unmonitored Workers

As for the unmonitored (sometimes unbadged) workers, such as secretaries, warehouse workers, drivers, guards, and construction workers, the potential missed doses would be from inhaling resuspended contaminated soils and ingesting contaminated materials while eating in a contaminated, previously considered uncontaminated, area (such as office and cafeteria). NIOSH should evaluate these potential missed doses.

5.1.2.10 Naval Reactor Facility Workers

Release data from the Naval Reactor Facility was deemed classified until recently. As the internal dose TBD indicates, "some workers' internal dose could have resulted from their support work at the NRF." NIOSH should evaluate the potential missed dose at the NRF for these workers.

5.1.2.11 Plutonium Monitoring

The TBD does not provide any historical information on the plutonium analysis methods used at INL. Table 5.4-1 indicates a Pu-239/240 MDA starting in 1964. Table 5.3.2-1 indicates that 18 Pu-239 urinalyses were performed in 1959. The site experts interviewed indicated that routine plutonium monitoring was not performed until the 1980s. There seems to be some confusion on this point. It is entirely possible that selective plutonium monitoring on workers was used at INL until 1980, but without this information, the dose reconstructors would not be able to assign missed internal dose due to plutonium intakes in the time period before 1980. NIOSH should provide information on plutonium monitoring in the TBD.

5.1.3 SL-1 Accident Dose Reconstructions

The TBD acknowledges that the highest sources of gamma and beta doses were the ICPP and SL-1. The criticality accident in the latter, which occurred at the end of 1960, caused many significant external exposures to workers involved in rescue and clean-up operations, but the INL TBDs treat this accident only sparsely. For instance, the Site Description TBD indicates the following:

> *The event released fission products (500,000 Ci in the building, and 1,100 Ci to the atmosphere) and created high-level radioactive contamination to 50 rad/hr around the ARA-II area. Initial recovery from the accident resulted in short term exposure exceeding 500 rad/hr to personnel in radiation fields. Extensive cleanup efforts followed, including complete dismantlement of the facility.* (pp. 49–50)

The Occupational Environmental Dose TBD states the following:

> *The amount of the release and the path that the cloud traveled from the reactor building was carefully monitored and well documented. All radiological doses to personnel involved in the rescue and cleanup of the reactor building were carefully controlled and documented. The SL-1 accident did not affect any other INEEL facility with the effluent of radioactive material.* (pg. 12)

The Occupational External Dose TBD states the following:

> *Experience following the SL-1 accident showed a wide variation of beta-to-gamma ratios and necessitated controlling both radiations rather than just the gamma. A set of as many as 18 badges could and in many cases was fastened on a belt around the worker to determine a beta:gamma ratio for each particular entry.* (pg. 16)

It is clear that the TBDs rely on the assumptions that (1) personnel dosimetry systems and dose assessment methodology following the accident were sufficient and accurate enough to calculate exposure doses to workers, and (2) the dosimetry record-keeping system was adequate to document dosimetry analysis and dose calculation results. However, these two assumptions are flawed, because dosimetry equipment of that era had limited capabilities to respond to an accident of such a large scale, and the dosimetry record-keeping system was still in its developmental phase. Additionally, there were close to 1,000 workers involved in that historical cleanup operation. It would be essential for NIOSH to identify high-risk (dose) jobs during the SL-1 rescue and cleanup process, and then provide specific claimant-favorable default dose values for each of these high-risk jobs. These high-risk jobs may include (1) fireman, (2) other first responder, (3) ambulance driver, (4) medical staff, (5) Hot Shop worker, (6) cleanup crewman, (7) machinist, (8) pipefitter, (9) crane operator, (10) RCT, and (11) guard. The missed dose could be composed of (1) missed beta dose, (2) missed gamma dose, and (3) missed internal dose. There may also be the potential of missed neutron dose for the rescue workers.

Susan Stacy's documentary book, *Proving the Principle* (Stacy 2001), devotes two chapters to describing the unfolding and the aftermath of the SL-1 incident. On page 147, it states the following:

> *The SL-1 was a mess. It hadn't been cleaned up at all. To clean it up, people had to make short trips inside and do limited tasks within a couple of minutes and then get out. Even though you suited up, those couple of minutes would expose you to*

> *your quarterly dose of radiation, and you couldn't go back in for three months or do any other work that could potentially expose you. Hundreds of people at GE, including those about to be transferred and many workers from other locations at the Site [and from Dugway], volunteered to take their quarterly radiation dose doing clean-up at the reactor. For many cleanup tasks, that was the only way of handling it." I don't remember anyone being particularly fearful of the risk. The time keepers were the HPs, who stood half-way down the stairway and banged on metal when someone's vacuum-cleaning stint was over. By November, the passage of time and removal of debris had reduced radiation levels.*

It seems that there are many discrepancies in getting to the real facts about the SL-1 incident and also the rescue operations and the aftermath cleanup activities, since this occurred almost 45 years ago. Over the years, there have been many complaints from INL workers and union workers about the lack of records and facts to assist in the dose reconstructions associated with this accident. The site expert, union worker, and retiree interviews at Idaho Falls raised similar complaints. The issue of "claimant favorable" for those claimants who participated in the rescue operation and/or in the aftermath cleanup activities must be addressed in a fair and reasonable way.

Proving the Principle continues to state the following:

> *Twenty-two of the people who had responded to the SL-1 alarm received radiation exposures in the range of 3 to 27 Roentgens total body exposure. Three of them received more than 25 R. The exposure guide that had been set up by IDO's prior emergency plan allowed rescue personnel a 100 R dose to save a life and 25 R to save valuable property.*

Obviously, INL was not prepared and organized in 1961 to deal with such a unique and deadly event. The IDO Report on the Nuclear Incident at the SL-1 Reactor (IDO 1962) states the following:

> *Following the incident, the routine service, of course, terminated; and it soon became apparent from the increased numbers of persons that were becoming involved with SL-1 rescue operations, etc., that a unique type of data processing system had to be instituted. The system would have to (1) maintain a radiation dosage record of those personnel brought in from other sites, (2) supply compiled information for appropriate officials and the person's permanent work location indicating the radiation dose received while in the SL-1 Area, and (3) provide a total radiation dose tabulation for all officials concerned in order to prevent an exposure in excess of guide values… In order to prevent persons who were involved not only in daily duties at the SL-1 Area, but also in daily duties at other NRTS Site Facilities from receiving excessive exposure, each individual's daily film badges, perhaps two or more, had to be processed and recorded. Next a report listing the individual's name, date and location of exposure, and amounts of beta and gamma radiation received had to be tabulated. This created a record retrieval problem which would have proven extremely difficult, if not impossible,*

> *without the central data processing system which maintains all radiation exposure records at the NRTS.* (pg. 39)
>
> *Following the incident, because personnel involved in the rescue operations were exposed to airborne radioactive materials, spot urine samples were collected from these people as soon as was possible. It was determined by gamma spectra analyses that the major portion of the radioactive material contained in the urine samples was iodine-131. Table 3.2 lists the sixteen persons who showed the highest urinary excretion of iodine-131 and the estimated thyroid dose that was calculated for each person…. In conjunction with the iodine-131 analysis, strontium-90 and cesium-137 analyses were performed. In all cases of the cesium-137 analyses, no activity greater than 15 milli-rem per year, infinity dose, was ascertained. Spot urine samples were collected from about 110 individuals at intervals during the first few days after the incident and were analyzed for strontium-90. Eight individuals whose urinary excretion of strontium-90 continued beyond the 10th day then submitted 24-hour urine samples for strontium 90 analyses. Table 3.3 lists the three persons who showed the highest strontium 90 dose and the estimated strontium- 90 dose to the bone critical organ…. Six persons involved with the SL-1 Incident reported for whole body counting during the first week following the incident. Identification of iodine-131, barium-lanthanum-140, and cesium-137 was made by gamma spectra of all six o f these persons. However, the amount of iodine-131 made it impractical to do much further total body counting until most of the iodine-131 had left the body.* (pp. 40–41)
>
> *After the recovery of the first victim from the reactor operating room of the SL-1, a radiation survey was made of all personnel participating in the rescue operation. Eleven of the persons surveyed were found to be highly contaminated, many exceeding a radiation level of 10 R/hr on their extremities; and, as a result, they were directed by an IDO Health Physicist to be taken to the Gas Cooled Reactor Experiment (GCRE) Plant for decontamination.* (pg. 42)

According to the IDO report, during the rescue operation, the monitoring equipment used were not capable of reading the high dose rates in the SL-1 reactor building.

> *At about the third or fourth step the Juno Radiation Detector which the MTR H.P. was carrying pegged at 25 R/hr. The Fire Captain and the MTR H.P., therefore, evacuated from the area stopping at the guard house. The total estimated time for this entry was one and one-half to two minutes.* (pg. 23)
>
> *At the top of the stairway, the door to the reactor operating room was found open. The Phillips H.P. held a high range Jordan Detector in the doorway and observed that the radiation detector pegged on the 500 R/hr scale. Next, with the Assistant Fire Chief holding a light in the doorway, in order to better illuminate the reactor room, the Phillips H.P. took a brief look into the room. Because either the light was dim or the mask's eyepieces were fogged, the Phillips H.P. did not see very*

> *much of the reactor room and none of the ceiling. He was able to observe some wreckage in the reactor room, but saw no personnel. They then hastened down the stairs and out the building via a door on the southeast side of the corridor leading from Building 602 to the reactor building.* (pg. 25)
>
> *At about 10:00 P.M. the MTR H.P. at the SL-1 was authorized to allow personnel to enter radiation fields greater than 200 R/hr by the enroute IDO Chief, Site Survey Branch; it was stipulated that Scott Air-Paks should be worn by all people involved in penetration operations. In view of this authorization, plans were made for another entry into the SL-1 Area. It was decided by the health physics staff at the scene that the planned entry should be halted when a radiation rate of 500 R/hr was observed, because this was the maximum dose rate which could be read on the high range Jordan Radiation Detector….. At the top of the stairs a dose rate of 500 R/hr was observed, and it was estimated that a radiation field of 500–700 R/hr existed about two feet inside the operating room. A quick observation made of the reactor room indicated one person positioned on the floor near the Motor Control Center (MCC) and moving, and another person positioned between a shield block and the reactor head. Radiation fields of 500 to 600 R/hr were estimated to be near the MCC and greater than 1000 R/hr directly over the core. The two CEI personnel then departed from the reactor room to obtain assistance and equipment for a rescue. The entire entry and return operation took approximately three minutes.* (pg. 26)

Table 3.1 on page 44 of IDO 1962 summarizes the distribution of gamma radiation exposures to workers involved in the SL-1 rescue and reactor stabilization operations from January 5 to January 31, 1961. This table shows the following:

(1) A total of 577 workers had participated in the operations
(2) 413 workers received gamma doses ranging from 0 to 300 mR
(3) 71 workers received gamma doses ranging from 300 mR to 900 mR
(4) 66 workers received gamma doses ranging from 900 mR to 3 R
(5) 18 workers received gamma doses ranging from 3 R to 12 R
(6) 6 workers received gamma doses ranging from 12 R to 25 R
(7) 3 workers received gamma doses over 25 R
(8) The highest gamma dose received by any worker was 27 R

There is no information given on the assumptions used in calculating these gamma doses. There are no data provided for internal doses, skin doses, and extremity doses.

Tables 3.2 and 3.3 of the IDO report also summarize the thyroid doses and the bone critical organ doses estimated for some workers with airborne intakes. However, this report does not provide information for the gamma, thyroid, and critical doses to workers involved in the SL-1 cleanup effort after January 31, 1961.

The 1963 technical paper (Horan 1963) entitled, *The Health Physics Aspects of the SL-1 Accident*, provides a table summarizing the whole-body and thyroid doses received by emergency personnel, including the following categories:

(1) AEC Health Physicist
(2) Contractor Supervisor
(3) Contractor Health Physicist
(4) AEC Project Officer
(5) Cadre Supervisor
(6) AEC Physician
(7) AEC Nurse
(8) Support Patrolman
(9) Support Health Physicist
(10) Army Support

As a "lesson learned," the paper states the following:

> *Disaster planning at the NRTS had been geared primarily to criticality type of maximum credible accidents involving the release of thousands of curies of fresh fission products or iodine-131 to the atmosphere. Many of the unique types of problems experienced or suggested by the SL-1 accident had not been considered: performing recovery operations in radiation fields of hundreds of R/hr, medical treatment and decontamination of highly contaminated survivors and casualties, performing field operations around the clock for an indefinite period of time. Supplies and equipment, with but a few exceptions, were adequate. One of the first lessons was that survey instruments with a maximum range of 500 R/hr are inadequate for emergency use. Instruments with a maximum range of 5000 R/hr should be available. Available health physics personnel were depleted due to overwork rather than overexposure before all equipment had been committed, and this despite the full support of 5 NRTS contractors and radiological assistance from 4 AEC or military organizations outside the State of Idaho. During the first 11 days, 81 health physicists were utilized in the field and over 130 other personnel to provide them with field or laboratory support. Many individuals worked in excess of 120 hr per week, and yet sufficient personnel were not available to do the innumerable data collecting and research items which were desirable. The twenty-six man radiological detachment from the Army Chemical Warfare Center at Dugway, Utah, provided early and effective field support.* (pp. 184–185)

The INL Site Profile TBDs only provide minimal treatment of the SL-1 accident. But it is known that during the SL-1 excursion, 500,000 Ci of fission products were released in the reactor building and 1,100 Ci to the atmosphere. The accident also caused high-level contamination of up to 50 R/hr around the ARA-II area. There were more than 1,000 workers involved in different phases of the rescue and cleanup operations. It is imperative for NIOSH to provide specific guidance to dose reconstructors in calculating different missed doses for the SL-1 workers.

For the initial rescue operation (first phase), NIOSH should evaluate the missed occupational environmental doses to unmonitored and monitored workers (support workers), who arrived at the SL-1 facility but stayed outside the reactor building. These workers would be exposed to the fission products released in the surrounding atmosphere, and also the high-level radiation field of 50 R/hr. It is plausible that these workers were exposed to short half-life fission products that were not considered in the TBDs.

NIOSH should evaluate the missed occupational internal and external doses to rescue workers, such as firemen and health physicists, who either were involved in the discovery of the accident, made entries into the reactor building to retrieve the victims, or performed radiation surveys. There is no doubt that these workers would have received high level bodily contamination and inhalation intake of the fission products (500,000 Ci) released inside the building. The workers also had the potential of significant intake of uranium and its daughters and exposures to very high-level beta/gamma radiation fields that exceeded 1,000 R/hr. The other workers (such as the nurse or the doctors) who did not enter the reactor building, but made physical contact with the victims and equipment used to retrieve the victims, would have been contaminated as well.

For the reactor dismantling operation (second phase), NISOH should evaluate the missed occupational environmental, internal, and external doses to workers, monitored or unmonitored, who were involved in retrieving the reactor core, fuel elements, and fuel fragments. For instance, the crane operators and maintenance workers would have been exposed to the high gamma radiation fields from the reactor core when the top cover of the reactor building was lifted, so that the reactor core could be retrieved and dropped into the shielded cask. NIOSH should also evaluate the missed occupational internal and external doses to workers who were involved in the examination and decontamination of the reactor core at the ICPP Hot Shop and Hot Cell.

For the reactor building cleanup operation (third phase), NIOSH should evaluate the missed occupational environmental, internal, and external doses to workers, monitored or unmonitored, who were involved in entering or not entering the reactor building to perform cleanup activities. These workers would have been exposed to high-level beta/gamma radiation fields and subject to significant inhalation intakes of fission products and uranium.

In view of all technical difficulties and equipment shortcomings in the SL-1 rescue and stabilization operations, NIOSH should evaluate the equipment used during the SL-1 incident and determine the uncertainties associated with the dose rate estimation, gamma dose assessment, airborne intake estimation (since there was no air sampling data), thyroid count, critical organ dose assessment, extremity dose assessment, and stay-time determination for various personnel categories. The results of the uncertainty evaluation would be helpful for the dose reconstructions. NIOSH should provide adequate information and explicit guidance in determining various types of missed dose for workers who participated in different phases of the SL-1 operations.

In addition, NIOSH should also evaluate the possibility of missed neutron dose received by the workers during the rescue operations. Even though the Occupational External Dose TBD states that, “the high dose locations where most of the gamma and beta dose is received, such as the ICPP and SL-1, do not have associated neutron dose” (TBD, pg. 25), it is quite clear that during the initial rescue operation of the SL-1 accident, there was no consideration given for neutron measurements of worker exposures when they were making entries into the reactor building where fuel rod fragments and debris were scattered all around the control room floors.

5.1.4 Occupational External Dose Issues

In the Occupational External Dose TBD, NIOSH presents thoroughly the history of the development of worker dose limits from the 1930s to the present. The TBD evaluates the external dosimetry programs, record-keeping practices, and different dosimetry systems (films, pocket ionization chambers, TLDs, neutron track emulsions, and albedo) used historically at INL. The TBD presents information (based on Reilly 1998) on the characteristics, calibration, and uncertainties associated with the dosimetry systems. The TBD also provides radiation fields measured at major INL facilities in 1998 with their relative biases. NIOSH summarizes all information associated with potential missed external doses in two separate categories (photon missed dose and neutron missed dose) for monitored workers in Tables 6B-1 and 6B-2.

However, NIOSH did not evaluate facility field logs, RWPs, facility survey reports, incident reports, occurrence reports, contamination reports, PEQs, and ALARA records. The TBD does not provide evaluation of special exposure events, such as the SL-1 incident, to determine missed doses for monitored or unmonitored workers (e.g., nurses) during rescue, stabilization, recovery, and cleanup operations. NIOSH interviewed INL Dosimetry Department staff and requested individual dosimetry records and documentation. NIOSH also interviewed some workers and retirees as part of an outreach program after the TBD was done.

As in other instances cited in this report, the authors of the Occupational External Dose TBD display confidence in the past and current INL radiological protection programs and implementation, the accuracy of the external dosimetry programs, including beta/gamma and neutron systems, and the dosimetry record-keeping systems. The TBD states the following:

> *It was INEEL policy that personnel expected to receive any radiation dose or personnel whose work was centered at the site were assigned a radiation monitoring badge.* (pg. 8)

> *The INEEL dosimetry organization developed a set of basic administrative practices in 1951, which have changed somewhat as the technologies of ionizing radiation dosimetry and recordkeeping have changed.* (pg. 9)

> *When there has been a question about a dose value being assigned to an INEEL worker, a Personnel Exposure Questionnaire was normally initiated.* (pg. 11)

It is important to point out that many records and data pertinent to the worker claims are not included historically in the workers' files, including personnel contamination reports, PEQs, RWPs, incident reports, and occurrence reports.

NIOSH has included the following historical dosimetry standards and site administrative practices information in the TBD:

(1) The TBD identifies the way that INL reports external doses for workers:

> *INEEL has reported doses as penetrating and nonpenetrating. The penetrating dose corresponds to the deep dose equivalent, and the nonpenetrating dose plus the penetrating dose corresponds to the shallow dose equivalent.*" (TBD, pg. 6)

(2) The TBD summaries the history of Federal dose limits for occupational workers:

a. *1949* – 0.3 R per week or 15 R per year (recommended by NCRP-7)

b. *1957* – (5N-18) rem as maximum allowable dose (NCRP)

c. *1958* – 3 rem per quarter or 15 rem per year (AEC)

d. *1960* – 3 rem per quarter or 12 rem per year (President Eisenhower)

e. *1971* – 5 rem per year (NCRP 39)

f. *1971* – defined the concept of deep and shallow dose equivalent indexes (ICRU)

g. *1985* – introduced ambient dose equivalent, directional dose equivalent, individual dose equivalent penetrating, and individual dose equivalent superficial (ICRU)

h. *1981* – required the monitoring threshold as 100 mrem effective dose equivalent (DOE 5480.1A)

i. *1985* – specified the measurement of deep and shallow dose equivalents at depths of 10 mm and 0.07 mm, respectively (ICRU)

j. *1990* – redefined the concept of dose equivalent to equivalent dose, quality factor to radiation weighting factor, and generated new factors (ICRP)

The NIOSH approach in identifying the missed external doses at INL makes the following assumptions:

(1) The INL site radiological protection programs, external dosimetry programs, record-keeping systems, laboratory analytical programs, equipment calibration programs, quality control programs, and training programs were effective and dependable. This assumption that there was no administrative flaw in the implementation of these programs in protecting workers from external radiation was found to be unsubstantiated by the Tiger Team and DNFSB site inspection findings and also comments from site experts.

(2) The information provided in the claimant files is complete for dose reconstruction. During site expert interviews, many past and current workers complained that there are dose records missing from their personal exposure file.

(3) The dosimeters were accurate in measuring external worker dose within the projected uncertainties. This assumption is valid only if the dosimeters were calibrated correctly to measure the intended radiation fields, but there were documented deficiencies in INL's calibration programs.

(4) The workers had worn their dosimeters correctly in such a way that external dose was measured accurately. This assumption is not entirely valid, especially in varying high beta/gamma fields or in high-risk maintenance jobs, when personnel had to work in tight spaces very close to different sources. For example, sensitivity of dosimeters to low energy photons is angular-dependent. As a result, the dosimeters would under-measure the gamma doses. The TBD does not evaluate missed worker dose due to lack of multi-badging or faulty badge placement in high, varying, or oblique beta or gamma fields.

(5) There is no other scenario that would cause missed external dose to workers at INL. The approach to missed dose in this TBD precludes the potential for missed beta/gamma doses, due to inconsistent and less protective work practices in the early years.

(6) NIOSH believes that the claimant-favorable approach to estimate missed external doses (both photon and neutron) is to use the following equation:

$$(N \text{ x } MRL/2),$$

where N is the number of "zero" doses recorded. The MRL/2 approach is not claimant favorable.

(7) There is no stand-alone missed beta dose for workers working at INL, such as low beta fields not measured by dosimeter (shallow dose with skin cancer concern) or very high beta fields measured as gamma (deep) dose. This is a questionable assumption (see Taulbee 2002).

NIOSH believes its missed gamma and neutron dose values are claimant favorable, as the Occupational External Exposure TBD states the following:

> *The missed dose for dosimeter results less than the MRL is particularly important for earlier years when MRLs were higher and dosimeter exchange was more frequent. One option to calculate the missed dose described in NIOSH (2002) is to estimate a claimant-favorable maximum potential missed dose where MRL/2 is multiplied by the number of zero dose results.* (pg. 32)

The use of MRL/2, or LOD/2 (Taulbee 2002), for calculating the missed dose values is questionable. This averaging approach may be adequate from a statistical standpoint for evaluating total missed dose for worker population, but at best, this approach is claimant-neutral for an individual. It is not claimant favorable for an individual worker, who may have received all his missed doses in the upper half above the average (MRL/2). It would be more appropriate to use the value of MRL (instead of MRL/2), i.e., 95% (2 sigma). In addition, the use of such an equation for missed dose calculation requires the compilation of accurate MRL values and conservative uncertainties for all dosimeters and film badges used at INL.

5.1.4.1 Beta/Gamma Dose Issues

For photon missed dose, NIOSH presents, in Table 6B-1 of the External Dose TBD, maximum annual missed dose values for different films (e.g., 552 Dupont film, 558 Dupont film, 508 Dupont film) and for different TLDs (LiF, LiF in Teflon, Harshaw Two-Chip, Panasonic Four-Chip), based on the MRLs for six different time periods; (1) 1951–1958, (2) 1958–1966, (3) 1966–1974, (4) 1974–1975, (5) 1974–1985, and (6) 1986 to present. The missed dose values are provided for different dosimeter exchange frequencies (weekly, biweekly, monthly, quarterly, and semi-annually).

For beta and gamma organ dose evaluations, the TBD provides the following instructions:

> *For photons prior to 1981 the conversion factor from exposure to organ dose should be used. For 1981 and after, the conversion factor from deep dose equivalent to organ dose should be used.* (TBD, pg. 34)

However, the TBD provides no discussion, approach, information, method, or specific instruction for calculating missed beta dose for INL facilities. The TBD does provide some information and data on the beta radiation fields at INL in Section 6.3.4.2 (pp. 23–25). Table 6-5 lists the beta dosimeter thicknesses and associated under-reporting with the following, somewhat cryptic, instruction.

> *The fraction of beta dose measured shown in Table 6-5 is the average as described above. To determine the corrected beta dose, divide the non-penetrating result from the dosimetry system by the values in the last column of Table 6-5. The reported dose will likely be somewhat higher than this because the calibration probably did not consider such a correction and reported the dose for the calibration exposure* (TBD, pg. 25).

This recommendation tries to capture beta radiation causation of skin cancer. The uncertainty for beta dose is estimated to be 50% at one sigma. This table is not included as part of the missed dose calculation or discussion. It is not clear whether NIOSH believes that there is no missed beta dose at all at INL facilities or the contribution from missed beta doses have already been included in the gamma missed dose calculation. In either case, NIOSH needs to clarify and justify its position in the TBD.

As quoted earlier, INL reported doses as penetrating or non-penetrating. The penetrating dose corresponds to the deep dose equivalent, and the non-penetrating dose plus the penetrating dose corresponds to the shallow dose equivalent. It is not clear how a dose reconstructor should allocate the total missed photon dose calculated using the above equation; should this total missed dose be added to both the shallow and deep doses?

The photon missed dose defaults are to be applied to the dose reconstruction according to the following conditions:

> *Missed photon dose for INEEL workers would occur where (1) there is no recorded dose because workers are not monitored or the dose is otherwise unavailable, or (2) a zero dose is recorded for the dosimeter systems for any response less than the site dose recording threshold (the MRL)* (TBD, pg. 32).
>
> *Some individuals who might occasionally visit site facilities but did little work with radiation, had badges at several different facilities. It is not appropriate to base missed doses on the multiple badges issued. Early on at INEEL, the badge change frequency was not the same for everyone. Workers with low probability of exposure were placed on a longer change cycle than those with more chance of exposure. Therefore, missed doses should be based on the actual change frequency for a person, and the frequency can be determined from the individual's data package* (TBD, pg. 9).

NIOSH also gives instruction to DOE staff on what to provide a claimant for the dose reconstruction:

> *DOE provided dosimetry information for a former INEEL worker, whose dose reconstruction is underway, should include a dose summary for the employment period and a copy of each weekly, monthly, quarterly, etc., form which will also show the work location, so the individual file could be several inches thick in hard copy. Each sheet is redacted so only the person of interest's name and applicable information are visible. This file provides the recorded information as to the exchange period for the person of that time period.* (pp. 9–10)

NIOSH has included the following beta/gamma dosimetry information in the TBD to demonstrate the adequacy of its missed dose approach. However, this information shows significant weaknesses in several key areas.

(1) The TBD indicates that the practice at INL is to subtract environmental radiation levels from the reported gamma doses.

> *These badges were usually stored at the respective operational area entrance security gate for INEEL facilities. Control badges, which are used to subtract background radiation, have also been located there. This practice may lead to subtracting environmental radiation from site activities reducing the reported doses. Environmental radiation levels have been monitored for most of the life of the INEEL, originally with film badges and later with TLDs. Table 6-1 presents results of this monitoring at facility fence-line locations near the security gates which can be added to individual's dose history or used for non-radiation workers working at the site.* (TBD, pp. 8–9)

If the fence-line dosimeters were kept at the same place as the control dosimeters, then the following could happen: Say a worker at ARA had an annual dose of 500 mrem in 1960, and that his dosimeters read a total of 726 mrem and 226 mrem was subtracted from the control dosimeters over the year. The TBD indicates that the workers dose will be calculated as 500 + 226 x (2000/8760) = 552 mrem. However, since the "real" background for the year is around 100 mrem, the workers dose should be 500 + 226 – 100 mrem = 626 mrem. The NIOSH environmental dose subtraction method, therefore, is not claimant favorable.

(2) The TBD provides some sample reporting forms used in early days in 1958 and 1959 (Figures 6-1 and 6-2) and explains how the worker doses were recorded in these forms. The TBD, however, does not provide a history of the methods used at INL in recording different external doses in the worker files, such as penetrating, non-penetrating, shallow, deep, and neutron doses. NIOSH should provide instructions on how to reconcile the worker doses using these different methods, and also how to add the missed doses during the dose reconstruction.

The TBD indicates that, "the personnel monitoring badges have always been considered more reliable than pencil dosimeters; so after the film badge results became available, the daily pencil readings were no longer considered doses of record." In one case, NIOSH shows the large difference between the two readings: "In Figure 6-1, the pencil readings totaled 820 mR and the badges reported 0 mR for 18 badges" (TBD, pg. 11). NIOSH indicates these discarded pencil dosimeter reading values "can be recovered from the earliest forms for a worst-case estimate of dose" (TBD, pg. 11). Perhaps NIOSH should conduct a study on how extensive are these differences in pencil dosimeter and film badge doses. This is especially relevant based on the numerous reports from site experts that the film badge dosimeters they wore while working at the INL facilities were underestimating the actual external doses they received (see Attachment 3).

(3) The TBD surveyed different beta/gamma personnel monitoring systems at the INL site. These systems include the following:

a. 1951–1958: NRTS Self-Service System – The film badges used sensitive and insensitive DuPont type 552 films with 2 windows, one open and the other with a 1 mm cadmium filter. The badges were processed weekly. The type 552 film has a threshold level of about 30 mR. DuPont type 558 films, with a threshold level of 10 mR, were used at two reactor areas.

b. 1958–1966: Multiple-Filter NRTS Film Badge – The film badges used DuPont type 508 films with three filters, including 1 mm cadmium, 0.013 mm silver, and 0.5 mm aluminum. The badges were processed biweekly or monthly, with the exception of the high-dose areas where weekly processing continued. A minimum reporting level (MRL) of 10 mrem was used for both beta and gamma radiation.

The TBD indicates that the densitometer was zeroed using the "blank film" – here taken to be a film that had not been used in the field. This would mean that this "blank film" was used as the control film. This, in turn, would mean that the "fence-line dosimeter" or the "access-gate dosimeter" was not used to subtract the background from the occupational dose. This practice of using the blank film for zeroing the densitometer is claimant favorable, in that the film dosimetry service found a low background area for its film store.

c. 1966–1974: Lithium Fluoride Teflon TLD System – This TLD system used a lithium fluoride disk with two 13 mm Teflon disks and a 1 mm cadmium filter. The badge could read 30 mR on a quarterly basis. "For normal monitoring, only the open window TLD was read and considered penetrating dose unless it read more than 125 mrem, in which case the shielded TLD was also read." In 1968, the monitoring period was increased from 3 to 6 months. In 1972, some processing was increased to an annual basis.

d. 1974–1986: Harshaw Two-Chip TLD System – This TLD system used two LiF TLD chips, with one covered by aluminum and the other by Mylar. The aluminum-covered chip provided penetrating doses at a tissue depth of nominally 1 cm. The beta dose was calculated from the difference between the two chips. The calculation was accurate only for the beta energy used in calibration. The practice was to read only the open window chip to determine if the non-penetrating dose was above 15 mrem and thus, reportable. If the threshold was exceeded, both chips would be read.

e. 1986–present: Panasonic Four-Chip TLD System – This TLD system used three lithium borate and one calcium sulfate TLD elements with plastic and aluminum filtration. The MRL used was 15 mrem beta and gamma in 1986, 30 mrem beta and gamma between 1986 and 1989, and 15 mrem for gamma and 30 mrem for beta until 1993. Since 1993, the MRL is 10 mrem.

(4) The TBD indicates the necessity of using multiple badges in some high beta radiation fields.

> *Experience following the SL-1 accident showed a wide variation of beta-to-gamma ratios and necessitated controlling both radiations rather than just the gamma. A set of as many as 18 badges could and in many cases was fastened on a belt around the worker to determine a beta:gamma ratio for each particular entry.* (TBD, pg. 16)

However, the TBD does not provide adequate information on multiple badging or specific guidance on how to apply it to dose reconstructions.

(5) The TBD describes the calibration programs at INL for beta and gamma radiation. In 1959, INL used a radium source for gamma calibrations and uranium for beta calibrations. From 1975 on, a 2–5 Ci Sr-90/Y source was used to calibrate TLD badges. In 1981, a phantom was used in the calibration facility. In 1983, a natural uranium source was used for beta calibration. NIOSH lists uncertainties associated with beta and gamma

dosimeter calibrations historically at INL in Table 6-3. Several important associated observations include the following:

> *Separation of penetrating dose from non-penetrating dose was an issue in 1957 (Bennett 1957) and again in 1976 (Jenson 1976), particularly for ICPP where strong high energy beta fields were not unusual.*
> (TBD, pg. 19)

> *The 1991 Tiger Team Review of the INEEL site indicated that the INEEL contractor and the Idaho Operations Office using the same sources for calibration led to a conflict of interest or an advantage in DOELAP tests.*
> (TBD, pg. 19)

NIOSH did not evaluate the adequacy of these radium and uranium sources used for beta and gamma calibration. There is no information on whether the calibration sources address the full range of the beta and gamma fields at INL facilities.

(6) The TBD discusses the radiation fields at the INL facilities based on an INL study (Reilly 1998). It also shows that the percentage bias of the beta and gamma measurements lie within a +27% to -43% range.

a. NIOSH identifies that these biases may have caused missed dose corresponding to low-energy photons at all INL facilities, especially at RWMC and at TRA radiography facility.

b. NIOSH provides a list of selected beta and gamma energies for various INL facilities in Table 6-6 as IREP inputs for use in dose reconstructions.

Given this information, NIOSH does not provide specific guidance in compensating for the biases determined.

(7) The External Dose TBD discusses beta radiation fields and dosimetry:

> *Beta radiation fields are usually associated with activation or fission product radioactivity that is outside of a container such as a spill or only lightly shielded or in hot cells. High beta fields were not unusual at the INTEC where large quantities of fission products exist. Pure high energy beta fields in some locations, particularly in the exhaust stream, have caused dosimetry problems because the badge shielding or instrument packages did not provide a full 10 mm tissue equivalent coverage and thus beta fields would be measured as gamma fields.* (TBD, pg. 23)

> *Beta field dosimetry became fairly accurate with the definition of DOELAP requirements in the early 1980s. Prior to that beta monitoring systems had various flaws, primarily in a detector too thick to give a good surface result or one that was covered with extra material. Calibration*

> *was to high energy betas from either uranium or strontium. The dose from low energy betas will be missed altogether if the beta energy is not sufficient to penetrate the detector cover and will be underreported if the beta energy is not sufficient to penetrate the entire detector.* (TBD, pg. 23)
>
> *Table 6-5 provides the cover and detector thicknesses for the beta badges used at the INEEL… To determine the corrected beta dose, divide the non-penetrating result from the dosimetry system by the values in the last column of Table 6-5. The reported dose will likely be somewhat higher than this because the calibration probably did not consider such a correction and reported the dose for the calibration exposure* (TBD, pg. 25)

Given these discussions, NIOSH does not provide any guidance in compensating for the potential missed beta doses.

(8) The TBD discusses the uncertainties associated with gamma measurements, saying that, "A realistic estimate of total uncertainty for photon dosimetry is about 35% at one sigma" (pg. 34). The absence of information from inter-comparisons means that the evaluation of the photon dosimetry uncertainties is more subjective than objective. The fact that the largest dose contribution was from photons with E > 250 keV will reduce the uncertainty. It also should be mentioned that the uncertainty depends on the dose recorded. For lower doses, the uncertainty can be as high as 100%, while for higher doses, 35% should be a reasonable estimate.

The TBD also states that 9 sets of 5 dosimeters were irradiated with combinations of 10, 20, or 30 mrem of Ra gamma and U beta with measurability within ±12 mR with 95% confidence. The standard deviation should be about ±3 mR instead of ±12. This applies to the system introduced in 1958.

(9) The TBD discusses the uncertainties associated with beta measurements: "The uncertainty for beta radiation is somewhat larger, an estimated 50% at one sigma" (pg. 34). A beta dosimetry system will only show reasonable uncertainties if the system is calibrated with the same beta energy spectrum as expected for the radioactive source that will be measured. This was not the case at INL, with the exception of the project for the fabrication of shielding using depleted uranium. This, alone, should make the uncertainty higher than 100%. More seriously, the dosimeter will only estimate dose rates for geometries similar to the calibration geometry. In reality, there was no comparison between the facility beta exposure and the calibration set-up. A thorax dosimeter can only give a response, such as "yes there was a beta dose" or "no there was no beta dose," and sometimes not even this was possible. A wrist dosimeter might give a more consistent answer, if it can be shown that it was "looking" at the beta source. The statutory requirement of a claimant-favorable dose reconstruction process is achieved by (1) giving the benefit of doubt when there are unknowns, and (2) defining uncertainties for measured data and selecting the 99th percentile values of a Monte Carlo distribution.

5.1.4.1.1 Completeness and Quality of INL Gamma and Beta Dosimetry and Record Keeping Programs

In addition to the 1991 Tiger Team and DNFSB findings listed in the above sections, there are several other findings specific to the INL dosimetry programs which call into question the completeness and quality of the dosimetry results and records. NIOSH should investigate these items and evaluate their impacts on the dose reconstructions.

(1) The 1991 DOE-HQ Tiger Team found that the INL Nuclear Accident Dosimeters were deficient, because the overall gamma/neutron response was within ±40%, which did not satisfy the ±25% requirement specified in DOE 5480.11 (Vol. 2, pg. 4-175).

(2) The 1991 DOE-HQ Tiger Team found that the INL personnel dosimetry program was in serious deficiency for several reasons (Vol. 2, pp. 4-176 and 4-177):

   a. Absence of a centralized, integrated dosimetry program

   b. Absence of field application and characterization studies to respond to the site dosimetry needs

   c. Absence of adequate technical staff for technical support and development functions to maintain a quality dosimetry program

   d. Absence of any calibration sources for daily functioning of the dosimetry processing and developing element correction factors and quality control

   e. Absence of reviews for personnel dosimetry and related procedures, characterization of site radiation conditions to assure the adequate coverage in the range of exposures and energies and type of radiation anticipated

   f. Absence of changes in algorithms necessary to correct dosimetry assessment for the intrusion of different radiation conditions

(3) During the site expert interviews, past and current workers at INL facilities provided first-hand information about potential missed dose scenarios and deficiencies in personnel protection programs and dosimetry record keeping. Even though there is a sentiment that the INL radiological protection programs and the advancement of equipment and techniques have made dramatic improvements over the past two decades, the missed dose problems due to these deficiencies in the early years must be addressed in a fair and reasonable manner.

5.1.4.1.2 Penetrating and Non-Penetrating Doses

Historically, after the early 1950s, INL categorized gamma and beta doses as "penetrating" dose and "non-penetrating" dose, while in the early 1950s, beta-gamma (combined) dose was categorized as non-penetrating/penetrating. Neutron dose was maintained and reported separately. Current dosimetry records report "shallow" and "deep" doses, with the notation that penetrating (deep) doses include neutron and gamma exposures. Records provided by INL in support of dose reconstruction include "shallow" doses, "deep" doses, and "extremities" doses. Neutron doses are provided separately, but are also included in the "deep" doses.

The procedures and algorithms used in the film badge dosimetry service in the early days underestimated the Hp(10) dose, because the low-energy photons reaching the dosimeter were considered beta radiation. Surprisingly, the film service then added this beta dose (to the skin) to the "deep" dose, making that practice claimant favorable. However, the External Dose TBD also requires the dose reconstructor correctly to consider only the "deep dose" as Hp(10). However in doing so, the low-energy photon contribution to Hp(10) is lost.

To be claimant favorable, the NRTS (an early name for INL) calculated the beta dose and the gamma dose, summed these two together, and recorded the result on the worker files as a whole-body dose. The current dose reconstruction process is applying correctly the gamma dose as the effective dose. The problem is that this gamma dose alone is not claimant favorable, as the information on dose due to low-energy gammas (E < 100 keV) has been lost.

The TBD states that, "..it was agreed that all soft radiation would be considered as beta, and all the penetrating radiation as gamma for the purposes of reporting. This created a large safety factor in favor of the employee, as the soft gamma produced excessive darkening of the film emulsion…"

This administrative decision made by the dosimetry system, which discounted the contribution of low-energy photons to Hp(10), was not in favor of the workers. On the contrary, low-energy gammas also cause absorbed doses in the ICRP 60 organs. Due to the high number of Compton scatters and the small mean free path for photons at low energies, the "effective dose per photon" at say 60 keV is greater than that for a 1.2 MeV photon. Low-energy beta radiation will not cause a dose to the ICRP 60 tissues organs, with the exception of the skin, but there may be a few cases when the dose due to very high-energy betas to the testes and breast are relevant.

It is very difficult to estimate the reduction in the dose recorded for the workers resulting from the film service's decision. For reasonably high-energy photon fields, such as for reactor work or reprocessing plants, a 15%–30% underestimation of the effective dose may be possible.

5.1.4.1.3 Correction for Beta Dose

For the film dosimetry system and the Harshaw two-chip TLD system after it, it seems that INL considered only two irradiation possibilities; high-energy photons and beta radiation. However, OCAS IG-001, the external dose reconstruction guideline (Item 2.1.1.2), states the following:

> *Within the NIOSH-IREP probability of causation program, there are three photon energy bands; 1) below 30 keV, 2) 30 to 250 keV, and 3) above 250 keV. Therefore, some separation of the dose from each energy band is required.*

> *At most of the larger facilities, multi-shielded film badges or multi-element TLDs have been used since the mid 1960s. Since only three energy bands are used in the probability of causation calculations, the differences between various filter doses can provide insight into the gross energy distribution at the facility. To the extent possible, these differences should be used to estimate the relative energy distributions in earlier years when only two element film badges were used. If individual energy distribution information is not available for two element film badges, the open window dose should be used as a claimant friendly estimate of the 30 to 250 keV dose.*
>
> *When an estimate (of the energy spectrum) based on radionuclide inventory (the type of radionuclide) is conducted, some consideration should be given to the degree of Compton scattering that would contribute to the 30 to 250 MeV energy range*.

Considering the entire open window dose to be due to beta radiation is not claimant favorable. The main problem is that the filtered part of the dosimeter will not see the low-energy photon radiation (below 100 keV), and will under-report the dose for photons between 100 keV and approximately 200 keV. When it is assumed that a two-element film dosimeter is looking at photon radiation only, the following steps are made to calculate the dose:

(1) If the net optical density (NOD) under the two filters is about the same, then the dose is due to high-energy photons (E> 250 keV)

(2) If the NOD under the open window is higher than that under the silver filter, then an estimate of the energy of the photon radiation is made and the dose is estimated using the response curve of the film; a curve similar to Figure 6-7 (pg. 15)

In the case of INL, this procedure (or a similar procedure) was not followed. The INL procedure considers that the open window measures the beta dose only, and the filtered part (also called the shielded part in the INL documents) of the dosimeter measures only the high-energy photons. The question now is how to estimate the low-energy photon dose from the available information. As the “beta doses” were estimated using the uranium daughter beta source for calibration, it is not possible to estimate the low-energy photon dose directly from the estimated beta dose. In fact, most of these “beta doses” are really low-energy photon doses.

As it seems that the original optical density information was recorded, it would be possible to calculate the low-dose information for a few cases. In addition, since gamma spectroscopy information for some facilities is available, it might also be possible to check whether there was a significant presence of photons in the $30 < E < 100$ keV range. If it can be shown from a few cases that a significant part of the photon dose was “missed,” then a global “correction factor” should be multiplied to the individual deep dose estimate, much in the same way the factor of 1.119 was applied in the SRS TBD site profile.

5.1.4.1.4 Angular Dependence Correction Factor for Gamma Dose

NIOSH should provide an angular dependence (anatomic geometry) correction factor for external photon doses. This correction factor would be used to account for the bias of a dosimeter worn at the neck level and the higher doses received by tissues/organs below the waist. The angular dependence of the sensitivity of the dosimeter is most pronounced at low-photon energies (SC&A 2005, pp. 143–145).

