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

June 10, 2005

Mr. David Staudt
Center for Disease Control and Prevention
Acquisition and Assistance Field Branch
Post Office Box 18070
626 Cochrans Mill Road – B-140
Pittsburgh, PA 15236-0295

Re: Contract 200-2004-03805, Task 1, Supplement to SCA-TR-TASK1-0004:
Draft Review of the Hanford Site Profile

Dear Mr. Staudt:

S. Cohen & Associates (SC&A, Inc.) is pleased to submit our draft review of the Hanford Site profile. Please note that the review does not explicitly address the most recent NIOSH revision of the Occupational Medical Dose dated April 11, 2005. We will address this issue during our extended review cycle.

Sincerely,

John Mauro, PhD, CHP
Project Manager

cc: P. Ziemer, PhD, Board Chairperson
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Draft

ADVISORY BOARD ON

RADIATION AND WORKER HEALTH

National Institute of Occupational Safety and Health

Review of NIOSH Site Profile

for the

Hanford Site, Richland, Washington

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

SCA-TR-TASK1-0004

Prepared by

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June 2005

Disclaimer

S. COHEN & ASSOCIATES: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program***	Document No. SCA-TR-TASK1-0004
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Task Manager: _____ Date: _____ Joseph Fitzgerald	Supersedes: N/A
Project Manager: _____ Date: _____ John Mauro	

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ATTACHMENTS

ACRONYMS AND ABBREVIATIONS

ACL	Administrative Control Limit
AEC	Atomic Energy Commission
AEW	Atomic Weapon Employer
ALARA	As Low As Reasonably Achievable
ALI	Allowable Limit on Intake
Battelle	Battelle Northwest Laboratories
Bq	Becquerel
Bq-sec/m^3	Becquerel - Seconds Per Cubic Meter per Becquerel Released
BHI	Bechtel Hanford, Inc.
BNI	Bechtel National, Inc.
Board	Advisory Board on Radiation and Worker Health
CAM	Continuous Air Monitor
CDC	Centers for Disease Control and Prevention
CFR	*Code of Federal Regulations*
CEDE	Committed Effective Dose Equivalent
CHG	CH2M Hill Hanford Group, Inc.
Ci	Curie
CP	Cutie Pie
D&D	Decontamination and Decommissioning
DOE	Department of Energy
DOELAP	Department of Energy Laboratory Accreditation Program
DOE-RL	Department of Energy Richland Operations Office
DF	Decontamination Factor
Dpm	Disintegrations Per Minute
DR	Dose Reconstructor
DTPA	Diethylenetriaminepentaacetate
DU	Depleted Uranium
Du Pont	E.I. Du Pont De Nemours and Company
EALENR	Elevated Ambient Level of External Neutron Radiation
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ERC	Environmental Restoration Contractor

EU	Enriched Uranium
EV	Electron Volts
FFTF	Fast Flux Test Facility
FHI	Fluor Hanford, Inc.
FP	Fission Products
GM	Geiger-Mueller
GOK	God Only Knows
GSO	General Service Operator
HAPO	Hanford Atomic Products Operations
HEDR	Hanford Environmental Dose Reconstruction
HEU	Highly Enriched Uranium
HIE	Hanford Internal Exposure (database)
HLW	High-level Waste
HMPD	Hanford Multipurpose TLD
HP	Health Physics
HRRP	Hanford Radiological Record Program
HTLTR	High-Temperature Lattice Test Reactor
HVAM	High-Volume Air Monitor
ICRP	International Council of Radiation Protection
IDP	Internal Dosimetry Program
IMBA	Integrated Modules for Bioassay Analysis
INEEL	Idaho National Environmental Engineering Laboratory
IREP	Interactive RadioEpidemiological Program
IRF	Intake Retention Fraction
KeV	Kiloelectron Volts
LOD	Limit of Detection
LVAM	Low-Volume Air Monitor
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDL	Minimum Detectable Level
MeV	Million Electron Volts
Mint	Tritium
MPBB	Maximum Permissible Body Burden