5.1.4.1.5 Restating Beta Dose as Gamma Dose

The TBD indicates that if individual energy distribution information is not available for two-element film badges, the open window dose should be used as a claimant-favorable estimate of the 30 to 250 keV dose. Most of the time, while performing normal work as a plant operator or supervisor, the workers film badge or TLD will not get close enough (less than 1 meter) to surfaces emitting beta radiation to detect a beta dose. A worker (such as a pipefitter or welder), however, may for short intervals bring his dosimeter closer to a surface contaminated with beta emitters. Therefore, it is not claimant favorable to state that the entire dose measured in the open window is due to the beta dose. It is more correct to say that it is from photons in the range of 30 keV < E < 250 keV.

5.1.4.1.6 Photon Spectrum Split

As the External Dose TBD indicates, NIOSH is assuming a 25%/75% split between low- and high-energy bands for the photon spectrum; SC&A recommends a more claimant-favorable split of 50%/50%, similar to that used in the SRS TBD. Whatever the split of the photon spectrum, the question is; How will the dose for the energy region 30 keV < E < 250 keV be calculated? As no photon open window dose was recorded, the dose in this energy range cannot be directly estimated. The difficulty of assigning correct doses can be illustrated by considering two extreme examples:

(1) A worker receives a dose of 200 mR from a 60 keV source. The open window information would be turned into a beta dose to the skin. The filtered window would record little or no dose, say 30 mR.

(2) A worker receives a dose of 200 mR from a 662 keV source. The photon energy is degraded by Compton scattering, so that in reality he receives 100 mR from photons in the energy range 30 keV < E < 250 keV and 100 mR from photons in the energy range E > 250 keV. The open window dose is turned into a beta dose to the skin. The filtered window dose registers around 150 mR, as some of the lower energy photons are filtered out.

Considering that the dose reconstructor does not know the photon energy spectrum the worker was exposed to, what dose value would be assigned for the 30 keV < E < 250 keV energy region, and what dose would be assigned for the E > 250 keV region?

5.1.4.1.7 Immersion Doses

Immersion in a cloud of beta emitters may happen periodically at INL facilities and lead to important consequences for internal dosimetry. The internal dose to the individual may be higher than that measured by a personnel dosimeter by one or more orders of magnitude. A finger dosimeter would not provide the required exposure information, as it would be filtered and therefore could not be used to estimate beta dose. A wrist dosimeter would give a better estimate, however, if it were located on the side of the wrist facing the beta source.

It was said that during the teleconference with NIOSH experts that the film or TLD monitors would record the dose received due to immersion in a cloud of gamma emitters. The dosimetry system is calibrated for AP irradiations. The dose recorded on a dosimeter due to a semi-infinite cloud irradiation would be approximately half of the actual dose received. NIOSH should, therefore, consider a weighting factor of 2 for immersion dose.

5.1.4.1.8 High-Risk (Dose) Jobs

The TBD indicates that there are facilities at INL (e.g., INTEC) where high beta fields exist. NIOSH should develop a list of high-risk (dose) jobs and provide corresponding beta/gamma dose rates and worker job doses. This information will be helpful for dose reconstructions for personnel who had worked in such jobs and areas. Working in areas where there are fragments or "hot particles" of fission products, for example, during the cleanup of a reactor destruction experiment or SL-1 accident, may lead to the deposition of hot particles with high beta dose rates (above 50 rad hr^{-1}) on the clothing and, possibly, directly on the skin of the face or hands. The beta radiation emitted from these hot particles will not be detected by the film or TLD dosimeters. For workers at fuel element or reactor cleanup operations, for small localized areas, the beta dose could be as high as 100–1,000 rads when calculated over a working day. For claimants with skin cancer, location and job-specific information should be taken into account.

5.1.4.1.9 Extremity Dose

NISOH should evaluate the potential for missed extremity dose for workers (such as operators, pipefitters, machinists, mechanics, or instrument technicians) working in facilities, where highly contaminated equipment, piping, instruments, valves, and systems resulted in exposures in confined spaces to hands. This is especially true for facilities, such as reactors, fuel processing cells, and waste storage systems.

5.1.4.1.10 Discrepancies Between PIC and Film Reading

Many facilities at INL contained areas with high beta radiation rates. However, as the distance between a worker and a planar beta-gamma source increases, the beta dose rate will fall off more quickly than the gamma dose rate. Therefore, at a typical working distance from contaminated surfaces, the beta dose rate will be lower than the gamma dose rate.

PIC dosimeters in the early years were fairly rugged, heavy devices, with steel cylindrical walls, and, therefore would record little contribution from beta radiation (perhaps a factor of 100–1,000 times lower than an open window film dosimeter would record). Additionally, PICs are known to have a larger uncertainty than films or TLDs, and are relatively fragile and easily disturbed. For example, when a PIC is dropped on the ground or otherwise shocked, the reading can shift anywhere, even off the scale. In addition, some PICs are "leaky" and will discharge quicker (show higher doses) than they should.

The INL workers interviewed have a confidence problem with the film dosimeter readings (see site expert interview notes in Attachment 3). However, if the PIC readings and the film readings are compared over a period of 1 year, and it is seen that the average comparison of PIC and film doses differs by more than say 30%–50%, this is evidence that the "deep dose" as reported by the film dosimetry service may have underestimated the Hp(10) dose.

Workers consistently had access to their PIC results, and management periodically reported their film badge results to them. In later years, personnel were issued an annual radiation exposure report. Hence, personnel were able to make their own comparisons between PIC and film badge readings. Interviewed site experts expressed concern regarding the underestimation of actual exposure by film badges. For example, a site expert recalled that he received 500 mR on his PIC, yet his film badge result was 100 mrem for the same period of time. When radiological control personnel were asked to explain the difference between the two readings, they indicated that the PICs were likely responding to the beta exposure. The β to γ ratios can vary from 1 to 1 to as high as 25 to 1 depending on the facility. The TBD should compare PIC verses film badge data (i.e., shallow and deep) and ensure that all the dose has been captured by the film badge. It is important to note that some PICs were worn for only the length of the job, so the discrepancy between readings of the two dosimeter systems cannot be explained by drifting.

An accurate estimate of the total "missed dose" may not be possible, but a rough idea of the "photon missed dose" could be obtained for each facility by the following:

(1) Calculating the average dose and collective dose measured by film dosimetry shortly before the change to TLD dosimetry

(2) Calculating the average dose and collective dose measured by TLD dosimetry shortly after the change from film dosimetry

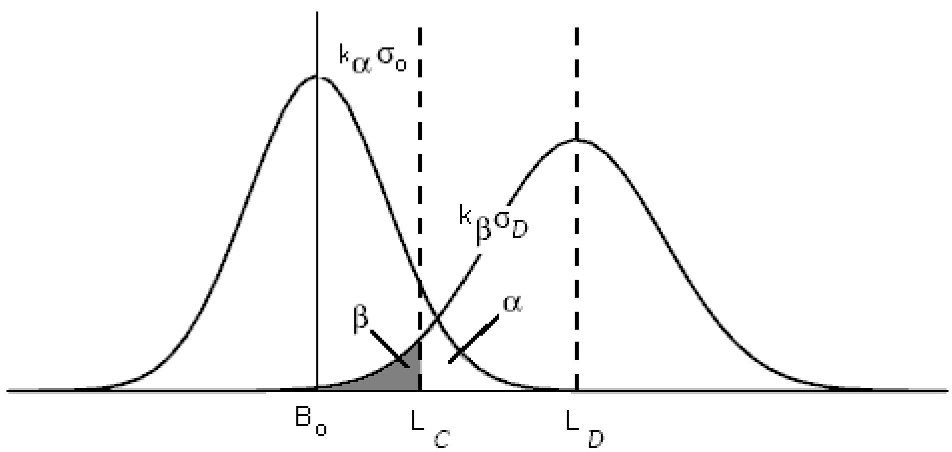
(3) Comparing the two and possibly use the difference as a direct estimate of the "film lost photon dose"

5.1.4.1.11 <u>Minimum Detection Limit</u>

Determination of the minimum detection limit (MDL) takes the dosimetry system uncertainty into consideration. The mathematics of this process is discussed in the External Dose TBD. When using state-of-the-art dosimetry systems today, with a 20 mrem (0.02 mSv) MDL, there will be a number of "false positive" results. As stated in the TBD (pp. 13–15), the necessary information to determine the MDL is not given in the NRTS file, but a value of 30 mrem chosen by INL for Type 552 film would have been an optimistic MDL. A value of 40 mrem (as adopted by SRS) would be more realistic for high-energy photons recorded many years ago. Unless actual reports on tests of the MDLs of the INL dosimetry systems are available and are favorably evaluated, the more conservative SRS values could be used as surrogate MDLs for INL.

The following is based on *Determining the Lower limit of Detection for Personnel Dosimetry Systems*. (Health Phys., 62(1): 2-9;1992.)

Recording a result of a dosimeter as a dose when the dosimeter was not irradiated is known as a Type I error. Most laboratories set their acceptance of a Type I error at 5% when calculating the MDL for a given assay. That is, for any value that is greater than or equal to the MDL, there is a 95% confidence level that it represents the detection of a true dose. The MDL (L_D) for the film dosimetry system may be derived from the following equations (see figure below).



$$L_C = k_\alpha \sigma_o + B_o$$

$$L_D = L_C + k_\beta \sigma_D$$

If $k_\alpha = k_\beta = k$, and for low values of σ_μ (note: see equation 14 of the above paper), then L_D may be considered to be approximately (for a 95% confidence level)

$$L_D = 3.4 \sigma_o$$

Unfortunately, the values cited for the 10, 20, and 30 mrem irradiations may not be used to estimate σ_o. For this calculation, approximately 20 or more non-irradiated monitors should be evaluated. The standard deviation of the optical densities may be used and then converted to a standard deviation in dose by multiplying by the optical density-to-dose coefficient for high-energy photons. This coefficient, from figure 1 of the report, is about 10/0.02 = 500 mrem per optical density unit. The L_D may then be calculated by multiplying the standard deviation in dose by 3.4.

The TBD mentions only the uncertainties due to the densitometer: ± 0.02 optical density units. The uncertainties due to the densitometer reading are the smallest and best known of all the uncertainties involved in the process. The largest uncertainty is the variation of the optical density due to the film development process. Very strict quality control measures help to reduce these uncertainties, especially variations between film batches. For example, the temperature of the film baths in one laboratory is controlled to 20°C ± 0.3°C, and even so, due to the continued use of the developing chemicals, the uncertainty in the optical density at 0.06 is ± 0.02. The unevenness of the film emulsion adds a further 0.01 optical density units to the uncertainty at this level.

These uncertainties do not take into consideration the use of the dosimeter in the field, with the additional associated uncertainties of energy and angular dependence, temperature and humidity considerations, etc. Looking at contemporary and modern film dosimetry systems, for high-energy photons, the MDL for the INL dosimetry system could not have been less than 30 mrem, and 40 mrem (as used by the SRS profile) would be considered more likely and claimant favorable.

5.1.4.1.12 Minimum Reporting Level

NIOSH uses the detection threshold levels of the film badges and the TLDs as the MRL values for the missed dose calculation. However, the information NIOSH used to determine these MRLs is not complete and sometimes not supported. For instance, there is no information provided for the threshold levels for the original LiF Teflon TLDs and the Atlas TLDs. For the Panasonic TLDs, the information is quite confusing for dose reconstruction. NIOSH should re-examine the issue and provide more supporting information and a clearer explanation. In addition, the TBD does not provide any information concerning the uncertainties associated with these threshold levels. NIOSH should provide guidance to the dose reconstructors in assessing MRL uncertainties to ensure that the missed dose calculations are claimant favorable.

5.1.4.2 Neutron Dose Issues

For assessing missed neutron dose, the TBD adopts the sentiment that the INL neutron protection programs, dosimetry program, and record-keeping systems were reasonably complete and effective. This is shown in the following selection of quotations from the External Dose TBD:

> *Most INEEL workers are not exposed to neutrons and so are not badged to measure neutrons.* (TBD, pg. 25)
>
> *Sources of neutron exposure include neutron sources at the instrument calibration laboratories and 14-MeV neutron generators used to characterize waste. For these spectra, the NTA works reasonably well* (TBD, pg. 25)
>
> *Individuals who have the potential to receive neutron dose currently wear albedo badges, and experience has shown that most do not receive significant doses.*" (TBD, pg. 25)

> *Most of the reactors built at the INEEL had no beam ports. Thus the neutrons were generally well contained away from the workplace. The reactor core environment is characterized not only by high neutron levels, but also by very high gamma levels. The* ***gamma shielding is often water and concrete which are also very good neutron shields****. The neutron fields in the energy spectrum for reactors and lower will be attenuated much more quickly in concrete or water than will the gamma fields. This is not true for lead or iron, but these are usually not used as gamma shields where neutrons also exist. Thus* ***neutron fields are generally not a problem at an enclosed reactor****.* [Emphasis added] (TBD, pg. 25)

In Table 6B-2 if the TBD, NIOSH provides maximum annual missed dose values for different neutron dosimeter types (NTA and TLD) based on laboratory MRLs for three different time periods; (1) 1951–1958, (2) 1958–1976, and (3) 1976 to present. NIOSH believes these missed dose values are claimant favorable. The TBD provides these instructions for using the missed dose table:

> *To calculate the missed dose, the reconstructor must first determine if the person worked near neutrons and which category of neutrons. This can best be done by looking for the work location and whether a worker or others in the badge reporting group were assigned any neutron dose equivalent. The work location code for TRA where the MTR operated is 4 (also 40 to 45). If no neutron dose was assigned to him/her or to co-workers for several months, the dose reconstructor should assume that the person was not exposed to neutrons so no neutron dose would be missed.*
>
> *If a worker was likely exposed to neutrons, the reconstructor should assign missed neutron dose equivalent using Table 6B-3 for the times when workers did not have reported neutron dose. For the period when NTA film was used, the dose should be multiplied by 1.25 for all facilities except the MTR experimental floor and by 2 for the MTR experimental floor when the MTR was operating between 1953 and 1970. Then the dose equivalent is apportioned into the IREP groups using Table 6B-3.* (pg. 33)

The TBD gives an example (the only example in the 6 TBDs) to demonstrate how the missed neutron dose for a worker at MTR can be calculated. This represents a commendable strength in the TBD that would have been welcome elsewhere as well.

> *For example, if in 1955 a person was an experimenter at the MTR, and 7 of the weekly badges recorded a total of 185 mrem neutron dose equivalent, the missed dose would be 315 mrem [(52-7)×14÷2] so the total dose by the badges would be 500 mrem. Because the badge only sees about one-half the MTR neutron dose equivalent (from Section 6.3.4.3.1), the total dose equivalent is 1 rem. To convert the 1 rem received from neutrons on the MTR experimental floor to equivalent dose, multiply the total dose equivalent by the last column of Table 6B-3 to get 200 mrem to the <10-keV group, 60 mrem to the 10- to 100-keV group, 700 mrem*

to the 0.1- to 2-MeV group, and 280 mrem to the above 2 MeV group for a total equivalent dose of 1.24 rem (pg. 33).

For neutron organ dose evaluation, the TBD provides the following instruction:

> *The calculated neutron dose equivalent to organ dose in each energy group should be multiplied by the conversion factors from ambient dose equivalent to organ dose for AP irradiation from Appendix B of NIOSH 2002* (pg. 34).

NIOSH has included the following neutron dosimetry information in the TBD:

(1) The TBD indicates that neutron monitoring was not automatic for workers at INL facilities. As stated on page 17, "Kodak nuclear track emulsion-Type A (NTA) was used for neutron monitoring when the field radiation protection staff **requested** it. NTA responds to neutrons with energies above 500 to 800 keV, for which the proton recoil tracks leave enough energy to expose at least three (four in some references) grains of emulsion." [Emphasis added]

(2) This type of neutron detector responds to neutrons with energies above 500 to 800 keV. The TBD assigns MRL values used for neutron track emulsions Type-A (NTA) at INL for two different time periods:

a. Before 1958, MRL = 14 mrem
b. After 1959, MRL = 20 mrem

NIOSH's approach in the determination of these two MRLs is not sound, as they are based on the lowest recorded neutron doses in several dose cards.

(3) The TBD identifies that there were missed neutron doses below the NTA energy threshold of 0.5 to 0.8 MeV, particularly at plutonium facilities. Hankins (TLD) dosimeters were used at INL to capture thermal neutrons. The TBD provides INL facility neutron correction factors (FNCF) for 1981.

(4) The TBD states that assuming INL changed from NTA to Hankins albedo dosimeters in 1976 is claimant favorable.

(5) The TBD describes that the calibration of the NTA was done with a 30-Ci PoBe source in 1958. The uncertainties assigned were at the 90% confidence level. In order to compensate for the energy limitations of these emulsions, neutron pencil dosimeters were also used. In 1982, an AmBe source was used. In 1993, INL changed to an unmoderated Cf-252 source for calibration. The TBD lists in Table 6-4 the sources of laboratory bias in neutron calibration historically at INL.

> *Alpha particles from the americium or polonium interact in the reaction 9Be(α,n)12C, and generate a broad spectrum of neutrons up to about 11 MeV (**mean energy about 5 MeV**).* [Emphasis added.] (TBD, pp. 19–20)

(6) The TBD evaluates neutron radiation fields and measured neutron doses at INL facilities. 1n 1979, the TBD indicates that five people received neutron doses between 0.5 and 1 rem, and 79 received measurable doses below 0.5 rem. The TBD also represents (Figure 6-11) that only 54 out of 1,461 neutron dosimeters measured reportable doses in 1995, and that only 6 of the 54 reportable doses were above 35 mrem. The MRL for neutron TLDs was taken as 15 mrem.

(7) The TBD presents in Figure 6-12 the results of a relative bias study on neutron field measurements.

(8) The TBD identifies the sources of neutron exposure at INL, including the instrument calibration laboratories, neutron generators, and reactors. However, the major contributors are the Material Test Reactor (MTR) and the Test Area North (TAN) Fuel Storage Casks:

a. MTR – The MTR, which operated from 1952 to 1970, had beam ports and neutron beams extending onto a research floor.

> *The exception to the above discussion is the MTR* [Material Test Reactor]*, which operated from 1952 to 1970 and had beam ports and neutron beams extending onto a research floor. Also in this category are the Zero Power Physics Reactor (ZPPR) and Transient Reactor Test (TREAT), both at Argonne West. Some neutron surveys of the MTR experimental floor have been recovered (Sommers 1959, 1962; Hankins 1961), but these individually do not provide all components of the radiation field.* (TBD, pg. 27)

The supplement to this TBD describes in detail a 1961 neutron field study at MTR by Hankins. The TBD acknowledges that MTR personnel would have missed dose for neutrons with energies below 0.5 to 0.8 MeV. To correct for this missed dose for exposures on the MTR experiment floor, the TBD recommends multiplying the NTA results by 2 ±0.2 for a Monte Carlo dose reconstruction or by 3 for the less accurate worst-case dose reconstruction. The TBD also concludes that the total neutron dose equivalent is 0.58 ± 0.48 of the gamma dose equivalent on the MTR experiment floor. NIOSH believes many workers wearing NTA film badges would receive gamma dose at locations other than on the MTR experiment floor while the reactor was operating. For example, an RCT may be covering jobs with only beta gamma fields, and a craftsperson may be servicing pumps carrying radioactive water and not receive any neutron dose. The TBD recommends using a lower multiplying factor of gamma dose equivalent; 1.6 or 2.1. This approach is not claimant favorable.

b. TAN Fuel Storage Casks – Some fuel storage casks were located on a storage pad at TAN. The dose rates were 25–30-mrem/hr gamma and 40-mrem/hr neutron. The metal casks attenuate the gamma radiation, but not the neutron field, by a significant amount. The casks were moved to the Warm Shop for tests, where they remained for about 2 weeks before being moved back out to the pad. Neutron levels were discovered in the offices by an area TLD albedo system. Six people were not wearing albedo neutron badges. The estimated dose equivalent for full time occupancy is less than 50 mrem for each cask evolution taking into account attenuation by distance between the casks and office and by shielding from building concrete.

(9) The TBD summaries the typical neutron personnel dosimeter parameters important to $H_p(10)$ performance for the dose reconstruction in Table 6-7. NIOSH identifies the most important parameter as the difference between calibration and workplace neutron energy spectra.

(10) Table 6B-3 of the TBD lists the locations at the INL facilities where it believes exposure to neutron radiation is credible. These locations include instrument calibration laboratories, waste characterization facilities (RWMC, SWEPP, 14-MeV neutron generator), neutron source research laboratories, and reactors (MTR, ZPPR, and TREAT).

(11) The TBD presents the neutron weighting factors (function of energy) used to replace the neutron quality factors. With these weighting factors, INL corrected the dose equivalent to a newer dose quantity:

> *Table 6-8 lists the calculated fractions of dose equivalent in the IREP energy groups and the conversion factors from dose equivalent to equivalent dose for INEEL spectra. The ratios of average radiation weighting factor to average quality factor for the IREP energy groups have some variation, particularly for the 10-100 keV group where the energy dependence of the fluence is radically different for the fission and 14 MeV source than for the reactor spectrum. The lower part of the table lists the recommended default values for the dose equivalent fractions and quality factor corrections.* (TBD, pg. 31)

(12) The TBD estimates the uncertainties associated with neutron measurements as follows:

> *The total uncertainty for neutrons is probably larger at about 60% at one sigma. The cause of the greatest uncertainty for neutrons is the variation of dose caused by an organ's position in the body. For 1 MeV neutrons, the dose facing the source is about a factor of 1000 higher than the dose on the back side of a 30 cm diameter sphere of tissue- like material. In a work environment, the direction of the neutrons may be unknown, but it is often from many directions which reduces the impact of this uncertainty driver.* (TBD, pg. 34)

(13) NIOSH provides a TIB supplement to this TBD. This TIB reanalyzes the neutron field measurements performed by Dale Hankins using Bonner Balls in 32 locations around

MTR at INL in 1961. The Hankins data were reanalyzed using detector responses calculated by Hertel and Davidson for 171 energy groups from thermal to 17.3 MeV. Neutron dose equivalent (mrem/hr) values were determined for these field measurements and are listed in Table 1 and Table 2. The comparisons (ratios) of the different components of the neutron dose equivalent rate are provided in Table 3. The NTA neutron dosimeters used at MTR respond only to neutrons above 0.5 to 0.8 MeV. The TIB also presents the dose equivalent rate, the fraction of the dose equivalent, the ratios of the neutron ambient dose equivalent and the neutron dose equivalent, and the corresponding NTA response in each of the IREP energy regions in Table 4.

In summary, the TBD draws the following conclusions:

(1) NTA dosimeters were used to monitor workers for neutron exposure between 1951–1976

(2) Monitoring of workers was performed only at the request of radiation protection staff

(3) Minimum reportable levels (MRLs) for neutron exposures were defined at 14 mrem (1951–1958) and 20 mrem (1959–1976)

(4) Before 1982, calibration of NTA dosimeters was performed with PoBe source with a broad neutron energy range of up to 11 MeV with a mean energy of 5 MeV

(5) Principal sources of neutron exposure to INL workers were the instrument calibration laboratories and 14 MeV neutron generators. Because most of the INL reactor fuel cores were "shielded" by water and concrete and had no beam ports, neutron fields created by reactors were ". . . generally not a problem."

5.1.4.2.1 Minimum Reporting Level

NIOSH did not use the detection threshold levels of the NTA emulsion and the albedo TLD dosimeters as the MRL values for the missed neutron dose calculation. Instead, NIOSH reviewed a limited number of neutron dose data sheets to identify the lowest dose assigned to the worker. This lowest dose was then used as the MRL value. For instance, NIOSH reviewed one data sheet from March 1958 and determined that the MRL, for the period between 1951 and 1958, was 14 mrem, because that was the lowest reading of the 10 neutron readings on the data sheet. In the other instance, NIOSH reviewed three data sheets, one from November 1959, another from January 1962, and the third from April 1959. These three data sheets show a total of six neutron readings; 10, 20, 20, 20, 20, and 40 mrem. NIOSH determined the MRL value to be 20 mrem. This approach is not thorough and supported. Ten neutron readings in one data sheet from March 1958 could not represent reasonably the MRLs from 1951 to 1957. It is possible that the MRL was higher before 1959. Similarly, it is not reasonable to use six neutron readings to represent all neutron measurements between 1959 and 1976. NIOSH needs to evaluate the history of MRL values in those years in more detail.

In addition, for NTA film dosimeters, NIOSH's values of 14 mrem and 20 mrem appear low and are inconsistent with generic values given for NTA dosimeters, as well as values cited by other

DOE facilities with similar neutron source terms. For example, Section 2.2.2.1 of OCAS-IG-001 provides the following generic MRL value as given in the following statement:

> *Watson (1959) discussed the neutron monitoring practices at the Hanford facilities and reported the limit of detection/reporting limit for the neutron dosimeter in the early 1950s was 90 mrem.*

For the Rocky Flats Plant, Table 6-17 of ORAUT-TKBS-0011-6 cites the following period-specific limits of detection.

Period	LOD
1951–1958	400 mrem
1959–1960	128 mrem
1961	120 mrem
1962–1963	369 mrem
1964	320 mrem
1966–1970	120 mrem

Several major DOE sites, including Hanford and Savannah River, have acknowledged the limitations and deficiencies of NTA film, along with the recommendation of using location-specific neutron-to-photon ratios as a surrogate means for assigning neutron exposure. As for the albedo TLDs, NIOSH does not provide any information concerning the detection threshold values or MRL used from 1976 to the present time. NIOSH should provide supporting data for the MRL value of 15 mrem listed in Table 6B-2.

Similarly, the TBD does not provide any information concerning uncertainties associated with these MRL values. It is very important for NIOSH to provide guidance to the dose reconstructors in arriving at a set of claimant favorable MRLs for the missed dose calculations, especially when the approach used to determine the MRLs is not very sound.

5.1.4.2.2 Failure to Properly Address Neutron Exposures at INL Reactor Facilities

The TBD presumes that neutron exposures at INL's reactors are ". . . generally not a problem" and may, therefore, be ignored. But INL had a total of 52 reactors, most of which were experimental/prototypes in design. For example, in the 1950s, INL began testing of experimental reactors as part of its Aircraft Nuclear Propulsion (ANP) Program. Between 1953 and 1961, three reactor assemblies were used for a variety of tests that evaluated reactor control systems, various nuclear fuels, and the consequences of potential system failures. The testing program for the three HTRE (Heat Transfer Reactor Experiment) assemblies was designated as Initial Engine Tests (IETs). There were a total of 26 IETs in all. Due to the very nature of unprecedented engineering designs and ANP test objectives, these reactor prototypes were designed for high power densities, and with minimum shielding and neutron moderation.

It is understandable that there are difficulties in assessing neutron spectra for this complex array of reactor facilities/operations for the purpose of defining correction factors for NTA film dosimeters. However, the current approach for assessing neutron exposures at INL reactor facilities is clearly inadequate. NIOSH should revise the missed neutron dose approach in this TBD.

5.1.4.2.3 Neutron Calibration Deficiencies

Based on information presented in ORAUT-TKBS-0007-6, NTA dosimeters were exclusively calibrated by a PoBe source, which generates high-energy neutrons. While this may provide a suitable calibration for select dosimeters used to monitor workers at these calibration laboratories, for other neutron sources with lower energy spectra, neutron doses must be expected to be significantly under-recorded (if recorded at all). In brief, NTA dosimeters must be calibrated to a neutron spectrum that closely resembles that of a given workplace. NIOSH needs to evaluate the missed neutron dose to workers because of this calibration deficiency.

5.1.4.2.4 Completeness and Quality of INL Neutron Dosimetry and Record-Keeping Programs

Similar concerns about the completeness and quality of INL dosimetry exist for neutron measurements as for gamma and beta measurements. The opening discussion of Section 5.1.4.1.1 and the following two sections on gamma and beta dosimetry apply as well to neutron dosimetry. Again, NIOSH should investigate the deficiencies mentioned with respect to detection of all three types of radiation.

5.1.4.2.5 Uncertainty Estimation

The main difficulty in assigning uncertainties to neutron measurements is the strong energy dependence of the neutron detection systems. To estimate the uncertainties for the INL dosimetry systems, it is necessary to know how these systems were calibrated. External Dose TBD Table 6-2 (page 18) gives the Facility Neutron Correction Factors (FNCF) by facility; however, it is not clear from the text if these factors were previously applied to the recorded neutron dose, or whether they should be applied now by the dose reconstructor. It is not explained in the text how these FNCFs were obtained. If the albedo dosimeter were well "calibrated" for each INL workplace, then the uncertainties applied for neutron dosimetry in the field (and not in the calibration laboratory) would be on the order of 70%–100%.

5.1.4.2.6 Neutron Organ Dose

For the dose reconstructor to be able to assess the organ doses, it is necessary to make a neutron spectrum breakdown for each INL facility with neutron fields into five energy groups; $E < 10$ keV, $10 < E < 100$ keV, $0.1 < E < 2.0$ MeV, $2.0 < E < 20.0$ MeV, and $E > 20.0$ MeV. It seems that the MTR has been well characterized in terms of neutron spectra. However, similar information on other reactor projects, including ZPPR and TREAT, is missing. NIOSH should provide information and guidance for dose reconstruction for workers at the latter two reactors.

5.1.4.2.7 High-Risk (Dose) Jobs

The TBD has indicated that sources of neutron exposure at INL include neutron sources at the instrument calibration laboratories and 14-MeV neutron generators used to characterize waste. NIOSH believes most of the reactors built at INL had no beam ports, and neutrons were generally well contained and kept away from the workplace. NIOSH also believes that neutron fields will be attenuated much more quickly in concrete or water than will gamma fields, so neutron fields are generally not a problem at an enclosed reactor. However, the TBD identifies MTR, APPR, and TREAT as potential problem neutron sources because of their beam ports and neutron beams. In fact, other reactor facilities, like ETA and ATR, also had neutron beams. NIOSH should treat these facilities as well.

In addition, NIOSH should develop a list of high-risk (dose) jobs at INL facilities, and provide corresponding neutron dose rates and worker job doses. This information will be helpful for dose reconstructions. Since there are also many employees working at these facilities who may not have been monitored by wearing neutron dosimeters, it is essential for dose reconstruction to develop missed dose for unmonitored workers.

5.1.4.2.8 Multiplying Factors for Missed Neutron Dose

For calculating missed neutron dose when the MRL for NTA film is taken from Table 6B-2, the TBD indicates the neutron dose should be multiplied by 1.25 for most workers or by 2 for workers on the MTR experimental floor. NIOSH should provide data to support these two factors.

In addition, the TBD recommends assigning 50 mrem of neutron dose to the few office workers without neutron dosimetry at the TAN Warm Shop office for each period they were exposed to the fuel storage casks. NIOSH should provide data to support this fixed missed neutron dose default value.

5.2 OTHER OBSERVATIONS

In addition to the specific findings and observations that appear in other sections of this report, there are several general observations that can be made. These can be distilled into the following two broad observations:

Observation 1: The content and organization of the INL Site Profile TBDs are quite uneven, where useful default values and specific guidance for dose reconstructors are often buried among general discussions about site activities, history, and technical information about dosimetry, which may or may not have practical application. In addition, there are an insufficient number of INL site-specific TIBs to provide guidance similar to those available for the Savannah River Site. There is only one (in the occupational neutron missed dose section) example given in the TBDs to help dose reconstructors apply the guidance. In some cases, there is no explanation why a default value is recommended and why it is deemed by NIOSH to be claimant favorable. The INL dose reconstruction workbook, which NIOSH provided to SC&A after the site profile TBD review process began, is just a large spreadsheet containing dose equations and default values in various cells. It is not readily comprehensible or user-friendly. There are no help pages to assist dose reconstructors in walking through data entry, default values and parameters selection, and dose calculation process. We also understand that there is no accompanying manual to the spreadsheet. While we know that the spreadsheet is intended for viewing and use only by the cognoscenti performing claimant reviews, some guidance and explanation should be provided to ensure that it is being used correctly.

Observation 2: The four authors of the INL TBDs and TIB are the most site-specific experienced and knowledgeable technical experts working on the development of the INL site profile. These four authors were intimately involved in the development, management, and implementation of the INL Radiological and Dosimetry programs in the past and the present. They understandably demonstrate confidence in the INL radiological protection programs. They relied heavily on their own experience and knowledge, and past INL environmental reports and dosimetry records in developing the TBDs, and used many of the reports that they personally authored as reference material. They did not interview any present or retired workers before they completed their work. They did not review any safety-related audit reports conducted by DOE-HQ and DNFSB, but they did interview some of the current INL staff and requested all dosimetry records.

6.0 OVERALL ADEQUACY OF THE INL SITE PROFILE AS A BASIS FOR DOSE RECONSTRUCTION

The SC&A site profile review procedure calls for both a "vertical" assessment of a site profile for purposes of evaluating specific issues of adequacy and completeness, and a "horizontal" assessment of how the profile satisfies its intended purpose and scope. This section addresses the latter objective by evaluating (1) how, and to what extent, the site profile satisfies each of the five objectives defined by the Advisory Board for ascertaining adequacy; (2) the usability of the site profile for its intended purpose (i.e., to provide a generalized technical resource for the dose reconstructor when individual dose records are unavailable); and (3) generic technical or policy issues that transcend any single site profile that need to be addressed by the Advisory Board and NIOSH. As mentioned in the Introduction, the practice of addressing the same items from several different perspectives has led to some redundancy in the report.

6.1 SATISFYING THE FIVE OBJECTIVES

The SC&A review procedure, as approved by the Advisory Board, requires that each site profile be evaluated against five measures of adequacy; (1) completeness of data sources, (2) technical accuracy, (3) adequacy of data, (4) site profile consistency, and (5) regulatory compliance. The SC&A review of the INL Site Profile finds that the profile generally satisfies these objectives, although several shortcomings and potential issues of varying significance need to be addressed. Many of the issues involve a lack of sufficient conservatism in key assumptions or estimation approaches, incomplete analyses of data, or incomplete reflection of operational or dosimetric history. Key issues are summarized below and in Table 6-1 (which is a duplicate of Table 1-1), which provides a matrix representation of the identified issues sorted according to the SC&A findings and observations. Detailed evaluation and discussion of these issues is provided elsewhere in this report.

An "X" in the table indicates significant shortfalls in meeting the corresponding review objectives for the indicated topics in the INL Site Profile. These shortfalls have been discussed either within the text of the findings themselves, or, in many cases, in special sections that address one or more of these shortfalls. The first column of the table indicates the primary place within the report that treats each issue. The last column of the table presents three categories of potential related regulatory non-compliance concern for the listed issues. These three categories are defined again briefly as follows:

- Category 1: Least challenged by any deficiencies in available dose/monitoring data are dose reconstructions for which even a partial assessment (or minimized dose(s)) corresponds to a probability of causation (POC) value in excess of 50%, and assures compensability to the claimant.

- Category 2: The use of upper bound values is limited to those instances where the resultant maximized doses yield POC values below 50%, which are not compensated. For this second category, the dose reconstructor needs only to ensure that all potential internal and external exposure pathways have been considered.

- Category 3: The most complex and challenging dose reconstruction represents claims where the case cannot be dealt with under one of the previous two categories and a detailed analysis is required.

Table 6-1: Issue Matrix for the INL Site Profile

Descriptions (a)	Issue Classification	Objective 1: Completeness Of Data	Objective 2: Technical Accuracy	Objective 3: Adequacy of Data	Objective 4: Site Profile Consistency	Objective 5: Regulatory Compliance (b)
Issue 1: (5.1.1.1) Routine Airborne Releases	Finding (5)	X	X	X	X	Category 2
Issue 2: (5.1.1.2) Episodic Airborne Release	Finding (6)	X	X	X	X	Category 2
Issue 3: (5.1.1.3) Direct Gamma Exposures	Finding (7)	X	X	X	X	Category 2
Issue 4: (5.1.2.1) Completeness and Quality of INL Internal Dosimetry Programs	Finding (8)	X	X	X	X	Category 3
Issue 5: (5.1.2.2) High-Risk Jobs (Internal Exposure)	Finding (9)	X				Category 1
Issue 6: (5.1.2.3) Calibration of Internal Dosimetry Analytical and Monitoring Equipment	Observation		X		X	
Issue 7: (5.1.2.4) Changes of Internal Dose Limits	Observation		X			
Issue 8: (5.1.2.5) High Fired Plutonium and Uranium Intakes	Finding (10)	X				Category 1
Issue 9: (5.1.2.6) Skin and Facial Contamination	Observation	X				Category 3
Issue 10: (5.1.2.7) Breathing Rates	Observation	X				
Issue 11: (5.1.2.8) Non-Occupational Worker Elimination of DU Background	Finding (11)	X	X	X		Category 2
Issue 12: (5.1.2.9) Unmonitored Workers	Observation	X				
Issue 13: (5.1.2.10) Naval Reactor Facility Workers	Observation	X				Category 2
Issue 14: (5.1.2.11) Plutonium Monitoring	Observation	X	X	X		Category 1
Issue 15: (5.1.3) SL-1 Accident Dose Reconstructions	Finding (1)	X	X	X		
Issue 16: (5.1.4.1.1) Completeness and Quality of INL Beta/Gamma Dosimetry and Record Keeping Programs	Finding (8)	X		X	X	Category 3
Issue 17: (5.1.4.1.2) Penetrating and Non-Penetrating Doses	Finding (4)	X	X	X	X	Category 3
Issue 18: (5.1.4.1.3) Correction For Beta Doses	Observation		X		X	
Issue 19: (5.1.4.1.4) Angular Dependence Correction Factor for Gamma Dose	Observation	X				
Issue 20: (5.1.4.1.5) Restating Beta Dose As Gamma Dose	Observation		X			
Issue 21: (5.1.4.1.6) Photon Spectrum Split	Observation		X		X	
Issue 22: (5.1.4.1.7) Immersion Dose	Observation	X	X	X		
Issue 23: (5.1.4.1.8) High-Risk Jobs (Beta/Gamma Exposure)	Finding (9)	X				Category 1
Issue 24: (5.1.4.1.9) Extremity Dose	Observation	X				Category 2
Issue 25: (5.1.4.1.10) Discrepancies between PIC and Film Reading	Observation	X				
Issue 26: (5.1.4.1.11) Minimum	Observation			X	X	

Table 6-1: Issue Matrix for the INL Site Profile

Descriptions (a)	Issue Classification	Objective 1: Completeness Of Data	Objective 2: Technical Accuracy	Objective 3: Adequacy of Data	Objective 4: Site Profile Consistency	Objective 5: Regulatory Compliance (b)
Detection Limit						
Issue 27: (5.1.4.1.12) Minimum Reporting Level (Beta/Gamma)	Finding (3)	X	X	X		Category 2
Issue 28: (5.1.4.2.1) Minimum Reporting Level (Neutron)	Finding (3)	X	X	X		Category 2
Issue 29: (5.1.4.2.2) Failure to Properly Address Neutron Exposures	Finding (2)	X	X	X	X	Category 1
Issue 30: (5.1.4.2.3) Neutron Calibration Deficiencies	Finding (2)	X	X	X		Category 3
Issue 31: (5.1.4.2.4) Completeness and Quality of INL Neutron Dosimetry and Record Keeping Programs	Finding (8)	X	X	X	X	Category 3
Issue 32: (5.1.4.2.5) Uncertainty Estimation for Neutron Doses	Observation	X	X	X		
Issue 33: (5.1.4.2.6) Neutron Organ Dose	Observation	X				
Issue 34: (5.1.4.2.7) High-Risk Jobs (Neutron Exposure)	Finding (9)	X				Category 1
Issue 35: (5.1.4.2.8) Multiplying Factors for Missed Neutron Dose	Observation	X	X	X	X	Category 3

Table Notes:

(a) Report section numbers discussing the issues are given after the issue number.

(b) Category 1: Least challenged by any deficiencies in available dose/monitoring data are dose reconstructions for which even a partial assessment (or minimized dose(s)) corresponds to a probability of causation (POC) value in excess of 50%, and assures compensability to the claimant.

Category 2: The use of upper bound values is limited to those instances where the resultant maximized doses yield POC values below 50%, which are not compensated. For this second category, the dose reconstructor needs only to ensure that all potential internal and external exposure pathways have been considered.

Category 3: The most complex and challenging dose reconstruction represents claims where the case cannot be dealt with under one of the two other categories.

6.1.1 Objective 1: Completeness of Data Sources

The breadth of data sources used as a basis for the INL Site Profile is evident in the 287 reports, papers, and other documents cited as references, including a number of authoritative historical documents dating back to the start of site operations in the early 1950s. Based on a review of the INL Site Description TBD (Rohrig 2004s), it is evident that NIOSH effectively compiled and characterized activities and operations at 14 areas and 101 facilities and processes. In fact, this review cites the breadth of operational data provided in several places as a strength. Also noteworthy is the use of the minimum reporting levels as a simple approach (Taulbee 2002) for calculating missed external doses for workers that are provided in Attachment 6B of the Occupational External Dose TBD (Rohrig 2004e). Notwithstanding the general excellence of the data sources, SC&A identified a number of areas to be deficient:

(1) The workers interviewed by SC&A as part of the site expert interview process (Attachments 2 and 3) characterized the site as having areas and jobs capable of delivering acute doses to workers. SC&A found that the TBDs lack separate characterization and in-depth consideration of high-risk/dose (acute) jobs at INL facilities. Although extensive descriptions of key operations and processing facilities are included

in the TBD, high-risk or high-dose jobs in the INL facilities have not been evaluated and potential missed doses are not considered.

(2) SC&A found a lack of characterization of potential missed neutron exposures to workers at the INL reactors in the Occupational External Dose TBD, even though INL had a total of 52 reactors, most of which were experimental/prototype in design. These reactors often were built with high power densities and with minimum shielding and neutron moderation. It is inadequate to presume that there are no missed neutron doses at INL reactors.

(3) SC&A found a lack of characterization of contaminated soil or materials stored outdoors at INL facilities, such as RWMC or ICPP, in the Occupational Environmental Dose TBD. For example, dry contaminated evaporation ponds at the reactor and fuel processing facilities are not characterized. The contaminated soils or materials that are present may be resuspended by blowing winds and vehicular activities, and inhaled by unmonitored employees working outside the facility buildings. In addition, the TBD did not evaluate the adequacy of the environmental monitoring instrumentation and validity of its collected data.

(4) SC&A found a lack of characterization of potential worker exposures at the High-Level Liquid Waste Tank Farms (at TAN, ICPP, and TRA), and in remediation and waste management in general. The list of radionuclides provided for those operations is incomplete and increases the potential for missed dose.

(5) The authors of the TBDs demonstrate their general confidence with the radiological protection practices, environmental monitoring programs, internal dosimetry programs, external dosimetry programs, analytical laboratory programs, quality assurance programs, and above all, the dosimetry record-keeping systems at INL over the entire operating history. Perhaps as a consequence, the TBDs do not consider potential missed doses due to deficient work practices and missing worker dose records. NIOSH did not request and review field data and facility specific records, including field logbooks, RWPs, PEQs, incident reports, occurrence reports, and contamination reports that may provide data and information for missed worker doses. It is important for NIOSH to validate that all pertinent records (such as incident report, personnel contamination reports, over-exposure reports, doses received at other DOE facilities due to temporary assignments, and other records essential for the dose reconstruction) are included in the worker files provided by DOE to the claimants.

(6) SC&A also found an inadequate characterization of worker internal exposures. The site profile evaluated the recorded worker internal doses only for the period between 1992 and 2000, which does not cover the more problematic early years of the INL site operation from 1949 to 1991. Instead, the TBD evaluated urinalysis results for 1959, 1960, and 1961, and whole-body counting results for 1963. The TBD also provides in vitro for urine samples and in-vivo MDAs for the entire site history. These data are not sufficient to provide any comparison or support for the missed worker internal doses.

(7) The TBDs do not characterize or provide any information on the potential missed worker doses due to extremity exposure, skin contamination, facial contamination, and ingestion. Many of these potential missed worker doses could be found at various facilities, such as ICPP and SMC.

(8) The TBDs also lack characterization of the potential missed internal and external doses for the hundreds of rescue and cleanup workers involved with the SL-1 accident that occurred in January 1961. It is a fact that the equipment used and the radiological control policies in that era were not as advanced and protective as those currently available. The TBDs should provide adjustment factors for stay-time used, dose field estimates, internal dose results, external dose readings, and contamination level estimates.

6.1.2 Objective 2: Technical Accuracy

There are a number of issues identified in the course of this review that may be classified as deficiencies in technical accuracy:

(1) The derivation of the background value of 0.16 µg/L used for subtraction from each urinalysis result of uranium prior to assessment of occupational internal dose for SMC radiation workers is not technically sound. The baseline background (population) intake value was determined by a study of urine samples submitted by non-radiation employees working at the SMC facility. A better approach would be to use urine samples from non-INL people in the Idaho Falls area, far removed from any sources of radioactivity. This approach would not create a suspected bias due to uranium intake through various pathways (inhalation and ingestion) by non-radiation workers while working at the SMC facility. During the site expert interviews, the dosimetry staff indicated that they tried to use residents from the Idaho Falls area, but no one was willing to sign a liability waiver form. In a subsequent study, they used 16 non-radiation workers from the CFA. In addition, the selected background value (0.16 µg/L) is significantly higher than the national average background value.

(2) There are deficiencies in the neutron calibration program at INL. Due to the use of a PoBe source for neutron calibrations, dosimeters would significantly under measure neutron doses from sources with lower energy spectra. These deficiencies could cause significant missed neutron doses.

(3) The technical accuracy of the approach suggested for estimating missed internal doses for workers exposed to plutonium or uranium whose intakes (mainly inhalation) were assessed mainly by in-vivo counts is not scientifically established. The approach is not persuasive in view of the varying age and isotopic composition of plutonium or uranium at the INL site (similar to Hanford).

(4) The overall approach of the Occupational Internal Dose TBD is to use the significant radionuclides for ICPP processed fuels as the most limiting (bounding) source terms for the calculation of missed worker internal doses. These source terms were suggested to be well-tagged with beta-emitting radionuclides, which allowed beta/gamma-detecting

CAMs to be used at ICPP. As a result, all possible alpha contamination or internal exposures would have been detected and monitored. In reality, there are facilities and areas at INL, such as the calciners, where alphas sources predominate and beta-emitting radionuclides cannot be used as a tag. Hence, the overall approach reported in the TBD is not sound and bounding, because it ignores the fact that there were shortcomings in the earlier CAM systems and also deficiencies in the internal exposure control work practices at ICPP and other INL facilities. Many site experts interviewed indicated that there were significant incidents where internal over-exposures and personnel contamination occurred at INL facilities that were not monitored or documented.

(5) The MESODIF model (mesoscale isopleths) used in the Occupational Environmental Dose TBD to determine the ground radionuclide concentrations from airborne releases due to routine operations or episodic events from INL facilities was found to be deficient by the DOE-HQ Tiger Team in 1991 (DOE-HQ 1991). These concentrations were used to calculate the default worker intake values in different INL facilities. Therefore, the validity of default intake values provided in Tables 4-1 through 4-12 is jeopardized. NIOSH should evaluate the deficiencies in the model and determine the associated uncertainties. Specifically, the TBD did not evaluate the doses to workers outdoors associated with episodic releases to the atmosphere.

(6) The Occupational Environmental Dose TBD uses the fence-line TLD results provided in the environmental monitoring data reports to determine the direct gamma doses from airborne releases and their cumulative ground depositions to personnel working outdoors. This approach assumes all workers working outdoors at a specific facility would receive an average direct gamma dose from normalized ground concentrations. If the assumption were valid, the fence-line TLD results should be adjusted by multiplying a weighting factor to account for uncertainties in TLD sensitivity and geometry. However, this approach is not entirely valid, because it does not take into account the most limiting scenarios, i.e., (1) outdoor workers may become immersed in the plume of routine or episodic releases from the facility stack; (2) outdoor workers may inhale resuspended cumulative ground radionuclide depositions; and (3) the cumulative ground concentrations inside the fence line are generally higher than that near the fence line. The fence-line TLDs are too far from the bounding source terms to represent the actual direct gamma doses received by the outdoor workers. Therefore, this TBD approach is not claimant favorable.

(7) The approach in calculating the missed external doses (both gamma and neutron) by using the minimum reporting level equation is not claimant favorable. This suggested approach is based on OCAS-IG-001 (Taulbee 2002), which assumes a statistical average dose value (MRL/2) for all workers. However, the dose reconstruction for a particular worker, especially in a compensable case (with cancer), should use the bounding dose value (i.e., MRL instead of MRL/2). In addition, the development of the MRL values was not comprehensive in the Occupational External Dose TBD, and NIOSH's MRL values of 14 mrem and 20 mrem appear low and are inconsistent with generic values given for NTA dosimeters, as well as values cited by other DOE facilities with similar neutron source terms and dosimeters.

6.1.3 Objective 3: Adequacy of Data

(1) Questions regarding data adequacy, where they arise in the SC&A evaluation, have largely focused on the adequacy of available facility release data and fence-line TLD dose values for estimating occupational environmental doses to unmonitored outdoor workers, who may have been immersed in release plumes, inhaled resuspended contaminated soils, and exposed to direct gamma radiation from accumulated ground depositions.

(2) Since the Occupational Internal Dose TBD does not use dosimetry data, SC&A focused on the adequacy of processed fuel characteristic data for estimating missed internal doses to unmonitored workers, who more likely have had missed doses due to inconsistent monitoring in the field in the early monitoring programs, from high-risk jobs, or unplanned intake or contamination incidents. The same is true for missed gamma and neutron doses.

(3) SC&A also found that the TBDs do not fully explore and develop procedures and guidelines to dose reconstructors that would lead them to focus on gaps in environmental, internal, and external doses that could lead to a significant underestimate of worker dose. Input from interviewed site experts indicates that there were situations where reactor workers were not provided neutron dosimeters or were not monitored on a continual basis, and where processing facility workers were not monitored when they had positive nose smears. Data either presented in the TBDs, or on which the TBDs are based, cannot be considered adequate unless an evaluation is conducted of the comprehensiveness of the neutron-monitoring and bioassay programs, and to what extent existing dose estimation assumptions and methodologies address this potential missed dose.

(4) The lack of actual worker bioassay data during the 1949−1991 period represents an important area in which adequacy of data is of concern. There are no data provided in the TBD concerning bioassay of high-fired plutonium and uranium. There may be workers with potentially high exposures during that period that cannot be adequately reconstructed, especially when worst-case dose reconstructions are required. Lack of knowledge of uncertainties in the actual bioassay techniques and instruments used to quantify internal dose and the minimum detectable activity (MDAs) represents an area of data inadequacy that can lead to significant underestimates of worker dose in this period.

(5) The source term list of radionuclides provided for the current and past environmental restoration and waste management projects at INL is incomplete and may contribute to missed dose. Risks of exposure to radionuclides that workers have encountered when retrieving and processing RWMC's stored TRU waste have not been adequately addressed.

(6) INL documented worker exposure incidents and unusual occurrences in facility record files. Although INL has significant quantities of personnel monitoring data, as well as field radiological control data, there are considerable gaps in the information. In many cases, some of this information may not be kept in the worker exposure files. In addition,

there are problems with the adequacy of data, particularly with regard to worker intake, high beta exposure, and extremity dose for high-risk (dose) jobs in some facilities, such as ICPP and reactors.

(7) INL maintained worker exposure records by including (some) dosimetry results and facility incident reports in the worker files. There are problems with the completeness of these worker files, however, with regard to worker intake results, external dose results, extremity doses, and contamination reports.

6.1.4 Objective 4: Consistency Among Site Profiles

(1) While INL, Hanford, and the Savannah River Site (SRS) had some missions that were similar, marked distinctions existed and continue to exist in facility design, operations, operational history, and radiological practice. NIOSH has appreciated this distinction and tailored its TBD assumptions and analytic approaches to the unique histories and conditions at the three sites, while mirroring those assumptions and approaches where justified. Both the Hanford and SRS site profiles predate the INL Site Profile; therefore, NIOSH benefited greatly from the early efforts at Hanford and SRS, and was able to remedy many of the apparent inconsistencies, especially in the SRS TBDs.

(2) Attachment 4 of this SC&A report provides, in tabular form (Tables A1–A4), an evaluation and comparison of the default assumptions for each element of exposure (i.e., occupational medical dose, occupational internal dose, occupational external dose, and occupational environmental dose) of INL, Hanford, and Savannah River. The lapses in consistency noted by SC&A include inconsistent methodologies and assumptions regarding external, internal, and environmental dose for almost identical monitoring and exposure conditions at the three sites. This comparison shows that the INL TBDs did not provide as much characterization of the default internal dose parameters and external exposure factors as the Hanford and SRS TBDs.

(3) An extensive comparison was performed by SC&A to compare and contrast the methodologies used in the INL, Hanford, and SRS TBDs to determine external dose. This comparison focuses on the methodologies and assumptions associated with dose assessments and the derivation of values used to obtain a POC for individual claimants. A detailed analysis is provided in Table A-4 of Attachment 4 to this report. This table notes where the INL, Hanford, and SRS site profiles differ or agree on a number of important assumptions. In summary, where inconsistent approaches or methods exist, they typically represent some lapses present in the INL TBD, despite NIOSH's experience with Hanford, SRS, and earlier reviews.

6.1.5 Objective 5: Regulatory Compliance

NIOSH has complied with the hierarchy of data required under 42 CFR Part 82 and its implementation guides. However, SC&A has identified some significant shortcomings of the data used in the review process of the INL Site Profile that may lead to dose reconstructions that are not claimant favorable. It is especially crucial for NIOSH to re-evaluate the technical basis of the missed dose assumptions used in the TBDs.

6.2 USABILITY OF SITE PROFILE FOR INTENDED PURPOSE

SC&A has identified seven criteria that reflect the intent of the Energy Employees Occupational Illness Compensation Program Act of 2000, the Final Rule, and the regulatory requirements of 42 CFR Part 82 for dose reconstruction. Because the purpose of a site profile is to support the dose reconstruction process, it is critical that the site profile assumptions, analytic approaches, and procedural directions be clear, accurate, complete, and auditable (i.e., sufficiently documented). SC&A used the following seven objectives to guide its review of the INL Site Profile to determine whether it meets these criteria:

- Objective 1 − Determine the degree to which procedures support a process that is expeditious and timely for dose reconstruction

- Objective 2 − Determine whether procedures provide adequate guidance to be efficient in select instances where a more detailed approach to dose reconstruction would not affect the outcome

- Objective 3 − Assess the extent to which procedures account for all potential exposures and ensure that resultant doses are complete and are based on adequate data

- Objective 4 − Assess procedures for providing a consistent approach to dose reconstruction, regardless of claimants' exposures by time and employment locations

- Objective 5 − Evaluate procedures with regard to fairness and the extent to which the claimant is given the benefit of the doubt when there are unknowns and uncertainties concerning radiation exposures

- Objective 6 − Evaluate procedures for their approach to quantifying the uncertainty distribution of annual dose estimates that is consistent with and supports a DOL POC estimate at the upper 99% confidence level

- Objective 7 − Assess the scientific and technical quality of methods and guidance contained in procedures to ensure that they reflect the proper balance between current/consensus scientific methods and dose reconstruction efficiency

The following addresses these objectives:

(1) The INL Site Profile does a very good job in the Occupational Medical Dose TBD in assessing the potential organ doses to workers who received medical x-ray examination over the entire operating history. The default organ dose values can be easily identified and used from Table 3A-1 (Rohrig 2004m). One potential opportunity to improve the validity of the dose assessment is to include a multiplying factor accounting for uncertainties in equipment parameters.

(2) The INL Occupational Environmental Dose TBD presents a thorough compilation of the airborne releases from routine operations and episodic events at INL facilities. The TBD also compiles extensive information on the fence-line direct gamma doses for different INL facilities. The default worker inhalation intake and direct gamma doses to workers can be determined and used from Tables 4-1 through 4-13 (Peterson 2004). The missed environmental doses from the airborne releases and fence-line gamma doses should not be significant. However, the TBD has not considered several important potential missed environmental dose streams, including unmonitored workers inhaling resuspended contaminated soils or materials.

(3) The INL Occupational Internal Dose TBD (Rich and Wenzel 2004) does not mention the use of ORAUT-OTIB-0011 (Siebert 2004) for calculating doses from tritium and estimating missed doses from this nuclide. There was a small amount of tritium produced at INL facilities in early years. Dose reconstructors should be alerted to the use of this technical information bulletin (TIB).

(4) The INL TBD does not address the opportunity to use surrogate (i.e., reference mix) radionuclides when data are not available for a less commonly encountered radionuclide and thus the 95^{th} percentile cannot be applied in estimating the upper bounds of a like dose.

(5) The INL Internal Dose TBD does not mention the use of an approach recommended for other similar DOE facilities when determining maximum dose. This approach is provided by NIOSH in ORAUT OTIB-0002 (Rollins 2004).

(6) The Occupational Internal Dose TBD does not provide bioassay data for the operating years before 1992. For the purpose of compiling data needed to reconstruct internal doses based on historical operation, NIOSH amassed a considerable amount of data describing radionuclides and operations at the various facilities and their associated processes. However, NIOSH does not give adequate (or explicit) guidance to dose reconstructors on how to navigate through the complex mix of radionuclides required to reconstruct historical internal exposures to workers. There are opportunities for improvement in the data sets and instructions to the dose reconstructors with respect to reconstructing internal exposures.

(7) The INL Occupational External Dose TBD does not provide external dose data for the operating history of the INL site. For the purpose of compiling data needed to reconstruct external doses (gamma, beta, and neutron), NIOSH evaluates the MRL values of different dosimetry systems (films, TLDs, and neutron dosimeters) used at INL facilities. In addition, the TBD provides information on external radiation fields at different facilities. However, NIOSH does not give guidance to dose reconstructors on how to use these data to reconstruct historical external exposures to workers from high-risk jobs, unplanned over-exposure incidents, and deficient monitoring in the field. There are opportunities for improvement in the data sets and instructions to the dose reconstructors with respect to reconstructing external exposures. One important

opportunity for improvement is to include instructions to the dose reconstructors on how to treat missed beta exposures.

(8) The INL Occupational External Dose TBD does not provide guidance to dose reconstructors in assessing missed neutron doses at INL reactors.

6.3 UNRESOLVED POLICY OR GENERIC TECHNICAL ISSUES

A number of issues were identified that are common to the INL, Hanford, and SRS site profiles and, in some cases, represent potential generic policy issues that transcend any individual site profile. These issues may involve the interpretation of existing standards, how certain critical worker populations should be profiled for historic radiation exposure (e.g., construction workers and early workers), and how exposure itself should be analyzed (e.g., treatment of incidents and statistical treatment of dose distributions). NIOSH indicates that it may develop separate TIBs in order to address some of these generic issues. The following presents those issues identified in the INL Site Profile Review that SC&A believes represent transcendent issues that need to be considered by NIOSH as unresolved policy or generic technical issues.

(1) Direction on the applicability and usability of the TBDs and/or TIBs to individual dose reconstructions is absent.

(2) Adequacy and completeness of worker records are essential to claimant-favorable dose reconstructions. None of the site profiles address this issue or give direction on resolving missing records.

(3) Site expert testimony indicates that many workers moved from one plant to the next on the same site, creating a complicating factor in determining overall exposure. Establishment of an accurate worker history is crucial in such cases. This will be especially difficult to accomplish in cases of family-member claimants, where the survivors cannot be expected to have a good grasp of where the worker was stationed and when.

(4) Statistical techniques used in the application of the data to individual workers should be considered. However, using statistical averages may not be claimant favorable, since in most compensable cases, they would not provide the upper bound for missed worker doses.

(5) Dose from impurities and/or daughter products in radioactive material received and processed at sites should be assessed.

(6) Assumptions on solubility, breathing rate, and ingestion should be addressed.

(7) A correction factor for external gamma doses should be considered to account for angular dependence of dosimeter sensitivities.

(8) Direction with respect to consideration of incidents and high-risk (dose) jobs in individual dose reconstructions should be provided.

(9) Availability of monitoring records for subcontractor and/or visitors and potential exposure while working on or visiting a facility should be ascertained.

(10) Dose to construction workers and other early workers should be assessed.

(11) Unique exposure conditions for decontamination and decommissioning workers should be considered. The relative impact of each of these items on dose reconstruction is site-specific and requires independent evaluation in each TBD.

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ATTACHMENT 1: CONFERENCE CALL WITH NIOSH AND SC&A

Teleconference of June 29, 2005
to Discuss SC&A Questions/Comments Regarding INL Site Profile[8]

Introduction

SC&A sent its initial list of questions concerning the INL Site Profile to NIOSH on June 16, 2005.[9] On June 29, 2005, the individuals listed below, from SC&A, NIOSH, ORAU, and the ORAU subcontractor, Intrepid, discussed the questions during a teleconference. SC&A summarized the responses to each question and invited all the participants to review and comment on the contents. Some comments were provided and have been incorporated into the summary. The questions (designated by "Q") and responses (designated by "R") are arranged by TBD number following a few general comments.

Teleconference Participants

SC&A
Nicole Briggs
John Hunt
John Mauro
Steve Ostrow (moderator)

NIOSH
Greg Macievic
Tom Tomes

ORAU
Ed Scalsky

Intrepid (sub. to ORAU)
Henry Peterson
Bryce Rich
Norman Rohrig

Questions and Responses

General

(1) Q: The content and organization of the INL Site Profile TBDs are quite uneven, where, often, useful default values and specific guidance for dose reconstructors are buried among general discussions about site activities and history and technical information about dosimetry, which may or may not have practical application. In addition, there are no INL site-specific TIBs to provide guidance, similar to those available for the Savannah River Site, for example. What guidance is given to the dose reconstructors to use the INL Site Profile TBDs? Are workbooks or other summary guidance documents provided to the reconstructors that extract the useful information from the TBDs and present it in a concise, accessible form?

[8] *The lab has changed names and acronyms several times over the years; INEEL (old) and INL (new) both refer to the same national laboratory.*
[9] *Letter from John Mauro (SC&A) to Dr. Lewis Wade (U.S. Dept. of Health and Human Services), June 16, 2005.*

R: The dose reconstructors are not provided with any special instruction per se in the use of the TBDs, but they have workbooks available. NIOSH[10] thought it had provided the INL notebook to SC&A at a meeting a few weeks ago. [note – subsequent to the teleconference, SC&A checked and determined that it had received the INL notebooks, which are in Excel spreadsheet format.]

(2) Q: Section 2.1 (pg. 13) of the Site Description TBD notes that the Naval Reactors Facility (NRF) is exempted under the EEOICPA. However, it should be recognized that workers not connected with the NRF had the potential of receiving exposures from the NRF through various pathways (e.g., airborne, skyshine, ground-shine), especially since the NRF is centrally located on the site. In addition, apparently, some INL personnel also worked at NRF, where they could have received an exposure. For example, Section 5.1.3 (pg. 9) of the Occupational Internal Dose TBD states: "It is possible that some workers' internal dose could have resulted from their support work at the NRF." No information on missed internal, environmental, or external doses from NRF is provided in the TBDs. Nor is information of operational and episodic releases from NRF provided. How should a dose reconstructor deal with INL individuals exposed from NRF activities whether located outside or inside of the NRF boundaries?

R: Some exchange of personnel between NRF and the rest of the site took place in the early days of operation, but the badging system was effective and recorded exposures appropriately. NIOSH considered exposure contributions from NRF to personnel outside the NRF area and concluded that it was not a significant factor. For example, of the 114 site-wide releases listed for 1955, only one, of 310 Ci, was attributable to NRF. The Historical Dose Evaluation database lists the NRF contribution to overall dose in the "other" category, with less than a 0.4% contribution.

(3) Q: The INL Health and Safety Laboratory published annual reports on environmental surveillance, external dosimetry, radiation detection, internal bioassay, quality assurance, personnel dosimetry recordkeeping, and research and development. It is not clear why: only the 1960 Annual Report is used in the Medical Dose TBD; only the 1958 and 1962 Annual Reports are used for the Environmental Dose TBD; and only the 1958, 1959, 1960, 1961, 1962, 1963, 1968, 1970, and 1971 Annual Reports are used for the Internal Dose TBD. No justification is given for including some years and excluding others. Are the chosen years supposed to be representative? If so, why?

R: NIOSH looked at all the Annual Reports that were available (they were not issued every year) and cited only the ones that provided information cited in the TBDs.

[10] *"NIOSH" is used to denote a response from NIOSH, ORAU, or Intrepid.*

Introduction (ORAUT-TKBS-0007-1, Rev. 0, 5/7/04)

No Questions

Site Description (ORAUT-TKBS-0007-2, Rev. 1, 7/28/04)

(1) Q: TBD Section 2.3.1 (pg. 27) on the ICPP Criticality Accident of October 17, 1978 states that: "The atmospheric Protection System (APS) at INTEC … significantly reduced particulate emissions, and filtered all releases associated with the criticality event." Noble gases, for example, would not have been filtered out; how were exposures from such radionuclides determined?

R: NIOSH looked at the incident report and considered the resulting exposure in the Environmental TBD. The evaluation included exposure from noble gases.

Occupational Medical Dose (ORAUT-TKBS-0007-3, Rev. 0, 5/28/05)

(1) Q: TBD Section 3.3 (pg. 5) and Attachment 3A (pg. 10) recommend 200 mrad (Kathren 2003) as the default value for entrance air kerma for pre-1954.

(a) What is the uncertainty associated with this default value?

(b) For the pre-1954 period, why is the assumed lateral view air kerma the same as the PA view air kerma while the former are more than 40% greater than the latter for periods after 1954 (Attachment 3A)?

(c) Kathren 2003 (Table 3.3-1) recommends 500 mrad as the default lateral view entrance air kerma value for the pre-1970 period. Why is a lower value of 200 mrad recommended in the TBD?

(d) In addition, the default values for the other periods in Kathren 2003 are significantly higher than the default values recommended in this TBD. For example, Kathren recommends 100 mrad for PA view and 250 mrad lateral view between 1970 and 1985 while the TBD recommends 52 mrad for PA view and 74 mrad for lateral view between 1954 and 1990. What is the basis for the lower default values? Which values should a dose reconstructor use to obtain claimant favorable results?

R: (a) As stated in the last line on page 7 of Section 3.5 of the TBD, the uncertainty is 30% at one sigma, which is a typical uncertainty value for medical x-ray equipment

(b) This is a "slip-up." No lateral views were taken in the pre-1954 time period. The entry should be replaced by "N/A"

(c) See the previous answer for the pre-1954 period of the table. For later periods, NIOSH reported posterior-anterior and lateral exposures (i.e., 52 and 74 mrad) based on actual equipment settings and practices and, therefore, did not use Kathren's default values; (d) See the previous answer.

Occupational Environmental Dose (ORAUT-TKBS-0007-4, Rev. 0, 3/30/04)

(1) Q: The TBD focuses primarily on inhalation dose from airborne releases; no consideration is given to inhalation doses from resuspension, oro-nasal doses, ingestion doses, and external doses due to cumulative deposition of these airborne releases. How should a dose reconstructor deal with these missed doses? What about skyshine from high level gamma sources in adjacent facilities?

R: NIOSH considered dose from resuspension of radioactive material and determined that, in the worst case, the contribution through this pathway is not significant; death from asphyxiation would occur before deleterious effects from radiation. Oro-nasal and ingestion (transferred from the lungs into the GI tract, not encountered by eating contaminated material) pathways are considered in determining total exposure. Skyshine contributions to external doses are recorded by TLDs. In general, the individual facilities within INL are relatively isolated from one another and no site is downwind of another so that activities within one facility have little effect on other facilities. For example, the SL-1 accident disposal area produces some wind-swept activity, but it is not transported outside the facility fence.

(2) Q: TBD Section 4.1 (pg. 6) states:

> *This TBD also addresses direct gamma doses resulting from facility operations. In general, these doses, if not controlled by management, increase with time and create a facility background dose. At INEEL, these facility background doses were recorded by film badges infrequently and inconsistently before 1970 and by thermoluminescent dosimeters (TLDs) on a routine basis since 1972. These facility background doses, or facility fence-line doses, as they are sometimes called, are a nebulous indication of a dose that workers could receive if they inhabited the facility. INEEL facility fence-line doses (minus background) are presented for 11 locations.*

(a) Does "facility background dose" result only from general facility operations, or should the facility background dose be the sum of doses resulting from facility operations and environmental deposition from neighboring facilities?

(b) Can the fence-line doses represent the facility background doses for workers occupying the facility inside the fence-line?

(c) When the background is subtracted from the fence-line doses, is it the "natural background" dose that would be present in the absence of INL? If so, how and where is it measured?

R: (a) A TLD cannot distinguish between radiation from the facility and from neighboring facilities

(b) Yes. All operational releases are from 250 ft stacks, so environmental doses are relatively uniform throughout the site; (c) Background measurements are taken in Idaho Falls, 50 miles from the site.

(3) Q: TBD Section 4.1 (pg. 6) states:

ICPP airborne effluents have been attributed to creating the maximum INEEL boundary dose. Considering this fact, it should be suspected that ICPP airborne effluent would also be responsible for the maximum INEEL worker doses. Calculations performed for the INEEL TBD show that although ICPP airborne effluent is the most radiologically significant release at INEEL, the impact to workers is significantly below the allowable and acceptable limit.

Since this TBD is concerned primarily with doses due to airborne effluents, it is logical to assume that if "ICPP airborne effluents have been attributed to creating the maximum INL boundary dose," then those effluents would also be responsible for the maximum offsite environmental doses to the surrounding population.

(a) How could the doses resulting from ICPP airborne effluents be accounted for as the maximum INL worker doses?

(b) Is there a possibility that a worker might be working in a facility subject to higher airborne effluents from a neighboring facility other than the ICPP?

(c) Is there any consideration given to episodic releases in a facility other than ICPP?

(d) Shouldn't the maximum worker doses be the summation of occupational environmental doses from airborne effluents, environmental doses from facility background (cumulative deposition), direct beta-gamma doses during various operational and episodic releases, and internal doses from ingestion, inhalation, and oro-nasal breathing?

R: (a) Evaluations have shown that the ICPP airborne effluents, which contained a significant amount of long-lived isotopes, were the greatest contributors to the environmental dose

(b) The facility sites are quite isolated from each other and ICPP was the most important contributor to the environmental dose; therefore, it is unlikely that a worker might be exposed to greater airborne effluents than from ICPP

(c) Yes. All identified episodic releases from all facilities were considered and actual meteorological data were used; (d) Yes, they are determined this way.

(4) Q: TBD Section 4.1 (pg. 8) states:

All of the airborne releases at INEEL that have occurred since the beginning of the Site were reviewed and analyzed ... This request ... was to evaluate the radiological impact to INEEL boundary individuals from airborne releases that had occurred since the beginning of operations at the Site. With the help of NOAA, which had hourly meteorological data from 1956 to that time, analyses were completed for all airborne releases that occurred at INEEL. The radiological consequences for an adult, a child, and an infant were calculated with Version 4 of the Radiological Safety Analysis Computer program RSAC-4 (Wenzel 1990). The results of the study were published in the Idaho National Engineering Laboratory Historical Dose Evaluation (DOE 1991[a]*); this TBD refers to that report as the INELHDE. All releases considered for that report are the basis for the releases considered in this TBD. In addition, all the releases documented in the INELHDE, operational and episodic, have been independently reviewed and found, with minor modifications, to be substantially appropriate.*

(a) What are the "minor modifications" to the releases used in this TBD?

(b) Why are these releases, which were used to determine doses to offsite populations outside the site boundary, "substantially appropriate" for onsite workers, who are largely inside the site boundary and inside the fence-line of a facility? In reality, source terms considered for offsite population dose evaluation tend to eliminate short-lived radionuclides. These short-lived radionuclides could be significant dose contributors to onsite workers.

R: (a) NIOSH modified releases following examination of SC&A's evaluation of the INELHDE. In addition, following analyses to determine the isotopes making the greatest contribution to exposure (releases x ICRP68 dose conversion factors), the original set of 56 isotopes was reduced to nine. These nine account for over 95% of the total dose

(b) Some short-lived isotopes were eliminated from the original set of 56 isotopes since decay reduced their contributions by the time they reached ground level following release from high stacks.

(5) Q: TBD Section 4.2 (pg. 10) states: "For worker dose reconstruction, the analyst should use default values for the calculation..." The TBD then states in Section 4.2.1 (pg. 10): "The annual inhaled quantities (Bq/yr) provided in Tables 4-1 through 4-8 for each of the 8 facility areas are based on known and published INEEL annual airborne emissions"; the same section (pg. 11) goes on to say:

> *In 1968 and 1969, formal Environmental Monitoring Reports (EMRs) report alpha, beta, and I-131 concentrations that can be correlated with Table 4-3 (CFA) values. EMRs between 1970 and 1990 were reviewed for data that could be used for this correlation. Results of the comparison, showing that the calculated values of Tables 4-1 through 4-8 are in reasonable agreement with measured values reported by EMRs, are provided in Table 4-9. Figure 4-4 illustrates the variation of the INEEL Environmental Monitoring sampling results for the 9-year period 1978 through 1986. The figure also illustrates the close correlation of environmental sample results acquired at distant communities and those acquired at the INEEL facilities and the effect of foreign nuclear tests and the Chernobyl reactor accident on INEEL environmental sampling results.*

(a) How does this correlation support the use of offsite airborne releases to calculate the inhalation intake for workers at various INL facilities exposed to onsite (facility-specific) airborne radionuclide concentrations?

(b) How should a dose reconstructor deal with doses from ingestion intake, oro-nasal intake, resuspension of ground deposition, and direct radiation from ground deposition other than inhalation intake from airborne effluents?

R: (a) As mentioned before, releases were through high stacks so that the environmental exposures throughout the INL site were fairly uniform

(b) As mentioned before, all applicable exposure pathways were considered in the dose determination.

(6) Q: TBD Section 4.2.1 (pg. 10) states:

> *Meteorological dispersion factors applicable to each INEEL facility were picked from the annual average mesoscale dispersion isopleths of ground-level air concentrations as published in the environmental monitoring reports published for INEEL as described in INELHDE (DOE 1991*[a]*, Appendix B) ... When an isopleth for a given year is chosen for a particular facility, such as SPERT, that isopleth is assumed to apply to PBF, SPERT-I, SPERT-II, etc. If a facility was between two isopleths, the higher-valued isopleth was chosen. Yearly isopleth values for each of the*

eight facilities have been extracted from the annual environmental monitoring reports and converted from the normalized annual concentration (hr^2/m^3) to concentrations (Bq/m^3) and multiplied by 2.4E+3 m^3/yr (the amount of air breathed occupationally per year) to produce activity inhaled per year (Bq) for an occupational individual. These are listed in Tables 4-1 through 4-8 for each of the facility areas. The annual inhaled quantities (Bq/yr) provided in Tables 4-1 through 4-8 for each of 8 facility areas are based on known and published INEEL annual airborne emissions.

(a) When average dispersion factors or average mesoscale dispersion isopleths are used, are these calculated ground-level facility annual concentrations claimant favorable?

(b) Do these ground-level facility annual concentrations represent cumulative concentrations (i.e., annual deposition plus residual from past years)?

(c) Are these concentrations fence-line concentrations?

(d) How are the concentrations used for onsite workers at various locations of the facility under consideration?

(e) What are the uncertainties associated with these average dispersion factors and mesoscale isopleths?

(f) Why are only annual inhaled quantities (Bq/yr) considered?

(g) How should a dose reconstructor deal with ingestion uptake, direct radiation from deposition, oro-nasal contribution, and inhalation intake from resuspension of airborne deposition?

R: (a) The given isopleths are for ground-level concentrations and are appropriate to use

(b) No; they represent annual concentrations and do not include contributions from deposition from previous years (found to be negligible). Fallout is a more significant dose contributor than what is released from a particular facility

(c) Concentrations are at the referenced facilities. Releases are from high stacks, so the whole site is covered relatively uniformly

(d) Tables 4-1 through 4-8 give isotopic concentrations by year for selected facilities

(e) As stated in the TBD section, the uncertainty is a factor of two for operational releases and a factor of three for episodic releases

(f) [question withdrawn]

(g) previously answered.

(7) Q: This TBD (Section 4.2.1, pg. 10) uses an annual breathing rate of 2.4 x 10^3 m^3/yr rather than the 2.88 x 10^3 m^3/yr (1.6 m^3/hr for 1800 hr/yr) of ICRP 68 or the 8.0 x 10^3 m^3/yr (RAC2002, pg. 38) of the NCRP. The TBD assumption appears less claimant favorable than the ICRP or NCRP assumptions; please comment.

R: The chosen breathing rare is applicable to a worker exposed to the environment, who is not breathing hard from strenuous labor. NIOSH indicated that this choice of breathing rate was made uniformly for consistency for all the evaluated sites.

(8) Q: Section 4.3 (pg. 22) states:

> *External radiation dose at a facility can be created by direct radiation from two sources: direct beta/gamma radiation from the facility, or gaseous effluents released from the facility or from adjacent facilities. In general, direct beta/gamma radiation from the facility will increase with time because the general contamination of the area will increase. In addition, as a facility ages, radioactive sources tend to accumulate at the facility, causing the general background to increase with time.*

(a) Were cloudshine and skyshine from adjacent facilities considered as potential sources of external radiation dose?

R: Cloudshine and skyshine were considered wherever applicable; e.g., the RWMC. Contributions via these exposure pathways would have been recorded by worker TLDs.

(9) Q: It is not obvious that airborne releases from various facilities of INL would be consistent at different locations within the site. The question remains whether using the average annual concentrations as the basis for dose reconstruction is a claimant-favorable approach or not. Also, can these average annual concentrations be used for different individual workers, who worked at different times, in different locations, and performed different functions within a specific facility?

R: As discussed, this TBD gives environmental exposures at various facilities on the site. If a particular worker stayed primarily in one facility, the corresponding dose would be applied. If the worker moved around a lot or it is not known where he worked at a particular time, the dose reconstructors would apply the maximum dose for the different possible locations.

(10) Q: Does this TBD take the same approach as the External Dose TBD, i.e., adding back the detection limit reading for film badges and TLDs?

R: No. The reported values are the difference between readings from the TLDs onsite and those in Idaho Falls (background measurement).

(11) Q: TBD Section 4.3.2 (pg. 24) states:

> *INEEL facility air-immersion (beta-gamma) doses could be calculated from the noble gas and halogen portions of the operational releases, and, if applicable, from the noble gas portion of the applicable episodic releases. This calculation should be unnecessary because these releases would be recorded in the fence line TLD doses presented in Table 4-13.*

The TLDs used at that time were not sensitive enough to record beta doses to the skin from noble gas and halogen plumes. What instructions are given to the dose reconstructor to compensate for this missed dose?

R: The dose reconstructor would multiply by a factor of two as instructed in ORAU Procedure 6, pages 100–101.

(12) Q: TBD Section 4.4 (pg. 24), in discussing uncertainty associated with operational releases, states: "Discussions with the authors of the INELHDE suggest that operational releases, which were monitored, could be low by a factor of not more than 2. When the annual normalized ground level concentration values are applied to the operational releases, the uncertainty could be increased."

(a) Should the dose reconstructor multiply the doses from operational releases by 2?

(b) What is the justification for the factor of 2 multiplication? Is it truly claimant favorable? What is the error analysis associated with this factor?

R: (a) and (b) The dose reconstruction process calculates "mean" doses, then applies appropriate uncertainty factors, like the factor of two mentioned in this section. The Crystal Ball Monte Carlo program is used to analyze uncertainties and a factor of two was found to provide a conservative margin.

(13) Q: TBD Section 4.4 (pg. 24), in discussing uncertainty associated with episodic releases, states: "In spite of the original effort to be 'reasonably conservative' in the exposure estimates, some of the authors, however, have stated that the release considered for a particular episodic event might be low by as much as a factor of 3."

(a) Should the dose reconstructor multiply the doses from episodic releases by 3?

(b) What is the justification for the factor of 3 multiplication? Is it truly claimant favorable? What is the error analysis associated with this factor?

R: (a) and (b) See the response to question 12 above.

(14) Q: There are no TIBs providing guidance for the INL dose reconstruction.

(a) How should a dose reconstructor use Tables 4-1 through 4-13?

(b) There are only nine radionuclides listed in Table 4-1 through 4-8; how should a dose reconstructor compensate for annual intake from other radionuclides?

(c) How should a dose reconstructor calculate ingestion uptake?

R: NIOSH produces TIBs in response to dose reconstructors requesting assistance; the latter has not occurred for the INL dose reconstructions

(a) Instructions are provided in the TBD. Tables 4-1 through 4-8 provide yearly operational intakes by isotope for eight different areas. If a person's location for a particular year is not known, the dose reconstructor should select the values for the location giving the maximum exposure. Tables 4-10 and 4-11 provide episodic intakes for several facilities. The dose reconstructor would sum intakes for operational and episodic exposures

(b) This has been answered in Question 4(a)

(c) As mentioned previously, ingestion exposure from eating outside material is not significant. However, the dose model does account for material inhaled then cleared from the lungs to the digestive system (a negligible contribution, though).

(15) Q: Table 4-13 (pg. 37) presents facility fence direct gamma values by year and location as TLD – background. It appears that background is subtracted from the total dose to identify the portion attributable to INL operations. It is not clear, however, how the background is determined. Is it "natural background" from a location far removed from the site or background measured near the site, where radiation attributable to site activities would have added to the natural background?

R: As previously discussed, background is taken in Idaho Falls, far removed from the INL site.

Occupational Internal Dose (ORAUT-TKBS-0007-5, Rev. 0, 10/12/04)

(1) Q: Consistent with the Savannah River and the Hanford site profile documents, there is no consideration given to oro-nasal doses from airborne releases. How should a dose reconstructor deal with these missed doses?

R: This was discussed in the response to Question No. 1 in the Occupational Environmental Dose TBD section.

(2) Q: Table 5.1.4-1 (pg. 10) lists radionuclides of concern by year of dose assignment.

(a) Are these lists reduced from longer lists? If so, what compensations are made to account for the missing internal doses?

(b) In the CEDE column, what do the various CEDE values for each year represent? Which CEDE value should a dose reconstructor use for a specific radionuclide of concern?

R: The table is included primarily to provide perspective to the dose reconstructor rather than for use in determining claimant exposure. (a) The table includes all doses assigned for the nine year time frame for the listed facilities; (b) The CEDE values listed by year and facility represent actual recorded internal doses for individuals. For example, the six numbers listed for the SMC facility in the year 2000 represent recorded internal doses for six different people.

(3) Q: TBD Section 5.2.2 (pg. 12) states that "routine bioassay of radiation workers has been conducted since the beginning of the site," and Section 2.1 of the Site Description TBD indicates that the National Reactor Testing Station (NRTS) was established in 1949. However, Table 5.2.2-1 (pg. 13) of the Occupational Internal Dose TBD begins the bioassay data in 1953. The table does not appear complete.

(a) What was the bioassay practice between 1949 and 1952, and where is the bioassay data presented?

(b) The reference list does not appear to be complete. Many missing reports lead to uncertainties in conclusions. Between 1953 and 1960, the table shows annual bioassay requirement; some data, however, indicate bi-annual bioassay results. From 1973 to 2001, there are no actual bioassay data collected or referenced. Why? Where are data from RCIMS, as mentioned in TBD Section 5.2.3 (pg. 13)?

R: (a) Construction began in 1949, but nuclear operations did not begin until 1951. 1953 data is the earliest that the ORAU team could find

(b) Table 5.2.2-1 is a reference table summarizing the history of bioassay practices at INL by year. Dose reconstructions would rely on actual bioassay results in a claimant's files. In general, bioassays were performed depending on a person's work details.

(4) Q: TBD Section 5.3.1 (pg. 19) states: "Generally workers were asked to submit to bioassay whenever they were in an area in which a CAM alarmed."

(a) What are the set points for alarms?

(b) Could there be missed doses due to high set points?

R: (a) The set points are not known, but, in practice, were set considerably below permissible levels. The alarms signaled that a problem may have been developing and triggered collection of bioassays. Few of the bioassay samples produced detectable uptake levels

(b) No. See the response to the preceding question.

(5) Q: TBD Section 5.3.1 on air monitoring states (pg. 20):

> *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} µCi/cc for beta) x 10 ratio of general area to breathing zone concentration x 1000 hours of undetected exposure = approx 100 DAC-hrs or approximately 2 x 10^{5} pCi intake.*

Please comment on our interpretation of this paragraph: It appears to apply to the situation where a worker may have been exposed to airborne radiation, but the area was unmonitored and there is no bioassay data available for the worker (either the worker did not submit a sample, or the record is not now available). The approach given, of assuming exposure to a minimum level of detection activity, would seem to be valid only if air sampling was conducted, a zero reading was obtained, and no bioassay was performed. Absent air sampling and a bioassay, any level of activity could have been present and the approach is then not claimant favorable.

(a) What is the basis for 0.01 MPC as the MDA for chronic activity?

(b) Why is the ratio of general area to breathing zone concentration equal to 10? Is this ratio for indoor or outdoor activity?

(c) What is the basis for 1000 hours as the duration of undetected exposure?

R: (a) SC&A's interpretation that some areas were not monitored is faulty. INL recognized the importance of air sampling and monitored most areas accordingly, even the cafeteria. However, NIOSH recognized that the quoted paragraph may not be totally clear and will rewrite it when revising the TBD

(b) The 0.01 MPC value is an estimate derived from counting data available from air monitoring system filters

(c) Published reports indicate that the factor of 10 ratio is appropriate; (d) 1,000 hrs per year is seen as a conservative, claimant-favorable assumption since workers move around and do not stay in one place all day throughout the year. NIOSH thought that this assumption is also used in the other site profiles.

(6) Q: Table 5.3.2-1 (pg. 21) has sets of columns labeled "statistically significant" and "highest results." Please define and explain these headings.

R: This table has been taken from 1959–1961 annual reports. These reports do not state what statistical assumptions were used, but it is thought that at that time two sigma above background was standard. The table indicates that of the thousands of bioassays taken each year, only a few yielded statistically significant results.

(7) Q: TBD Section 5.3.2 (pg. 22), which discusses the change from "no positive exposure reported" to "no reportable levels recorded" records, is not clear about what was the minimum reportable level for bioassay sample analyses. How should a dose reconstructor treat these records during the dose reconstruction process?

R: The section from which the quotation is taken is just for information for the dose reconstructor. The minimum reportable levels are provided in subsequent tables.

(8) Q: Table 5.4-2 (pg. 24) shows internal dosimetry in-vivo MDAs for fecal samples.

(a) What is the sample size? (mg or cc?)

(b) The footnote instructs "when sample size is not identified in individual's records, assume the activity is that excreted per day." Please clarify.

R: (a) The sample size is not given in the table; the sample analyzed is the total sample provided by the individual. In most cases, the claimant's records would contain sample size information

(b) Assume that the sample referred to is the total amount excreted per day.

(9) Q: The last paragraph of TBD Section 5.6.1.1 (pg. 29) on the SMC (Specific Manufacturing Capability) project to produce DU (depleted uranium) armor for tanks discusses natural background uranium excretion, and states that "Urine samples submitted by SMC nonradiation worker in 1987, 1994, and 1998 were assumed to represent nonoccupational elimination of the SMC worker population."

(a) Why are only SMC nonradiation workers considered?

(b) Couldn't the SMC nonradiation workers have been exposed to uranium deposition from airborne releases from SMC and other facilities? Shouldn't a wider population be used to determine the natural background uranium uptake?

(c) Is this type of nonoccupational elimination approach used for other INL facilities?

R: (a) The SMC program was run independently from the other INL programs and considered only its own people for establishing background

(b) NIOSH considered that, but using only the SMC population to determine background is consistent with the approach taken elsewhere on the site and at other sites (e.g., Fernald)

(c) No. SMC is the only facility handling depleted uranium.

(10) Q: Table 5.6.2-1 (pg. 30) indicates that some airborne incidents at various INL facilities released noble gases and halogens.

(a) Was any consideration given to external radiation exposure from these releases?

(b) How should a dose reconstructor estimate these doses?