MPC	Maximum Permissible Concentration
mR	Milliroentgen
mrad	Millirad
mrem	Millirem
mrep	Millirep
m/s	Meter Per Second
MW	Megawatts
nCi	Nanocurie
NCRP	National Committee on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NRC	Nuclear Regulatory Commission
NTA	Eastman Kodak Nuclear Track Film Type A
OBT	Organically Bound Tritium
OCAS	Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
P-10	Tritium
PA	Posterior-Anterior
PAM	Portable Alpha Monitor
pCi	Picocurie
PFG	Photofluorography
PFP	Plutonium Finishing Plant
PIC	Pocket Ionization Chamber
PNL	Pacific Northwest Laboratory
PNNL	Pacific Northwest National Laboratory
POC	Probability of Causation
Ppb	Parts Per Billion
PPE	Personnel Protective Equipment
Ppm	Parts Per Million
PUREX	Plutonium-Uranium Extraction
QAPP	Quality Assurance Program Plan
RAC	Risk Assessment Corporation
RadCon	Radiological Control

RAS	Retrospective Air Sampler
RCT	Radiological Control Technician
REDOX	Reduction-Oxidation
REX	Radiological Exposure (database)
RF	Resuspension Factor
Rockwell	Rockwell Hanford, a Division of Rockwell International
RPP	Radiation Protection Program
RPR	Radiation Problem Report
RU	Recycled Uranium
RWP	Radiation Work Permit
SC&A	S. Cohen and Associates
SRD	Self-Reading Dosimeter
SRS	Savannah River Site
SWP	Special Work Permit
TBD	Technical Basis Document
TEPC	Tissue Equivalent Proportional Counter
TIB	Technical Information Bulletin
TLD	Thermoluminescent Dosimeter
TLND	Thermoluminescent Neutron Dose
TRU	Transuranic
μCi	Microcurie
UNC	United Nuclear Corporation
UO_3	Uranium Trioxide
WESF	Waste Encapsulation and Storage Facility
WHC	Westinghouse Hanford Company

1.0 EXECUTIVE SUMMARY

S. Cohen and Associates (SC&A, Inc.) evaluated the following documents related to historical occupational exposures at the Hanford Site: ORAUT-TKBS-0006-1, *Technical Basis Document for the Hanford Site – Introduction* (Scalsky 2004ba); ORAUT-TKBS-0006-2, *Technical Basis Document for the Hanford Site – Site Description* (Selby 2004); ORAUT-TKBS-0006-3, *Technical Basis Document for the Hanford Site – Occupational Medical Dose* (Scalsky 2003); ORAUT-TKBS-0006-4, *Technical Basis Document for the Hanford Site –Occupational Environmental Dose* (Savignac 2003); ORAUT-TKBS-0006-5, *Technical Basis Document for the Hanford Site – Occupational Internal Dose* (Bihl 2004); and ORAUT-TKBS-0006-6, *Technical Basis Document for the Hanford Site – Occupational External Dosimetry* (Fix 2004). The evaluations focused on the completeness, technical accuracy, adequacy of data, and compliance with stated objectives, as stipulated in the *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004) approved by the Advisory Board on Radiation and Worker Health (Advisory Board) on March 18, 2004. (A fifth objective, "consistency among various site profiles," was limited to a comparison with the Savannah River Site Profile.) Although SC&A is aware that there was a recent Rev. 01 to ORAUT-TKBS-0006-4 in April 2005, this Rev. 01 has not been evaluated in this report.

In addition, SC&A evaluated and made use of technical information bulletins (TIBs) that relate to the Hanford Site Profile:

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

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

The National Institute for Occupational Safety and Health (NIOSH) Technical Basis Documents (TBDs), which together constitute the NIOSH site profiles for specific U.S. Department of Energy (DOE) and Atomic Weapons Employer sites, are designed to support the conduct of individual dose reconstructions under the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA). This is accomplished by compiling and analyzing data such as those related to facility operations and processes over time, radiological source term characterization, chemical and physical forms of the radionuclides, historic workplace conditions and practices, and incidents and accidents involving potential exposures. As the support contractor to Advisory Board, SC&A has been charged with independently evaluating the approach taken in NIOSH site profiles (encompassing TBDs and supporting TIBs) to gauge their adequacy, completeness, and validity. This information will be used by the Advisory Board to advise the Secretary of Health and Human Services on the scientific validity and quality of dose reconstruction efforts performed.