R: (a) Yes. Releases though 250 ft (76 m) stacks (primarily Kr) produced doses that were recorded as environmental exposures throughout the site

(b) Guidance is provided in the External Dosimetry TBD.

(11) Q: TBD Section 5.6.2.1 (pg. 30) on high-enriched fuel storage states that "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 2×10^{-9} μCi/cc." The same request for clarification applies here as for question 5(a) in this section; how is this approach valid in the absence of airborne monitoring?

R: See the response to question 5(a). NIOSH also noted that this section of the TBD is being rewritten.

(12) Q: Section 5.6.4 (pg. 33) discusses the Radioactive Waste Management Complex (RWMC). Several major floods over the years and operational practices, especially in the earlier days of dumping waste into open trenches and pits, resulted in spreading of radioactive contamination across the facility. Not only are the workers at the RWMC and adjacent facilities vulnerable to airborne releases, they are also subject to potential pathways for ingestion, oro-nasal intake, and external/environmental exposure to high level radioactive waste materials. Where is guidance on accounting for exposure through these pathways?

R: This has already been discussed. [see Question 1 of the Environmental Section]

(13) Q: Section 5.6.4 (pg. 33): As recently as 1995 and 1996, there were reported incidents of skyshine at Pit 9 due to disposal of high level gamma-emitting waste in Pit 17. This skyshine not only affects the onsite Pit 9 radiation workers, but also the offsite non-radiation workers in the Pit 9 Administration Area. Did the TBD team evaluate all these exposure pathways and source terms for the RWMC facility?

R: Yes. The exposure pathways are well-documented and were considered. TLDs on badged workers would have recorded the exposures and those readings can be used to determine exposure to unbadged workers in the same area.

(14) Q: Tables 5.6.4-1 (pg. 34) and 5.6.4-2 (pg. 35) show the total volume and total curies of waste inventory at the RWMC facilities. Have these number been compared with 1995 and 1996 RWMC Historical Data Task reports? These inventory and concentration values are very significant and could cause significant internal doses to workers. How are these values used?

R: The values appearing in Tables 5.6.4-1 and 5.6.4-2 are straight out of the RWMC Internal Dosimetry technical basis document. NIOSH believes that these values were taken from the RWMC Safety Analysis. In any event, the values given in the TBD tables are primarily to indicate to the dose reconstructor the magnitude of the radioactive inventory and would not be used in an actual calculation to determine exposure to a claimant.

(15) Q: TBD Section 5.6.6.1 (pg. 35), on the TRA reactors, states: "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."

(a) There is no Table 5.1.1-1. Reference should probably be made to Table 5.4-1.

(b) If actual data is absent, why should MDLs be used? They may not be claimant favorable, since, as in episodic releases, exposure levels possibly may be above MDL. A preferable approach may be to use surrogate data (similar worker in a similar situation); please comment.

R: (a) Yes; NISOH corrected this discrepancy in the revision of this TBD that is currently under development

(b) NIOSH is in the process of generating a co-worker exposure database for all sites that could be used in cases where data are absent. This task will eventually result in individual TIBs for the various sites.

(16) Q: TBD Section 5.6.6.1 (pg. 36) on the TRA reactors, states: "The claimant-favorable position is that there could have been some halogens and particulate radionuclides released along with the noble gases." It continues: "Table 5.6.6.1-1 contains too many radionuclides for efficient dose reconstruction."

(a) Is I-131 considered the most limiting radionuclides in the list of radionuclides of concern in Table 5.6.6.1-1?

(b) Why is the factor of 1.6 used as the weighting factor to account for the total iodine dose? The total activity fraction is 0.834.

(c) What compensation is provided in the weighting factors to account for the rest of the iodine contributions? Are there other radionuclides in addition to those in this table?

R: (a) I-131 produces the greatest dose of the iodine isotopes, which, themselves, produce almost 95% of the total dose. Hence, I-131 is a good surrogate for the total dose

(b) An earlier version of the table explained the origin of the factor of 1.6. A comparison of iodine doses, using ICRP-68 conversion factors, shows that I-131 contributes about 60% of the total dose; the reciprocal of 60% gives a multiplication factor of 1.6

(c) The multiplication factor of 1.6 accounts for the dose contribution from the other iodines, which account for about 95% of the total dose.

(17) Q: TBD Section 5.6.6.1 (pg. 37), on the TRA reactors, states: “The claimant-favorable position is to assign a missed dose based on the MDA for in-vivo counting (see Table 5.4-3) for Ag-110m and Ta-182.” Since noble gases are involved, should the dose reconstructor take into account the external dose from these gases?

R: Noble gas contributions are recorded by film badges or TLDs.

(18) Q: TBD Section 5.6.6.2 (pg. 37), on the TRA laboratories, states: “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.” How should a dose reconstructor account for ingestion and oro-nasal doses?

R: This has already been discussed. [see Question 1 of the Environmental Section]

(19) Q: TBD Section 5.7 (pg. 37), on missed dose, states: “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 [5.7.1]. If no detectable external or internal dose information is recorded, only the environmental dose should be included.” What approach should be taken if the claimant is a radiation worker, but the claimant’s file does not show positive external dosimeter readings? Should the default internal missed dose be applied to this worker?

R: No. If there is no external dose, there also is no internal dose.

(20) Q: Table 5.7-1 (pg. 38) presents missed dose default assumptions recommended to be used by the dose reconstructors. The recommendations, however, are not very clear on how to apply these default assumptions. In the pre-1960 period, the assumptions are based on urine gross beta results. Then, between 1961 and 1980, they are based on whole body counting results. From 1981 on, they are based on bioassay results. There are significant inconsistencies in the foundation of these assumptions and approaches. How does a dose reconstructor reconcile these inconsistencies involving ingestion only and ingestion plus inhalation/oro-nasal results?

R: Historically, this table has been a problem with respect to clarity. However, dose reconstructors have not registered any complaints about its use. The co-worker data base process may supersede this table in the future. [See response to question 15 in this TBD.]

(21) Q: TBD Section 5.8 (pg. 39), on unmonitored workers, states: "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."

(a) This is confusing. Isn't a worker who wore a personnel dosimeter a worker who was monitored?

(b) In the early days, the reporting levels were much higher due to lower sensitivity of the instrumentation. Should compensation be given to this worker by adding the reporting level or a portion of it to the total worker dose?

R: (a) "Monitored" in this sentence refers to a worker who underwent a bioassay for internal dose. This will be clarified in a future revision

(b) The co-worker data base process should supersede the TBD guidance. [See response to question 15 in this TBD.]

(22) Q: A July 2003 report prepared by SC&A under contract to the Centers for Disease Control and Prevention (Contract No. 200-2002-00367) revealed that the airborne emissions associated with several of the Initial Engine Tests (IETs 3, 4, and 10) of the Aircraft Nuclear Propulsion (ANP) Program, as estimated by the INEL-HDE (Historical Dose Evaluation) task group, were likely to have been underestimated as follows: IET 3 – underestimate of total radionuclide release by up to a factor of about 3; IET 4 – underestimate of noble gases by up to a factor of about 16, halogens by up to a factor of about 7, and solids by a factor of up to about 2; and IET 10 – underestimate of total radionuclide releases by up to a factor of about 7. It is suggested that occupational environmental doses associated with these revised estimates of airborne releases from the ANP Program be taken into consideration in dose reconstructions; please comment.

R: NIOSH looked at the SC&A report supplied in the Advisory Board meeting in Boise in August 2004. The report, however, was only a "comment" report and not a final report. NIOSH revised the TBD where it felt revision was justified. NIOSH would like to see the final report if it has been produced. [It was subsequently determined that the report has not been finalized; SC&A will make it available to NIOSH when it is released.]

Occupational External Dosimetry (ORAUT-TKBS-0007-6, Rev. 0, 4/6/04)

(1) Q: TBD Section 6.3.1 (pg. 8) mentions that "control badges which are used to subtract background radiation, have also been located there [at each operational areas entrance security gate]. This practice may lead to subtracting environmental radiation from site activities reducing the reported doses."

(a) How would the dose reconstructor know if environmental doses were subtracted from the reported doses, and what he is then supposed to do in either case?

(b) Since the measured "environmental dose" could include contributions from air-borne radionuclides after release of fission products, what procedure is followed to assure that that the same dose is not ascribed twice to a worker?

R: (a) Environmental doses were subtracted from all reported doses for individuals. The dose reconstructor would use Table 6-1 yearly facility fence dose values (by year and facility) to determine direct gamma exposure by multiplying by (2000 WH/y)/(8760 hr/y). This value would be included in the person's total dose

(b) [no notes taken]

(2) Q: TBD Table 6-1 (pg. 9) presents the gamma measurements recorded by the facility fence-line dosimeters; this table may be more helpful in the Environmental Dose TBD.

(a) It is not clear what "TLD – Background" in the table's title means, especially as TLD monitoring must have begun in the 1970s.

(b) The background value for 1952–1972 (not given) is probably equal to the average exposures for the years 1973–2002 (123 mrem/year). Explain in a table footnote.

(c) It is not clear if the values stated for 1952–1972 are average values. Consider, for example, the SL-1 accident at the ARA II site in 1961. The dose at the facility fence for that site in that year must have been much higher than the 226 mrem (per year) given in the table.

R: (a) [no notes taken]

(b) The "1952–72" row entries represents annual doses at each facility

(c) Film badges in the SL1 area were moved far away from the reactor site after the accident and, therefore, do not reflect contributions from the accident in the years after the accident occurred. The dose reconstructor would assume 226 mrem/y for each year in that range.

(3) Q: TBD Figure 6-4 (pg. 12), a dose report, shows an entry of 120 rads from beta radiation. Although there was a second, less sensitive, film included in the package, this is a very high dose, out of the range of modern film dosimetry systems with a maximum net optical density of about 4.5. OCAS IG-001 states that the saturation optical density of the Dupont film 502 was around 2.8. The TBD should discuss upper detection limits and how the dose reconstructor should account for high doses in excess of those limits.

R: The original document gives 1000 R as the upper range of detection. Therefore, the dose of 120 rads is within range.

(4) Q: TBD Section 6.3.2.1 discusses the film badge system used initially at INL, and says: "Type 552 film has a threshold level of about 30 mR, and type 558 film has a threshold level of about 10mR" (pg. 13). The minimum detection level of film dosimetry depends strongly on the energy of the radiation. 10 mR would be possible for low energy photon radiation, but not for energies above about 200 keV, which are characteristic of what was experienced at INL.

R: A high-energy radium source was used for calibration. NIOSH maintained that published reports indicate that a 10 mR minimum reporting level (MRL) was achieved. NIOSH has since supplied supporting documented from the "O-drive" to SC&A.

(5) Q: TBD Section 6.3.2.7 (pg. 17) discusses nuclear track emulsion-Type A (NTA) neutron detectors. The second paragraph begins with the statement that "the minimum dose assigned was 14 mrem," then presents anecdotal justification based on some 1958 readings. The following paragraph, discussing April 1959 data, ends with the statement: "These values suggest an MRL of 20 mrem." It is not clear what value the dose reconstructor should assume, or whether assuming either 14 or 20 mrem would be claimant favorable.

R: The 14 mrem value applies through 1958 and the 20 mrem value applies thereafter. The protocol for reading NTA detectors changed in the 1958 to 1959 time frame. SC&A noted that the minimum dose assigned was much higher at other DOE sites; for example, it was taken as 50 mrem at the Savannah River Site.

(6) Q: Table 6-2 (pg. 18) presents INEEL Facility Neutron Correction Factors (FNCF) for Hankins dosimeters for several of the site facilities. It is not clear from the accompanying text whether the data recorded in the workers' records were

already adjusted based on the FNCFs or whether the dose reconstructor should make the adjustments.

R: The adjustments have been made in the recorded data. NIOSH stated that it would clarify this point in a future revision of the TBD.

(7) Q: TBD Section 6.3.2.8 (pg. 18), when discussing the date of changeover from NTA to albedo neutron monitoring, states that "dose reconstructors should make the claimant-favorable assumption that this transition occurred in October 1976." Why is this assumption claimant favorable? If the albedo monitors recorded higher doses than the NTA monitors, then shouldn't readings from the latter be adjusted upward appropriately?

R: The multiplier factor applied to the NTA is greater than one, so the assumption is claimant favorable.

(8) Q: Table 6-3 (pg. 20) lists estimated (based on judgment) percentage laboratory calibration uncertainties for beta/photon dosimeters. How were these uncertainties factored into reported dose in a claimant-favorable fashion?

R: Individual uncertainties are factored into the IREP process of deriving a claimant-favorable dose.

(9) Q: Figure 6-8 (pg. 21) shows two curves, but they are not labeled. Also, the probability density should fall to zero at zero MeV.

R: The curves are not labeled. Number "1" is probably measured values and number "2" calibration values. NIOSH thought that it was permissible that the graph not go to zero at zero MeV since there are some low energy neutrons present.

(10) Q: Table 6-4 (pg. 21) lists sources of anticipated laboratory bias in calibrating neutron dosimeters. Some of the rows indicate that "recorded dose of record is likely too low." How is that finding factored into deriving claimant-favorable dose numbers?

R: This finding has not been factored explicitly and separately into determining doses. Rather, all uncertainties are folded into the IREP program.

(11) Q: Table 6-6 (pg. 23 – note that Table 6-6 precedes Table 6-5, which is placed on page 25) shows IREP beta and photon energy groups for different INL facilities. In all cases, save the last, the split between the lower and higher photon energy groups is given as 25% to 75%. This assumption is not as claimant favorable as the SRS or Hanford photon spectrum assumption of a 50% to 50% split (see, for example, SRS ORAUT-TKBS-0003, Table 5.3.4.1-1). Why was the 25% to 75% split chosen for INL?

R: NIOSH will look into this. [subsequent to the teleconference, ORAU team member Intrepid determined that while Savannah River used the 50:50 spectrum mix, Hanford and other sites used the 25:75 mix adopted for INL. Intrepid believes the latter mix is appropriate since the unshielded dose fraction in the upper energy group (> 250 keV) represents over 95% of the total dose, and that after passing through some shielding, an assumption of 75% in the upper group is a reasonable approximation of that dose mix.]

(12) Q: TBD Section 6.3.4.2, on beta radiation, refers to Figure 6-10 (pg. 24) as indicating that the beta exposures can only be appropriately applied to the skin and not to the breast or testes. The figure, however, shows that 100% of the nuclides will contribute to the 0.07 mm dose (skin), but that also at least 30% could contribute to the 1 mm (100 mg/cm^2) dose at the closest parts of the breast and testes. Therefore the potential effect of beta exposures to the breast and testes should not be disregarded. Also, it is not clear whether the general shallow dose procedures developed by NIOSH should be applied at INL, namely the "Addendum to External Dose Reconstruction Procedure: Shallow Dose Calculations for Complex-Wide Cases," ORAUT-PROC-0006 Rev. No. 00 PC-2.

R: It is not appropriate to apply beta surface dose to the testes and breast at the 1 mm level. That there is some beta contribution at that depth is factored into the dose reconstruction process.

(13) Q: Table 6-5 (pg. 25) of the TBD presents data to allow determination of correction factors for the recorded beta doses to compensate for the underreporting associated with dosimeter thickness. It is not clear, however, how the table values were obtained.

R: The correction factors account for the dosimeter covers and dosimeter thicknesses. A dose reconstructor would divide the reported dose by the factor given in the last column.

(14) Q: TBD Section 6.3.4.3.1 (pg. 27) states that: "The exception to the above discussion [no beam ports – low potential for neutron exposure] is the MTR, which … had beam ports and neutron beams extending onto a research floor. Also in this category are the Zero Power Physics Reactor (ZPPR) and Transient Reactor Test (TREAT), both at Argonne West." Why was the Neutron Radiography (NRAD) TRIGA reactor (described in Section 2.4.13 of the INL Site Description TBD, ORAUT-TKBS-0007-2) not included as well?

R: The TRIGA is excluded since it was installed in 1977, after NTA dosimeters were no longer used.

(15) Q: TBD Section 6.5.2 (pg. 33), on neutron missed dose, states: "If no neutron dose was assigned to him/her or to co-workers for several months, the dose reconstructor should assume that the person was not exposed to neutrons so no neutron dose would be missed." It is not clear if these instructions pertain to all workers or only workers in the designated neutron exposure areas (MTP, ZPPR & TREAT). For those workers in the neutron exposure areas, it may not be claimant

favorable to disregard a series of zero measurements since the workers may have been exposed to ongoing low levels of radiation below levels of detection. Also, it appears that the term "several months" could be interpreted in various ways by the dose reconstructors, resulting in different missed neutron dose estimates.

R: "Several" may be interpreted as more than three. The dose reconstructor examines a copy of the dose report, which includes co-worker data along with the claimant's exposure. If they all have zero neutron dose recorded, then the claimant probably also should have a zero neutron dose. Since neutrons are only present at a few known areas, it would be inappropriate to ascribe a neutron dose to the claimant.

(16) Q: TBD Section 6.7 (pg. 34) discusses uncertainties and ascribes the following uncertainties: photons – 35% at 1 sigma; beta – 50% at 1 sigma; neutrons – 60% at 1 sigma. It is not clear how these values were obtained or how the reconstructor should apply them to ensure claimant-favorable results.

R: These are "best guess" numbers, not calculated values. Some recent field measurements and Hanford results support these assumptions for uncertainties. Uncertainties are incorporated into the IREP

ATTACHMENT 2: ADVANCE QUESTIONS FOR SITE EXPERT INTERVIEWS

INL Plant Site Expert Interview

External Dosimetry

(1) The MDL for a dosimeter would be affected by the exposure geometry and potential shielding of the dosimeter by the body or other aspects of the geometry of exposure relative to the badge location. How was angular dependence accounted for in non-traditional exposure situations? Were angular dependence issues considered with neutron dosimetry as well?

(2) Were there any studies done on the differences in badge response between the calibration sources and the actual radionuclides in the field? Are these differences in source term accounted for in calculation of dose from dosimeters?

(3) Who was responsible for performing tests on new badge systems? How were these tests documented? Can we obtain a copy of these documents?

(4) How were individuals directed to wear their beta/gamma dosimeter? How were they directed to wear their neutron dosimeter?

(5) What were the procedures for assigning dose due to a lost or damaged dosimeter?

(6) Where were badges stored historically? Were individuals allowed to take them home?

(7) How did you measure slow neutron dose prior to the TLND? What was the effectiveness and detection limit on this system?

(8) Were temporary badges ever assigned to individuals on the routine monitoring program? If so, under what conditions? How were the results accounted for when assigning external dose of record?

(9) When is/was extremity dosimetry assigned to workers?

(10) Are/were there conditions where partial body exposures (e.g., skin, lens of eye, gonads, chest, etc.) would have significantly exceeded whole body exposure measured by the routine dosimeter? If so, how did INL address this issue?

(11) Under what conditions, do/did INL use multiple dosimetry? How was the whole body dose assigned in these circumstances?

(12) Are/were dosimetry requirements varied for high-risk jobs, such as during sampling of process material or maintenance?

(13) Was/Is there an area dosimetry program at INL? If so, when and how were these results used?

(14) Was timekeeping ever used as a mechanism to limit internal or external dose? If so, under what conditions and where are the records located?

(15) Were zeros ever entered into badge records when workers did not turn in their badges? If so, how common was the practice? Was it confined to some periods of operation?

(16) What is/was the process for assigning dosimeters to subcontractors and visitors?

(17) Are you aware historically of unsanctioned practices by workers with respect to their dosimeter (e.g., putting them on sources, not wearing them, etc.)? For instance, were there times when workers would take off their dosimeters when the maximum limit was being reached so that they could continue to work? If yes, was this kind of practice more common in some areas than in others?

(18) Are there audits or assessments (especially historical) of the INL external dosimetry program available? If so, can we get a copy of these documents?

(19) Can you recommend resources (i.e., technical reports, books, films, etc.) that may be helpful in understanding the historical operations of the site?

INL Site Expert Interview

Radiological Field Operations

(1) What was the radiological control organizational structure at the inception of the plant? How has it changed over time?

(2) What is the current radiological organization's structure?

(3) Were employees frequently transferred between the different processes or did they maintain the same job for a majority of their career?

(4) How much movement was there between INL and other DOE sites?

(5) How does this organization interact with subcontractors and/or the environmental restoration contractor?

(6) What has been the primary regulatory driver (e.g., 10 CFR 835) over the period of operation at INL?

(7) There are a number of areas at the INL plant where the DOD conducted operations. What did these operations involve and where were they conducted? How much support did INL workers provide to these operations?

(8) Do you provide RCT, radiological engineering, dosimetry, and instrument support for subcontractors and construction support? How do you control the radiological practices of subcontractors? How is/was this different in the past?

(9) When were RWP/SWPs used at INL? What types of RWP/SWPs were used (i.e., job specific, routine)? Were early work permits focused on job specific coverage or did they include routine work also? Did they have sign in sheets?

(10) If the data were available, is there a mechanism to associate workers on a particular job with survey/air sampling data taken for that job? Why or why not?

(11) How were/are RWP/SWP requirements such as the following determined?

- PPE Requirements
- Bioassay Requirements
- Dosimetry Requirements
- Not to exceed radiation and contamination levels
- RCT coverage
- Entry/Exit requirements

(12) In the absence of RWP/SWPs, how were the above requirements determined and communicated to the workers?

(13) How have the posting and contamination control limits changed over the period of operation of the site?

(14) Were there always common procedures for posting, contamination control, and determining radiation protection requirements? In other words, was the process for determining survey frequency, dosimetry requirements, PPE, etc. consistent over time?

(15) How did you historically determine monitoring for individuals who work all over the site (i.e., maintenance)?

(16) How are/were spread of contamination and other small occurrences documented?

(17) How were incidents involving personnel exposure documented? Where are these records located? What percentages of incidents are classified?

(18) What are the radionuclides of concern from a field perspective by building?

(19) When did the site start receiving recycled fuel?

(20) In terms of ALARA classification, can you name some high-risk jobs?

(21) Are there areas on site where administrative offices back up to or are adjacent to radiological areas?

(22) What portable survey instruments were originally used at INL? What instruments are used today?

(23) Is there field characterization data? If so, where is this data located? How can we obtain a copy?

(24) How are/were portable survey instruments calibrated at INL? How has this changed over time?

(25) What sources (neutron, beta, gamma) were used for calibration over time?

(26) Are there documents or reports available on the instrumentation and air sampling equipment used now and in the past at INL? If so, how can we obtain a copy?

(27) What type(s) of air sampling equipment have been used in the past?

(28) When were breathing zone apparatus first used at INL? Is INL involved in DAC-hour tracking? If so, when did this begin?

(29) How was air sampling equipment calibrated historically? How has this changed over time?

(30) What sort of technology shortfalls have your identified with respect to field instrumentation and analytical abilities?

(31) Describe the engineering controls, administrative controls, and PPE that are commonly employed in each building handling radioactive material. How was this different in the past?

(32) How extensive is/was the use of Radiation Generating Devices (e.g., radiography sources, x-ray diffraction, etc.)? What types of devices were used? In what operations and buildings were these devices used?

(33) Are there field conditions that are particularly challenging to the field radiological control staff? If so, what?

(34) Have there been particle size and/or solubility class studies at INL?

(35) What is the relationship between INL and the state of Idaho?

(36) What is the extent of decontamination and decommissioning operations? What D&D activities have occurred in the past?

(37) Can you recommend resources (i.e., technical reports, books, films, etc.) that may be helpful in understanding the historical operations of the site?

INL Site Expert Questions

Environmental Monitoring

(1) How were these environmental releases documented? Where are these records located? Has INL published any internal reports on releases of radioactive material to the environment? If so, where can we obtain a copy?

(2) Do annual environmental reports include both routine and episodic releases? If not, please explain.

(3) Are you aware of historical particulate and gaseous releases from facilities to the

(4) What is the extent of outdoor contamination at INL in the soil, groundwater, vegetation, etc.?

(5) When did the environmental monitoring program for soil, groundwater, and vegetation begin? What did/does the monitoring protocol involve?

(6) What is the relationship between INL and the state of Idaho?

(7) What EPA regulated activities are currently occurring at INL?

(8) In some situations, liquid effluents can be found to deliver a higher dose to the public than airborne releases from a facility. Is this the case for occupational workers at the INL site? If so, explain.

(9) Have you ever calculated environmental doses to onsite workers? If so, where is this information documented?

(10) What radionuclides were released from INL facilities and when did these releases occur?

(11) Were there any significant episodic releases from INL?

(12) What controls did INL implement over time to reduce environmental emissions from the facility?

(13) Were there any burning operations at INL involving radioactive material? If so, please explain.

(14) Can you recommend resources (i.e., technical reports, books, films, etc.) that may be helpful in understanding the historical operations of the site?

INL Plant Site Expert Questions

Internal Dosimetry

(1) Historically, what was the basis for determining who was put on a bioassay program?

(2) Were there any employees that were exposed to radionuclides who were not on an adequate bioassay program? How was internal dose to these individuals determined?

(3) What were the bioassay requirements in areas where recycled uranium was handled? Were individuals tested for potential plutonium intakes?

(4) Historically, who determined the need for special bioassay samples?

(5) What areas had the greatest number of intakes historically? Why do you think this was the case?

(6) Based upon current/past decision levels related to in-vivo and in-vitro analyses, have you estimated a potential missed dose? If so, where is this information documented?

(7) How effective were early in-vivo counters for the detection of natural, depleted and enriched uranium? Were they used for the detection of other radionuclides of concern?

(8) Describe the interaction between internal dosimetry and the field personnel during an incident or occurrence.

(9) Have there been any particle size studies done at the INL facilities? If so, where are they documented?

(10) What default assumptions do you used when calculating internal dose (i.e., particle size, solubility class, date of intake, type of intake, etc.)?

(11) Are there areas at INL where there is a potential for an intake of high fired plutonium or uranium oxide? If so, how does an intake of high fired oxides affect your internal dose calculation?

(12) How extensively did INL Plant handle tritium? Was there potential exposure to tritides?

(13) Was there ever a time when air concentration data was used to determine internal dose? If so, how was the dose determined?

(14) Have you done any comparisons between air concentration data, in-vivo data and in-vitro data? If so, what were the results?

(15) Have certain isotopes between used as an indicator for the presence of other isotopes? For example, a bioassay program may focus on Cs-137 as the central radionuclide; however, when Cs-137 results are positive it may trigger a Sr/Y-90 bioassay.

(16) How did you account for intakes of a given radionuclide prior to the establishment of bioassay techniques for that given radionuclide?

(17) What is your involvement in incident response?

(18) What are the background levels of uranium and thorium in fecal and urine samples for the geographical area?

(19) What technology shortfalls have been identified in the current and historical bioassay programs?

(20) Are there audits or assessments of the INL internal dosimetry program available? If so, can we get a copy of these documents?

(21) Are thoron and radon an occupational exposure issue in any areas of the facility? If so, where?

(22) How involved were you in the development of the INL site profile? Did NIOSH/ORAU interview you? If so, when?

(23) Can you recommend resources (i.e., technical reports, books, films, etc.) that may be helpful in understanding the historical operations of the site?

INL Site Expert Questions

Radiological Records

(1) Are you aware of any code names encountered in the records and their meanings? If so, can we get a copy?

(2) What were the various terms used to refer to shallow and deep dose through time? How were these terms defined? Were tritium and neutron dose incorporated into whole body doses?

(3) Do you have a document describing the historical dosimetry records and the abbreviation used on these records? For example, a document that may define codes used to differentiate between extremity, whole body and multiple dosimeters. If so, can we obtain a copy of this information? If we have questions related to a specific record who can we direct these questions to?

(4) How complete are the dosimetry files (early and current) with respect to dosimeter results and calibrations, bioassay results, PIC data, time-keeping data, multiple dosimetry results, and investigation reports for lost or damage dosimeters? Are these records associated with an individuals personnel radiation exposure file, or are some of the records stored with field radiological control records?

(5) Who is in charge of maintaining field radiological records? How are these records stored? How retrievable is information such as work permits, early radiological surveys, and air monitoring records?

(6) How are incident and personnel contamination reports stored? Is there an incident database that summarizes all incidents at INL? If so, does it include classified incidents?

(7) How are off-normal situations such as a spill of radioactive material that is not likely to result in personnel exposure documented?

(8) How were/are the doses for subcontractor/construction tracked? Are these records stored separately from permanent employee records?

(9) When are radiation exposure reports issued to visitors, workers, subcontractors, etc? Has this always been the practice? For example, if a visitor from SRS came to visit INL and entered a radiation area, would INL provide a radiation exposure report to SRS on this particular employee?

(10) Has dose of record been modified for any reason over time?

(11) Did early contractors take monitoring records with them when a new contractor took over? For example, at Hanford DuPont took their dosimetry records and other records with them when their contract ended.

(12) Has the site destroyed any dosimetry, bioassay or field radiological records in the past? If so, please explain?

(13) What radiological exposure records are provided to NIOSH for the purpose of dose reconstruction? How often have you received follow up requests for additional information?

(14) Can you recommend resources (i.e., technical reports, books, films, etc.) that may be helpful in understanding the historical operations of the site?

INL Site Expert Questions

Medical Exams/Treatments

(1) What were the medical exam requirements for workers in the present and historically?

(2) What did the medical exams include?

(3) How frequently did INL employees have medical exams?

(4) How frequently were they given x-rays as a part of this exam?

(5) What type of x-ray equipment (i.e., manufacturer, model) was used historically?

(6) What type of x-ray equipment (i.e., manufacturer, model) is used currently?

(7) Was photofluorography used at any time at the INL?

(8) Were 4” x 5” films every used for x-rays?

(9) Did the radiation safety department ever perform a radiation survey or study to determine the exposure from medical exam x-rays? If so, what were these results?

(10) Who performed x-ray inspections on your equipment? Where are these records located?

(11) Are there other items that you would like to add that would help evaluate medical x-ray exposure?

(12) How many chelation procedures have been performed at INL?

(13) How many lung lavage procedures have been performed at INL?

(14) What is/was the criteria for using chelation therapy?

(15) When chelation therapy was done, where were the records stored? Did this information end up in the medical or radiation exposure file?

(16) Are there other items that you would like to add that would help evaluate internal exposure to radionuclides?

(17) Is it possible to review a sample medical record from a long term employee and ask some questions?

INL Site Expert Questions

Operations/Maintenance

General Questions

(1) Which plant did you work in and what tasks did you perform. Did these tasks involved radioactive material? If so, what type?

(2) Were employees frequently transferred between the different processes or did they maintain the same job for a majority of their career?

(3) How many hours a day and per week did you routinely work?

(4) Did you receive radiological worker training? If so, when did this training begin?

(5) What particular forms of radioactive material did you work with?

(6) Were you monitored with dosimeters (i.e., film badges or TLDs) when you were working on tasks where radiation exposure was likely?

(7) Where were you instructed to wear your dosimeter?

(8) Did you store your dosimeter onsite or take it home?

(9) Were there situations where a portion of your body was more exposed than others? If so, where was your dosimeter in relation to this exposure?

(10) Did you every wear pocket ionization chambers (also called PICs or pencils) at the plant?

(11) Did you participate in a urine or fecal bioassay program? If you received sampling, how often was it done i.e., monthly, quarterly, annually?

(12) If you had urinalysis or fecal bioassay, when were you told to submit the sample (e.g., after an event, Monday morning, etc.)?

(13) Were you asked to submit blood samples for the purpose of radiation monitoring? If so, please explain.

(14) Did you ever have a whole body or lung count and if so how often?

(15) Were you aware of the use of co-worker exposure data to assign a dose? If so, what problems or advantages do you see in doing this?

(16) Were your dosimeter and bioassay results made available to you routinely? When did this begin?

(17) What type of PPE was used throughout the operation of the plant in the various operations? Did you wear special protective clothing while working on the job? When did you start wearing shoe covers?

(18) How were PPE and special dosimetry requirements communicated to you prior to the use of radiation work permits? Were the PPE requirements consistent between different departments for the same type of task?

(19) Were there situations where an individual in the immediate vicinity of the work wore PPE yet an individual standing near that individual did not? If so, please explain?

(20) Where was support personnel (e.g., health physics technicians, foreman, security, etc.) located in relation to the immediate production lines, machining or maintenance?

(21) Were there showers or change rooms at the facility? Did you shower and change your work clothing before leaving for home?

(22) How were respirators assigned? Did an individual use the same respirator more than one time? If so, where was the respirator stored in between uses?

(23) Did you keep your respirator around your neck in production areas even when not in use?

(24) Did you support army operations onsite? If so, what was your role? What other INL employees provided support to these operations?

(25) What operations involved hands-on work with radioactive material verses being automated?

(26) Were you allowed to eat, drink or smoke in the immediate vicinity of the radioactive material? Were you allowed to have food or beverages at your immediate work location?

(27) Did you undergo medical exams and if so what did these exams involve?

(28) Did you receive medical chest x-rays as part of your medical screening while at INL and if so how often was that done? Do you remember any details about these x-rays such as film size?

Operations

(1) Describe the process for shipping and receiving of weapons, weapons components and SNM. Who was responsible for verifying inventory on incoming and outgoing shipments?

(2) What radioactive materials did INL handle? This would include Research and Development activities. What quantity of these materials were handled?

(3) When did INL start to process and/or store recycled uranium?

(4) What is the range of enrichment of uranium handled historically and currently at INL?

(5) How was material transported from one facility to another onsite?

(6) Were you ever involved in environmental restoration operations? If so, please explain.

(7) Where there machining operations onsite which involved radioactive material? If so, please explain.

(8) What were the high-risk jobs associated with the operation of the INL site?

Incidents and Accidents

(1) How are/were incidents documented at the INL Plant?

(2) Are there incident reports that are classified? If so, what percentage of the reports?

(3) How are/were minor occurrences such spills, clothing and personnel contamination, and area contamination spreads that could result in internal exposure documented?

(4) Were you ever involved in an accident, contamination spill, fires, or other incidents? If so, please explain?

(5) How frequent were personnel and/or personal effects contamination incidents?

(6) Are you aware of other major incidents that occurred at INL?

Environmental Releases

(1) Were there any significant episodic releases from INL?

(2) Were there radiological releases from INL facilities? If so, explain. When did this occur?

(3) Were you exposed to contaminants outside the facilities that might have contributed to your dose? If so, please explain.

Other Questions

(1) What chemical exposures have you received in conjunction with radiation exposure? Were these chemical exposures simultaneous with radiation exposure?

(2) Were you ever directed by management to not wear your badge while around radioactive material?

(3) Were there unauthorized practices occurring at the INL plant? If so, what were they?

(4) Are there other individuals that should be interviewed?

(5) Are there sources of historical documents you are aware of which may be beneficial to our review?

ATTACHMENT 3: SUMMARY OF SITE EXPERT INTERVIEWS CURRENT AND FORMER WORKERS

Interviews were conducted with 24 former workers and approximately 55 current Idaho National Laboratory (INL) personnel from operations, maintenance, security, laboratory support, environmental monitoring, medical, and radiation protection groups. Personnel represent experience at the site ranging from 1954 to the present. The interviews were conducted by Ms. Kathryn Robertson-DeMers (C.H.P.) and Dr. Desmond Chan from June 27, 2005, through July 1, 2005. The purpose of these interviews was to receive first-hand accounts of past radiological control and personnel monitoring practices at INL, to better understand how operations were conducted, and to hear worker concerns. Interviewees were selected by the EEOICPA site coordinator based on guidance provided by SC&A. Retired workers were selected based on their knowledge of historical operations with the assistance of the onsite worker representatives. Site experts selected represented a reasonable cross-section of production areas and job categories. Interviews with current workers were conducted onsite at INL. Retired workers were interviewed in the evenings at the local union hall.

Workers were briefed on the purpose of the interviews, provided with background information on the EEOICPA dose reconstruction program and site profiles, and asked to provide their names in case there were follow-up questions. Participants were reminded that participation was strictly voluntary. Notes generated as a result of the site expert interviews were reviewed by the Idaho Department of Energy Classification office to ensure classified or Unclassified Nuclear Information (UCNI) was not inadvertently disclosed.

INL facilities represented by the site experts interviewed included the following:

- Aircraft Nuclear Propulsion Program (ANP)
- Argonne National Laboratory-West (ANL-W)
- Army Reactor Area (ARA)
- Advanced Test Reactor (ATR)
- Battelle Energy Alliance, Inc. (BEA)
- Burn Out One Two (BOOT)
- Burial Grounds
- Bechtel BWXT Idaho (BBWI)
- Central Facilities Area (CFA)
- Chemical Research and Analysis (CRA)
- CH2M-WG Idaho (CWI)
- Engineering Test Reactor (ETR)
- Experimental Breeder Reactor I (EBR-I)
- Experimental Breeder Reactor II (EBR-II)
- Idaho Chemical Processing Plant (ICPP or INTEC)
- Loss of Fluid Test Facility (LOFT)
- Materials Test Reactor (MTR)
- Power Burst Facility (PBF)
- Radiological and Environmental Services Laboratory (RESL)

- Radioactive Waste Material Complex (RWMC)
- System Nuclear Auxiliary Power Transient (SNAP TRAN)
- Specific Manufacturing Capability Facility (SMC)
- Special-Power Excursion Reactor Test (SPERT)
- Test Area North (TAN)
- Test Reactor Area (TRA)

Also included were support personnel who worked throughout the site.

The job categories represented included the following:

- Chemical Plant Operators
- EEOICPA Coordinator
- Electricians
- Environmental Monitoring Staff
- External Dosimetry Staff
- Firefighter
- Operational and Support Health Physics
- Heavy Equipment Operator
- High-Level Waste Operators
- Industrial Hygiene and Safety Technicians
- Instrument Technicians
- Internal Dosimetry Staff
- Machinist
- Mechanical Engineer
- Mechanics
- Occupational Medicine Physician
- Pipefitters
- Radiological Control Technicians
- Radiological Control Technician Supervisors
- Radiological Records Staff
- Reactor Auxiliary Operator
- Reactor Operator
- Security Guards
- Shift Supervisors
- Transportation Workers
- Welders
- X-ray Technician
- Yardmen (Laborers)

Individuals interviewed were given the opportunity to review this summary for accuracy and completeness. This is an important safeguard against missing key issues or misinterpreting some vital piece of information. Although many individuals provided comments to SC&A, not all interviewees provided feedback on their interview summary. Some conflicting opinions arose related to documented policies for radiation protection versus actual practice. Both views are presented in the summary when available.

The information the workers provided to SC&A has been invaluable in providing us with a working knowledge of the site operations and the radiation protection program. All interviews have been documented and summarized below. The information provided is not a verbatim transcript, but a summary compiled from multiple interviews with many individuals, and with additional background information supplied to improve clarity. In addition, responses have been grouped under various categories to improve understanding. Editing by SC&A was limited to preserve some of the tone and flavor of the interviews, resulting in some redundancies and less-than-fluid writing in some places. Individuals have provided this information based on their personal experiences. It is recognized that these site expert recollections and statements may need to be further substantiated before adoption in the Site Profile TBDs. However, they stand as critical operational feedback to the dose reconstruction process. These interview notes are provided in that context; site expert input is similarly reflected in our discussion and, with the preceding qualifications in mind, has contributed to our findings and observations.

General Site Description

The INL covers an area of 890 square miles in the southeast corner of the State of Idaho. The INL site is approximately 32 miles wide and 39 miles long. The eastern site boundary is about 32 miles from the city of Idaho Falls. There are currently 530 buildings onsite. Each INL area is very much self-contained, with a full complement of utilities and services in each major area. There are about 900 workers at the Naval Reactor Facility (NRF) and 6,000 at INL. Currently, the INL site consists of INL (laboratory operations including ANL-W), Idaho Cleanup Project (ICP), and NRF. Recently, the Department of Energy (DOE) bought the privatized Advanced Mixed Waste Project from British Nuclear Fuels, Inc. (BNFL).

The Atomic Energy Commission (AEC) operations began at INL in 1949. Prior to AEC activities, the site was used as a naval proving ground for ordnance testing. The facility has undergone a number of name changes. In 1949, the site was named the National Reactor Testing Station (NRTS). It was renamed the Idaho National Engineering Laboratory (INEL) in 1978, the Idaho National Environmental and Engineering Laboratory (INEEL) in 1993, and the Idaho National Laboratory (INL) in 2005.

The INL site was divided into three distinct entities; INL, NRF, and Argonne National Laboratory – West (ANL-W). NRF was always a separate facility run by the military. ANL-W, which was formerly managed by the University of Chicago, was transferred under the jurisdiction of the Idaho DOE office in February of 2005. ANL-W and NRF had their own support staff (e.g., radiation protection, engineering, maintenance). There were times when maintenance personnel were lent to ANL-W or NRF as INL employees. NRF (the Navy) gradually separated, depending less and less on the NRTS support. ANL-W basically continued to use the NRTS/INEEL support, including the services of RESL.

In the last 18 months, the site has been undergoing transition from a single contractor to multiple contractors. The site has been split into the Nuclear Engineering Program, which is managed by

Battelle Energy Alliance, and the Environmental Restoration Program, which is managed by CH2M-WG Idaho (CWI) and Bechtel BWXT Idaho (BBWI). The Nuclear Engineering Program includes the laboratory, research and development, the Nuclear Regulatory Commission (NRC) operations, National Nuclear Security Agency (NNSA) operations, and a variety of other programs. Many of the facilities onsite have had multiple missions during their course of operation.

The current missions at the INL site include, but are not limited to, laboratory operations and environmental restoration. This has created a change of program perspective. There is a significant effort to consolidate old programs and facilities. For example, over the past 5 years, 200 buildings have been demolished; 130 of those buildings have been demolished in the last year or two. The program is in a continually changing ("flux") condition.

Facility Descriptions

The INL site has had multiple areas consisting of different programs. These areas included Experimental Breeder Reactor I (EBR-I), Argonne National Laboratory-West (ANL-W), including the Experimental Breeder Reactor II (EBR-II), Army Reactor Area (ARA) (with SL-1), Special-Power Excursion Reactor Test (SPERT), Central Facilities Area (CFA), the Idaho Chemical Processing Plant (hereafter, ICPP or INTEC, as it is known now), Test Reactor Area (TRA), Test Area North (TAN), Radioactive Waste Material Complex (RWMC), and the Naval Reactor Facility (NRF). There have been 52 nuclear reactors built onsite.

Although the NRF is located on the INL site, this facility is under the auspices of the Department of Defense (DOD). In addition to the DOD, the Office of Sciences, the National Nuclear Safety Administration (NNSA), and the Nuclear Regulatory Commission (NRC) have done work on the INL site. This work is outside the scope of the EEOICPA Program.

Argonne National Laboratory - West

The EBR-I area was involved with the experimental breeder reactor program from 1954 into the 1960s. ANL-W, containing the EBR-II and the Zero Power Reactor (ZPR) facilities, was the site of commercial grid reactor testing. ARA housed the Stationary Low Power Reactor No. 1 (SL-1 reactor) during its brief operation. SPERT was dedicated to testing for reactor excursions and reactor safety. CFA housed the support services, such as maintenance, transportation, heavy equipment, the laboratory, and health physics services.

Test Area North

The earliest program at the TAN area was the ANP, which operated from 1956–1961 for the purpose of testing reactors to power jet engines. There were two reactor engines, two rail cars, and four rails. The reactors used nickel chromium and ceramic fuel elements. The first reactor engine, P102, contained a horizontal reactor with zirconium moderator. The second engine, the Core Test Facility (CTF), contained a vertical reactor with a water moderator, and was situated on a railcar. The reactors engines were direct cycle, with air circulating through their compressors. When an engine was blocked off, the air passed straight through, was compressed through the reactor, and traveled back to the other torus. The air then passed through the turbine and out a 110-ft stack. Delay tanks and carbon filters were integrated to reduce releases from the stack. Chemical fuel was used to turn on the reactor, but once the reactor was going, it provided its own power. Since there was no containment around the reactor, essentially, this operation became a shielding study project.

For these two reactor testing facilities, the operator control room was built underground and located about 1.1 miles away from the engines. There was a tunnel, free of contamination, leading to the engine from the control room and a 110-ton shield built underground for an emergency bunker. The engine locomotive could drop down to the shielded bunker in case of accident.

The facility also performed intensive fuel meltdown testing. For example, tests were performed to determine the type of emission that would result from burn-up of nickel chromium and ceramic fuel. Monitoring stations were established out to 25 miles in all directions from the test and many detectors were placed around the site. Some radioactivity was released from the stack and deposited in the immediate area. Emissions were much higher with ceramic fuel elements than with nickel chromium fuel elements.

In the ANP Program P102 start-up, the operators overheated (temperature-wise) the reactor and the fuel elements became warped. Individuals had to enter the area and manually straighten the elements.

A huge (hanger-size) Hot Shop was built for maintenance and change out of the reactors; the largest hanger ever built for such tests. A test dolly built for equipment service purposes was used for maintenance, refueling, and servicing the jet engine. The hanger contained a large door for the dolly to go in and out. Although the contamination associated with the project was low, there were airborne releases from the units. When the P102 was moved to EBR, it experienced a mercury leak. Following an accident in January 1961, the SL-1 reactor core was transported to the Hot Shop in the TAN area for examination. The remote hot cell facilities in the Hot Shop were built for this type of work.

In 1961, President Kennedy cancelled the ANP Project, along with the Experimental Organic Cooled Reactor Project before it went to operation.

The Loss of Fluid Test (LOFT) facility was located in the TAN area of the site. Personnel encountered Sr-90 and Cs-137 contamination during breach-of-system maintenance operations, such as cutting pipes. One of the most recent uptakes of Sr-90 occurred last summer during Decontamination and Decommissioning operations. An additional hazard at LOFT was mercury.

The Burn Out One Two (BOOT) test was conducted to determine what would happen if fuel burned and completely melted down. The test was to be performed during zero wind conditions; however, airborne contamination drifted up the valley and then came back to the site due to the wind. The worst emission from this area was as a result of this test.

In the 1980s, the Specific Manufacturing Capability (SMC) facility was built in the TAN area. Operations at the SMC facility, which involved the machining of depleted uranium (DU), are considered classified. Chip fires occurred due to the pyrophoric nature of the uranium. An incident occurred when the elevated temperature resulting from a worker drilling a can containing DU reacted with the moisture inside the can to produce hydrogen gas. The hydrogen was ignited by the sparks, and flames shot up to the ceiling. This worker got burned all over his face and hands.

The Initial Engine Test (IET) Facility and the System Nuclear Auxiliary Power Transient (SNAP TRAN) project were also located in the TAN area.

Idaho Chemical Processing Plant (ICPP)

The ICCP, which began operations in 1953, was built for chemical processing of uranium from onsite and offsite fuel. This facility was originally designed as a pilot plant, but was put into routine operations. Nuclear fuel rods were dissolved in acid, which produced uranyl nitrates, fission products, and some transuranics. Solvent extraction was used to separate the uranium from other products. The uranium nitrate hexahydrate (UNH) underwent the denitrification process and was converted to powder, producing a fairly pure product. Samples were taken and analyzed through the process. Final sampling took place in Z-cell prior to shipment. Initially, UNH was shipped to Oak Ridge, and orange oxide was the eventual product of the process.

The ICPP was composed of several buildings with different functions:

Building	Operation
601	**Fuel and waste processing or reprocessing**
602	**Laboratory Services**
603	**Old fuel storage building**
604	**Rare gas collection plant**
627	**Remote Analysis Laboratory**
633	**Old Waste Calciner Facility**
637	**Laboratory Services**
640	**ROVER Dissolution Project**
659	**New Waste Calciner Facility**
661	**Security area**
666 (FAST)	**Fluorinel Dissolution Process and Fuel Storage**

The operations were divided into what is referred to as cells or wings. The cells ranged from A-Cell to Z-Cell, with the latter the final stage used for packaging of material for shipment. The radionuclide concentrations in each cell and the subsequent waste streams were dependent on the particular operations conducted in each stage. For example, A-, B-, and C-cells are associated with the presence of plutonium. The denitrification cells were associated with UNH and yellowcake; at times the yellowcake was visible in these cells. There have been two waste calcining facilities built for the processing of high-level liquid wastes. The New Waste Calcining Facility (NWCF) was built in the 1970s to replace the older Waste Calcining Facility.

A process (known as RALA) retrieved radioactive lanthanum from the fuel. This process required freshly irradiated fuel that had not been allowed to decay. As a result, there were iodine releases associated with this operation. The workers remembered that the temperature of some shipping casks coming over from MTR for the RALA Project was extremely high, boiling the water in the casks. The casks were also radiologically hot, since the radioactivity in the fuels from the MTR reactor hadn't decayed for any length of time. After the RALA Project, better insulated shipping casks were used to move radioactive fuels from Building 603 to Building 601.

Various other processes occurred at ICPP—neptunium and plutonium were recovered; Building 601 dealt with recycled uranium, which was dissolved in nitric acid; Navy fuel was processed; and the ROVER project was designed to process graphite fuel, where the fuel was burned and then dissolved in acid. The project involved multiple curies of U-233/234.

Raffinates were produced in the process of uranium extraction, and contained fission products as well as transuranics. These raffinates were transferred from Building 601 to the liquid tank farm for storage through a set of long pipes. Krypton and xenon gases were stripped in Building 604. The products were treated with nitrogen to separate out H_2, O_2, krypton, xenon, and argon. Then, H_2, O_2, potassium, and xenon were boiled off at different temperature ranges. INL had two calciner facilities to convert tank waste to crystalline power.

Test Reactor Area (TRA)

TRA began operations in the early 1950s for the purpose of material testing. The TRA was the location of the Engineering Test Reactor (ETR), the Advanced Test Reactor (ATR), and Materials Test Reactor (MTR). The Power Burst Facility (PBF) was an experimental reactor used to determine the consequences of reactor accidents; for example, fuel rod failure.

In general, the facilities in TRA were contained unless systems were breached. The highest external exposures seen in the reactor areas are among individuals who performed reactor top operations, canal operations, and reactor chemistry. The ventilation at this facility is well maintained. Emissions from the stack are filtered and monitored.

Material Test Reactor (MTR)

The MTR was the first reactor that went into operation. It was unique in that it has a neutron beam port for experimental work. MTR had very heavy concrete walls, due to concerns of possible air raid bombing. A huge graphite reflector outside the coolant tank was air-cooled. There was a beryllium reflector inside the tank with empty tubes that ran up into the reactor. There was also an aluminum reflector that held samples. All the reactors in that era were designed to handle the fuels on top of the reactors.

The MTR top assembly was shielded only by the water tank. To refuel or change fuel elements, workers had to pull the top assembly off. Fuel was pulled out of the core and moved under water; then the workers would drop it into the discharge chute and it would fall into the canal in the basement of the reactor building. During the time the fuel was in the chute, it was not in water. In all other operations, the fuel elements were kept under water. The workers used long hooks to move the fuel elements. Test fuel immediately removed from the reactor has a dose rate of approximately 10^6 R/hour, followed by rapid decay of short-lived fission products.

MTR had several horizontal neutron beam ports for experimental work, which could be plugged or unplugged to project beams to the ground floor area. The beams were controlled by temporary shielding, but they were detectable out to Highway 20. In one case, radiation leakage exposed a truck full of film, ruining the film. One beam had what was referred to as a neutron chopper. Two disks were tied together with two slots. The chopper speed could be adjusted to obtain a particular neutron energy. This unit was used to produce a neutron beam that could be used for analysis and research. There were some reported neutron leaks in areas of the reactor shield that were not intentional. During one occasion, leakage radiation set off the alarm and the area had to be evacuated.

Experimental Test Reactor (ETR)

The start-up of the ETR followed MTR. ETR was built to generate from 80 MW to 175 MW of power. Used fuel elements were handled under water. Initially, the reactor used flat plate fuel assemblies; however, these would ripple and collapse. As a result, the fuel design had to be changed. Fuel handling was done underwater. ETR supported the ANP program.

Neutron flux wires were put in the core and removed for analysis. These were typically removed by the reactor operators and/or analytical support. These wires would occasionally break and lodge in the heat exchanger. The heat exchanger room was only accessible during reactor shutdown. When broken wires lodged in the heat exchanger, this area had high dose rates. Eventually, there were heat exchanger problems that required repair, and workers entering the area to perform maintenance on the system were exposed to high dose rates.

Waste material was often stored in the canal. This waste had to be pulled up out of the canal water and placed in a cask. This operation required notification of personnel in the office areas, and that personnel not involved in the operation evacuate to the north end of the building to minimize their radiation exposure. The radiation alarms would sound as the waste was removed. The operations took a few minutes to complete. Rarely, a used experimental loop of stainless steel was pulled out of the top of the reactor by crane and lowered into the canal. In one incident, the loop was dropped and operators involved in the operation were over-exposed.

Advanced Test Reactor (ATR)

The ATR, for testing nuclear fuels, was the last reactor built and is the only one currently in operation. The ATR is a 250 MW light-water-cooled reactor with beryllium reflectors and 48-in long fuel at 93% enrichment. The fuel is handled from the top of the reactor. One of the missions of the reactor was to produce radioactive materials for commercial use. Capsules containing materials were inserted in the beryllium blocks of the reactor along the outer edge. Some of the radionuclides produced in these capsules included iridium and cobalt. There were no beam ports or neutron leaks at ATR. Hot cells were built for fuel element examinations and other testing. Operations of this type continue today.

There were two sets of operations personnel at ATR. The first was composed of non-exempt reactor operators responsible for operations in the canal area of the reactor, including moving fuel. Some operators were also responsible for routine maintenance, inspections, and cleanup. Exempt operators had jurisdiction over the tanks and the reactor proper, and were responsible for keeping the reactor up and operating, and inserting capsules into the reactor.

Primary system inspections and maintenance were conducted when the reactor was shut down. The time post-shutdown for re-entry into areas off limits during operations has varied. Initially, personnel were allowed to enter these areas a few hours after shutdown. The loops and primary system were not always depressurized prior to entry into these areas, resulting in higher dose rates. In the last 10–15 years, re-entry has not been allowed for 3-7 hours after shutdown and depressurization of the primary system and loops. During the same time period, an argon delay system was installed.

There were several safety hazards associated with the operation and maintenance of the ATR. Exclusion areas at the reactor include the heat exchanger area, the pit area, the area around the outer shim cylinder, the rod access area, the pipe corridor, and the Subpile room. The Subpile room was probably the most contaminated area at ATR, with contamination the result of byproducts that came from the reactor system and, sometimes the fuel. Leaks in pipes, valves and packing, and capsule ruptures also resulted in some contamination. Pressurized steam represented a substantial safety concern. In general, ATR operations and practices had the benefit of learning from mistakes observed at both ETR and MTR.

Radioactive Waste Material Complex (RWMC)

The RWMC represents one of the current active areas onsite. The RWMC was designed to dispose of waste, both in underground burial sites and in aboveground storage areas. Much of this waste was shipped to INL from the Rocky Flats Plant and Mound in barrels or burial boxes, and contained considerable amounts of plutonium in a variety of forms. At the burial site, the waste containers were dumped into what is referred to as pits or trenches and run over with heavy equipment. Site experts observed lids coming off some units, and causing surface and in-depth contamination. The primary radiological concern was contamination and airborne resuspension of particles.

There is an ongoing effort to retrieve these compromised barrels, repackage them, and ship them to an appropriate facility. Experience thus far has indicated that waste drums and boxes include plutonium chips.

Workforce Characteristics

The worker population in eastern Idaho was a closed population. There has not been a large movement in the last 30 or 40 years except during site management changes. In fact, workers are moving to the site rather than leaving the site. For example, workers from Rocky Flats have moved to Idaho to work at the INL site.

The mobility of the workforce onsite was dependent on the period of operation at INL and the contractor. At times there was a central maintenance pool located in the CFA area that provided support to the entire site. During other periods of operations, support staff was assigned to particular areas. Some maintenance personnel worked in teams and were assigned routinely to a particular facility. Larger facilities had dedicated maintenance support. With the current contractor, maintenance has been reorganized into a central group at CFA. Some support staff worked all over the INL site, including ANL-W and NRF. The non-exempt reactor workforce was more stable, because of the large investment in training. There was a higher turnover rate among exempt staff.

When there were particularly high dose rate jobs that required limited stay times, supervisors, security, and other work groups were utilized to perform simple tasks to prevent maintenance personnel from exceeding their allotted maximum dose; often the chosen personnel had no training for the job they were told to do. Maintenance personnel were saved for more difficult and intricate tasks. For example, in the 1970s, there was a secretary in the ICPP office sent by the management to perform a job in the Calciner facility. They gave her a heavy saw and told her to cut out a pipe for a valve replacement job. She had no idea what to do or how to do it. Other untrained workers would be sent in the hot cells without any training on how to follow hot cell entry requirements. Although there was RCT coverage, the RCT was responsible for multiple jobs during the day. These management decisions were made by the Operations Manager or Production Manager. These untrained employees in the Calciner facility and other high dose rate facilities could receive high doses in the range of 3,500 to 5,000 mrem per year.

The amount of overtime worked depended on the job function and the period of time. Historically, there were more overtime hours for maintenance, operations, and security. Part of this was due to the limited number of staff available for certain operations. Maintenance and operations personnel approximated the number of hours worked per week to be 40–45 hours. There were rules established that restricted operations to a maximum of 72 hours per week, unless there was a very special situation. Overtime was especially prevalent at ICPP when the calciner was in full operation in the late 1980s. The security guards worked up to 70-80 hours per week during the 1980s, due to limited staffing. Overtime is limited at the present.

Security

Security had a number of responsibilities, including guarding incoming material (e.g., material delivered to RWMC), escorting individuals, Special Nuclear Material (SNM) monitoring, facility inspections, escort of classified documents, and other routine security tasks, such as traffic control. These job responsibilities took security personnel all over the site.

Material receipt and guarding could be required for up to an entire shift prior to transfer to the appropriate facility. Shipments were received from all over the DOE complex and, in some cases, from the Navy and commercial entities. There were even foreign fuels shipped to the site.

Escorting individuals often resulted in questionable exposure conditions. For example, the guards were stationed just across the radiological rope from the 603 Cave with no air space between them and the work area. The cave door was left open and, while individuals inside were in airline respirators, guards outside were simply in their uniforms. Radiation postings were variable in some of the areas that guards were required to tour. Tours of the ICPP building involved visits every 30 minutes to the denitrification area, all corridors, the sampling areas, and analytical laboratories.

Tackle testing exercises resulted in access to areas where others did not routinely go, such as areas at the NRF facility and on facility roof tops. Guards noted that the original area outside the fence that was used for some exercises was unposted in the past, but is now posted. As this area was not anticipated to cause radiation exposure, many did not wear their film badge, as it was easily lost in maneuvers.

Several facilities in the DOE Complex, such as the Rocky Flats Plant, Brookhaven National Laboratory, and the Y-12 Plant, were periodically short-staffed. As a result, INL security personnel were sent to these sites to augment the existing staff for short periods of time. In addition, security personnel also participated in exercises at Lawrence Livermore National Laboratory.

Radiological Operations

Organization

The INL radiological control (RadCon) program was brought in from the Oak Ridge National Laboratory (ORNL). The air sampling program, the personal dosimetry program, radiation protection program, and some professionals were transferred from the ORNL programs.

Before 1997, there were multiple companies at the INL site. Former contractors at the site included WINCO, Allied Chemical, Exxon, Phillips Petroleum, Westinghouse, Lockheed Martin, EG&G, BBWI, and CWI. In the earlier years, Phillips Petroleum was the operating contractor of the main site. EG&G later took over this contract, except for the Idaho Chemical Processing Plant (ICPP), which Allied Chemical took over. Historically, the site could be divided into four entities—ANL-W, ICPP, INEL, and NRF—with a different contractor managing each different area. In October 1994, ANL-W was absorbed into INL proper, and in February 1, 2005, it was put under the jurisdiction of DOE-Idaho rather than DOE-Chicago. There are also several privatized facilities, like the Advanced Mixed Waste Project and the Pit 9 Project. Firemen and guards originally worked for the AEC.

Each contractor onsite was responsible for maintaining a Radiation Safety Program and was required to comply with the same regulations. Programs such as dosimetry, instrument calibration, radiobioassay services, and environmental monitoring were originally the responsibility of DOE-Idaho, but were eventually turned over to the contractor. There have been periods of time (i.e., during the Phillips Petroleum through the EG&G and CWI eras) when field RadCon personnel were matrixed out to facilities. Historically, individuals working on the backshift have reported to the Shift Operations Manager. Fundamentally, the approach to radiation protection should have been the same throughout all facilities; however, there were facility-level differences. For example, some facilities allowed individuals to wear street clothes under their PPEs, while others required company clothing. Some RadCon procedures were specific to a certain facility. Currently, field RadCon personnel report to operations at the particular facility, and are matrixed to the RadCon organization.

There is currently only one radiological protection program (RPP) at INL, applicable to subcontractors as well. The radiological control (RadCon) organization is currently composed of a program director, field RadCon managers, supervisors, Radiological Control Technicians (RCTs), radiological engineers, and a central technical support group. RadCon field operations support a variety of projects, including nuclear naval program testing, reactor outages, maintenance activities, decontamination and decommissioning, and routine area entry and egress. The RCTs are responsible for the hands-on radiological protection work, including routine and job-specific surveys, job coverage, ensuring workers comply with radiation protection rules, etc. Radiation Engineers in the field are mainly responsible for work planning, including determining internal and external monitoring requirements. The central technical support group provides procedure and technical basis development and oversight of the dosimetry, records, instrument calibration, and air sampling programs. For example, one manager indicated they used the central group to determine correction factors with respect to source strength difference for their monitoring devices (e.g., dosimeters). In 1974, the RCTs joined the bargaining unit. INL now hires rent-a-techs to fill some of the radiological control staffing needs.

Historically, workers at TRA and RWMC were often loaned to other facilities when needed. Sometime, workers themselves bid out to other facilities. With the transfer of the contract to CWI, this practice has stopped.

According to the RadCon staff, refueling at NRF did not use any workers from other INL facilities. There were construction workers who provided support services at NRF. NRF was a very clean facility in terms of radiation protection. In fact, from an air sampling standpoint, it is cleaner inside of the facility than outside.

Radiation Protection

Historically, worker and management mentality indicated a lack of concern for safety issues. The field managers had very little control over this situation. The operations staff was often instructed by management to do whatever they could during their shift to get things done. In the past, the private construction contractors came, got their job done, and got out. They were always in a hurry to get things done and pushed the workforce hard. This attitude remained consistent through the various changes in contractors.

Radiological control practices and policies were less stringent prior to the arrival of Westinghouse. After Westinghouse took over in 1984, they cleaned up most of the problems at ICPP. There was an improved contamination control program and better equipment was provided to field support. There was also a noticeable improvement in practices and worker knowledge of hazards in the workplace as training improved.

As late as the early 1980s, there was a casual attitude toward RadCon. In the last several years, the pendulum has swung the other way, and controls are much more restrictive. For example, there is excessive use of Personnel Protective Equipment (PPE), and Radiation Work Permits (RWPs) are more specific to a particular job. There were several reasons for these improvements, including health and safety audits, public pressure, and lawsuits. The change in attitude may be partially due to the DOE Tiger Team audits. The Defense Nuclear Facility Safety Board (DNFSB) also came to Idaho for several visits in the 1980s and 1990s. Some site experts felt that DOE-Idaho did little to encourage improvement in the safety program, and that in fact, DOE-Idaho was contractor-owned. Some of the improvements in radiation safety are attributed to improved instrumentation and policies. Site experts also noted that there have been improvements in training, conduct of operations, and chemical sampling.

From the 1950s to the 1970s, the ACL was 3,000 mrem/quarter. Dose was controlled on a weekly basis, with an allotted external dose of 300 mrem/week, or 600 mrem/2 weeks. These 2 weeks could be sliding 2 weeks. With the signature of the Plant Manager, a worker could get up to 900 mrem for a particular job. In the 1980s, the ACL was lowered to 1,500 mrem per year. It was lowered further to 1,000 mrem/year, and most recently has been dropped to 700 mrem/year. The goal was to maintain radiation exposure for an individual within 80% of the radiation limit. For some jobs, workers were authorized to receive 2-weeks exposure in 1 week.

The maximum allowable exposure for a given time was 900 mrem in 1 week. Limits could be reached within a matter of minutes for some jobs and required stay-time restrictions. Workers were typically reassigned to non-radiation work when they are bumping up against the ACL.

Field Procedures

In the 1950s, the INL site began to use Safety Work Permits (SWPs), which covered all aspects of safety concerns, including radiation, chemical, and industrial safety. Safety information provided on this form was of a more general type. Either the permit and/or the RCT communicated the Personnel Protective Equipment (PPE) requirements.

In 1984 or 1985, the site switched to Radiation Work Permits: both job-specific and general RWPs. In the early days, the permits were primarily job-specific and had to be closed out on a daily basis. The general RWPs were for routine work with constant radiological conditions. Routine verification surveys were carried out to ensure radiological conditions remained unchanged. The job-specific RWPs were for work with changing radiological conditions. In this kind of RWP, pre-job survey, ongoing survey, and/or post-job surveys would be done. There were also ALARA calculations performed to estimate job-specific dose. RWPs were used for cell entries at ICPP and other high-risk jobs around the site.

Initially the RCTs prepared the RWPs and determined the requirements. An RWP would be submitted to the foreman, the operations manager, and, lastly, the area manager for review and approval. Permits included PPE, job-specific dose limits, stay time, area dose rates, and other precautions as necessary. These requirements were determined based on process knowledge and experience. Dosimetry and bioassay requirements were not listed on RWPs until the 1990s. Currently, there is a more formal procedure for writing RWPs. Historically, the focus was on weekly limits. As a result, the original work permits did not have void limits. Radiological controls were based on past radiation and contamination surveys, field air sampling data, and former experience. RCTs would rely on pre-job air sampling to make adjustments to requirements for workers before entering an area. A RCT could initiate a 600 mrem limit (for 2 weeks) for a worker with signatures from the shift foreman, shift manager, and senior manager.

Radiation worker training was offered to employees at the site. The training addressed the different types of radiation; PPE; the concept of time, distance and shielding; and the proper way to don and doff personnel protective clothing. Annually, workers received refresher training on RWPs and PPE. There was a re-qualification process. There appears to be less radiation worker training now than before.

Radiation Characterization

At INL, the contaminants mainly consist of U, Pu, Cs, Am, Ce, and Sr. Uranium was associated with the dissolution and extraction processes at Building 601, SMC, ICPP, ATR, ARA, and SPERT areas. Plutonium was identified at many locations onsite, including ICPP, RMWC, ANL-W, the HFTF hot-cell facility, and the TRA. There were a number of places at the ICPP facility where plutonium and americium were identified, including A-cell, B-cell, C-cell, the Solvent Burner, and locations involved with the extraction process. There were trace amounts of plutonium in recycled uranium received onsite. Mixed Activation Products and Mixed Fission Products were predominant radionuclides of concern onsite, especially Cs-137 and Sr-90/Y. These are found in facilities throughout the site.

Radionuclides associated with particular processes included:

- At the test reactors, Co-60, Ir-192, and Cs-137 were the major concerns.
- The ROVER program processed graphite fuels with U-233/U-234. Uranium-233 was not uncommon in ORNL fuels process.
- Tritium was found at STRA, BDA, and in low levels at RWMC.
- At the ICPP radionuclides of concern include Sr-90/Y, Cs-137, Kr-85, Pu-238, Pu-239, Am-241, iodine, and xenon.
- At Cell 603 at ICPP, the primary radionuclide of concern was Cs-137.
- Depleted uranium is of concern at SMC.
- Just recently, a vial of U-233 was found in an old building.
- There was/is soil contaminated with plutonium, strontium/tritium, and cesium.
- Waste transported during clean-up projects resulted in potential exposure to a number of radionuclides.
- Analysis of waste water from injection wells indicated the presence of tritium along with Sr-90.
- Tritium was found in the waste-water pond east of TRA.

At the reactors, when sodium-clad fuel elements reacted with water, the fuel and fission products would leak to the water in the storage basin causing contamination spread. This would cause uptakes and chronic exposure problems for workers. General exposure rates were estimated at 1 to 2 mR/hr in limited areas in the fuel storage and transfer areas.

Although external exposure was the predominant radiation hazard, there were areas in the reactor that were contaminated as well; for example, there was contamination on the top of reactors and a lot of sealed capsules in aluminum tubes in the reflectors. There was a spider (basket) device in the end of each tube, so that coolant could go through. The capsules that held sample material would sometimes leak and cause contamination in the coolant water. The water was carefully monitored for unexpected contamination. Also, the tools used to manipulate items in the reactor pool were taken in and out of the reactor coolant water. In 1988/1989, there was a large release of tritium from the TRIGA Reactor in Building 604.

Both internal and external exposures were of concern at ICPP, with the former the primary concern. Pure alpha contamination was found at the Research and Development laboratories. The B-cell probably had the highest potential for internal exposure. Process Cell 601 was involved in the characterization and processing of uranium. ICPP had processed both enriched uranium and recycled uranium. Graphite fuels were also processed at ICPP. Fission products constituted the main radionuclides of concern and occurred in some areas with alpha emitters. There were in-process ambient sources like Cs-137. There were potential skin doses due to Sr-90/Y and Kr-85. At the calciner, the radionuclides of concern were Cs-137 for external exposure and Sr-90 and plutonium for internal exposures.

Most of the fuel processed onsite was aged fuel of 1–3 years decay, with the exception of fuel associated with the RaLa process, which processed fuel immediately upon reactor discharge. The Pu-238 and fission products were extracted and separated from uranium, and the majority of the extracted uranium product was shipped to the Y-12 Plant. There were concentrated fission products and transuranics in the aqueous raffinate. Raffinates, a significant exposure hazard, were transported to 50,000-gallon underground storage tanks. Later, a calciner was built to granulate the wastes. At ICPP, there was an aqueous raffinate spill when a crack formed in the hydraulic fluid line, producing dose rates greater than 500 R/hour up to 15 ft away.

Internal exposure was/is of primary concern at the RWMC. Pure alpha contamination was/is found at RWMC, due to disintegrated waste drums. There were/are plutonium products in the wastes from RFP.

Tritium was a hazard in limited areas onsite. It was formed as a tertiary fission product in the high-enriched fuels. There were also tritium experiments onsite. At LTSF, drums were being retrieved from a carbon steel vault and caused a tritium issue. In the 1980s, there were tritium leaks in the gas plant. The operators had to wear bubble suits (not a routine procedure) to work in the area. Workers involved in this operation indicated that there were no recorded doses associated with tritium exposure. There is a still a trace amount of tritium in the surface soil in the gas plant. The site discharged tritium-related wastes 600 ft into the aquifer during the initial years of ICPP operation. Although tritium was present onsite, it did not constitute a significant source of exposure.

Radiography

Radiation generating devices include x-ray diffraction units, several accelerators, radiography sources, and calibration sources. Citrix was a neutron generator facility that had boundaries established to limit access to certain areas. Outside these boundaries, the dose rate was generally less than 5 mrem/hr. Inside the boundaries, the dose rate was in the mR/hr to R/hr range.

Maintenance personnel would periodically modify experimental loops in the reactors and in cubicles where the cooling systems were located. These experimental loops were used to support Navy testing. Maintenance was also involved in repair of systems and reactor components. Radiographers would x-ray welds to insure they were intact. The process of welding and radiographing welds took a number of iterations. Portable x-ray machines and shielded Co-60 sources were used for radiography. Both the maintenance crew and the radiographer often received significant doses from the radiation in the cubicles over a short period of time. Sometimes the workers would exceed their dose limit of 100 mrem/week. They then would need approval from management to allow them higher exposure to finish the job. During reactor shutdown, use of radiography sources was common.

Radiological Contamination Control

Survey and posting requirements were/are based on the existing radiological conditions (e.g., contamination level) and potential problems in an area. In the beginning, different color codings designated contamination areas; red, yellow, and blue. Different radiation areas were also established. Prior to the implementation of the complex-wide posting requirements, INL designated contamination areas as Zone 1, Zone 2, and Zone 3. Contamination limits for each zone were as follows.

- Zone 1: 200–5,000 dpm/100 cm^2 β; 20–100 dpm/100 cm^2 α
- Zone 2: 5,000–20,000 dpm/100 cm^2 β; 100–2,000 dpm/100 cm^2 α
- Zone 3: >20,000 dpm/100 cm^2 β; >2,000/100 cm^2 dpm α

Personnel Protective Equipment (PPE) and Respiratory Protection requirements were dependent on the radiological conditions of the work area. Single, double, and even triple anti-Cs (including coveralls, shoe covers, boots, etc.) were worn, depending on the level of contamination. Wearing multiple layers of Anti-Cs helped reduce beta exposure. Waterproof Anti-Cs were worn when working in wet areas, such as at reactor facilities. Bubble suits were used in the mid-1980s for the ROVER project. Some work locations were so contaminated that even three sets of PPE did not protect a worker from skin contamination. Zone 3 represented the most contaminated areas.

- Zone 1: Shoe covers, gloves, modesty clothing, single pair anti-Cs
- Zone 2: Shoe covers, two pairs of gloves, modesty clothing, double set of anti-Cs, no respirator
- Zone 3: Shoe covers, two pairs of gloves, modesty clothing, double set of anti-Cs, respiratory protection

Later the criteria for determining PPE requirements changed. A single set of PPE was required in areas with 1,000 to 100,000 dpm/100 cm^2 β/γ. A double set of PPE with respirators was required in areas with contamination levels larger than 100,000 dpm/100 cm^2 β/γ. Plastics were used when liquid was present. Other safety factors, such as temperature, also influenced the PPE selected for a job.

Respirators were often used as a preventative measure, and were issued to workers. The workers were taught how to maintain their respirators, and the respirators were inspected routinely. In later years, there was a single-use respirator policy for radiation and a multiple-use policy for chemicals.

Respiratory protection ranged from half-mask respirators in the early days to airline respiratory protection for some of the hot cell jobs. Half-face respirators generally did not provide a tight seal. The respiratory protection was upgraded from a half-face to full face if the protection factor with the half-face was not high enough. In the early 1980s, INL started to change from routine use of half-face respirators to full-face respirators. For highly contaminated areas such as hot cells, airline respirators or Scott Air Paks with compressed air supplies were used in place of a full-face respirator for particularly high-hazard work.

At the reactors, workers and operators had to wear Anti-Cs, shoe covers, and gloves. Occasionally, respirators were used for some jobs. The bridge across from the entrance to the reactor top was normally roped off.

Personnel Protective Equipment use did not always seem to be consistent between workers on the same job or over the course of time. For example, respiratory protection for ICPP hot jobs initially required a full-face respirator. Airline respiratory protection was eventually implemented with some jobs. The site eventually discontinued maintenance of the airline system and switched to Positive Air Purifying Respirators (PAPRs). No respiratory equipment was used for routine sampling. Process hot cell work initially required only a PAPR. This was upgraded to an airline respirator for grinding, cutting, or use of pneumatic tools in the mid-1980s.

In the current work at RWMC, permanent staff may wear respirators; however, vendors do not. Historically, there were higher contamination levels in the work areas. Respirator equipment was not always used during sampling operations. In some situations, two workers in the same contamination area or high radiation area wore different levels and types of PPE. Management explained that there were different sets of rules that were applicable to each worker.

Radiological Control staff indicated that RCTs observed every individual on a major job. Workers indicated that the RCTs at different facilities approached their work differently. Some RCTs were very thorough in their coverage, while others were laid-back. Starting in the 1980s, coverage for high-risk jobs, such as hot cell entries, was constant. Per procedure, RCTs were expected to ensure that individuals were wearing their dosimeter and to control exposures within approved limits. Dosimeter usage was sometimes difficult to police, since they were often worn under Anti-Cs. Remote operations and shielding were used to minimize dose on high-level jobs. Process sampling in the operation area was handled in fully shielded gloveboxes in a remote part of the facility.

Self-survey was part of the doffing process for routine work. After finishing a job and prior to exiting, workers were required to change out their work clothes in the hot side and perform self-surveys before they went over to the cold side. For egress in a high-contamination area (HCA), RCTs were required to perform frisking before the worker was allowed to pass through the Personnel Contamination Monitor (PCM). Pre-job surveys were done to access areas prior to the start of high-risk jobs. Objects leaving the area required survey by an RCT. When they became available at INL, individuals would pass through portal monitors before they exited a control area or left the plant site.

Although eating, drinking, and smoking were not permitted in the immediate operations area or protected area, some site experts indicated that this was allowable in Radiation Buffer Areas. Operations maintained a coffee pot inside the control room, which was inside an RBA. The control room was in the lower level of the building and was next to the sampling room. Operators carried water and food through the upper-level corridor beneath some leaky pipes. Then they had to walk down the steps to the lower level and go through a major operating corridor to get to the control room.

Building 601/602 facility housed a cafeteria. Historically, workers were allowed to wear a lab coat over their working clothes and shoe covers during lunch to get to the cafeteria. This was no longer allowed as of the 1970s. The cafeteria was essentially surrounded by a contamination area. At one time, high contamination was spread all over Buildings 601 and 602, including to the cafeteria. A new cafeteria was eventually built in a separate building.

Dressing rooms were the demarcation line between the cold and hot side of the facility. Workers were required to change their clothes and shoes and put on full Anti-Cs (particularly laboratory workers). Some site experts indicated that workers ate inside the change room. Some site experts used to eat on the "clean" side of the change rooms, immediately adjacent to the area where workers wore their plant clothing or frisked out of contamination areas.

Historically, ICPP was very lax and contamination spread occurred on a daily basis. Some of the contamination spreads involved non-radiological areas. For example, in 1979/1980 management closed down the cafeteria due to a contamination spread all the way to the secretary station in the INL vice president's office. They found the contamination level at the secretary's chair reading 10,000 dpm beta and gamma. As a result of this incident, portal monitors were installed.

Site experts referred to the "ICPP shuffle," which was the practice of shuffling in gravel to remove contamination from shoes. Machinists reported that chips of contaminated metals were often present in the CFA machine shop.

Laundry services were available to pick up contaminated laundry and delivered clean laundry. Laundry workers used water-soluble bags, which could be placed directly into washers. A truck was used to transport laundry to and from the various onsite facilities. The truck was divided in half with the clean laundry on one site and the contaminated laundry on the other side. The contamination level on the laundry was generally low. The waste water was discharged to the leaching pond.

In general, contamination control was based on providing the worker with PPE and decontaminating the affected areas whenever feasible. These methods of controlling contamination, however, were not always effective. Personnel and personal effects contamination was an issue at many facilities, particularly ICPP. Discussions with site experts indicate that there were numerous personnel contamination incidents. The incidents were usually associated with hot cell entries, evasive maintenance activities, or release of material from contained systems. In fact, a majority of the site experts involved in hands-on work had been involved in one or more contamination incident. In some cases, they lost their shoes, clothing, and even their hair. Although nasal smears were a part of the contamination control program, not all individuals with positive nasal smears were required to receive special whole-body counts (WBCs), urinalysis, and/or fecal samples. In fact, if the worker was able to remove the contamination, the incident was not even documented.

Personnel Decontamination

When there was a personnel contamination incident, workers were required to scrub the contaminated area with soap and water; the RCT would have the individual scrub the affected area with abrasive material, which, in some cases, caused bleeding. Another method employed was putting tape over the contaminated area for the night. If the contamination was particularly fixed to the individual's skin, as a final step, $KMnO_4$ (the purple cow) was put on the area. Further scrubbing was sometimes necessary. There was some concern over the fact that the old decontamination room supervisor was mixing the solutions incorrectly. In the early days, there were no designated decontamination personnel at ICPP. Management would use different workers to perform decontamination. This approach was also used in other areas. Equipment was often decontaminated with the used of trichlorethylene (TCE). In the early years, if an individual got contaminated, they washed off the contamination as best they could and did not report it to keep themselves out of trouble.

Air Sampling

The purpose of air sampling at INL was to characterize the condition for workers, to verify posting, and determine the appropriate level of worker protection. Air sampling results were trended to identify low-level increases in concentration. An upward trend would result in an investigation of the cause. Most of the airborne radioactivity generated was the result of resuspension of contamination. As a result, contamination surveys were also trended to monitor for potential airborne hazards.

The air monitoring program has included fixed air sampling at ICPP, job-specific air sampling, real-time air monitoring, and lapel air sampling. Continuous Air Monitors (CAMs) were used routinely from the 1950s forward for the detection of unexpected increases in airborne radioactivity in occupied area of most facilities. In 1981, job-specific air sampling was implemented (it was not used before). Breathing zone air sampling was not used often at the INL. There was no air sampling in high-contamination areas according to some site experts.

The collection of krypton gas was performed for two to three batches every shift. There were CAMs placed in the facility to monitor the leaking. When there was enough in the air, the CAMs would sound the alarms. The operators would notify the RCTs and then continued working without stopping. The operators would wear bubble suits while bottling the krypton gas. Xenon gas was not collected, because there was no strategic value for it. There was no effective leak-testing of the bottles. The operators would just open the valve of the bottle to see whether it triggered the CAMs.

Instrumentation

One of the responsibilities of instrument technicians was to maintain radiation protection and criticality safety instrumentation. Instrumentation under their jurisdiction included PCMs, Radiation Area Monitors (RAMs), and Criticality Alarm Systems (CASs). Work involved calibration of RAMs and CASs. Collimated, directional point sources were used during the calibration process. The CASs in Building 601, Building 640, and Building 602 are General Atomic systems. Those at the FAST Facility are made by Victoreen. When the technicians performed calibration, they normally used a collimated, directional point source. Neutron sources at FAST used cadmium as an absorber. At the ROVER facility, neutron sources were doped with poisons.

Area Dosimetry

Area dosimeters were placed in different locations to monitor work areas and perimeters. MicroR meter surveys were also performed in low potential areas to ensure postings were correct and to evaluate any trends. If there was an elevated dose rate, occupancy restrictions in the area were considered and the need for personnel monitoring re-evaluated. Although there may have been an absence of neutron monitoring in some areas of ICPP, there were wall-mounted neutron systems used in the corridors of some buildings. Also located in these areas were emergency dosimetry systems.

External Monitoring

Characterization of external exposure as acute or chronic is dependent on the work location and job assignments, and individual interpretation of the meaning of acute and chronic. In general, acute radiation exposures are more prevalent than chronic exposures. Radiation exposure is better characterized for some individuals as fractionated, where there were high exposures for short periods of time (well within established standards) followed by periods of no exposure at all. For example, ICPP had remote operations, but not remote maintenance, and, typically, there would be a lot of high-dose maintenance jobs over a short period of time. In some situations, workers could perform one to two jobs in a year and reach their allotted radiation exposure limit. There were some facilities at which exposure is better characterized as chronic (e.g., SMC, work outside hot cells). Most radiation doses at the reactors were due to exposures during shutdown activities.

Maintenance and other personnel (health physics, operators, etc.) were involved in work leading to the receipt of higher total lifetime doses. For example, maintenance personnel interviewed indicated their cumulative exposures while working at the INL site ranged from 5–32 rem. Dose to maintenance workers has seen a large decrease over time. For example, a worker may have received 3–5 rem per year in the past, whereas his current annual dose may be approximately 100 mrem per year. In some cases, personnel exceeded the limits of the time period.

NRF, ANL-W, and INL used HSL dosimetry services initially. ANL-W has received radiobioassay, instrument calibration, and dosimetry services from HSL since the initiation of dosimetry services. In the early years, each entity used the same dosimeter. In the early 1960s, NRF became independent and eventually changed dosimeter types.

Calibrations

For calibration, INL used a panoramic Cs-137 source starting in the early 1990s. The source was stored in a lead pig. The source also had an uncertainty of ±20% with <28% drift. Prior to this, DOE performed Cs-137 calibration using a beam irradiator and supplied the calibration to the contractors for a short period of time. Californium-252 was used as a neutron calibration source. The neutron albedo dosimeter was not calibrated with a neutron source. The TLD reader was calibrated using Cs-137 irradiated TLDs. The referenced Cf-252 neutron source used for the past few years replaces an older AmBe source, which was maintained by DOE in the mid-1990s. The neutron source TLD irradiations were used as a quality assurance check of the neutron TLD processing program.

External Dosimetry Evaluation

In 1998, a study was completed (EGG-1-98-04 or EDF/RDR-98005) to characterize neutron and beta-gamma fields in the various workplaces to verify that the monitoring badges were appropriate for the radiation fields. INL bought a multi-sphere from Chalk River and a BTI B-G filed spectrometer for use in this study. Dosimetry personnel spent 2 years studying the response of neutron and beta/gamma badges. In 1998, the study underwent an independent review by Idaho State University. The technical basis for the monitoring systems used at the INL have been documented in *Technical Baseline for INEEL Personal* γ-β *Dosimetry* (INEEL/EXT 01-00636), *Technical Basis of the INEL Personal Neutron Dosimeter* (Report 960112), and *Technical Baseline for INL 6776 Personal Neutron Dosimetry* (RPT-131). Angular dependence was one of the items reviewed in these studies. Neutron spectral analysis was also included as a part of this study.

There have been many studies to characterize the neutron spectra at INL, such as at the MTR, ETR, and ATR reactors. A Bonner Sphere spectrometer technique was used to document the field strength and to calculate the n-γ ratio. The 3" to 10" sphere ratio was used to determine the neutron field strength.

Beta/Gamma Monitoring

The issuance of dosimeters began immediately in 1951 and calibration records are available back to October 1951. The INL site benefited from the experience gained at other DOE sites. Dosimetry calibrations were performed by AEC Health and Safety Division, currently named Health Physics Instrumentation Laboratory (HPIL). The AEC was responsible for all the dosimetry equipment, calibration, bioassay, and environmental monitoring until January 1989, when contractors took responsibility for personnel dosimeter services. Effectively, the same individuals continued to perform the dosimetry service. One change was the requirement for calibration using National Institute of Standards and Technology (NIST) traceable sources. At times, this required that the specialized calibration services be provided by outside facilities, such as the Pacific Northwest National Laboratory. DOE served in more of a quality assurance role. The dosimetry program is currently in compliance with the requirements of the Department of Energy Laboratory Accreditation Program (DOELAP). The dosimetry systems had good sensitivity over the years.

Personnel may have worked at a number of different facilities within INL proper, such as ICPP, ETR, ATR, MTR, RWMC, and CFA. If an individual was assigned to more than one facility (e.g., maintenance, heavy equipment operators, etc.), the individual would have had a separate dosimeter for each facility. Regularly, badges were stored at the guard gates at each facility. Rack control badges were used to determine background dose rates. For the last couple of years, workers have been allowed to take their dosimeter home. However, most workers continue to leave their badges in the storage racks provided in the guard gates. Workers were told to wear their dosimeter on their chest, between the neck and waist, since the inception of the dosimetry program. In cases of potential partial-body exposure, RCTs could evaluate whether individuals needed to reposition their dosimeters. Historically, this was communicated as a part of radiation training. All badges were designed to wear or clip on the chest. The neutron dosimeter was combined with the standard dosimeter.

Film badges and pocket ionization chambers (PICs) were used regularly by exempt support personnel associated with reactor operations. Prior to 10 years ago, a worker was required to wear a badge and a pocket dosimeter before going into a radiation area. Sometimes they also wore chirpers. For some period of time, the entire workforce wore dosimeters, including non-radiation workers, subcontractors, and construction workers. Currently, every worker entering a Radiation Buffer Area (RBA) is required to wear a TLD.

Visitor, subcontract, and construction worker doses were maintained with the permanent dosimetry records. There was also a visitor form that allowed individuals to request their exposure record. Dose received by other DOE complex employees may have been sent to the particular DOE site of concern. At INL, contractors (companies) received monthly reports on individual worker doses received. Historically, the REASON or EXCUSE code in the dosimetry databases identified visitors.