These TBDs are used by NIOSH, along with individual dose data provided by DOE and information gathered in interviews with claimants, to reconstruct doses for Hanford employees (including contractor and subcontractor employees). This review is designed to fulfill the objectives set by the Advisory Board for assessing the accuracy and adequacy of the Hanford Site Profile to serve as the main set of TBD documents that informs dose reconstruction for claimants. For instance, it provides the data on the limits of detection of radiation monitoring

methods as well as descriptions of facilities and processes that resulted in the worker exposures. The site profile also provides direction for assigning internal and external doses to monitored and unmonitored workers.

Hanford was and remains a complex operation involved in numerous missions, each of which has its own unique exposure hazards. Occupational risks of exposure to ionizing radiation are generally defined by Hanford's past and current missions:

(1) Production of nuclear weapons materials and nuclear energy research and development (1943–1990)

(2) Environmental restoration, waste management, nuclear material stabilization, and facility decontamination and decommissioning for permanent site closure (1990–2033)[1]

In the context of these missions, facilities of concern include:

- "Nine graphite-moderated, light-water cooled reactors were constructed near the Columbia River in the Hanford 100 Areas over a period of 20 years commencing in 1943 (Carlisle 1996). The production reactors were used to produce plutonium by irradiating metallic uranium fuel elements with neutrons during the fission reaction in the reactor core. Other defense-related radionuclides that were experimented with included: irradiation of thorium to produce ^{233}U, irradiation of depleted uranium to produce ^{240}Pu, irradiation of neptunium targets to produce ^{238}Pu, and irradiation of americium to produce medical grade ^{238}Pu."[2] Radiological hazards included external photon, beta, and neutron exposure from fission products and neutron radiation, and internal exposure to fission and activation products.[3]

- Seven physical testing, research, and demonstration reactors.

- Five chemical separation plants and associated fuel separation facilities, including the T and B plants, the REDOX plant, the PUREX plant, and U Plant, where radiological hazards included potential for internal and external exposure to a variety of radionuclides.[4]

- "Three facilities for fuel fabrication, i.e., the Uranium Metal Fuels Fabrication facility, the Uranium Metal Extrusion facility, and the Fuel Cladding facility. There were also two

[1] U.S. Department of Energy, *Performance Management Plan for the Accelerated Cleanup of the Hanford Site*, DOE/RL-2002-47, Rev. D., page ii.

[2] Selby, J, Technical Basis Document for Hanford Site – Site Description, ORAUT-TKBS-0006-2, Revision 00, PC-1, Oak Ridge Associated Universities, Oak Ridge, TN, December 29, 2004.

[3] U.S. Department of Energy, Hanford Site Waste Management Units Report, DOE/RL-80-30, Revision 12, January 2003.

[4] Chemical separation activities included: (1) Bismuth Phosphate (BiP04) Process (1944-1956), (2) REDOX Process (1952-67); (3) Solvent uranium extraction from waste tanks (1952-1958); (4) PUREX Process (1956-1972, 1983-1990); and (5) Radiocesium and radiostrontium solvent extraction from high-level tank wastes (1968-1985).

support facilities; the Uranium Storage and Oxide Burner facility and the Reactor Fuel Manufacturing Pilot Plant."[5]

- "Two plutonium finishing facilities, 231-Z (Plutonium Isolation Building) and 234-5Z (Plutonium 18 Finishing Plant Complex) operated at Hanford from 1945 to present. The latter is still involved in 19 plutonium stabilization efforts as a part of the Hanford cleanup program. Both of these complexes are located in the 200-W Area."[6]

- Twenty-one research, development, and testing facilities where a variety of exposures to radioisotopes occurred.