Neutron Monitoring

From 1951 to the mid-1970s, neutron monitoring was performed mainly by using Kodak nuclear track emulsion Type A (NTA) film. NTA film was used for measuring fast neutron exposure, and has a detection limit of 10–14 mrem, with an upper threshold of 20 rem at a ±30% accuracy (documented). Thermal neutron exposure was not monitored until the thermoluminescent neutron dosimeter (TLND) was implemented, with the exception of some boron-lined PICs used at MTR. The results were not integrated into the "legal" (RESL) dose of record. In 1975, the albedo dosimeter replaced film.

NTA was only effective for neutron energies of 0.5 MeV and above, and calculations were performed to estimate the total neutron dose. Sometimes, thermal neutron beams also had to be taken into consideration. Dale Hankins began his neutron dosimetry career at the MTR, published widely, and developed the multiple sphere Bonner Spectrometer system, including the 3-in/9-in ratio technique for determining the average neutron energy. Hankins moved to LANL, but left the technology at INL. The neutron exposure attenuation was significant. To monitor the real-time neutron dose, a neutron-to-photon ratio was applied to the standard PIC reading.

These neutron-to-photon ratios were determined by field RadCon through survey instrumentation readings.

Neutron monitoring devices were limited to specific areas of the site and not widely distributed to the workforce, as only a small percentage of the workers were involved in work resulting in neutron exposure. There were several areas where neutron surveys were performed routinely throughout the site. For example, neutron surveys were made on casks or drums shipped to or received from other sites. Many of the incoming casks or drums were filled with waste from the Rocky Flats Plant. On the average, the neutron dose rates were less than 10 mrem/hr. Neutron surveys were also conducted during reactor startup, around accelerators, during work with neutron-generating sources, and during work with fuel. Repackaging of plutonium nitrates normally produced very low neutron fields. Work with Cf-252 sources at TRA and ICPP required neutron dosimetry.

Neutron monitoring was assigned to individuals working at the reactor areas, chemical laboratories, RWMC (starting 2004), the Calciner Facilities, the Process Equipment Waste Tank Farms area, and Citrix. Historically, no allowable threshold was implemented for neutron exposure. NTA film was routinely used at the reactor areas and at the chemical laboratories. In the early days, neutron exposure was primarily observed at the reactors.

Some site experts believe the neutron monitoring program at INL has been inconsistent. For example, although the work has not changed, monitoring for neutron changed over the course of time. Workers noted that those outside the radiation boundary at RWMC do not participate in neutron monitoring, while those inside do. There are some disagreements between RadCon and other site experts as to whether neutron dosimetry was consistently used at ATR and ETR throughout the years. There was also some inconsistency between monitoring of permanent workers versus vendors, such as equipment handlers and excavators at RWMC. There was no routine neutron monitoring of some hands-on maintenance workers in ICPP.

About 15% of the current workforce (about 700 people) is monitored by neutron dosimeters. The potential for neutron exposure primarily exists among personnel who work with neutron sources, with neutron beams at the reactors, at the dry storage spent fuel area, at RWMC, and with the pulsed neutron generator. There is some neutron exposure from work with plutonium.

There was a potential for missed neutron dose in the early days, due to incomplete monitoring of the exposed population. For example, many laboratory analysts and chemists did not have neutron dosimeters, as they were not aware that there was an issue with neutrons. This lack of neutron monitoring could be verified by evaluating ambient neutron sources and cross comparing this information with dosimetry processing data.

Real-time External Monitoring

Pocket Ionization Chambers (PICs) and Electronic Dosimeters (EDs) were used in conjunction with film badges or TLDs for real-time monitoring. During high dose rate jobs, workers were given both a high-range and low-range PIC. The PICs had ranges of 200 mrem (TRA), 600 mrem (low-range ICPP), and 1,500 mrem (high-range ICPP). During routine operations, PICs were worn for up to 1 week. During high dose rate jobs, they would be used for the period of the job. Dose from PICs or EDs were used mainly for day-to-day tracking of worker dose to ensure regulatory compliance with dose limits. These units were also used in the estimation of dose when there were problems with film dosimeters (e.g., the film was completely black). If there was an unexpected exposure reflected by the PIC, the film badge would be pulled immediately for processing.

Timekeeping was also used as a means of keeping track of and controlling real-time external exposure. Dose rates and stay times were used for high-risk jobs (e.g., entry into hot cells). RCTs were responsible for establishing area dose rates and ensuring individuals were in compliance with stay-time rules. The RCTs would also perform ongoing surveys in job areas. Now, electronic dosimetry systems are used to alert workers to unexpectedly high doses. EDs are set by radiological control personnel to alarm at a certain dose. Once the dosimeter reaches this dose, an alarm sounds and the worker is expected to exit the area.

In the early days, pocket dosimeters were used and data collected by the Dosimetry group. These data were recorded together on the record sheets or cards. In 1958, film records were kept separately from pocket dosimetry data. The results from these are not available in the individual dosimetry files. RadCon field offices were responsible for assigning, reading, and maintaining PICs. The calibration check (they could not be adjusted, hence no calibration) of PICs was provided by Central Services. The pocket dosimeter or electronic dosimetry records from 1958 to recent times were kept as field records and would be difficult to retrieve for comparison to film badge or TLD records. Since 1998, they have been kept in the same computerized database.

Extremity Dosimetry and Multiple Badging

Multiple badges and extremity dosimetry have been in use since 1953. Multiple badges included dosimetry for extremities (e.g., finger rings), the upper trunk, the lower trunk, and any other location deemed necessary by RadCon personnel. "Routine" badges were worn in addition to the multiple dosimeters provided for a job. Workers wore their primary dosimeters customarily on their chest or at their belt level. In the case of multiple dosimeters, the highest dosimeter value was recorded as the dose of record. The results from all badges of a multiple pack were maintained in the individual's dosimetry file. The use of multiple badging was based on multiple high-level sources being present simultaneously in the work area, high-level point source with multiple workers, or other arrangements reflecting non-uniform fields.

Formal procedures are now in place for extremity and multiple badging, including when and how to use multi-badging. The multi-badging records were kept in the regular worker files. The highest dose recorded was assigned as the dose of record up until 1995. At that time, INL implemented weighting factors to calculate dose from multiple dosimetry systems.

There were situations, especially during maintenance activities, where there was partial-body exposure to workers. Many site experts were concerned about the potential in their jobs for extremity or nonuniform exposure. Multi-badging was used only in unusual conditions. For example, it was used during the SL-1 incident to monitor for high beta exposures. It was also sometimes used at the NWCF. Multi-badging was rarely used among maintenance and operations workers. While typically there was some shielding afforded to the whole body during maintenance jobs, workers were often required to reach inside an area or around a pipe or valve where dose rates were much higher. In some cases, work was performed in tight spaces in close proximity to high-radiation sources. In the past, management required workers to perform some high radiation jobs without multi-badging. The multi-badging is very important in assessing external dose, as some jobs involved different dose rates at different body levels in a high-radiation area.

Prior to the 1980 time period, extremity and multi-badging were rare. Laboratory personnel often wore extremity monitoring. The jobs which typically required extremity dosimetry included the following:

- Bottling of Krypton gas
- Maintenance at the NWCF
- Replacement of valve boxes in tank storage
- Entry into hot cells (starting in the 1990s)

Extremity dosimetry was implemented in the 1980s for crafts personnel. Not all operators were provided with extremity dosimetry in the earlier years.

Hot particles were sometimes an issue in the reactor areas. These particles can be created by core internals change out, preventative maintenance on the reactor systems, and remote movement of reactor components. Although the fuel was nearly always under water when moved, they were transferred in air to the canal. This process generated hot particles, which deposited in the work area.

Dosimetry Investigations

If there is a lost or damaged dosimeter, a Personnel Exposure Questionnaire (PEQ) was completed. They were also completed for badges showing unexpected doses. The PEQs were sent to RESL to review and perform investigations in cooperation with the field. The investigations included a review of field radiological control data for the time period in question. In addition to field data for a particular individual, co-worker data was used to estimate dose when dosimeters were found to be overexposed. Real-time dosimeters of other workers on the job were also used for the purposes of maintaining real-time external exposure when the PIC or ED malfunctioned. “Doses of record” were changed accordingly. The resulting dose assignment was documented, including the method of dose estimation (e.g., based on field data or co-worker data). When the reason for spurious results could not be determined, a conservative dose was recorded. RESL would also perform trending studies on individual dosimetry records, on groups of individuals, and on whole facilities.

Workers have expressed concern that the dose of record has underestimated the actual dose received. Workers consistently had access to their PIC results, and management periodically reported their film badge results to them. In later years, workers were issued an annual radiation exposure report. Numerous workers cited instances were their PIC went off-scale while working on a job. Workers did not specify which dosimeters went off scale. There were also a number of instances reported of excessive film exposure (i.e., black film). These situations were common during ICPP cell entries, reactor shutdown, and in the old calciner facility. Several site experts reported that dose received on PICs was approximately three times higher than the dose reported in their dosimetry file [presumably deep dose] in some cases. One site expert recalled that he received 500 mR on his PIC, yet his film badge result was 100 mrem for the same period of time. Another individual had a dose of record of 3–4 rem; however, he believes the dose to be twice that much. Site experts indicated that the TLD seem to be more representative of the dose than film badges.

Formal complaints have been made by workers regarding the lack of recorded dose on dosimeters. In these cases, RadCon investigated the situation by reviewing field records and interviewing workers. One possible explanation for discrepancies seen between PIC exposure results and dosimeter results is the substantial beta component in the field. It is not uncommon to see high energy β emitters causing PICs to read much higher levels. The same exposure detected by a dosimeter (i.e., film or TLD) would be broken down into penetrating and non-penetrating dose. This beta dose would be reflected in the non-penetrating rather than penetrating dose. As a result, the whole-body dose would be smaller than the dose from the PIC. In areas where gamma exposure was the primary hazard, there was more agreement between the PIC and the deep dose from the film badge. At some sites, this phenomenon would also manifest itself in survey meters. A correction factor was developed for closed- and open-window readings as a result to correct for this situation. These correction factors are dependent on the assumed β energy. β-to-γ ratios can vary from 1-to-1 to as high as 25-to-1. Electronic dosimeters more closely matched the dosimeter results.

There were no adjustments made to retrospective dosimetry records based on quality factor (QF) changes. Dose values were maintained consistent with the dosimetry standards at the time of measurement.

Internal Monitoring

Field or facility management personnel were always responsible for selecting individual bioassay programs. They were responsible for analyzing field results and identifying who to monitor, with the assistance of central radiological engineers with expertise in internal dosimetry, whether they were AEC/ERDA/DOE or the contractor. In the case of positive results, the internal dosimetry specialist would work with the field engineers to interpret the data and determine follow-up action. Internal Dosimetry was responsible for dose calculation and procedure development.

The site implemented urine, fecal, and whole-body counting as a part of the internal monitoring program. The monitoring requirements have changed over time. In the 1950s to the early-1960s, there was a routine bioassay program for radiological workers. In early 1964, INL found that it was not cost effective to perform extensive bioassay monitoring on all workers. In the 1970s, there was no scheduled or routine bioassay program for any workers; rather, sampling was triggered by an incident. Several facilities included extensive urine, fecal, and whole-body counting bioassay programs that began in the late 1970s and continued until the mid-1990s. In the late 1990s, this was changed to a job-specific, event-based bioassay program coupling with random sampling, which included fecal and urine sample analyses. Random sampling included workers performing high-risk jobs (e.g., 40 DAC/hour). For example, RCTs on high-exposure potential jobs would routinely be monitored. Records showed that any significant worker uptake would mainly relate to incidents. The event-based bioassay program considered risk factors like RWPs requirements, incidents, failed respiratory protection, exceeding the protection factor for the respiratory protection, and personnel contamination. Some baseline sampling was done for particular jobs.

Internal monitoring was dependent on the work locations and even the period of operation. Radionuclides evaluated with the internal monitoring program included fission and activation products, uranium, tritium (limited), and plutonium in later years. This type of sampling was less common in the reactor areas. Routine programs typically included annual WBCs, including on employees from ANL-W. If there was a positive WBC, fecal and urine analysis were submitted. WBCs were also used for intake follow-up and after a significant contamination incident. During the period of routine bioassay, workers submitted up to several urine and fecal samples per year, depending on the area of the site. When there was a potential exposure to iodine, the site also performed thyroid counts. Some site experts believe that with the implementation of the RWP, radiological conditions improved and a more routine bioassay program was implemented.

Site experts indicated they received the following internal monitoring, in practice, while at INL:

- In 1965, at TRA, there was routine urine analysis and WBC for workers.
- In the late 1970s or early 1980s, select personnel received semi-annual fecal and urine sampling with an annual WBC.
- In the mid-1980s, RWMC personnel received annual WBCs.
- Since the mid-1980s, ANL-W personnel have received annual WBCs.
- In the mid-1990s, personnel received an annual WBC and event-based in-vitro bioassay.
- Select personnel at the Special Manufacturing Capability (SMC) facility received depleted uranium urinalysis.
- Reactor personnel received only WBCs, and were not typically asked to submit urine or fecal samples.
- Support personnel, such as security guards and some crafts, indicated they only received a few WBCs while at INL, and were not placed on a routine monitoring program.

Individuals who worked in process areas received bioassay and WBCs. Exempt support personnel (e.g., engineers) associated with reactor operations did not receive routine internal monitoring (WBC, fecal, or urine sampling). Construction workers received the same internal and external monitoring as did site workers. There was no routine bioassay program for non-radiological workers; however, the site would periodically include these individuals in bioassay monitoring to ascertain background information and validate the assumption of no internal exposure potential.

The internal exposure program relied on worker protection and contamination prevention, such as engineered design features, administrative controls, wearing respirators, using ALARA principles, and wearing detectors.

In-Vitro Bioassay Sampling

A urinalysis program for some radionuclides was implemented with the start of operations. In the mid-1960s, the urinalysis program was discontinued, since experience indicated that it was ineffective in discovering internal uptakes. The routine whole-body counting program was continued, while the bioassay program went to an incident-based protocol. In the mid-1980s, an internal exposure bioassay program was set up for detecting plutonium and Sr-90. There was a potential for missed dose prior to the implementation of these bioassay programs, as these radionuclides were present in the work areas. The current bioassay program is random and confirmatory in nature. It is a selection-based program that is dependent on the characteristics of a worker's job.

In the mid-1980s, the Radiobioassay Laboratory fell behind in analyzing samples for approximately 1 year. Follow-up samples for positive results would have been substantially delayed, resulting in potential missed dose if an intake occurred. At times, the workers reported a delay in incident-based bioassay. Sometimes, contaminated workers were sent to the whole-body counter after their contamination or radionuclide uptake was already gone from their bodies.

There are about 1,600 to 2,000 in-vitro bioassay samples submitted per year. The numbers of positive results in more recent times have been low (10–30 positives per year). Because of recent remediation work at INL, the number of samples performed is expected to increase.

Special In-Vitro Bioassay Sampling

Field RadCon determined the need for special bioassay. "Triggers" (i.e., high air activity in the workplace, area contamination, and worker contamination) for special sampling were identified in operational procedures. The triggers varied over time. In 1979, if a worker in a contamination area had a nasal smear of 100 mrem/hr or more (a frequent occurrence), a bioassay would NOT be triggered. The potential missed (unrecorded) dose for a worker with this magnitude of uptake would be in the range of rems. No trigger levels were associated with field survey levels.

Specific trigger levels for bioassay seemed to be in conflict with actual practice in the field. Over 50% of the site experts interviewed had been involved in a skin contamination incident (including some facial contamination incidents) at some time during their employment at INL. These incidents were especially prevalent in ICPP, but were not limited to this area. When asked if there were follow-up bioassays after these occurrences (which included positive nasal smears), many workers stated that no special bioassays were performed.

In-Vivo Counting

Historically, INL had an extensive in-vivo counting program. In 1958, the site implemented annual WBCs for radiation workers when the first whole-body counter came online. INL used a mobile van, which traveled to work sites to perform counts. If the count was determined to be positive, the individual was sent to the fixed facility whole-body counter. As an example of the extent of the bioassay program, the site currently does 600–1,000 WBCs in a year.

Thyroid counting was common in the late 1950s. This data must be evaluated with caution, however, because there were some uncertainties in the method used to calibrate the thyroid counters.

Uranium

SMC and ICPP were associated with depleted uranium and enriched uranium, respectively. ICPP ran a process to convert enriched uranium (uranyl nitrate), recovered from spent nuclear fuel, to UO_3. Available information regarding operations at SMC was limited due to the classified nature of the work conducted there. The SMC area was primarily associated with a Type M depleted uranium form. For enriched uranium, there was an extensive bioassay program with fecal and urine sampling. The SMC facility did receive recycled uranium from other facilities.

Two uranium background studies were conducted to determine the natural uranium concentration in urine. In the late 1980s, prior to the operation of SMC, a uranium background study was performed on a limited number of SMC workers. The background level was determined in the range of 0.04 to 0.33 µg/L. From this study, an action level of 0.25 µg/L was adopted. This study used a small population, however, and there was also some question about whether the workers monitored had been previously exposed to uranium.

A second uranium background study was conducted using 16 individuals from the CFA. This represented a confidence level of 95% or 2σ (two standard deviations). These 16 people at CFA were used because they drank onsite water potentially contaminated with uranium. The background level was determined to be 0.045 ± 0.042 µg/L. From this study, an action level of 0.175 µg/L was adopted. Offsite people were asked to participate in the study to obtain a more reliable background level, but none were willing to sign a liability waiver. This baseline sampling study was documented in EDF 32-56 (November 11, 2002). New possible confidence levels would be 99% (3σ) for 27 people, or 99.99% (4σ) for 1,000 people.

Since plutonium operations were not a part of the INL site's early mission, the plutonium bioassay program was not initiated with the startup of the site, but added at a later date. Operations with the potential for plutonium and other transuranic exposures included ICPP and RWMC. For example, plutonium was dissolved in the process stream, which periodically lost its integrity and barrier. Monitoring was originally designed to look for β and γ-emitters. In the 1980s, the monitoring program began to evaluate bioassay samples for transuranics. The radiobioassay program has included analysis for both U-233 and Pu-238. Occasionally, there was a small uptake of Pu-238. The programs at RWMC (including the Advanced Mixed Transuranic Waste Treatment Program) presented a significant potential for uptake of transuranics. Transuranic waste stored in drums on the pads and in pits/trenches is currently being characterized, processed, and shipped offsite to the Waste Isolation Pilot Plant (WIPP) for permanent disposal. Recycled uranium was processed at some facilities at INL. Plutonium, neptunium and technetium were in parts per billion concentrations.

Some areas of the site contained high-fired uranium and plutonium oxide. The ROVER cell was used to process graphite space reactor fuel, resulting in the formation of high-fired uranium oxide. Some high-fired plutonium oxide may also have been shipped from Rocky Flats and disposed of at INL. At ICPP, carbonized fuel was processed such that high-fired oxides were formed. The fuel processed contained highly enriched uranium (~ 93% U-235). The recovery process resulted in a uranyl nitrate liquid product, which was converted to an oxide form for shipment.

There were several studies on high-fired plutonium and uranium bioassay and lung counting. Studies of bioassay elimination curves for this material indicated a lung retention of 300 days. Fecal sampling was determined to be the bioassay method of choice for high-fired oxides. In the case of plutonium, a urine sample might only contain small concentrations for the first few days after a known intake. INL staff developed the procedures for detecting high-fired plutonium oxide in fecal samples. Some of the pioneer work for detecting isotopes of plutonium was conducted by the Health and Safety Division (and successor organizations).

Tritium

There was very little potential for exposure to tritium, with the exception of some periods during the 1970s. Tritium monitoring, consequently, was performed on a limited basis. The estimate of total population dose from tritium at INL is about 7,100 mrem. There are no known radon and thorium issues at INL above standard background levels; there was no thorium handled at INL.

Internal Exposures

Positive bioassay results were not uncommon in the past among support workers. Most of the exposure received by general service support personnel was from the ICPP facility. There was also some exposure received in the TRA and at the Power Burst Facility (PBF). The old calciner and old solvent burner had the reputation for having the highest internal and external dose potential at ICPP. In the 1980s, skin contamination and internal exposure were major problems for workers.

There were a number of uptakes associated with operations at INL. Site experts collectively indicated that they had positive bioassay results for U-233, U-235, Pu-238, Co-60, and iodine. There was a general feeling that not all uptakes, particularly acute uptakes, were measured and documented in the individuals' dosimetry files.

Dose Assessment

Most internal dose calculations have been documented. Doses were determined based on WBC data, urine bioassay, and fecal bioassay. Detectable internal doses almost always came as the result of an incident. The best fit method for the radionuclides identified in the bioassay was applied. Parameters resulting in the conservative dose estimates were used. In some cases, the individual worker would provide several bioassays over a period of several months. The internal dosimetrist would then model the worker's intake of radioactive material. The internal dosimetrist would also review RWPs written by the field engineers and other pertinent monitoring data when evaluating an internal exposure.

No calculations of internal dose were completed with the use of air sampling data, except in unusual cases of dose reconstruction, where bioassay data was not available. There have been no comparison studies conducted between air sampling and bioassay-derived doses. The site did conduct a study (EDF 4510) on the use of Cs-137 as an indicator isotope for other radionuclides.

A study completed at the Calciner with a cascade impactor determined particle sizes ranged from 0.3 to 1 µm. The default particle size for internal dose calculations has been 1 micron AMAD. No recent particle size studies have been completed.

As new data became available, internal dose from acute intakes were periodically recalculated. Both Annual Effective Dose Equivalent (AEDE) and Committed Effective Dose Equivalent (CEDE) can be found in the dosimetry records. Over the years, INL has recalculated doses and converted them to CEDE where data is available. The worker's internal dose has been rolled up into their total lifetime doses, which include organ doses with weighting factors. The recalculation of internal doses often resulted in a reduction in dose, which was not adequately explained to the affected individuals.

The INL Internal Dosimetry technical basis document includes missed dose calculations based on detection limits for various radionuclides of concern. An estimated intake and internal dose are calculated and provided in the document.

Environmental Monitoring

There has been environmental monitoring onsite since DOE (and its predecessors) began operations. Between 1951 and 1993, environmental monitoring was the responsibility of DOE and analysis was performed at the Health and Safety Division, followed by the Health Services Laboratory, and the Radiological and Environmental Services Laboratory (RESL). The onsite monitoring was taken over by the site management and operations contractor in 1993. There were both in-perimeter (onsite) and outside perimeter (offsite) monitoring stations for airborne and direct radiation measurement. The State of Idaho also has its own monitoring stations onsite and offsite. Some of these locations are co-located with the site-managed locations. The State sampling locations have been in place for approximately 15 years. Site personnel believe fence-line TLD measurements represent the best records for longer-worker impact evaluation.

INL has collaborated with the National Oceanic and Atmospheric Administration (NOAA) from the beginning of the site operations. NOAA has provided the site with valuable meteorological data and analysis of atmosphere dispersion within the Snake River Plain.

The primary operations associated with airborne releases at INL included the ANP project (specifically Initial Engine Testing); the ARA (which was the site of the SL-1 Accident); reactor operations at the Reactor Testing Complex (Materials Test Reactor, Engineering Test Reactor and Advanced Test Reactor) and the Materials and Fuels Complex (Experimental Breeder Reactor II); the ICPP (including the Fluorinel and Fuel Storage facility and the RaLa program); the NRF; and several tests (such as the Fuel Element Burn Tests). The highest airborne releases occurred from the mid-1950s to the mid-1960s. Radionuclides released during operational and episodic releases included Cs-137, Sr-90/Y, I-129, I-131, I-132, I-133, Ar-41, Kr-85, tritium, transuranics, and other fission products. The RaLa processing runs at ICPP were the main source for I-131, but also resulted in the release of mixed fission products (such as Sr-90/Y and Cs-137). This was due to the very short decay time of the fuel prior to dissolution for reprocessing. The test reactor program resulted in releases of short-lived noble gases, such as Kr-88, Xe-138, Kr-87, Xe-135, and Ar-41. Measurable quantities of tritium have been released from TRA, NRF, and ICPP. Transuranic releases are associated with ICPP and RWMC. Cesium-137 and Sr-90/Y were released primarily as a result of fuel reprocessing. The radionuclide of most concern in relation to the NRF is Co-60; this isotope has been identified on the roof top. Roof-top contamination has not been limited to the NRF facility.

ETR, MTR, and ATR had de-gassing tanks. At MTR, 100% of the coolants went through the degassing tank by using a vacuum. At ETR and ATR, smaller de-gassing tanks (about 330 gallons) were used. They had to side-stream the coolants into the tank. Hydrogen and oxygen would be removed or degassed. Sometime, the gases would be blown out of the stack.

Particulate releases at ICPP in the early 1990s were tracked by basic surveys of the facility surroundings. Contaminated areas from fallout were flagged and put into the release maps. Particulate releases were observed as a result of a new steam cleaning process of the ICPP stack. The material released was believed to be Cs-137 attached to white insulation material. Measurable radioactivity was associated with these releases. An air protection system now filters through a High Efficiency Particulate Air (HEPA) filter system, which has prevented reoccurrence of particulate emissions like those that occurred from the stack steam cleaning process.

There are sources of surface water that flow onto the site, but no surface water flows off the site. Surface water percolates into the soil at playas located on the INL site, contributing to the recharge of the Snake River Plain aquifer. When reactors shut down, operators would purge the reactor system and fill the tank with clean water. The waste water was sent to a retention basin to allow for decay, and then discharged to an unlined pond, which allowed fluids to soak into the ground.

The Snake River Plain was formed as a result of lava flows. There are fractures in the basalt leading down to the aquifer. Some transport of strontium, tritium, and iodine to the aquifer has been observed. There are also some detectable concentrations of tritium in the drinking water wells in the CFA. Although there is some groundwater contamination, the site has always met the drinking water standards for radioactive materials. The U.S. Geological Survey has conducted extensive studies of the aquifer and contaminant transport.

The SL-1 incident caused the largest soil contamination of Cs-137 and Sr-90 at the ARA, with from 2–100 pCi/g Cs-137 after cleanup. The background level was 0.5 to 1.5 pCi/g Cs-137. Aerial surveys identify areas onsite with elevated soil concentrations. During decontamination and decommissioning of the old calciner in 1994 or 1995, the soil in the area was found to be highly contaminated. As the workers dug into the ground, each layer of soil was hotter than the previous layer. Finally, they entombed the facility with concrete.

Additional information on soil contamination (and the contemporary environmental monitoring program) is available in the *Final Environmental Impact Statement for Waste Operations* (ERDA-1536, September 1977). Since the initiation of Environmental Restoration, soil characterization data has become more accessible through the generation of the Environmental Data Warehouse Database.

There have been several efforts completed to characterize historical releases at INL and resulting doses to offsite individuals. From 1989 to 1991, a study referred to as the *Idaho National Engineering Laboratory Historical Dose Evaluation* (HDE) was performed. S. Cohen and Associates (SC&A) and Risk Assessment Corporation (RAC) conducted an extensive records review, and generated a document summarizing the historical environmental releases and monitoring program at INL. This report contained information on historical airborne releases from INL facilities, provided some background on the monitoring program, and gave information on the environmental regulations applicable throughout the period. Information provided includes stack monitoring data, fence-line TLD data, release fraction data, and uptakes by vegetation, cow, and fish.

The report identified that there were originally 26 meteorology monitoring towers ("met-towers") at the INL site operated by the National Oceanic and Atmospheric Administration (NOAA). Now there are 33 met-towers in use. Before 1969, archival hourly data from one or more site meteorological stations were generally available back to 1954. Doses to offsite personnel, including a maximum hypothetical individual, from airborne emissions were calculated using a multiple puff model to estimate the potential ground concentration from release points. The assessments in this report are based on both onsite and offsite monitoring data. Where available, site data was verified with State environmental data collected from co-located sites. The report includes an analysis of various fuel element burn tests. It does not contain soil and water release data because these are insignificant pathways for offsite releases. Although valuable data is included in this report, it does not estimate potential onsite exposure to individuals. The final HDE report was issued in 1991.

The Centers for Disease Control and Prevention (CDC) has been involved in a chemical and radiological dose reconstruction effort at INL since 1991. The Risk Assessment Corporation, on behalf of CDC, released *Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory* in October 2002.

Valuable resources for environmental data, containing soil and water release information, include the INEL Site Environmental Reports (DOE/ID-12082 series), quarterly and annual reports from the Health and Safety Division back to 1958, NESHAPS reports (DOE/NE-ID-10890 (05) series), the 1995 Title V Operation Permit Application, INL Site Comprehensive Environmental Response, Compensation and Liability Act Closure documents, the *Environmental Monitoring Program Report* (INEEL/EXT-01-00447, September 2001), the Environmental Data Warehouse Database, and the *Radioactive Waste Management Information*

System (DOE/ID-10054, 1993). Annual and/or quarterly reports were generated by the Health and Safety Division starting in 1951. These reports include environmental monitoring data and are available back to 1958 through the environmental monitoring group. The database includes historical and new environmental data to support environmental restoration work. EG&G Nevada also conducted aerial surveys of the INL site.

The State of Idaho has the Clean Air Act (CAA) primacy, except for National Emission Standards for Hazardous Air Pollutants (NESHAPs), Subpart H, *National Emission Standards for Emissions of Radionuclides Other than Radon from Department of Energy Facilities*. The Idaho Department of Environmental Quality conducts inspections of all regulated sources of air emissions. The Environmental Protection Agency requires air permits for radionuclide emissions, which stipulate High Efficiency Particulate Air (HEPA) filter testing, regular sampling of the warm waste pond, stack monitoring at the source, and continual ambient monitoring. The requirements in NESHAPs provide information on stack monitoring for specific radionuclides and their activities. The resulting site-wide dose from airborne emissions at INL is normally in the range of a few thousandths of a millirem (compared to the 10-mrem limit established in the regulations). Frenchman's Cabin is used as the location where the hypothetical maximally exposed individual resides for the offsite dose calculations. The last operating permit application submitted by INL to the State was the 1995 Title V Operation Permit Application, which was updated in 2001. A number of Permit to Construct applications and modifications have been submitted for specific sources in the past 10 years.

Medical Examinations

The initial medical program at INL consisted of a physician, nurse, and aid-man, who were there to offer medical assistance to employees, if necessary. When operations at INL moved from construction to operation, the medical services were consolidated under the AEC. The original AEC guidance for Occupational Medical Programs was set forth in AEC Appendix 0528, *Occupational Medical Program*. Management surveys of the Medical Branch were conducted by the Medical Branch, Health and Safety Division as early as 1966. After the Energy Research and Development Administration (ERDA) took over from the AEC, the ERDA Manual Appendix 0528-A, *Contractor Occupational Medical Program Handbook*, was issued in 1975. In 1978, DOE Order 5480.8A, *Contractor Occupational Medical Program,* replaced the ERDA manual. The current requirements are found in DOE Order 440.1A, *Worker Protection Management for DOE and Contractor Employees*. In addition to this requirement, there were Occupational Safety and Health Administration standards (29 CFR 1910) requiring mandatory examinations (i.e., for metal workers, fissile material handlers, hoisters, and respirator users). The medical program was turned over to site contractors by the DOE in 1978.

Medical examination requirements for workers have been based primarily on the physical work locations, materials an individual worked with (e.g., asbestos), and worker age. In the 1960s time frame, the frequency of exams was based on age and work location. Examinations were provided for individuals working with radiation, significant concentrations of toxic substances, or if there was a fitness-for-duty question. Medical Branch services were provided to all INL personnel. In the 1970s time frame, medical examinations were given to the whole work force. These exams continued for radiation workers up through the 1980s. Eventually, there were changes made in the program due to the legal climate resulting in performing physical exams only when there was a reason for one.

The following three excerpts from early AEC/ERDA Medical Services/Branch reports discuss exam schedules, and were described by the interviewees and provided to the SC&A team during the meeting:

> *During 1960, … a system to schedule laboratory work and physical examinations for all AEC and IDO contractor personnel was devised and placed into operation. All active medical records were reviewed to schedule employees for examination. Previously, the examinations had been scheduled either by one of the nurses in the plant areas or by request from the individual. This resulted in poor schedule control by the Medical Services Branch, and it was not possible to concentrate the examinations where the greatest benefit could be anticipated. …. The interval between complete physical examinations is dependent upon the employee's work assignment, age, and health status. Unless the examining physician specifies a shorter interval, the following schedule has been employed:*

Age	*Radiation Area Employees (Badged)*	*Non-Radiation Area Employees (Not Badged)*
18-24	*4 years*	*at age 30*
25-39	*3 years*	*5 years*
40-49	*2 years*	*3 years*
50-59	*1 year*	*2 years*
60 and over	*1 year*	*1 year*

> (Source: 1960 Medical Services report by George L. Voelz, M.D., Branch Chief)

> *The general schedule for physical examinations was revised during 1970 so that most employees would be examined at the same age. Prior to this, different schedules were applied for radiation and non-radiation areas. Experience has shown that with the level of radiation exposure at the NRTS, no increased incidence of medical problems is encountered in "radiation" workers. With the new schedule an employee may anticipate examination at time of hire, at age 25, 30, 34, 37, 40, every two years until age 62 and then annually. …."Lateral chest x-rays have been added to the routine PA every three years." … "A complete periodic physical examination at this time, in addition to the 'routine' physical and history, would include: chest x-ray, laboratory work – complete blood count, urinalysis and blood (chemistry) profile, pulmonary function testing, stool occult blood, PAP smear (optional), EKG and testing of visual and auditory acuity. Counseling and referral to private physicians follow in the event that abnormalities are identified.* (Source: 1970 Medical Branch report by J.H. Spickard, M.D.).

> *Employee Health Examinations (1976)*
>
> *Physical examinations are performed to determine the health status and physical fitness of employees in an effort to aid in job placement and work safety and as a guide to strengthening the preventive medicine aspects of the occupational medical program where the health and welfare of the individual are concerned … These physical examinations are of three types:*
>
> a. *Pre-employment: at time of hire*
> b. *Periodic: every year at age 45 and over and every two years if under 45 years of age (at the conclusion of calendar year 1976, 63% of all full-service INEL employees were under the age of 45 and 37% were age 45 or over.*
> c. *Termination: Upon termination of employment.*
>
> *The content of each type of physical examination is as follows:*
>
> a. *Pre-employment: Serology, Complete Blood Count, Urinalysis, Vital Capacity, X-Ray (PA & Lateral Chest), Audiogram, Orthorater (visual acuity, depth and color perception), height, weight, blood pressure, pulse, Electrocardiogram, physical examination.*
> b. *Long Periodic Physical*: In addition to items enumerated above for pre-employment physical, the Long periodic physical contains: SMA 12/60 blood chemistry profile, other lab tests deemed necessary by physician (e.g., GTT, Uric Acid, etc.)*
> **Alternate Long and Short Periodic done on each scheduled physical*
> c. *Short Periodic Physical: Hematocrit, Hemoglobin and White Blood Count, Urinalysis, Audiogram, Orthorater, height, weight, blood pressure, pulse, physical examination and other lab tests deemed necessary by physician.*
> d. *Termination Physical: Hematocrit, Hemoglobin and White Blood Count, Urinalysis, Audiogram, Orthorater, Interview – nurse or physician.*
>
> (Source: 1976 Annual Report – ERDA-ID Medical Division by John H. Spickard, M.D., Director)

In the 1992/1993 time frame, the medical program ceased performing physical examinations on employees who did not have a mandatory driver for a physical exam, and started offering health profiles, including a laboratory blood chemistry profile, to those employees, in lieu of a physical exam. Employees whose work requirements involve medical certifications or health surveillances, as identified by their management, continued to receive physical examinations per those mandatory drivers, such as DOT Drivers, Firefighters, Fissile Material Handlers, Hoisting and Rigging Workers, Human Reliability Program (HRP) personnel, QA/NonDestructive Examination Inspectors, Reactor Operators, Respirator Users, Security Guards, Security Police Officers (SPOs), Asbestos workers, Beryllium Workers, Cadmium Workers, Hazardous Waste Operators, Hearing Conservation Program, Laser Operators, Lead Workers, etc.

Chest X-rays

The technique used for chest x-rays prior to 1970 was a posterior-anterior view. Lateral and posterior-anterior view x-rays were given every 3 years from 1970 to approximately 1993. In 2000, pre-placement x-rays were discontinued. INL did not use photofluorography, and no 4 in x 5 in films have been identified by present medical staff. There have been no x-ray machines identified with this capacity.

In the 1965/1966 time frame, there was a medical van installed with equipment owned by the INL site. In the late 1970s to 1990, surplus x-ray equipment, including that from the van, was either moved to other INL facilities or declared as surplus for use by other facilities. Current staff is not aware of the make and model of x-ray equipment used from 1949 through the late 1970s. Current x-ray units were installed in 1990.

In 1980, the Idaho Food and Drug Administration was responsible for inspecting equipment every 2 years. Their reviews included checks on the collimator, beam quality, radiation output, field alignment, source-to-image distance indication, and the entrance skin exposure. The administrative program was also reviewed. Preventative maintenance was responsible for calibrating, inspecting, and auditing equipment. This preventative maintenance was completed twice per year, and included mR/mAs and kVp/mAs tests. An offsite x-ray equipment maintenance contractor provides preventive maintenance support on the INL x-ray units. In 1973, there was a test conducted by health physics personnel using a phantom, with TLDs placed on the surface and at a depth of 1.5 cm. Doses were determined for various techniques. The above documents are the primary sources available for occupational medical radiation exposure from x-rays.

Incidents and Accidents

In the earlier years, when an incident occurred, an Individual Event Report (IER) was generated. Later, the site recorded anomalies, personnel contamination, skin exposure, and over-exposure incidents on a Personnel Exposure Questionnaire (PEQ) form. For example, a lost dosimeter by a worker would trigger a PEQ process. PEQs were stored together and maintained with field RadCon records. Only the exposure assessment was maintained in the dosimetry record. This included primarily dosimetry information, such as organ doses, radioisotopes of concern, total doses for the events, and bioassay results. If there was a dose adjustment as a result of a PEQ evaluation, this was noted in the electronic dosimetry database. Anyone onsite could issue a PEQ, but they were usually generated from the Radiological Control Organization. A significant unplanned event would also be recorded in the Health Physics logbooks and on supporting surveys, air sampling, laboratory reports, etc. These records are stored with the field records.

There were a number of recorded and unrecorded incidents or unexpected events that occurred at INL over the history of the facility. The recording practices for these incidents appeared inconsistent. Prior to the arrival of Westinghouse at INL, there was a high threshold for defining incidents; Westinghouse subsequently tightened controls. Site experts were not sure what the trigger levels were for incident reporting. A significant unplanned event would result in a formal incident report, critiques, and lessons-learned reports. Interviewees indicated that, on occasion, personnel contamination incidents would seem to go unreported. Even when an incident or unusual event occurred, management did not effectively communicate the resulting internal and/or external doses to the workers. Eventually, individuals involved in an incident would receive a copy of the incident report 2 to 3 weeks after the events. In cases where no incident report was generated, this information was documented in the worker's annual individual exposure report once a year.

There were four major incidents at the INL site, including three criticality accidents and the SL-1 reactor accident. The first criticality accident occurred in 1959 at the ICPP. While siphoning of uranyl nitrate solution in a waste receiving tank, there were multiple excursions due to favorable (for criticality) geometry. During the evacuation of the building, airborne fission products resulted in individual exposures. The second criticality accident, which took place in 1961, also occurred at ICPP; it resulted from improper configuration of uranyl nitrate solution in a vapor disengagement vessel. There was only minimal dose to personnel as the result of airborne fission products. The last criticality accident occurred in G-Cell at ICPP in 1978 as a result of excess material in the lower disengagement section of the scrubber column. There were notable releases out the stack. The SL-1 reactor accident occurred on January 3, 1961. As a result of a manual withdrawal of a control rod, the reactor went supercritical. A discussion of the SL-1 incident is presented in a separate section.

Following the 1978 G-Cell criticality accident, the plant was shut down for 8 days, and the dissolver had to be rebuilt. This dissolver was basically a garbage can that collected concentrated waste. The cleanup work required remote access sawing. Pipefitters had to reach deep into the contaminated lead line. When the piping was cut and dropped into a barrel, dose rates were so high that the RCT called for immediate exit from the area. The RCT surveyed areas around the dissolver bottom, and the instrument read off scale in excess of 50 R/hr. The RCT personnel got everyone out of the area and put temporary shielding materials around the garbage can.

The ICPP facility was responsible for a lot of overexposure and personnel contamination incidents. This facility was originally designed as a pilot plant, but was put into routine operation. Work associated with the waste stream involved high beta dose fields. The $\beta:\gamma$ ratio was in the range of 4:1, or even higher, in the calciner cell or the off-gas cell. Some of these incidents led to uptake of radionuclides. Several examples of non-routine exposures or unusual events at ICPP, mentioned by site experts, are provided below.

- A worker was working in a multicurie cell and was grossly contaminated. The HP personnel were able to decon the worker's shoulders down to 200,000 dpm/100 cm^2. The area read 10 mR/hour. No bioassay sample was requested.

- One worker actually exited the plant with 365,000 dpm contamination on his knees, setting off the alarms at the gate. Eventually HP personnel went to the worker's house to ensure no contamination was taken home.

- A mechanic was instructed not to touch a manipulator, but he inadvertently did. The RCT had to scrub him down to10 mR/hr. There was no bioassay done on this mechanic.

- There was a gas bubble burst in the processing system and contamination was spread throughout the ICPP facility. There was an extensive cleanup effort.

- A pipefitter was leaning on a test train trying to remove it. The radiation meter went off scale. The test train was immediately replaced. As a result of this incident, the individual received his allotted dose.

- In the 603 Storage Building at ICPP, operators used buckets to transport contaminated waste water.

- In the tank farm, workers had to vacuum out and clean the facility due to a cyclone blow out (633). The workers had to work double shifts to get the cleanup done.

- In 1982, workers were dissolving Al-Zr fuel rods at ICPP. These fuel rods had to go through two separate solvents. They had problems with the mixture of O_2 and H_2. These fuel rods were about 40 ft long and were designed critically safe. This mixture problem caused an explosion so powerful as to lift the rods.

- There was a situation where a welder and machinist were burned out (i.e., exceeded dose limits) during work in the old calciner facility. This dose is not included in the lifetime dose, according to the affected individuals.

- Yellow uranium powder was found on the floor in E-cell.

- There were several incidents of positive nasal smears and facial contamination without follow-up bioassay.

There was a significant release of I-131, I-132, and I-133 at the ICPP from the RaLa Project. Seven individuals inhaled the airborne iodine, with a maximum intake of approximately 40 µCi. These individuals were monitored with thyroid counts and urine sampling for several months. Detailed incident reports were generated, which were initially placed into individual dosimetry files.

Routine maintenance of systems involved changing filters, which were later disposed of at RWMC. Dose rates around the filter and/or filter housing could be as high as 25 R/hour at 4 ft. As the Old Calciner Facility started to age, the cyclone started to leak and areas began to plug. This led to more personnel contamination incidents. Eventually, the Old Calciner Facility was shut down and the New Calciner Facility was put into operation.

Leaks in system components or during transfers appeared to be a repetitive issue. One worker was sent to clean up waste water in the basement, because someone left the scum line open. There were significant α and β emitters associated with the task of reducing the water level accumulated in the basement. An intake of radioactive material resulted from this incident.

Trucks from TRA were loaded with waste materials and sent to the evaporator for processing. Operations personnel were responsible for unloading the material out of a tank via a pressure system. There was a lot of waste water spilled out on the ground in the unloading process, and, as a result, the ground became very hot and contaminated. Garden hoses were used to hose down the ground with service water. This spread the contamination around and the ground had to be cleaned. The waste water was absorbed into the surrounding soil. Sometimes, the tank would blow out gallons of waste materials. As this was supposed to be a contained system, no respiratory protection was worn during this operation.

In the ICPP Calciner Facility, workers had to use friction saws to cut off valves that were contaminated with high levels of Cs-137, Sr-90/Y, and U-235. When they were drilling into the valves, sometimes filters were burned through, spilling contaminants, which would get all over their hands or faces. This often resulted in skin contamination. The airborne radioactivity level was also very high and may have been responsible for significant uptakes of radionuclides.