- Waste handling and storage facilities, one in each of the 200-W and 200-E areas, a trench facility, a settling tank area, an evaporator facility, chemical separations exhaust filtration facilities, and three liquid waste handling buildings, all providing a potential for external and internal exposure, as well as exposures via the environmental transport pathway.

- Some 2,710 waste disposal sites and burial grounds in the 100, 200, 300, and 1100 Areas, currently being characterized and remediated.[7] The preponderance of these sites poses radiation exposure risks.

- High-level radioactive waste (HLW) storage in 177 large underground tanks.[8] High-level radioactive tank waste stabilization and removal from underground tanks, scheduled for processing and disposal over the next 30 years, pose ongoing risks of exposure to radionuclides.

- An estimated 2,750 surplus facilities, many of which are contaminated with radionuclides, are either scheduled or are now undergoing deactivation, decontamination, and decommissioning.[9]

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

Based upon a review process, which included not only a review of the TBDs and supporting TIBs and documentation, but also interviews with the authors of the documents and site experts,

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[5] Scalsky, E. D., Technical Basis Document for the Hanford Site Introduction, ORAUT-TKBS-0006-1, Rev. 01, January 9, 2004, page 5.

[6] Selby, J., Technical Basis Document for Hanford Site –Site Description, ORAUT-TKBS-0006-2, Rev. 00, October 2, 2003, page 14.

[7] U.S. Department of Energy, Hanford Site Waste Management Units Report, DOE/RL-80-30, Revision 12, January 2003.

[8] Between 1944 and 1988, some 530 million gallons of high-level wastes containing more than 800 megacuries (uncorrected for decay) were generated at Hanford. High-level wastes stored at Hanford currently contain approximately 194 megacuries in 54 million gallons or 204,000 cubic meters.

[9] U.S. Department of Energy, Office of Environmental Management, Linking Legacies, Chapter Five, Surplus Facilities, http://legacystory.apps.em.doe.gov/text/link/link5.htm.

SC&A has identified a number of issues. These issues are sorted into the following categories, in accordance with SC&A's review procedures:

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

Following the introduction and a description of the criteria and methods employed to perform the review, the report discusses the strengths of the TBD, followed by a description of the major issues identified during our review. The issues were carefully reviewed with respect to the five review criteria. Several of the issues were designated as findings because they represent deficiencies in the TBDs that need to be corrected, and which have the potential to substantially impact at least some dose reconstructions.

1.1 SUMMARY OF STRENGTHS

For the purpose of reconstructing internal doses based on historical operations, NIOSH compiled an enormous amount of data describing the radioactive materials and operations at the various facilities and their associated processes.

Attachment D.3 of the internal dosimetry TBD includes a series of eight tables that provide guidance to dose reconstructors for assigning inhalation intakes of various radionuclides when the results of urinalysis are below the MDA. The intakes, in units of dpm/d, are normalized to an MDA of 1 dpm/d (based on a 24-hour urine sample). The tables also list cumulative intakes, in both dpm and pCi, based on exposure durations of 1 to 50 years. A second set of 11 tables provides similar guidance for whole body counters, normalized to an MDA of 1 nCi. These tables are helpful for dose reconstructions for claimants who worked at the Hanford Site, and are carried out to 50 years. According to the internal dosimetry TBD, plutonium urinalysis started in September 1946 (page 13), reliable uranium urinalysis started sometime in 1948 (page 24), and routine fission product urinalysis started in January 1947 (page 27). These urinalysis data were available in the late 1940s and generally provide a better means than air sampling data for the dose reconstructors to determine daily and cumulative intakes. However, some limitations of the data, discussed below, need to be factored in.

The use of the hypothetical intake described in ORAUT-OTIB-0002 (Rollins 2004) by NIOSH likely overestimates the dose to nonradiological workers and minimally exposed workers. For sites with reactors, such as Hanford, each claimant is assigned 28 radionuclides considered representative of potential sources of intake.

In compiling the atmospheric source terms for deriving outdoor occupational exposures, NIOSH made a concerted effort to compile the source term data needed to reconstruct the doses to unmonitored workers. This applies especially to the early period, prior to 1968.

1.2 SUMMARY OF FINDINGS

Finding 1: The NIOSH-derived neutron-to-photon dose ratios for use in pre-1972 neutron dose reconstruction are technically deficient and based on nonconservative assumptions, making them claimant unfavorable for use in dose reconstruction. For many Hanford workers, neutron exposure contributed a large fraction of the total dose derived from external radiation. In fact, when they are adjusted to account for the current International Commission on Radiological Protection (ICRP) neutron-weighting factor, neutron doses at the Hanford 200 and 300 Area plutonium facilities dominate the external dose. SC&A found various combinations of deficiencies that include: (1) the use of inappropriate data, (2) the use of incomplete or insufficient data, (3) the use of unconfirmed assumptions, and (4) the failure to account for critical variables, which limits the use of extrapolated data over time. It is also clear that historic neutron exposures to reactor workers in many areas are not adequately characterized.

Finding 2: The lack of bioassay data during the early period makes it difficult to properly quantify internal doses during that period. It is particularly a problem when dealing with the potentially high exposures that occurred during that time. Plutonium bioassay did not begin until September 1946; uranium bioassay did not begin until the first half of 1948. Fission product urinalysis data are unreliable until 1948. Uncertainties in the actual bioassay techniques and instruments used to quantify internal dose and the MDAs used in the years following 1946 need to be more thoroughly evaluated. Use of air monitoring data as a surrogate for worker intake during this early period is insufficiently substantiated, particularly given the lack of a basis for the assumed statistical distributions.

Finding 3: No guidance or direction for the dose reconstructor is provided regarding how adjustments are to be made or uncertainty factors calculated based on film badge and thermoluminescent dosimeter (TLD) error data provided in the TBD. In fact, no adjustments are recommended in recorded penetrating or gamma dose, with the exception of penetrating dose recorded for the two-element dosimeter used prior to 1957 for workers in the 200 Area. Likewise, adjustment factors are lacking for the large variety of exposure geometries experienced by workers at Hanford.

Finding 4: There is a significant potential for missed internal dose at Hanford that is insufficiently addressed in the TBD. Issues not adequately addressed include estimation of uncertainties for bioassay measurements prior to 1981, uncertainty corrections for whole-body counting prior to 1986 (and even default radionuclides until 1993), and potential contribution of radioactive contaminants in recycled uranium. The uncertainties in the case of plutonium in vivo counts are especially large. While the TBD recognizes the problem, the approach for dealing with them is not scientifically persuasive and does not appear to be consistently claimant favorable.

Finding 5: Modeling of occupational exposures due to Hanford environmental releases is not as claimant favorable as it should be, because the RACHET puff advection model is apparently not being applied to daily episodic airborne releases. Given that there were a number of relatively large short-term, ground-level, and elevated atmospheric releases at Hanford, it is important that these are modeled as hourly, not continuous annual releases, as indicated by Tables A-1 through A-21 of the TBD (Scalsky 2003). Lack of adequate parametric modeling of episodic releases also presents a significant potential for missed dose if releases are treated as continuous releases,

e.g., plutonium releases from the T and B reprocessing plants, ^{103}Ru and ^{106}Ru releases from the REDOX plant, and fission product releases as part of the Green Run and other operational release episodes.

Finding 6: The Tank Farm characterization in the TBD (Bihl 2004) is inadequate for dose reconstruction guidance in several respects. The list of radionuclides cited in the TBDs is incomplete, increasing the potential for missed dose. The site profile relies primarily on ORIGEN calculations to identify radionuclides that occur in large quantities and has not consulted field characterization data to verify the calculations (see Attachment 2 of this report). The TBD also does not reflect a complete description and characterization of past and current environmental restoration and waste management operations from which radiation exposure is likely to result.

Finding 7: Hanford was involved in both minor and major special campaigns, most notably those involving production of thorium and polonium. NIOSH needs to provide a detailed revision in the Hanford Occupational Internal and External Dosimetry TBDs to properly account for doses from the production of ^{233}U in the 100, 200 and 300 Areas, particularly in the 1960s to 1970s during peak production of ^{233}U. For workers exposed to thorium in the 1950s and 1960s, NIOSH needs to confirm such thorium exposures by urinalysis data for individual claimants, and dose reconstructors should carefully review potential doses in the 1960s and 1970s from irradiated thorium.