In 1979, there was a worker uptake incident at the shift laboratory in the ICPP. This laboratory was responsible for analyzing samples from the uranium separation process and waste production at B627. The wastes produced were very high level.

There was a time when the door flew open in the West Vent tunnel and caused a significant radiation release. In the tank farm, there were releases of materials to the soil with 20–30 mrem/hr.

In the early 1980s, at the solvent burner, waste water dripped on to the ground from the pumps. There were releases of airborne contaminants out of the stack. These types of incidents had been occurring since the 1970s. During one operation, the pumps were primed too much and the process line had not caught up, causing too much solvent to build up inside the line. When the burner fired up, it exploded and excess solvents went up 100 ft in the process line and out of the stack.

In late summer of 1981, one of the fans in the evaporator broke and plugged up the drain. Waste water had flooded the building. All operators were required to help decontaminate the area. For each of two entries per day, they would receive 300 to 400 mrem. The RCTs surveyed the area and found an average reading of 20 R/hr. When they got close to the drain, readings increased to 50 R/hr. Standard PPE and a dosimeter were worn by personnel. The doses they received for this incident were recorded in their exposure records. Cesium-137 and Sr-90/Y intakes resulted for some workers.

In about 1985, there was an excursion (incident) in the gas plant that blew up the bottom of a vessel storing krypton gas. All the CAMs went off in the plant. The operators ran into the facility to shut off all the valves. Later, by just looking at the instrumentation, they discovered that there was no krypton gas left in the system. Due to the absence of RadCon support, the operators were sent home. There was no bioassay sampling or contamination check. There were no doses reported or recorded for that incident. Critiques of the incident were held. The incident was not reported or recorded in the affected individuals' annual medical and exposure files.

High radiation exposures were not uncommon during reactor shutdown, where an individual could burn out in less than 15 minutes. There were also hot particles found in the reactor areas. Although it was not the primary hazard, contamination was present as well. The reactor top had a particular issue with contamination. Several examples of non-routine exposures or unusual events at TRA, mentioned by site experts, are provided below.

- A worker was working on the reactor top at ETR when the Continuous Air Monitor (CAM) sounded. The resulting contamination required that the walls be scrubbed up to 60 ft and the crane be decontaminated.

- A mechanic was working on a plug in the test train tube at MTR storage area. When he pulled the plug, his pocket dosimeter peaked and his chirper alarmed.

- A mechanic was reassigned to non-radiation work when his badge was observed to be black.

- The nozzle trench surrounding the core became contaminated when individuals worked in this trench.

- Workers at ATR were changing out the resin column and loaded the unit into a lead cask. They found the dose rates were close to 5 R/hour, forcing them to transfer the resin into another cask for disposal. They used a galvanized garbage can (55-gallon drum), put lead paper on the bottom, transferred the resin into the can, and then removed the can for disposal.

- A welder was assigned to re-weld a valve at the ATR. The stay time was limited to 15 minutes. Upon exit, he noted his film badge was entirely blackened. The foreman sent him back in for an additional 15 minutes.

- A welder was making a cut in the in-pile tube to remove it and became contaminated. Area alarms were set off.

- In the 1960s, an experiment was transferred from ETR to a transport cask. The experiment dropped out of the bottom of the cask, setting off radiation area monitors.

- In 1964, a worker assigned to the TRA Hot Cell became contaminated and was later found to have detectable radioactive material in his left lung.

- In June 1984, there was a large spill at ATR. Workers were sent in to decontaminate the canal. It was so hot in the canal that a worker could not stay there for more than 30 seconds. After the cleanup, the workers were found to have exceeded their dose limit for that month. They were reassigned to CFA initially, but then got assigned to work on another spill in another facility.

- There was a continuous release of argon from MTR while it was operating, because the graphite reflector was air cooled.

- MTR/ETR had Ag-110m, Hf-181, Ta-182, Cr-51, and Hg-203 releases, which are typical of activation products at reactor experimental loop experiments. Rupture of experimental and sample lines resulted in spread of these products.

- There was a neutron chopper with a large neutron beam. A fire started in the paraffin at the MTR.

In the 1964–1965 time frame, during a reactor shutdown at ATR, two welders went to weld the gas annular inside the reactor vessel. Some wooden boards were put on top of the pipes, so that the welders could slide in to do the welding. They were wearing 3 sets of anti-Cs, shoe covers, cotton coveralls, gloves, and half-face respirators, because it was extremely hot (radioactive) inside the vessel. Later on, while they were welding, they found they were all wet inside and out. It turned out that the reactor operators had been ramping up the reactor, without checking or realizing that the welders were still working in the reactor vessel. The water started to rise up in the vessel to be ready for the reactor fuel. The individuals were grossly contaminated and required extensive decontamination. This incident was not reflected in the involved personnel's dosimetry files.

Incidents also occurred at other facilities. In the late 1980s at the FAST facility, there was a contamination-spread issue involving antimony in dissolved fuels that spread all over the facility. Some fires associated with machining of depleted uranium occurred at the SMC facility. The latest event associated with an uptake occurred last summer at LOFTS. Some workers showed positive bioassay results for Sr-90/Y. The WBC results turned up negative. At the Custom Processing Facility, a glass vessel with plutonium and nitric acid exploded causing significant alpha contamination of a cell. One worker received a significant exposure as a result of the accident. There was a Type B investigation related to this incident.

An individual received chelation therapy in the 1960s for an uptake of transuranics at the TRA facility. Union officials have documentation to support this. Another chelation, recorded in the medical files, occurred in 1993 after a person at ANL-W working with Am-241 in a glovebox was struck with the material after the flask broke. He was administered one very low chelation dose, based on TRU criteria, intended to calm him down. The INL dosimetry organization performs internal dose calculations to assist in the determination of when chelation is performed. Because ANL-W was under the auspices of DOE-Chicago, its processes were not known to the DOE-Idaho Occupational Medical Program at the time and, therefore, communication issues were a concern. These issues have now been rectified due to the recent contract consolidation.

The PEQs are not provided to NIOSH as a part of the claimant packet. It is uncertain whether IERs were permanently preserved, or whether they are being provided to NIOSH.

SL-1 Accident

In January 1961, there was an excursion at the SL-1 reactor, which resulted in the death of three operators. The only evidence initial responders had that an excursion had taken place was very high dose rates. Initial responders included firemen, security guards, and radiation protection personnel. Employees were exposed from the initial response, cleanup, and examination of accident personnel and reactor debris.

At 8:00 AM on January 3, 1961, a fire alarm sounded in the SL-1 complex and the CFA fire station. A firefighting engine company of five firefighters and the Assistant Fire Chief responded to the SL-1 reactor building and looked around. The reactor was shutdown at the time. Eventually they made their way to the furnace room, where they found the activated alarm. There was no smoke or fire, so it was considered a false alarm. The fire alarm was reset and the fireman returned to the station. At 3:00 PM that same day, there was another fire alarm at the SL-1 complex. Firefighters went to investigate again and found the same fire alarm sounding next to the furnace. Again, there was no smoke or fire to be found and they reset the alarm. At this point, they requested that the alarm maintenance crew come to fix the alarm. They would not be available until the next day. The firemen reset the alarm and returned to the station once again.

At 9:18 PM that night, a third fire alarm (generic alarm # 2-2-1 designated as the SL-1 complex) sounded at SL-1. At the time, it was -18°F outside and the firemen were not enthused about responding to yet another false alarm. They responded to the SL-1 site, where there was no guard present in the guard house at the time, and noticed there was an alarming personnel radiation monitor in the guard house. A phone call was made to the control room with no answer. It was unknown how many personnel were present on back shifts. The firemen then used their pass key to enter the complex. They first checked the alarm, which had given the previous false alarms. They found that this alarm was not sounding.

The fireman entered the front door of the SL-1 reactor building to check the fire alarms in that area, as they thought the alarm may be coming from the reactor area. They went into the hallway and headed directly to the reactor control room. At that point, they noticed the dose rate in the control room area was 25 R/hour on the hand-held detection instrument they were using. Immediately, they saw the high-radiation lights were all flashing on the reactor control console. The three reactor operators were not in the control room and could not be found in the area. The firemen ignored all these signs. They were outfitted in Scott-Air Packs, standard fireman response gear, and had a radiation dose rate meter. They started up the stairway leading to the reactor floor. There were radiation alarms sounding and the meter they had went off scale, indicating a dose rate of greater then 250 R/hr when they were half way up the staircase. They thought the meter was broken, so they went to get another meter, which indicated the same results. They took one quick look inside the reactor floor, noting one person down a short distance from the entry door. At this point they knew something was terribly wrong. They retreated at this point and called the AEC dispatcher.

A health physicist was requested, and the one available was at TRA, which was 18 miles away. It took him about 30 minutes to respond, as he had to find a vehicle and get high-reading instruments. When they went back to the reactor floor, they found debris and the first victim not far from the entrance. The dose rate at the entrance to the reactor was estimated at 6,000 R/hour. The first two victims were lying on the floor. The third victim was not immediately located, as he was pinned to the roof by a control rod. They knew they were in trouble, so they all got out of the building quickly. The Assistant Fire Chief immediately called the dispatching center.

A team of personnel that were in charge of operating the complex was sent out from Idaho Falls as a rescue operation team. The AEC recognized this as an emergency incident and authorized 100 R for lifesaving and 25 R for saving equipment. The operators and co-workers volunteered to assist in the recovery of the bodies. The dose rates were so high that entry time was limited to a matter of seconds. The first rescue team was merely able to enter and lay a blanket on the floor, so that the second team could place the victims on a blanket and exit. The third team removed the victims to the outside, where radiation levels were much lower. The second individual was removed by plant workers from on top of the reactor several hours later.

The site doctor responded from Idaho Falls in a government vehicle. The nurse responded to the scene in an ambulance from CFA. The first individual removed from the complex was placed in the ambulance with the nurse and was driven to Highway 20, where they were met by the site doctor. The victim was pronounced dead about 30 minutes after being retrieved. While transporting the first victim to meet the doctor, the nurse could not detect any vital signs, but she thought she saw movement in the victim.

After the first two victims were out, they used a remote camera to look for the third victim, who was found pinned to the ceiling. A cherry picker equipped with a torch and a stretcher was later used to remove his body. The beams on the roof of the reactor were cut. The third victim was dropped into the stretcher and removed via a freight door to the outside of the building. This was done some time later.

The victims were sent to ICPP and iced down in large sinks for preservation for further examination. This inadvertently led to some decontamination of the bodies. The Army doctors dismembered the bodies for analysis. This was the first time a human body had received such a high dose of radiation in peacetime. The bodies were eventually placed in caskets lined with lead, and their caskets were entombed in concrete for shielding purposes. No radiation could be detected outside the concrete.

In addition to resulting in high external dose rates, there was a considerable amount of personnel and property contamination that resulted from the accident. The six firemen and others who initially responded to the accident were sent to a laboratory building 1.5 miles away. When they got there, the personnel monitor at the entrance was activated by their mere presence. They were so contaminated that they were asked to remove their clothes outside in the -18°F weather prior to entering the facility for decontamination. Their film badges were taken at the entrance for immediate processing.

Following the SL-1 incident response, those involved were asked to submit daily samples (fecal and urine) for a 30-day period. After this time period, they submitted a urine sample twice a year and a fecal sample once a year. There were no explanations or debriefing by the management after the sampling or the incident. There were approximately 32 individuals involved in the initial response team. Among the initial response team were six firemen.

The ambulance was highly contaminated after transporting the victims. It was decontaminated and returned to the firemen for use at EBR-II. Eventually it was declared surplus and transferred to the local community of Blackfoot.

The SL-1 reactor was allowed to sit for a few months after the recovery of the last body. At this point, a massive effort to remediate the site was undertaken by General Electric. The dose rates were still very high, though they had dropped since the initial recovery. All types of individuals, such as engineers, maintenance personnel, photographers, etc., were brought in to support the cleanup effort. There were over 1,000 individuals involved over the course of the cleanup operations. Operations were carefully planned, and remote handling was used wherever possible. As a result of the high beta doses, a belt with multiple dosimeters was used during the SL-1 cleanup of the site.

A team of GE engineers was sent to cut the roof off the building and remove the reactor core for analysis. The core was pulled and put into a transport cask. It was then taken to the TAN Hot Shop to be washed down, decontaminated, and examined. Criticality experiments were performed on the fuel to see if it would go critical if water was added. They bore a hole in the bottom of the vessel, causing particulate matter to be released. Control rods were placed through the top of the vessel. The reactor vessel bulged out as a result of the accident; it was hung in the Hot Shop for examination. The bulging of the reactor vessel was evaluated through remote measurements. Personnel removed the head of the vessel, took all the fuel out, and shipped the fuel to ICPP for processing.

Medical personnel were aware that some workers had exceeded their dose limits prescribed for the time. One fireman responding that night received 18 R; however, this is not reflected in his radiation exposure reports. Other responders reported the same discrepancy in their dose records. No one they know of came to the Medical Office and complained about sickness due to these overdoses. Those firemen responding to the incident have not suffered a lot of adverse side effects. The nurse died of cancer a number of years later. As of 2005, of the six responding firemen, it is believed that three are still surviving today.

High-Risk Jobs

There were numerous higher-risk jobs associated with both routine and special operations at INL. These jobs were associated with high contamination, airborne radioactivity, and high radiation areas. It was not uncommon to encounter elevated radiation fields during maintenance and operations. Many of the operations onsite, especially those at the ICPP, required individuals to enter areas with very high dose rates. For example, at the New Calciner Facility, individuals worked in areas with 50 mR/hr to 50 R/hr dose rates. As a result of high exposure potential, stay time in these areas was limited. In some cases, the individual could receive the authorized limit for the job (up to 900 mrem with management approval) within a few minutes. Simpler jobs, such as tool placement, etc., would have to be performed by management, engineers, security, and other administrative-type personnel. The skilled craftsmen were saved to complete tasks that required special skill. Individuals were pulled from other areas of the plant, if necessary.

As a result of the high levels of contamination in the cells and the high dose rates associated with work in these cells, some of the highest doses received onsite were associated with the ICPP facility. The maintenance and repair of this facility required a variety of jobs, such as removal and replacement of the piping and filtration systems, welding, radiography, electrical work, changing bulbs, etc.

Operations personnel were responsible for all kinds of tasks across the site, including work with waste streams. They would prepare areas for maintenance prior to the beginning of a job by performing decontamination, positioning lead shields, and/or cleaning up waste water as necessary. Operations were also responsible for collection of samples for analysis. At times, these samples were the result of process upsets. Other high-risk jobs mentioned by site experts included the following:

- Filter change outs at facilities, particularly ICPP
- Repair jobs at the calciner facilities
- Process cell entries at ICPP
- Shutdown activities at TRA
- Maintenance work in the resin bed

Many of the operations required were manual, rather than remote. Additionally, the original reactors were not well designed, which likely resulted in higher exposures than at ATR. Engineering controls, such as maintaining negative pressure or providing shielding, were utilized to minimize occupational exposure and release to the environment. The hot shops and change rooms maintained negative pressure. Radiation Work Permits were also used. Continuous Air Monitors (CAMs) and Radiation Area Monitors (RAMs) set up in different areas indicated any changes in radiological conditions. When either of these alarms sounded, individuals were told to exit the area immediately.

The collection of krypton and xenon gas samples was a source of significant exposure to operations. The primary dose associated with this operation, assuming PPE was intact, was beta dose. Krypton escaped into the work place and through the stack as a result of leaks in valves and pipes. Workers had to use high pressure to push krypton gas into a cask or a bottle. About only 45%–50% krypton would be collected into the casks or bottles. One of the methods used to reduce β exposure at this and other facilities was wearing three pairs of anti-C clothing. There was a period of time when they placed plastic over the TLDs; the effect on the recorded shallow dose is not known.

There were 6-in to 8-in pipes all over the basement room housing the calciner vessel. The radiation field was very high, in the range of 50 to 60 R/hr. Workers had to obtain special approval from management to work this job, because the expected dose was 5 rem. The vessel was very hot, in terms of both temperature and radiation. Temperature of 530°C made it difficult to work around the vessel. There was a spray head and a wing nut inside the bottom of vessel that required maintenance, but there was limited space to work in. In addition, there were oil and airborne contamination hazards around the vessel. The maintenance job involved reaching in for the wing nut, thus bringing the body close to the vessel; consequently, stay time was very limited. Those involved wore several layers of coveralls, anti-C clothing, gloves, etc. Personnel wore a

TLD, a pocket dosimeter, and extremity dosimetry. Individuals involved feel their record does not adequately reflect the exposures they received.

In the mid-1980s, ICPP began the ROVER Project, which involved graphite fuels. As a result, graphite contaminants were found all over the facility. The primary concern was strictly α emitters. Worker protection required a special PPE and worker decontamination program. After worker decontamination and before exiting the facility, nasal swipes and then a WBC were done on the workers to ensure they were not contaminated.

In about 1986/1987 time frame, an equipment operator was working in a subpile room at the ATR Facility. The task was to take out some pipe covering and lead shielding in the room underneath the reactor. The stay time was limited to 10 minutes, and the total job dose was 900 mrem. Those involved wore a TLD and PIC on the chest. and two pairs of Anti-Cs. The exposure from this job was not reflected in the dosimetry record of personnel involved.

In the early 1990s, maintenance personnel had to remove fuel and feed nozzles remotely in the New Calciner Facility. The removed nozzles were taken to be decontaminated and then to the hot machine shop to be cut apart. One of the kerosene nozzles in the calciner vessel was plugged up. They did not have the machinist to do that kind of job, so that they called the TRA machinist to do it. In this special task, no respirator was used. This was not a routine job and the individuals involved were not proficient in the operation. After the calciner stage, the calcine products were transported to the storage tank by piping. In this tank, there were dissolved stainless steel, cesium, and strontium. There were also slurries from the fuel dissolution, extraction, and de-nitration processes. The operators added nitrate, followed by organics, to the slurries in the process. They then added water to extract out uranium and plutonium. The denitration process was electrically heated and the products were reduced to powder. These powders were packed in sealed 2-kg cans or sealed overpacks. There was a lot of contamination around this facility, and personnel used paint to fixate the contamination. A 20-lb air purge was done at one point to remove air bubbles, during which a 25 R/hr field was measured around the system. Workers were wearing a single set of Anti-Cs and no respiratory protection.

The Advanced Retrieval Program in the RMWC involves retrieving buried waste drums. Most of the waste drums in Pit 4, where they are currently working, are still in good shape. The airborne radioactivity levels are in the range of 15,000 to 50,000 DAC-hour. To contain this airborne material, a tent made of Herculite fabric was build over the pit. Drums in Pit 9 are in very bad shape and are expected to cause more of a radiological hazard. Rocky Flats wastes are stored in the buried waste drums being retrieved.

Safety Analysis

There were technical review standards and a "Safeguards Committee" that reviewed all operations in the early days. However, detailed written procedures were not produced. After ATR was built, extensive procedures for reactor operations were developed. Detailed written criticality safety procedures were not available until the later years, although criticality evaluations were performed. There were procedures in place to ensure criticalities did not occur.

For example, highly enriched uranium fuel elements were stored in wooden racks. Each rack was limited to six full assemblies.

A Hazard Report was generated for MTR. This was the predecessor of the Safety Analysis Report (SAR). A Safeguards Report was generated for ETR in the mid-1950s. This is similar to a design basis document. An SAR was generated for ATR.

Field Radiological Control Records

A number of pertinent field radiological control records have been generated over the period of operation at INL. These records include Safe Work Permits (SWPs), Radiation Work Permits (RWPs), general area radiation and contamination surveys (with maps), job-specific radiation and contamination surveys, Health Physics/Radiological Control logbooks, air sample sheets, etc. The Health Physics/Radiological Control logs included information on the daily radiological activities and unusual occurrences. Field radiological control records were facility-based. For large generators, record types (e.g., RWPs) are stored together. For smaller generators, there may be multiple record types covering a particular period in the same box.

All personnel contamination incidents were documented on a PEQ after their implementation, whether it was reportable or non-reportable. These were also used to document skin dose. RadCon maintained a PEQ file. Spread of contamination incidents were recorded in logbooks and on survey reports.

Currently, radiological field records are stored in the RadCon office of the facility. Surveys, air samples, and other documents are reviewed routinely by the RCT foreman. These records are typically maintained at this area for less than 1 year. Like records are boxed and sent to the central radiological records area. Contents of boxes submitted are inventoried. The retention period for these records is 75 years. ANL-W records were maintained by ANL-W while they were under the jurisdiction of the University of Chicago. As many of the operations at SMC are classified, this facility maintains its own records. Field records are boxed separately from dosimetry records.

Dosimetry Records

AEC originally had the responsibility for maintaining dosimetry records. This provided some continuity between the numerous INL contractors. The reporting protocol for dosimetry data was based on the standards of the time. The initial standards used at INL were the AEC 0524 Manual and the AEC 0525 Manual, which provided guidance on dosimetry and reporting. Subsequent revisions, including ERDA Chapter 0524, DOE Order 5480.1 Chapter XI, and DOE Order 5480.11, were the next standards that were implemented. The DOE Radiological Control Manual was implemented as mandatory guidance, followed by 10 CFR 835. The RadCon manual was required by contract up until about 5 years ago, and is currently a DOE Standard.

There were code names used for dosimetry records, including INL site area codes, contractor codes, reason codes, and excuse codes. There is an official list of acronyms used across the site to refer to facilities, programs, etc. There was a historical dosimetry report for film badge usage prepared by Foster Cipperly. This report documents information on the dosimetry and records program up to a certain time frame. It had been provided to NIOSH.

Badge numbers were used as personal identifiers for some dosimetry records at the site. These badge numbers were not reused with one exception; the INL Site Managers/AEC/ERDA/DOE Field Office Managers/Presidents were always given the highest badge numbers (i.e., 1, 2 and 3). Dosimetry badge numbers were used for dose records and were unique. Dose records were stored under this number.

Historically, INL categorized doses as penetrating (deep) dose, non-penetrating (shallow) dose, and neutron dose. In the early 1950s, beta-gamma (combined) dose was categorized as non-penetrating/penetrating. Thereafter, the dosimetry algorithm called for separate categorization of non-penetrating and penetrating doses. Neutron dose was maintained and reported separately from the penetrating and non-penetrating dose. Current dosimetry records report "shallow" and "deep" doses, with the notation that penetrating deep doses would include neutron and gamma doses. Records provided in support of EEOICPA include "shallow" doses, "deep" doses, and "extremities" dose. Neutron doses were provided separately, but were also included in the "deep" doses.

PIC and film badge results were initially recorded together on the same form. The dosimetry group eventually went to a logbook recording system, and the primary dosimeter results were maintained separately from PIC or electronic dosimeter results.

Some bioassay records or results were maintained by the contractors during the early years, but incorporated into the records of the central dosimetry group. However, the AEC Health and Safety Division and its successors maintained the records of WBCs, bioassays, and internal dose evaluations that were conducted under their cognizance. In the late 1980s, the contractors took over the responsibility and continued with the existing records.

Offsite exposure reports from personnel monitoring at other DOE complex facilities are received and integrated into the individual's dosimetry file. There are several file cabinets full of exposure records received from other facilities. Offsite exposure data was supplied to NIOSH when available.

NRF records were maintained by this facility and were not intermixed with other INL records. Some workers may have worked for both the DOE contractors and NRF facility. NRF doses are excluded from the compensation program, as they are associated with the Department of Defense.

Individual dosimetry records were never destroyed, and there were no field records destroyed, with one notable exception. There was an incident at EBR-II where several boxes of records were contaminated. These records were thought to have been recopied and the originals stored or buried. There are subsets of records that have been removed from the site. When GE left the ANP, they took the worker exposure records with them. GE had a 4-digit number coding system for each individual worker. GE may have sent all the worker exposure records to ORNL. INL does have the exposure records for individual workers involved with SL-1, including those who were working for GE at that time. The information available in the INL files was mainly for individuals involved with the SL-1 reactor examination (Hot Shop at TAN) and the cleanup. Personnel came from ORNL and other sites to support the SL-1 cleanup and went back to ORNL after they completed their tasks. SL-1-related exposure records for individuals involved in the cleanup work were kept in boxes stored in a room at the CFA 690 Building.

As the DOE record policy has changed over the years, records may have been destroyed. For example, operational reports were disposed of after the established records retention period of the time. Personal dosimetry records had a retention period of 75 years and are intact. At INL, historical analytical data were thrown away at one point in time. They were reconstructed by memory and knowledge, in addition to personal files. Data was documented in summary and operational reports.

Medical Records

Existing staff have had the opportunity to review historical medical records. These records were not as extensive as the records maintained in more recent times. Without sufficient regulatory guidance in earlier years, each contractor employee given a medical exam had different requirements and medical forms. This included the particular elements of the exam. There were two flaws found in historical paper or microfilmed records.

(1) Occupational medicine was not a specialty, and this type of information was not emphasized.

(2) The quality of documentation was sometimes poor, due to the handwriting and subsequent microfilming of the original documents.

These historical records have been kept in a system of records and are organized according to patient name. The electronic medical record system began in 1995. All digitized examination records include examinations and radiology interpretation (i.e., lateral, posterior-anterior, ankle, chest, etc.).

Radiological control incidents were not included in the medical records unless medical intervention occurred. For example, if an individual was involved in an incident where a vial with I-131 was broken, there would be no mention of this in the medical record unless the individual received treatment.

Unauthorized Practices

Deliberate violation of RadCon requirements was considered a serious impropriety and management treated these allegations very seriously. If found guilty, an employee could be punished by discharge.

In the 1950s and 1960s, some individuals would take off their dosimeters (to reduce the measured exposure readings), because they wanted overtime work. Conversely, there were other rumors that some workers would put their badges in high radiation areas to "cook" them, so that they would not have to work more hours in radiation areas. RCTs did not verify that individuals were wearing their dosimetry except prior to "hot jobs." Research and evaluations into these practices were conducted by facility management, but these rumors were not substantiated with facts. The RWP was the vehicle to control the work process. Historically, most people followed the rules, and badge misplacement was not commonly done deliberately. Sometimes the workers could have inadvertently moved the badge during work. INL procedure required a PEQ be completed when a worker went into a radiation area without proper badging. If individuals forgot their dosimeters, however, they would often not tell anyone, because they would get in trouble.

In 1978, at the 603 Sludge Facility, workers were observed hiding badges in a lead pig. This incident was immediately reported to supervision and warnings were issued to those workers. In another incident, a person was found asleep in a contaminated area; he was fired as a result.

In the beginning of the INL site, operational management allowed the RCTs to act as safety cops. Periodically, individuals were found eating, smoking, and chewing gum in radiologically controlled areas. Occasionally, RCTs would catch workers taking off their respirators in a contamination area to have a cigarette break. One time, a worker was found fishing in the waste water settling pond at TRA. His boat flipped over and the worker was up to his neck in waste water. If an individual was caught, an occurrence report was filed and disciplinary action was taken.

Although ALARA was practiced at INL, supervisors knew they had an authorized “bank” of 5 rem per year per worker, and considered this exposure as a resource for operational efficiency.

EEOICPA Program

The EEOICPA Program at INL began in July 2001. Initially, some data were not available or in place for claim support. INL has a Records Storage Facility, which currently contains 44,340 boxes of records. Included are approximately 220 boxes of exposure/dose records that have now been entered into an electronic system, and approximately 1,400 boxes of personnel files that are accessed regularly for EEOICPA. Some of the ANL-W records were managed by the INL Records Program, including radiological dosimetry records. The INL Dosimetry Office decided not to use or scan the field records, because they were felt to be not required for the purposes of the EEOICPA Program.

Dose records provided to NIOSH are submitted using batch numbers. The information provided includes dosimeter data for each badge cycle, in-vivo and in-vitro bioassay data, incident files, and a summary dosimetry report. Records provided to NIOSH are only redacted for Privacy Act reasons. For example, if there are records with more than one name on them, the record will be redacted, so that it contains only the information of the claimant. The EEOICPA Office includes medical and dosimetry records as requested by the Department of Labor (or as needed to help verify employment) for INL claimants. For the purpose of dose reconstruction, there is incomplete data and missing context of data during some periods of time. For those cases where medical records are incomplete or inaccurate, DOE has asked medical staff to make a guess of what the medical doses were. There is no technical basis for this guess.

Many workers shared by NRF (Naval Reactor Facility) and INL were construction-type workers. They were monitored by DOE and shared dose exposure data with NRF. NRF dose data were excluded from the data provided to NIOSH under the EEOICPA. The electronic database housing dose information has not been provided to NIOSH. Some claims contain over 2,000 pages of exposure records. NIOSH has not requested and is not provided with field radiation records from INL.

Documents associated with the SL-1 accident have been provided to the NIOSH Advisory Board on Radiation and Worker Health. Additional information on the SL-1 incident was published in 1963 in the *Health Physics Journal* by the individual then in charge of radiation monitoring, environmental monitoring, and the dosimetry processing program.

Release data from the Naval Reactor Facility were deemed classified until recently; therefore, the HDE report assumed values for environmental releases from this facility. In 2004, NRF provided environmental release and emission information to INL. NIOSH requested NRF environmental release documents from INL more than a year ago. These documents were redacted, but NIOSH has indicated no further interest in this data. The group within NIOSH requesting this data is unknown. INL has scanned and provided this data to the SC&A team.

Radiological Control, DOE, and EEOICPA personnel interviewed believe these NIOSH individuals are very qualified to develop the INL Site Profile Technical Basis documents. INTREPID, a contractor to NIOSH, participated in interviews with the current INL and DOE radiological control staff and reviewed historical records. A number of documents were copied from the 1950s, 1960s, 1970s, and 1980s. The INTREPID team included technical experts with a long history of working at INL, which union officials believe to be a conflict of interest. ANL-W was initially included in the INL TBD; however, ANL-W now has its own TBD.

Chemical Exposure

In addition to being exposed to radiation, selected workers were also exposed to the following chemicals:

- Beryllium
- Asbestos
- Cadmium
- Lead
- Mercury
- Chromium
- Iodine
- Hydrofluoric acid from the denitrification process
- Nitric acid
- TCE

At times, there was very little concern with chemical exposure. Workers were allowed to handle solvents barehanded. There was no safety control and no chemical monitoring, and workers often had no idea of what to do with chemicals. For example, historically at ICPP, if the acids fumes were not overcoming individuals, the area was not monitored and PPE was not used. In the reactor areas, the primary concern was with radiation exposure. Engineers would approve engineering analysis of a job based on the acceptance limits of chemicals versus radiological hazards. The wet method for asbestos removal was not immediately implemented at the site. Brooms and dust pans were used to clean spills. Respirators were not worn.

Asbestos blankets were used to control the spread of sparks from welding. They also were used as a kind of shielding. Instrument technicians, welders, electricians, machinists, maintenance workers, operators, and craftsmen often all worked in the same areas. They were using Inconal or Carpenter 20 (proprietary alloy) for piping to shield high temperature heat. Welders would build the shielding using these alloy materials. Among the other hazards at INL were the reactor beryllium reflectors, which had to be hand-filed in some cases.

Miscellaneous

- Historical medical records are of little value in determining cancer effects from radiation. This is primarily due to the limited latency period for cancer since 1995. Medical records also do not document industrial hygiene issues effectively. In fact, many industrial hygienists took their records with them when they left the INL site. Similarly, many site employers took their records with them when they left.

- Medical personnel believe that facility releases were not a significant issue for onsite workers. Mortality rates have shown no excess for INL workers above background levels.

- Plant engineering support personnel were responsible for fixing things on the reactor, performing safety and core analysis, and developing concepts for new reactors.

- There have been formal audits performed through the operation of INL. Early audit reports were documented in operational reports. These reports were likely shipped to the Federal Records Center in Seattle. There were many audits performed on the dosimetry record system for the INL site. These audits were conducted by groups such as the DOE Tiger Team, DOE-ID, and DNFSB.

- There was incineration of low-level wastes at the Waste Experiment Reduction Facility in the later 1990s. Advanced Mixed Waste Treatment Program is not an incinerator.

- There are currently about 2,000 claims submitted from INL site workers. Many claims from ICPP personnel have been turned down so far by DOE and NIOSH.

- There was a bone cancer problem at TRA in the 1960s.

ATTACHMENT 4: CONSISTENCY BETWEEN HANFORD, SRS, AND INL SITE PROFILES

Table A-1: Occupational Medical Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	Savannah River Site	Hanford
Frequency of chest x-rays (Default)	Before 1954 (PA): Annual 1954–1969 (PA): New hires; Radworkers – 18–24 (every 4th year); 25–39 (triennial); 40–49 (biennial); over 50 (annual); Non-radworkers: 30–39 (every 5th year); 40–49 (triennial); 50–59 (biennial); over 60 (annual). 1970–1976 (PA+Lat): New hires; At age 25, 30, 34, 37, 40; 40–62 (biennial); over 62 (annual). 1977–1978 (PA+Lat): New hires; less than 45 (biennial); over 45 (annual). 1979 to 1/1990 (PA): New hires; high risk only (every 4th year). 2/1990 to present (PA+Lat): New hires; high risk only (every 4th year).	One annual x-ray procedure for each year or partial year.	Posterior-Anterior View: Before 1946 – 1/1982: Pre-employment, annual, and termination 1/82–1/83: Pre-employment, annual, and termination for over 50 years; Biennially for 40–49 years; Every third year for 39 years or younger. 1/83–3/90: Biennially for over 50 years; Every third year for 40–49; and Every 5 years for 39 years and younger. 3/90 – present: Every 5 years Lateral chest x-rays also given periodically prior to 4/1997.
Organ Dose Conversion Factors	Obtained from ICRP 34 (1982)	Obtained from ICRP 34 (1982)	Obtained from ICRP 34 (1982)
IREP Radiation Rate	Not indicated in the TBD.	Acute	Acute
IREP Radiation Type	Not indicated in the TBD.	Photons, 30–250 keV	Photons, 30–250 keV
IREP Dose Distribution Type	Not indicated in the TBD.	Constant	Constant
Total uncertainty	30% (no multiplier indicated)	30% (x-ray dose multiplied by 1.3 and entered as a constant)	30% (x-ray dose multiplied by 1.3 and entered as a constant)
Conversion Factor from PA to Lateral	No conversion factor indicated. Specific air kerma (mrad) for PA and Lateral used: Pre-1954: 200 (PA) and 200 (Lat); 1954–1990: 52 (PA) and 74 (Lat); 1990–Present: 53 (PA) and 76 (Lat).	2.5	2.5

Table A-1: Occupational Medical Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	Savannah River Site	Hanford
Chest Thickness	Chest thickness not indicated. Distance accounts for body thickness and space between person and film: PA View: 30 cm. Lateral View: 40 cm.	PA View: 26 cm Lateral View: 34 cm	PA View: N/A Lateral View: N/A
Substitute dose conversion factors for thyroid, eye/brain, ovaries and analogues, testes, and uterus	In the absence of collimation information, substitute DCFs used for the pre-1970 time frame (Table 4.0-1, Kathren 2003). For skin doses, a backscatter factor is taken from NCRP 102 Table B-8.	Substitute view and organ DCFs applied to minimally collimated beams prior to 1970. (Scalsky 2004, pg. 50)	Use DCFs for lung for all other organs in thoracic cavity; for organs in abdomen, use DCFs for the ovary (Scalsky 2003, pg. 10)
Analog organ for Thymus	Lung	Lung	Lung
Analog organ for Esophagus	Lung	Lung	Lung
Analog organ for Stomach	Lung	Lung	Lung
Analog organ for Bone Surface	Lung	Lung	Lung
Analog organ for Liver, gall bladder, spleen	Lung	Lung	Ovary
Analog organ for Remainder Organs	None (Specific organ doses are provided for skin.)	Lung	Ovary
Analog organ for Urinary/bladder and colon/rectum	Ovary	Ovary	Ovary
Analog organ for Eye/brain	Thyroid	Thyroid	Thyroid
	Posterior-Anterior View X-ray Techniques[1,2]		
<1946	Site was not in operation. INEEL began operations in 1949 (TBD#2, pg. 12).	Site not in operation.	kVp: Unknown mAs: Unknown SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 120 mR
2/1946 – 12/1950	Site was not in operation until 1949. <u>1949–1950:</u> kVp: unknown mAs: unknown SSD: 153 cm SID: 183 cm Filtration: unknown ESE: 200 mR	kVp: 80 mAs: 30 SSD: 152 cm SID: 183 cm Filtration: 1.5 mm Al ESE: 108 mR	kVp: 80 mAs: 25 SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 79 mR

Table A-1: Occupational Medical Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	Savannah River Site	Hanford
1/1951–4/19/59	1951– 1954: kVp: unknown mAs: unknown SSD: 153 cm SID: 183 cm Filtration: unknown ESE: 200 mR 1954 on: kVp: 90 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 52 mR	kVp: 80 mAs: 30 SSD: 152 cm SID: 183 cm Filtration: ESE: 108 mR	kVp: 80 mAs: 10 SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 79 mR
4/1959 – 12/1970	kVp: 90 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 52 mR	kVp: 80 mAs: 30 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 108 mR	kVp: 80 mAs: 10 SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 40 mR
1/1971 – 1/1983	kVp: 90 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 52 mR	kVp: 110–120 mAs: 10 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 44 mR	kVp: 80 mAs: 10 SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 40 mR
1/1983 – 7/1985	kVp: 90 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 52 mR	kVp: 110–120 mAs: 10 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 44 mR	kVp: 100 mAs: 10 SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 35 mR
8/1985 – 3/1990	kVp: 90 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 52 mR	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 100 mAs: 10 SSD: 72” (183 cm) SID: 183 cm Filtration: 2.5 mm Al; 4.0 mm Al for CONX Type 12 ESE: 35 mR
3/1990 – 4/1997	kVp: 100 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 53 mR	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 6.7 SSD: 72 “ (183 cm) SID: 183 cm Filtration: 4.0 mm Al ESE: 21 mR

Table A-1: Occupational Medical Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	Savannah River Site	Hanford
4/1997 – 2/1998	kVp: 100 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 53 mR	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 10 SSD: 183 cm SID: 183 cm Filtration: 4.0 mm Al ESE: 17 mR
2/1998 – 5/1999	kVp: 100 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 53 mR	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 5 SSD: 183 cm SID: 183 cm Filtration: 4.0 mm Al ESE: 11 mR
5/1999 – present	kVp: 100 mAs: 300 SSD: 153 cm SID: 183 cm Filtration: 2.5 mm Al total ESE: 53 mR	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 5 SSD: 183 cm SID: 183 cm Filtration: 4.0 mm Al ESE: 11 mR
Photofluorography			
Technique Factors	Photofluorography was not used at INEEL Site.	kVp: 100 mAs: 60 SID: 102 cm Filtration: 2.5 mm Al ESE: Applies from 1951–1957	kVp: 80 to 100 kVp mAs: not specified SID: 102 cm Filtration: 2.5 mm Al ESE: 1.53 R Applies 1945 to 1962

1 Refer to Scalsky 2004, pages 41–47 for SRS x-ray technique discussion.
2 Refer to Scalsky 2003, page 18 for Hanford x-ray technique summary.
3 N/A = not applicable; PA = posterior-anterior; LAT = lateral; kVp = kilovolt potential; mAs = milliampere-second; SSD = source-to-skin distance; SID = source-to-image distance; ESE = entrance skin exposure

Table A-2: External Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Missed Photon Dose Application	Applies to workers with no recorded dose because they weren't monitored or their results are unavailable; and workers who have a zero recorded dose (Rohrig, 2004, pg. 32).	Applies to workers with no recorded dose because they weren't monitored or their results are unavailable; and workers who have a zero recorded dose (Scalsky 2004, pg. 111).	Applies to workers with no recorded dose because they weren't monitored or their results are unavailable; and workers who have a zero recorded dose, (Fix 2004, pg. 75).
Missed Photon Dose Methodology	Divide the MRL by 2, and multiply by the number of zeros and not monitored periods (Rohrig 2004, pg. 32). Table 6B-1 (Rohrig 2004) provides potential maximum annual missed photon dose by period of use.	(1) For a claimant-favorable maximum potential missed dose, use the limit of detection (LOD) multiplied by the number of zero doses (Scalsky 2004, pp. 111 and 238) (2) Divide the LOD/2, and multiply by the number of zeros and not monitored periods; (Scalsky 2004, pg. 242), or (3) Missed doses are added to measured doses and treated as a constant.	Divide the MDL by 2, and multiply by the number of zeros and not monitored periods (Fix 2004, pg. 75). Table 6E.6 (Fix 2004), provides potential maximum photon dose by year.
IREP Dose Distribution Type for missed photon dose	The percentage bias for gamma measurements at INL is plotted in Figure 6-9 (Rohrig 2004, pg. 22). The results lie within +27% to -43%. The TBD estimates that a realistic total uncertainty for photon dosimetry is about 35% at one sigma, which is consistent with the relative bias results. (pg. 34).	(1) When using the LOD/2 methodology, a lognormal distribution with a geometric standard deviation of 1.52 in Parameter 2 of the IREP input is used (Scalsky 2004, pg. 116). (2) When simply adding the missed and measured dose, a constant is used.	Lognormal distribution with a geometric standard deviation of 1.52.[1] The assessment at Hanford is based on the assumption that uncertainties from individual sources followed independent lognormal distributions. For each uncertainty source, a factor is assigned reflecting bias (B) and a 95% uncertainty factor (K); the uncertainty factor was determined so that the interval obtained by dividing and multiplying by this factor would include 95% of all observations (Fix 2004, pg. 27).

Table A-2: External Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Missed Neutron Dose Application	Assign a missed neutron dose equivalent using Table 6B-2 for the times when workers do not have reported neutron dose. For the period when NTA film was used, the dose should be multiplied by 1.25 for all facilities except the MTR experimental floor or by 2 for the MTR experimental floor when the MTR was operating between 1953 and 1970. Then, the dose equivalent is apportioned into the IREP groups using Table 6B-3. (Rohrig 2004, pg. 33).	Assign a missed neutron dose if there is neutron monitoring between 1958 and 1962, if there is neutron monitoring in 1971 or later, or there is indication of use of the 17 keV calibration curve for interpretation of beta/gamma film. Also applies to those who worked with Cf or Cm, maintenance workers, those involved in the PuAl target campaign, and those on routine plutonium bioassay. If the recorded neutron dose is greater than the calculated dose, the calculated dose is used (Neton 2003).	Assign a missed neutron dose if the individual worked in a facility with a potential for neutron exposure. The vast majority of neutron dose to Hanford workers was received at the 200 West Area Plutonium Finishing Plant (PFP) facilities (pg. 74.) There is potential for significant missed dose in the 300 Area plutonium laboratory (308, 309, 324), the 100 Area reactor facilities (i.e., reactors B, D, F, H, DR, C, KW, KE), the 300 Area accelerator (3754B), the calibrations facilities (3745, 318) and the Fast Flux Test Reactor (pg. 3). (Fix 2004).
Missed Neutron Dose Methodology	Use MRLs for neutron dosimeters because the neutron dosimeters were calibrated with neutron sources that had energies similar to those encountered in the workplace and most of the neutrons to which workers were normally exposed had energies greater than the 500 to 80-keV threshold of the NTA film dosimeters. Divide the MRL by 2, and multiply by the number of periods (week or month or quarter) (Rohrig 2004, pg. 32). Table 6B-2 (Rohrig 2004) provides potential maximum annual missed neutron dose by period of use. Multiplying factors are also applied here.	A neutron-to-photon ratio is applied to missed and recorded photon dose for nonmonitored worker and workers with inadequate neutron monitoring (i.e., prior to 1971). The upper 95% value is used for the maximizing technique. The geometric mean value is used for the best-fit technique (Scalsky 2004, pp. 240–241). After 1970, the assignment of missed dose is based on the limit of detection provided in Table E-10 (Scalsky 2004, pp. 241–242). It appears that an ICRP 60 correction factor is applied to missed dose; however, this is unclear in the TBD (Scalsky 2004, pg. 110).	A neutron-to-photon ratio is applied to missed and recorded photon dose for nonmonitored worker and workers with inadequate neutron monitoring. The upper 95% value is used for the maximizing technique. The mean value is used for the best-fit technique (Fix 2004, pp. 75–77).