Finding 8: The TBD is incomplete with respect to remediation and disposal sites. Although NIOSH has included descriptions of key production and storage facilities, they have not addressed the numerous environmental waste streams and cribs that have been cleaned up in the past at Hanford disposal sites (e.g., ERDF). These areas pose radiological risks to those workers involved in the remediation and disposal process. Also, as these areas continue to age, the radionuclides of concern may be different from those in the original operations. Dose reconstructors need to take into account the risks associated with these areas at the Hanford site and the variability in radionuclide concentrations.

Finding 9: The method of locating, evaluating, and integrating incident data into the dose reconstruction is not clear in the Hanford TBDs. The Hanford occupational internal dose TDB (Bihl 2004) gives no specific information as to the spread of contamination in the reactor building, 231-Z Plutonium Isolation Facility, concentrator buildings, and uranium metal fabrication shops during the period 1943–1946. NIOSH should search for records that can provide additional information on doses resulting from accidents and incidents.

1.3 OPPORTUNITIES FOR IMPROVEMENT

Oro-Nasal Breathing: NIOSH should take into account oro-nasal breathing in the estimation of inhalation and ingestion doses. The dose conversion factors for light and heavy breathing should take account of the fact that many workers switch from nasal to oro-nasal breathing as the work becomes heavier. An upward adjustment to the percentage of heavy exercise and the consideration of oro-nasal breathing would ultimately increase the total uptake of radioactive material and be more claimant favorable given the uncertainties involved.

Addressing High-Risk Jobs: NIOSH should create a list of high-risk jobs and incidents for consideration as a complement to the site profile to inform the dose reconstructor during the individual dose reconstruction process and develop likely maxima and uncertainties for high-risk job categories.

Dose Calculation Example: NIOSH should provide in the TBD text an example of a hypothetical individual 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. The components of environmental dose should be consistent between DOE facilities. Hanford has provided a superior RATCHET puff advection model that is especially helpful in the evaluation of acute episodic releases.

Recycled Uranium: The dose contribution from trace radionuclides in recycled uranium 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 trace radionuclides in recycled uranium and develop bounding conditions that can be applied to DOE facilities, including Hanford.

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 investigate worker accounts. First-hand experience and association with the Hanford Site enable them to provide original perspectives and information concerning site practices and exposure histories. A limited amount of worker input has been incorporated into the latest versions of the TBD.

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.

2.0 SCOPE AND INTRODUCTION

> *The Hanford site encompasses an area of approximately 600 square miles. The site was chosen in January 1943 and construction of the facilities at Hanford commenced in March of that year, when several major nuclear facilities were started. These included 105-B (Production 3 Reactor); 221-T (Separation Facility); 305 (Test Pile); and 313 (Fuel Element Fabrication). The production reactors designed to produce plutonium by irradiating metallic uranium were constructed in the 100 Area located along the Columbia River on the north side of the Hanford site. The separation facilities were built in the 200 Area located on a high plateau in the center of the Hanford site. The fuel fabrication facilities, test reactors, and research and development laboratories were built in the 300 Area located on the south side of the Hanford site. The facilities represent more than 500 major facilities where nuclear activities were conducted.* [10]

By 1990, all defense production at Hanford was halted. Since 1990, the dominant risks of radiation exposure stem from fissile material stabilization, environmental restoration, and waste management activities. The U. S. Department of Energy's environmental mission at Hanford is scheduled for termination by 2033.[11]

Under the Energy Employees Occupational Illness Compensation Program Act of 2000 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* (42 CFR Part 82), the Advisory Board on Radiation and Worker Health 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, Inc.) has been charged under Task 1 to support the 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 documents related to historical occupational exposures at the Hanford Site: ORAUT-TKBS-0006-1, *Technical Basis Document for the Hanford Site – Introduction* (Scalsky 2004ba); ORAUT-TKBS-0006-2, *Technical Basis Document for the Hanford Site – Site Description* (Selby 2004), ORAUT-TKBS-0006-3, *Technical Basis Document for the Hanford Site – Occupational Medical Dose* (Scalsky 2003); ORAUT-TKBS-0006-4, *Technical Basis Document for the Hanford Site – Occupational Environmental Dose* (Savignac 2003); ORAUT-TKBS-0006-5, *Technical Basis Document for the Hanford Site – Occupational Internal Dose* (Bihl 2004); and ORAUT-TKBS-0006-6, *Technical Basis Document for the Hanford Site – Occupational External Dosimetry* (Fix 2004). Although SC&A is aware that there was a recent Rev. 01 to ORAUT-TKBS-0006-4 in April 2005, this Rev. 01 has not been evaluated in this report. SC&A, in support of the Advisory Board, has critically evaluated the Hanford site Technical Basis Documents (TBDs) in order to:

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[10] Selby, J., *Technical Basis Document for Hanford Site – Site Description*, ORAUT-TKBS-0006-2, Rev. 00, October 2, 2003, page 7.

[11] DOE/RL-2002-47, Rev. D.

- Determine the completeness of the information gathered by NIOSH in behalf of the site profile with a view to assessing its 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 that characterized the facility and its operations and the methods prescribed by NIOSH for its use of these data in dose reconstruction. The review was conducted in accordance with the objectives stated in *Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004). 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 influential in deriving dose reconstructions, bridging uncertainties, or correcting technical inaccuracies. This review does not explicitly address the issue of radiation exposures to cleanup workers and decommissioning workers, as that is not addressed in the TBDs.

The six TBDs serve as site-specific guidance documents used in support of dose reconstructions. These site profiles provide the health physicists who conduct dose reconstructions on behalf of NIOSH with consistent general information and specifications to support their individual dose reconstructions. This report was prepared by SC&A to provide the Advisory Board with an evaluation of whether and how the TBDs can support dose reconstruction decisions. 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 lens of whether dose reconstructions based on the TBD would provide for robust compensation decisions.

The basic principle of dose reconstruction is to characterize the radiation environments to which workers were exposed and determine the level of exposure the worker received in that environment through time. The hierarchy of data used for developing dose reconstruction methodologies is dosimeter readings and bioassay data, coworker data and workplace monitoring data, and process description information or source term data.

In accordance with directions provided by the Advisory Board and with site profile review procedures prepared by SC&A and approved by the Advisory Board, 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 Hanford Site Profile as a Basis for Dose Reconstruction

Based on the issues raised in each of these sections, SC&A prepared a list of findings, which are provided in the executive summary. 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 nine findings, but had **no** observations.

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.) Section 6.0 provides in summary form a list of the issues, and to which objective the particular issue applies.

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.

3.0 ASSESSMENT CRITERIA AND METHOD

S. Cohen and Associates (SC&A, Inc.) is charged with evaluating the approach set forth in the site profiles that is used in the individual dose reconstruction process. These 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 2004). This review is specific to the Hanford Site Profile and supporting technical information bulletins; however, items identified in this report may be applied to other facilities, especially facilities with similar source terms and exposure conditions. The review identifies a number of issues and discusses the degree to which the site profile fulfills the review objectives delineated in SC&A's site profile review procedure.

3.1 OBJECTIVES

SC&A reviewed the site profile with respect to the degree to which technically sound judgments or assumptions are employed. In addition, the review identifies assumptions by NIOSH that give the benefit of the doubt to the claimant.

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 applicable to the development of the site profile. The two elements examined under this objective include: (1) determining if the site profile made 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 data are available in site technical reports or other available site documents for particular processes, and if the TBDs have not taken into consideration these data where it should have, 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 of the data collected by NIOSH to the development of the site profile. Additionally, SC&A evaluated records publicly available relating to the Hanford Site and records provided by site experts.