Table A-2: External Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
IREP Dose Distribution Type for missed neutron dose	No information on distribution with standard deviation is provided. Fractions of neutron dose equivalent were calculated. The conversion factors from dose equivalent to equivalent dose for INL spectra were also calculated. The ratios of average radiation weighting factor to average quality factor for the IREP energy groups showed variation. Default quality factor corrections were provided in Table 6-8 (Rohrig 2004, pg. 32).	Lognormal distribution with a geometric standard deviation of 1.52.[1]	Lognormal distribution with a geometric standard deviation of 1.52.[1]
IREP Exposure Rate	No information is provided.	Acute for beta and photon Chronic for neutron (Scalsky 2004, pp. 87 and 235, respectively).	Acute for beta and photon Chronic for neutron (Fix 2004, pp. 8, 59, and 69, respectively)
IREP Radiation Type (default)	Photon, 30–250 keV Electron, > 15 keV, Neutron, 0.1–2, 2–10 MeV (Rohrig 2004, pp. 23 and 48)	Photon, 30–250 keV Electron, > 15 keV, Neutron, 0.1–2 MeV (Scalsky 2004, pp. 49, 236, and 237, respectively)	Photon, 30–250 keV Electron, > 15 keV Neutron, 0.1–2 MeV (Fix 2004, pg. 29)
Organ dose conversion factor	For photons prior to 1981, the conversion factor from exposure to organ dose should be used. For photons 1981 and after, the conversion factor from deep dose equivalent to organ dose should be used. For neutron doses in each IREP energy group, the conversion factors from ambient dose equivalent to organ dose for AP irradiation from Appendix B of NIOSH 2002 should be multiplied. (Rohrig 2004, pg. 34).	For the maximizing approach, a value of one is used (TBD, pg. 61). For the best-fit analysis, the dose conversion factors in the external dosimetry guide for the relevant exposure geometry. OCAS-IG-001 Appendix A (NIOSH 2002) contains a detailed discussion of the conversion of measured dose to organ dose equivalent, and Appendix B contains the appropriate dose conversion factors (DCFs) for each organ, radiation type, and energy range based on the type of monitoring performed. (Scalsky 2004, pg. 242)	The dose conversion factors for each, organ, radiation type, and energy ranged from OCAS-IG-001 are used. If the exposure geometry cannot be determined, default values are found in Table 6E-9 (Fix 2004, pg. 77). No separate value is provided for the maximizing approach.

Table A-2: External Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Exposure geometry	No information is provided for exposure geometry such as those for Hanford and SRS. For neutron exposure, it is assumed that the worker irradiation is in an AP geometry. (Rohrig 2004, pg. 34).	Default exposure: Likely non-compensable workers - 100% AP Compensable worker – 50% AP, 50% ROT Compensable supervisor – 50% AP, 50% ISO. Dose reconstructor has the option to choose the most appropriate exposure geometry for the individual. (Scalsky 2004, pg. 242)	Default exposure: Likely non-compensable workers - 100% AP Compensable worker – 50% AP, 50% ROT Compensable supervisor – 50% AP, 50% ISO. (Fix 2004, pg. 77)
Photon Adjustment Factors (Recorded Dose)	No adjustment for photons is provided.	Multiply by 1.119 for years prior to 1987. Multiply by 1.039 for 1987. No adjustment is needed post-1987 (Scalsky 2004, pg. 238). Note: Taylor et al.(1995) indicates that the 1.119 adjustment factor should be applied through 1985 and the 1.039 adjustment factor should be applied for 1986. No correction is required for 1987 and after.	No adjustment for the multi-element dosimeter, TLD, or gamma dose. For 200 Area plutonium workers prior to 1957, the 20% of the open window dose is added to the penetrating dose (Fix 2004, pg. 73).
IREP Dose Distribution Type for recorded photon dose	No information is provided for IREP dose distribution type.	Constant. The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included. [1]	Constant.[1]
Recorded Neutron Dose Adjustment Factor (Prior to 1971 – SRS; Prior to 1972 Hanford)	Table 6-8 lists the recommended default values for the dose equivalent fractions and quality factor corrections for adjusting recorded dose (Rohrig 2004, pg. 31).	NTA film is considered inadequate for use in dose reconstruction due to the energy dependence. The missed neutron dose approach is applied for this period of time. If the measured dose from the NTA is greater than the calculated dose, this value is used and the ICRP 60 conversion factor is applied (Scalsky 2004, pg. 238).	NTA film is considered inadequate for use in dose reconstruction due to the energy dependence. The missed neutron dose approach is applied for this period of time (Fix 2004, pg. 48).

Table A-2: External Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Recorded Neutron Dose Adjustment Factor (7/78–12/83)	See above.	In order to calculate the dose input for the IREP, Table E-1, the recorded neutron dose must be separated into neutron energy groups as shown in Table E-3 and subsequently converted to ICRP 60 (1990) methodology (Scalsky 2004, 235–238).	When using the four-chip HMPD during the period of its use from July 1978 through December 31, 1983 in Hanford 200 and 300 Area plutonium facilities only, multiply the recorded neutron dose by 1.35. At all other times, divide the dose into the facility specific neutron energy bins, and multiply by the ICRP 60 conversion factor (Fix 2004, pg. 74).
Recorded Neutron Dose Adjustment Factor (1/72–6/78, 1/84 – present)	See above.	In order to calculate the dose input for the IREP, Table E-1, the recorded neutron dose must be separated into neutron energy groups as shown in Table E-3 and subsequently converted to ICRP 60 (1990) methodology (Scalsky 2004, 235–238).	Divide the recorded neutron dose into the facility specific neutron energy bins, and multiply by the ICRP 60 conversion factor (Fix 2004, pg 74).
IREP Dose Distribution Type for recorded neutron dose	Various for different IREP energy groups. See Table 6-8 (Rohrig 2004, pg. 32).	Constant. The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included. [1]	Constant[1]
Shallow Dose Adjustment Factors	Shallow dose adjustment factors are not addressed in this TBD.	Shallow dose adjustments factors are not addressed in the TBD or SRS TIBs.	Shallow dose adjustments factors are not addressed in the TBD.
Low-energy photons (< 30 keV)	INL reported doses as penetrating and non-penetrating. The penetrating dose corresponds to the deep dose equivalent and the nonpenetrating dose plus the penetrating dose corresponds to the shallow dose equivalent.	1954–1981 Subtract the reported deep dose from the shallow dose for plutonium workers. 1982–present. Plutonium workers are those individuals that worked in 321M, 221H – B line, 221F – B line, 772F, 235F, 773A, 736A, and other plutonium storage areas (Neton 2004). (For testicular, breast, or skin cancer)	The stated Hanford practice to include 1/5 of the shallow dose based on a 16-keV calibration to the deep dose for Hanford plutonium facilities workers could resolve this source of potential under-response around 17 keV (Fix 2004, pg. 26). For 200 Area workers prior to 1957, the 20% of the open window dose is added to the penetrating dose, (pg. 14).

Table A-2: External Exposure Default Assumption Comparison for INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
IREP Dose Distribution Type for recorded shallow dose	Not included in the TBD.	Shallow dose is addressed from a technical perspective in the TBD, but no direction is provided to the dose reconstructor (Scalsky 2004, pg. 97).	Not included in the TBD.
IREP Radiation Type for recorded dose	Specific to the particular facility for beta, photon, and neutron dose. For example, in the reactor area 100% of the beta doses is assumed to be >15 keV, 75% of the photon dose is >250 keV, and 25% of the photon dose is 30–250 keV (Rohrig 2004, pg. 23).	Specific to the particular facility for beta, photon, and neutron dose. For example, in the reactor area 100% of the beta doses is assumed to be >15 keV, 50% of the photon dose is >250 keV, and 50% of the photon dose is 30–250 keV (Scalsky 2004, pg. 98).	Specific to the particular facility for beta, photon, and neutron dose. For example, in the reactor area 100% of the beta doses is assumed to be >15 keV, 75% of the photon dose is >250 keV, and 25% of the photon dose is 30–250 keV (Fix 2004, pg. 29).

[1] These parameters were obtained from review of several dose reconstruction IREP input sheets.

Table A-3: Comparison of Default Assumptions for Internal Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Particles Size (default)	5 micron AMAD (default) for RWMC (TBD, pg. 34). 2.4 micron AMAD (default) for SMC (TBD, pg. 29). No default values are provided for other INEEL facilities.	5 micron (Scalsky 2004, Section 4.0, Attachment D)	5 micron (Bihl 2004, pg. D-10)
Intake Type (default)	Chronic (TBD, Table 5.7-1).	Chronic (Scalsky 2004, Section 4.0, Attachment D)	Chronic (Bihl 2004, pp. 7–9)
Default Excretion Volume	Not included in the TBD.	1.4 liters/day (Volumes less than 1.4 liters/day are corrected by normalizing the actual volume to 1.4 liters/day. Samples recorded as activity per 1.5 liters are not corrected.) (Scalsky 2004, pg. 70)	Uses a urinary excretions value of 0.2 ug/d for elemental analyses, 0.15 dpm/d for ^{234}U and ^{238}U and essentially anything detected for ^{235}U (Bihl 2004, pg. 27)
Solubility Class	The default assumption of M or S would be appropriate, based on the most claimant-favorable result to the organ in question (TBD, pg. 10).	For the maximizing approach, the most claimant-favorable solubility type for the organ of interest is used. For the best-fit approach the most appropriate solubility type can be used (Scalsky 2004, pg. 85).	For the maximizing approach, the most claimant-favorable solubility type for the organ of interest is used. For the best-fit approach the most appropriate solubility type can be used. Inhalation class and lung absorption type for uranium is found in Bihl 2004, Table 5.2.5-3, pg. 24).
Intake Date for Hypothetical Intake (excluding tritium)	Not included in the TBD.	Acute inhalation on January 1 in the first year of employment (Scalsky 2004, pg. 85; Bracket 2003, pg. 3).	First day of employment or the first day of operation of the facility where the worker was assigned. For separation plants, chronic intakes would apply from either the first day of work for the worker or the start-up of the plant, December 1944 for T Plant and April 1945 for B Plant (Bihl 2004, pg. 8).
Tritium Missed Dose Application	Not included in the TBD.	Assigned to workers monitored for external dose, but having no bioassay. For workers not in the dosimetry or bioassay-monitoring program, the missed internal dose is based on environmental intake only. Scalsky 2004, pg. 84; Duncan 2003, pp. 6 and 12)	Assigned to workers who worked in 108-B, the 300 Area Test Reactors, and in some cases where work location was unknown or variable. Those who never wore a dosimeter and had no bioassay results were assigned environmental doses (Bihl 2004, pp. 21–22).

Table A-3: Comparison of Default Assumptions for Internal Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Basis for Tritium Missed Dose	Not included in the TBD.	Dose calculated based on the tritium reporting level for a particular time period (Scalsky 2004, pg. 67; Duncan 2003, pg. 6).	Tritium urinalysis was not perfected until 1961. Liquid scintillation counting for tritium likely was started in 1958 (Bihl 2004, pp. 21–22). From 1949 to 1960 the MDA was 5 uCi/L and from 1961 to 1981 the MDA was 1 uCi/L. Later in 1982 the MDA changed to 10 dpm/ml and in 1991 to 20 dpm/ml, (Bihl 2004, pg. 22). Tritium intakes were accounted for as part of external dose until about 1986–1987 (TBD does not explain methodology), when they were entered in the dose database as internal dose (Bihl 2004, pp. 12 and 22).
Hypothetical Intake Application	Not included in the TBD.	Applied to claims with non-metabolic and digestive tract cancers (Scalsky 2004, pg. 85; Bracket 2003, pg. 2).	Applied to individuals who wore a dosimeter but did not have any bioassay (Bihl 2004, pg. 48).
Basis for missed internal dose from radionuclides other than tritium	(1) If claimant file include 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 in recorded, only the environmental dose should be included (TBD, pg. 37). (2) The probability that a worker received 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	(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose (Scalsky 2004, pg. 84; Bracket 2003, pg. 2). (2) For those individuals with external monitoring but no or limited internal monitoring, an annual missed tritium dose and environmental dose from uranium, plutonium, and ^{131}I are assigned as internal dose. It is also reasonable to pick a fission or activation product that produces the largest dose to the organ of interest (Scalsky 2004, pg. 84; Bracket 2003, pg. 8). (3) Highest five intakes for various nuclides are applied to those individuals with non-	(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose, (Bihl 2004, pg. 48) (2) For those individuals with external monitoring but no or limited internal monitoring, the approach was year dependent. For 1947 through 1952, daily intakes at 10% of the respiratory protection required value for 40 hours/week were assumed. Iodine was assumed to be at 0.1 times the vapor index. For 1953 through 1988, daily intakes were based on an exposure to airborne concentrations at 10% of the limiting air concentration for four hours per week, (Bihl 2004, pg. 49). (3) From 1989 through the present, a daily exposure

Table A-3: Comparison of Default Assumptions for Internal Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
	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 (TBD, pg. 39). (3) 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 (TBD, pg. 40).	metabolic or digestive system cancers (Bracket 2003, pg. 2).	at 5% of the limiting air concentration for 4 hours per week was assumed, (Bihl 2004, pg. 50). (4) For monitored workers with no confirmed intake, a maximum intake is determined by using the MDA of the last sample as an upper bound (Bihl 2004, pg. 47).
Radionuclides included in the Hypothetical Intake	Not included in the TBD.	^{241}Am/^{241}Pu (M), ^{244}Cm (M), ^{60}Co (S), ^{137}Cs (F), ^{237}Np (M), ^{238}Pu (M), ^{239}Pu (M), ^{90}Sr (F), ^{234}U (F), and ^{238}U (F) (Bracket 2003, pg. 9)	Variable by facility and organ of interest. Alpha intakes are assigned for the Plutonium Finishing Plant (PFP), the 200 Area Fuel Separations Plants, U-Plant, C-Plant, the 300 Area Fuel Fabrication Facilities, 209E, 120, 324, 325, 327, the Tank Farms and evaporator facilities (0.5 times the alpha intake), and where work location is unknown or highly variable. Alpha intakes are based primarily on ^{234}U or ^{239}Pu. Beta/gamma intakes are assigned for all facilities *except* PFP, 209E, 120, the 300 Area Fuel Fabrication Facilities, 108-B, and U-Plant. Tritium intakes are assigned for the 108-B Building, the 300 Area Test Reactors, and in some situations where work locations are unknown or variable. The particular beta/gamma radionuclide and its solubility class are determined based on the organ of concern. For

Table A-3: Comparison of Default Assumptions for Internal Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
			some facilities and periods of time it is specified (Bihl 2004, pp. 51–52).
Default Activity Ratios Pu Mixture	Not specified in the TBD.	Ten-year old 12% plutonium mix (Scalsky 2004, pg. 66).	Not specified in the TBD.
Activity Fractions for other Mixtures	Activity fractions are provided for ATR fuel gaseous releases in Table 5.6.6.1-1 (TBD, pg. 36). Activity fractions are provided flor SMC DU uranium isotopes in Table 5.6.1-1 (TBD, pg. 28).	Activity fractions are facility dependent. The activity fractions are taken from the Internal Dosimetry Technical Basis Manual (WSRC 1990). The information for these ratios was obtained from safety analysis reports, personal interviews, open literature, etc.	Activity fractions are provided for uranium mixtures, Table 5.2.5-3, page 24, weapons and fuel grade plutonium, Table 5.2.1-3 page 16, and recycled uranium impurities., Table5.2.5-2, page 24. Default mixtures based fission product urinalysis was developed by time period and organ of concern (Bihl 2004, pg. 10, Attachment D).
Radionuclides of Concern for Monitored Workers	Radionuclides of concern are identified for each of the eight INEEL facilities (TBD, Tables 5.6.1-1 to 5.6.6.1-2, pp. 28–37).	Radionuclides of concern were based on the in-vivo and in-vitro bioassay data of the individual (Scalsky 2004, pp. 66 and 67). Although the TBD provides activity fractions in Attachment A, it is not clear how these activity fractions are used in dose calculations.	Radionuclides of concern were based on the in-vivo and in-vitro bioassay data of the individual, or the minimum detectable activity for a particular radionuclide. Radionuclide assumptions varied by facility and organ of interest (Bihl 2004, pg. 13).
Tritium Dose for Monitored Workers	Tritium is not identified as a key radionuclide of concern at INEEL facilities.	Based on the reporting level if the tritium bioassay is less than this level, or the actual bioassay result if it is greater than the reporting level. Organically Bound Tritium and Stable Metal Tritides are not considered (Bracket 2003, pg. 6).	Tritium urinalysis was not perfected until 1961. Liquid scintillation counting for tritium likely was started in 1958 (pp. 21–22). From 1949 to 1960 the MDA was 5 uCi/L and from 1961 to 1981 the MDA as 1 uCi/L. Later in 1982 the MDA changed to 10 dpm/ml and in 1991 to 20 dpm/ml (pg. 22). Tritium intakes were accounted for as part of external dose until about 1986–1987 (TBD doses not explain methodology), when they were entered in the dose database as internal dose (pp. 12 and 22). (Bihl 2004, pp. 12 and 22)

Table A-3: Comparison of Default Assumptions for Internal Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Internal Dose for radionuclides other than tritium	See Table 5.7-1: defaults table for missed dose. This table provides the default assumptions for calculating missed dose of personnel working at a specific INEEL facility and during a specified time period (TBD, pp. 38–39).	Based on either actual bioassay values or detection levels for bioassay techniques. For non-metabolic cancers, the maximizing approach is used (Scalsky 2003, pg. 85).	Based on either actual bioassay values for positive values. Based on a chronic intake over the entire exposure period with the last sample assumed to be at the MDA (Bihl 2004, pg. 47).
Basis for pre-bioassay program doses	Not included in the TBD.	Not included in the TBD.	Air concentration tolerance or limits, (Bihl 2004, pg, 7)
Ingestion	Not included in the TBD.	Not included in the TBD.	Assigned during periods were air sampling was used to determine internal dose. The quantity is based on the air concentration level or on the guidance provided in *Estimation of Ingestion Intakes* (NIOSH 2004). (Bihl 2004, pg. 8)
Surrogate Radionuclide in IMBA for ^{65}Zn/^{95}Zr	Not included in the TBD.	^{137}Cs used as a surrogate. Surrogate Adjustment factor = 2.43. (Brackett 2003, pg. 9)	Not included in the TBD.
Surrogate Radionuclide in IMBA for ^{106}Ru/^{144}Ce/^{95}Nb	Not included in the TBD.	Radionuclides not available in IMBA. ^{90}Sr used as a surrogate. Surrogate Adjustment factor = 7.25 (Brackett 2003, pg. 9).	Not included in the TBD.
Surrogate Radionuclide in IMBA for ^{242}Cm/^{252}Cf	Not included in the TBD.	Radionuclides not available in IMBA. ^{244}Cm used as a surrogate. Surrogate Adjustment factor = 1.09 (Brackett 2003, pg. 9).	Not included in the TBD.
IREP Radiation Types for Hypothetical Intake	Not included in the TBD.	Alpha Beta: >15 keV Tritium: < 15 keV (Bracket 2003, pp. 8 and 12)	Alpha[1] Beta: >15 keV[1] Photon: > 250 keV[1] Tritium: < 15 keV[1]
IREP Dose Distribution Type	Not included in the TBD.	Constant (Brackett 2003, pg. 12)	Constant[1]
Internal Dose Uncertainty	Not included in the TBD.	For the missed dose assignments, the value entered includes the uncertainty. [1] No direction is provided to the dose reconstructor for dose assignments based on monitoring data.	For the missed dose assignments, the value entered includes the uncertainty. For dose assignments based on monitoring data, the following values can be applied as a standard deviation: (1) 0.3 times the MDA or reporting level, or (2) 0.5 times the MDA for chest counting.

Table A-3: Comparison of Default Assumptions for Internal Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
			Actually report errors can be used if available (Bihl 2004, pg. 46). For air concentration data, a triangular distribution with zero as the minimum, the derived values as the mode, and twice the mode as the maximum is used (Bihl 2004, pg. 7).
Other Comments	Personnel employed in the Naval Program are not included in this TBD (TBD, pg. 7). However, through the years, NRF has participated in limited coordination of radiological protection programs and site support services. Some workers' internal dose could have resulted from their support work at the NRF (TBD, pg. 9).	None.	Informs the dose reconstructor of limited use radionuclides such as ^{14}C, ^{232}Th, radon, ^{90}Y, ^{227}Th, ^{227}Ac, and ^{32}P (Bihl 2004, pg. 32)

[1] These parameters were obtained from review of several Hanford dose reconstruction IREP input sheets.

Table A-4: Comparison of Default Assumptions for Environmental Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Application	Environmental doses for personnel working at various INL facilities are calculated using facility-specific intake values and fence-line TLD direct gamma dose values (Tables 4-1 to 4-13, TBD).	Apply the annual internal and external environmental dose for each full or partial year of employment for the maximizing approach. Dose reconstructors are instructed to use only the maximum annual intakes in Table C-17 for the maximizing approach (Scalsky 2004, pg. 179). For the best-fit approach, modifications can be made for partial year of employment. No environmental dose is assigned if the background is not subtracted from the workers badge (Scalsky 2004, pg. 62).	Environmental doses are assigned to personnel with no bioassay and no evidence of having worn a dosimeter at the Hanford Site (Bihl 2004, pg. 48).
Sources of Environmental Releases Considered	The TBD focuses mainly on airborne gaseous effluent releases and direct beta/gamma radiation from INL facilities. All releases considered in the Idaho National Engineering Laboratory Historical Dose Evaluation (DOE 1991a) are the basis for the releases considered in the TBD (TBD, pg. 8). Eight facility areas have been chosen for TBD analysis: TAN, ICPP, TRA, RWMC, CFA, SPERT, ARA, and ANL-W (TBD, pg. 10).	The TBD heavily references the Cummins (1991) and CDC (2001) documents, and dose not include many of the base assumptions from those reports in the TBD. It is apparent that releases from the reactors and separations areas were considered.	T-plant particles and iodine, B-Plant particles and iodine, REDOX particles and iodine, PUREX particles and iodine, Z-Plant particles, reactor noble gases, and tritium from 108B Building (Savignac 2003, pg. 18).
Source Term Basis	*Idaho National Engineering Laboratory Historical Dose Evaluation Report (DOE 1991*[a]*). Identification and Priorization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory (RAC 2002). Version 6 of the RSAC code (Wenzel and Schrader 2001*) is used extensively in the current report to provide onsite concentrations due to episodic releases as well as other evaluations (TBD, pg. 8).	*Radioactive Releases from the Savannah River Plant 1954–1989* (Cummins 1991), *Savannah River Site Dose Reconstruction Project Phase II: Source Term Calculation and Ingestion Pathway Data Retrieval, Evaluation of Materials Released from the Savannah River Site* (CDC 2001), SRS meteorology data, SRS environmental reports for 1993–2001.	Hanford Works environmental reports; Methods for Estimating Radiation Doses from Short-Lived Gaseous Radionuclides and Radioactive Particles Released to the Atmosphere During Early Operations at Hanford (Till et al. 2002).

Table A-4: Comparison of Default Assumptions for Environmental Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Methodology	For routine operations, atmospheric dispersion of releases was modeled using annual average meteorological conditions. Dispersion of episodic releases was generally modeled using actual hourly wind speed and direction data at the time of release (BOE 1991). Progressive Gaussian	Gaussian model (Scalsky 2004, Section 3.1.1)	Puff advection (RATCHET) model (Savignac 2003, pg. 14)
Type of Releases	The TBD included two types of releases: (1) normal operational releases, and (2) episodic releases that generally are of short duration (e.g. criticality) (TBD, pg. 6). The TBD claims that these releases potentially represent unrecorded or missed does, either as direct gamma or beta-gamma from immersion in the radioactive gaseous cloud, for those individuals who do not have personal dosimetry to record the dose, or as internal doses from inhalation.	The TBD heavily references the Cummins (1991) and CDC (2001) documents, and dose not include many of the base assumptions from those reports in the TBD.	Calculations included routine and identified non-routine releases. Estimates include inhalation of radionuclides in air, direct external radiation from plumes, and physical contact with particulate radionuclides on skin.
Ventilation Rate (m3/year)	Not included in the TBD.	2,400 (default); Adjustments can be made for light and heavy work (Scalsky 2004, pg. 162).	2,400 (default); Based on 1.2 m^3/hour ± 0.4 m^3/hour (Savignac 2003, pg. 16)
Exposure Time (hours/week)	Not included in the TBD.	40 with a 1.25 conversion factor to increase the exposure time to 50 hours/week (Scalsky 2004, pg. 61).	40 (Savignac 2003 pg. 24)
Mobile Workforce	Information not included in the TBD.	Assign the maximum dose listed for any area onsite.	Information not included in the TBD.
Facility Specific Workforce	Intake values are listed for eight specific facilities including TAN, ICPP, TRA, RWMC, CFA, SPERT, ARA, and ANL-W (TBD, pg. 10).	Assign the maximum dose listed for any area onsite for the maximizing approach. Assign an area specific environmental dose based on the work location of the worker for the best-fit approach (Scalsky 2004, pg. 61).	Information not included in the TBD.
Radionuclides Considered for External Dose	No specific radionuclides considered in the TBD. Direct gamma values were taken from fence-line TLD reading (TBD Table 4-13).	^{41}Ar, (Scalsky 2004, pg. 60)	^{41}Ar, ^{131}I, ^{106}Ru (Savignac 2003, pp. 19 and 23)

Table A-4: Comparison of Default Assumptions for Environmental Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
Radionuclides Considered for Submersion Dose	Noble gas and halogen (TBD, pg. 24).	^{41}Ar, (Scalsky 2004, pg. 59)	^{41}Ar, page 17, ^{131}I, ^{3}H Kathy – can't find evidence that these last two belong here.
Submersion DCF	Not considered. TBD states that air immersion doses would be recorded in the fence-line TLD doses presented in Table 4-13.	Assumed values from the Federal Guidance Report 12 (EPA 1993). (Scalsky 2004, pg. 60)	Federal Guidance Report No. 13, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides*, 1999.
Radionuclides Considered for Internal Dose.	Facility Annual Intake (Bq/yr) due to normal operations provided for Ce-144, I-131, Pm-147, Pu-238, Pu-239/240, Ru-106, Sr-89, Sr-90, and Y-91 (TBD Tables 4-1 to 4-8). Specific Incident Intake (Bq/event) due to criticality provided for Rb-89, Sr-91, Sr-92, Y-92, Y-93, Te-133, I-131, I-133, I-134, I-135, Cs-138m Ba-139, La-141, La-142, and U-234 (TBD Table 4-10). Individual Test Intake (Bq/event) due to Special Tests provided for Sr-89, Sr-90, Y-91, Zr-95, Ru-103, Ru-106, I-131, Ce-144, and Pr-143 (TBD Table 4-11). Individual Test Intake (Bq/event) due to Initial Engine Tests provided for Rb-89,Sr-89, I-131, I-133, I-135, Cs-138, and U-234 (TBD Table 4-12).	^{3}H,^{131}I, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{234}U, ^{235}U, and ^{238}U (Scalsky 2004, pg. 51)	^{3}H,^{131}I-^{131m}Xe, ^{144}Ce-^{144}Pr, ^{137}Cs-^{137}Ba, ^{239}Pu, ^{103}Ru-^{103m}Rh, ^{106}Ru-^{106}Rh, ^{90}Sr-^{90}Y, ^{95}Zr-^{95}Nb (Savignac 2003, pg. 8)
Soil	Not included in the TBD.	Density = 1,600 kg/m3 Surface Factor = 0.08 Resuspension Factor =1E-9/m (Scalsky 2004, pg. 59)	Not included in the TBD.
Liquid Effluents	Not included in the TBD.	Not included in the TBD.	Not included in the TBD
Organ Dose Conversion Factor	Not included in the TBD.	1.0 is used in the maximizing approach. The organ dose conversion factors in the external dosimetry guide for the relevant exposure geometry are used in the best-fit analysis (Scalsky 2004, pg. 61).	Not included in the TBD.
IREP Rate	Not included in the TBD.	Chronic (Scalsky 2004, pg. 61)	Chronic[1]

Table A-4: Comparison of Default Assumptions for Environmental Exposure at INL, Savannah River Site, and Hanford

Description of Assumption	Idaho National Laboratory	SRS	Hanford
IREP Radiation Type	Not included in the TBD.	Photon, 30–250 keV ^{41}Ar , 100% photon, > 250 keV (Scalsky 2004, pp. 60 and 61)	Photon, 30–250 keV[1]
IREP Dose Distribution Type	|Not included in the TBD.	Constant. Doses and intake quantities provided with a 50th percentile and a geometric standard deviation. A 95th percentile for the source term is estimated as 25% greater than the 50th percentile (Scalsky 2004, pg. 60).	Constant. Doses and intake quantities provided with a geometric mean and standard deviation. There is no direction on how these values should be entered into IREP.
Special Considerations for Uranium and Plutonium	No special considerations for uranium and plutonium in the TBD.	The isotope yielding the maximum organ dose was assumed at 100% rather than applying a mixture (Scalsky 2004, pg. 59).	Not applicable.
Other	Dose reconstruction for individual whose location is unknown should use intakes provided by Table 4-3 (CFA) and exposures for ICPP as provided in Table 4-13. The values suggested will maximize the resultant individual dose.	1955 values are assigned to 1952, 1953, and 1954 (Scalsky 2004, pg 54)	The four chemical separations plants, T Plant, B Plant, REDOX Plant and the PUREX plant, along with the plutonium handling Z-plant are shown in Figure 4.1.1 to be the most important release points at Hanford (Savignac 2003).

[1] These parameters were obtained from review of several Hanford dose reconstruction IREP input sheets.

ATTACHMENT 5: ISSUE RESOLUTION MATRIX FOR FINDINGS AND KEY OBSERVATIONS

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
1	ORAUT-TKBS-0007-4	5	Issue 1: (5.1.1.1) Routine Airborne Releases - Source terms provided require improvement for use in determining the worker intake from airborne releases at different INL facilities. The data NIOSH uses do not take into account the deficiencies in the environmental monitoring equipment and their locations, and, in addition, NIOSH does not assess the uncertainties associated with the meteorological dispersion model used for the INL site. Most importantly, the source terms do not account for worker inhalation of resuspended contaminated soils and materials around the INL facilities.	45		
2	ORAUT-TKBS-0007-4	6	Issue 2: (5.1.1.2) Episodic Airborne Release - The airborne releases associated with several of the Initial Engine Tests of the Aircraft Nuclear Propulsion (ANP) Program were likely to have been underestimated by factors ranging from 2 to 7. Also, NIOSH did not evaluate the uncertainties associated with the deficiencies in air monitoring equipment.	55		
3	ORAUT-TKBS-0007-4	7	Issue 3: (5.1.1.3) Direct Gamma Exposures – The fence-line TLD measurements are not adequate for reconstructing direct gamma doses to personnel working outdoors at and around a specific INL facility inside the fence-line boundary, because they do not take into account the most bounding scenarios.	57		
4	ORAUT-TKBS-0007-5	8	Issue 4: (5.1.2.1) Completeness and Quality of INL Internal Dosimetry Programs - The identification and determination of missed internal	73		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			dose for workers are heavily influenced by the assumption of confidence, but SC&A found this premise to be unsupported after examining several critical DOE-HQ Tiger Team and DNFSB site audit reports. In addition, many site experts interviewed by SC&A indicated that there were significant deficiencies and inconsistencies in radiation work practices throughout the operating history of the INL facilities. These observations jeopardize the validity of the TBD approaches in reconstructing missed worker internal doses.			
5	ORAUT-TKBS-0007-5	9	Issue 5: (5.1.2.2) High-Risk Jobs (Internal Exposure) - NIOSH did not evaluate comprehensively the facility and field data to identify and separate out the high-risk or high-dose jobs for worker internal exposures. This information is essential for dose reconstructors to fill in the data gap when dose records in a claimant's file are not complete.	77		
6	ORAUT-TKBS-0007-5	O	Issue 6: (5.1.2.3) Calibration of Internal Dosimetry Analytical and Monitoring Equipment - The TBD does not provide any information on the calibration procedures, sensitivities, and standards of the internal dosimetry analytical equipment and monitoring instrumentation. The 1991 DOE Tiger Team findings show the deficiencies in these areas. NIOSH should evaluate the uncertainties and impacts on the internal dose assessment results associated with the deficient calibration programs at INL.	78		
7	ORAUT-TKBS-0007-5	O	Issue 7: (5.1.2.4) Changes of Internal Dose Limits - Inconsistent work practices were prevalent in the early years of the INL operation and may have led to significant missed dose to workers. NIOSH should evaluate the impacts of these dose limit	78		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			changes over the operating history of INL to see whether there were missed doses in the early years when the radiation protection policy was less protective and inconsistently implemented.			
8	ORAUT-TKBS-0007-5	10	Issue 8: (5.1.2.5) High-Fired Plutonium and Uranium Intakes - The TBD did not evaluate the hazard associated with high-fired plutonium and uranium at the INTEC (ICPP) and RWMC facilities. High-fired Pu-238, Pu-239, and uranium are not easily dissolvable, nor do they readily break into very small particles. They also emit some gamma rays and neutrons. Similar to the treatment of recycled uranium, NIOSH should evaluate the lung dose for intake of high-fired uranium and plutonium oxide particulates (alveolar deposition).	78		
9	ORAUT-TKBS-0007-5	O	Issue 9: (5.1.2.6) Skin and Facial Contamination - This TBD does not consider incidents with workers having skin contamination, facial contamination, and positive nasal swipes in the INL facilities. These kinds of problems would be compounded by the deficiencies in air sampling systems and ineffective respiratory protection programs. Guidance should be provided to a dose reconstructor to account for the missed dose due to the unaccounted uptake.	79		
10	ORAUT-TKBS-0007-5	O	Issue 10: (5.1.2.7) Breathing Rates - The TBD assumption appears less claimant favorable than the ICRP or NCRP assumptions.	79		
11	ORAUT-TKBS-0007-5	11	Issue 11: (5.1.2.8) Non-Occupational Worker Elimination of DU Background - The derivation of the background value of 0.16 µg/L used for subtraction from each urinalysis result of uranium prior to assessment of occupational internal dose for SMC radiation workers is not technically	79		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			sound. The baseline background (population) intake value was determined by a study of urine samples submitted by non-radiation workers at the SMC facility. A better approach would be to use the urine excretion samples by non-INL people in the Idaho Falls areas. NIOSH should consider this subtraction from urinalysis results as a missed internal dose.			
12	ORAUT-TKBS-0007-5	O	Issue 12: (5.1.2.9) Unmonitored Workers - The potential missed doses for unmonitored workers would be from inhaling resuspended contaminated soils and ingesting contaminated materials while eating in a contaminated, previously considered uncontaminated, area (such as office and cafeteria). NIOSH should evaluate these potential missed doses.	80		
13	ORAUT-TKBS-0007-4/5/6	O	Issue 13: (5.1.2.10) Naval Reactor Facility Workers - As the internal dose TBD indicates, "some workers' internal dose could have resulted from their support work at the NRF." NIOSH should evaluate the potential missed dose at the NRF for these workers.	80		
14	ORAUT-TKBS-0007-5	O	Issue 14: (5.1.2.11) Plutonium Monitoring - The TBD does not provide any historical information on the plutonium analysis methods used at INL. It is entirely possible that selective plutonium monitoring on workers was used at INL until 1980, but without this information, the dose reconstructors would not be able to assign missed internal dose due to plutonium intakes in the time period before 1980. NIOSH should provide information on plutonium monitoring.	80		
15	ORAUT-TKBS-0007-4/5/6	1	Issue 15: (5.1.3) SL-1 Accident Dose Reconstructions - The TBDs do not evaluate the potential missed internal and external doses or the	80		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			associated uncertainties for the over 1,000 rescue and cleanup workers involved with the SL-1 accident that occurred in January 1961. There was a high potential for significant exposures, because the equipment used and the radiological control policies in place in that era were not as advanced and protective as those in current use. The TBDs should develop adjustment factors related to stay time, dose field estimates, internal dose results, external dose readings, and contamination level estimates.			
16	ORAUT-TKBS-0007-6	8	Issue 16: (5.1.4.1.1) Completeness and Quality of INL Beta/Gamma Dosimetry and Record Keeping Programs - The identification and determination of missed external dose for workers are heavily influenced by this assumption of confidence, but SC&A found this premise to be unsupported after examining several critical DOE-HQ Tiger Team and DNFSB site audit reports. In addition, many site experts interviewed by SC&A indicated that there were significant deficiencies and inconsistencies in radiation work practices throughout the operating history of the INL facilities. These observations jeopardize the validity of the TBD approaches in reconstructing missed worker external doses.	96		
17	ORAUT-TKBS-0007-6	4	Issue 17: (5.1.4.1.2) Penetrating and Non-Penetrating Doses - NIOSH should re-evaluate the missed gamma dose, due to the deficiencies in the procedures and algorithms.	96		
18	ORAUT-TKBS-0007-6	O	Issue 18: (5.1.4.1.3) Correction For Beta Doses – NIOSH should develop a method to consistently account for uncertainties in dosimetry readings. Claimant-favorable correction factors should be developed for beta dose reconstruction.	97		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
19	ORAUT-TKBS-0007-6	O	Issue 19: (5.1.4.1.4) Angular Dependence Correction Factor for Gamma Dose - NIOSH should provide angular dependence (anatomic geometry) correction factors for external gamma doses, particularly for low-photon energies, where the angular dependence of the sensitivity of the dosimeter is most pronounced. These correction factors are used to account for, for example, the bias introduced by a dosimeter worn at the neck level and the higher doses received by tissues/organs below the waist.	99		
20	ORAUT-TKBS-0007-6	O	Issue 20: (5.1.4.1.5) Restating Beta Dose As Gamma Dose - It is not claimant favorable to state that the entire dose measured in the open window is due to the beta dose.	99		
21	ORAUT-TKBS-0007-6	O	Issue 21: (5.1.4.1.6) Photon Spectrum Split – NIOSH should provide guidance assigning dose values for the 30 keV<E<250 keV and E>250 keV regions.	99		
22	ORAUT-TKBS-0007-6	O	Issue 22: (5.1.4.1.7) Immersion Dose - The dose recorded on a dosimeter due to a semi-infinite cloud irradiation would be approximately half of the actual dose received. NIOSH should, therefore, consider a weighting factor of 2 for immersion dose.	100		
23	ORAUT-TKBS-0007-6	9	Issue 23: (5.1.4.1.8) High-Risk Jobs (Beta/Gamma Exposure) - Site experts interviewed by SC&A classified INL as an "acute dose" site, with a significant number of facilities, operations, experiments, and occurrences providing the possibility of personnel receiving dangerous levels of radiation. NIOSH did not evaluate comprehensively the facility and field data to identify and separate out the high-risk or high-dose jobs for worker external exposures. This	100		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			information is essential for dose reconstructors to fill in the data gap when dose records in a claimant's file are not complete.			
24	ORAUT-TKBS-0007-6	O	Issue 24: (5.1.4.1.9) Extremity Dose - NISOH should evaluate the potential for missed extremity dose for workers working in facilities where highly contaminated equipment, piping, instruments, valves, and systems resulted in exposures in confined spaces to hands.	100		
25	ORAUT-TKBS-0007-6	O	Issue 25: (5.1.4.1.10) Discrepancies between PIC and Film Reading – NIOSH should compare PIC versus film badge data (i.e., shallow and deep), and ensure that all the dose has been captured by the film badge. It is important to note that some PICs were worn for only the length of the job, so the discrepancy between readings of the two-dosimeter systems cannot be explained by drifting.	100		
26	ORAUT-TKBS-0007-6	O	Issue 26: (5.1.4.1.11) Minimum Detection Limit – NIOSH should re-evaluate the approach in determining the MDL of the dosimetry system by taking into account the system uncertainties.	101		
27	ORAUT-TKBS-0007-6	3	Issue 27: (5.1.4.1.12) Minimum Reporting Level (Beta/Gamma) - NIOSH does not provide adequate information supporting the use of chosen detection threshold levels to represent the MRL values for gamma film badges and TLDs. The use of MRL/2 as the missed external dose for dose reconstruction per OCAS-IG-001 is not claimant favorable for claims where the probability of causation value is close to 50%. In addition, NIOSH should re-evaluate the MRL values used and provide more supportable default values.	103		
28	ORAUT-TKBS-0007-6	3	Issue 28: (5.1.4.2.1) Minimum Reporting Level (Neutron) - NIOSH's approach for determining the MRL values for NTA emulsion film is not	108		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			thorough or supported. For example, NIOSH uses 10 neutron readings in one data sheet from March 1958 to determine the MRL values for the period between 1951 and 1957, and 6 neutron readings to represent all neutron measurements between 1959 and 1976. Furthermore, the use of MRL/2 as the missed external dose for dose reconstruction per OCAS-IG-001 is not claimant favorable for claims where the probability of causation value is close to 50%. In addition, NIOSH's MRL values of 14 mrem and 20 mrem appear low and are inconsistent with generic values given for NTA dosimeters, as well as values cited by other DOE facilities with similar neutron source terms and detectors. NIOSH should re-evaluate the MRL values used and provide more supportable default values.			
29	ORAUT-TKBS-0007-6	2	Issue 29: (5.1.4.2.2) Failure to Properly Address Neutron Exposures - INL had a total of 52 reactors, most of which were experimental/ prototype in design, which typically operated with high-power densities and with minimum shielding and neutron moderation. It is unjustified to presume that there are no missed neutron doses. In addition, there are deficiencies associated with neutron calibrations. Due to the use of the PoBe source for neutron calibration, dosimeters would significantly under-measure neutron doses from sources with lower-energy spectra. NIOSH should re-evaluate the entire approach in the TBD to account for potential missed neutron doses.	109		
30	ORAUT-TKBS-0007-6	2	Issue 30: (5.1.4.2.3) Neutron Calibration Deficiencies - Due to the use of the PoBe source for neutron calibration, dosimeters would significantly under-measure neutron doses from	110		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
			sources with lower energy spectra. NIOSH should re-evaluate the approach in the TBD to account for potential missed neutron doses.			
31	ORAUT-TKBS-0007-6	8	Issue 31: (5.1.4.2.4) Completeness and Quality of INL Neutron Dosimetry and Record Keeping Programs - The identification and determination of missed neutron dose for workers are heavily influenced by this assumption of confidence, but SC&A found this premise to be unsupported after examining several critical DOE-HQ Tiger Team and DNFSB site audit reports. In addition, many site experts interviewed by SC&A indicated that there were significant deficiencies and inconsistencies in radiation work practices throughout the operating history of the INL facilities. These observations jeopardize the validity of the TBD approaches in reconstructing missed worker neutron doses.	110		
32	ORAUT-TKBS-0007-6	O	Issue 32: (5.1.4.2.5) Uncertainty Estimation for Neutron Doses – NIOSH should explain how the FNCFs were obtained and provide instruction to dose reconstructors on how to apply them.	110		
33	ORAUT-TKBS-0007-6	O	Issue 33: (5.1.4.2.6) Neutron Organ Dose – NIOSH should provide neutron spectrum information and guidance for organ dose reconstruction for workers at ZPPR and TREAT.	110		
34	ORAUT-TKBS-0007-6	9	Issue 34: (5.1.4.2.7) High-Risk Jobs (Neutron Exposure) - NIOSH did not evaluate comprehensively the facility and field data to identify and separate out the high-risk or high-dose jobs for worker neutron exposures. This information is essential for dose reconstructors to fill in the data gap when dose records in a claimant’s file are not complete.	111		

Table A-5: Issue Resolution Matrix for INL Findings and Key Observations

Comment Number	TBD Number	Finding Number	Issue Number and Description	SC&A Page No.	NIOSH Response	Board Action
35	ORAUT-TKBS-0007-6	O	Issue 35: (5.1.4.2.8) Multiplying Factors for Missed Neutron Dose – NIOSH should provide data to support the two multiplying factors (1.25 and 2) and the fixed missed neutron dose default value of 50 mrem.	111		

Note: O-Observation