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 which occurred at Hanford. The goal of this objective is to first analyze the data according to sound scientific principles, and then to evaluate this information in the context of compensation. If, for example, 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 Hanford TBDs were compared to the Savannah River Site (SRS) TBDs. The SRS Site Profile is appropriate for comparison as the sites had similar missions. 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 SC&A to evaluate 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 TBD 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 and the completeness of available dose data, as well as on the accuracy/uncertainty of data.

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. 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. 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, 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, highest 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 either to deny or to compensate. 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:

> *. . . 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.*** [Emphasis added.]

In order to achieve the five objectives described above, SC&A reviewed each of the six TBDs, their supplemental attachments, and TIBs, giving due consideration to the three categories of dose reconstructions that the site profile is intended to support. The six Hanford TBDs provides well-organized and user-friendly information for the dose reconstructor when adequate data were available to do that comprehensively.

ORAUT-TKBS-0006-1, *Technical Basis Document for the Hanford Site – Introduction* (Scalsky 2004ba), explains the purpose and the scope of the site profile. SC&A was attentive to this section because it explains the role of each TBD in support of the dose reconstruction process. 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. 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. 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.

ORAUT-TKBS-0006-2, *Technical Basis Document for the Hanford Site – Site Description*, (Selby 2004), is an extremely important document because it provides a description of the facilities, processes, and historical information that serve as the underpinning for subsequent Hanford TBDs. Specifically, this document describes 58 facilities and processes and their associated source terms that are relevant to dose reconstruction. SC&A's review of this section specifically addresses whether all the potentially important site activities and processes are described and whether characterization of source terms is complete/sufficient to support dose reconstruction.

ORAUT-TKBS-0006-3, *Technical Basis Document for the Hanford Site – Occupational Medical Dose*, (Scalsky 2003), provides a set of procedures for reconstructing the radiation exposures of workers from medical radiographic procedures that were required of employees at the Hanford site. SC&A reviewed this section for technical adequacy and consistency with other NIOSH procedures and the SRS site profile.

ORAUT-TKBS-0006-4, *Technical Basis Document for the Hanford Site – Occupational Environmental Dose*, (Savignac 2003), provides background information and guidance to dose reconstructors for reconstructing the doses to unmonitored workers outside of the facilities at the site 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.

ORAUT-TKBS-0006-5, *Technical Basis Document for the Hanford Site – Occupational Internal Dose*, (Bihl 2004), presents background information and guidance to dose reconstructors for deriving occupational internal doses to workers. This section was reviewed with respect to background 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 dose, and the methods recommended for characterizing uncertainty in the reconstructed internal doses.

ORAUT-TKBS-0006-6, Technical Basis Document for the Hanford Site – *Occupational External Dose*, (Fix 2004), presents background information and guidance to dose reconstructors for deriving occupational external doses to workers. This section was reviewed with respect to background 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. We also reviewed the recommendations for converting external dosimetry data to organ-specific doses, the methods recommended for use in the reconstruction of missed external doses, and the methods recommended for characterizing uncertainty in the reconstructed external doses.

In accordance with SC&A's site profile review procedures, SC&A performed an initial review of the six TBDs, their supporting documentation, and the two TIBs. SC&A then submitted questions to NIOSH with regard to assumptions and methodologies used in the site profile. These questions are provided in Attachment 1. A conference call was then conducted with the Chairman of the Advisory Board, staff members of NIOSH and ORAU, and the SC&A team to allow NIOSH to provide clarifications and to explain the approaches employed in the site profile. A summary of the conference call is provided in Attachment 2.

Site expert interviews were conducted to help SC&A obtain a comprehensive understanding of the radiation protection program, site operations, and environmental contamination. Attachments 3, 4, and 5 provide summaries of the interviews conducted by SC&A by teleconference or in person in the Hanford area during the course of this review. The site experts included current and former staff from radiation control, operations, environmental monitoring, maintenance, and other support organizations. These interviews were conducted during the course of the Hanford site profile review. Each summary is a paraphrase of conversations held with a number of site experts rather than a verbatim transcript. Their statements have been grouped into categories to provide a linkage with various portions of the Hanford Site profile. References to specific site experts have been omitted for privacy reasons. These 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.

An extensive comparison was done between the methodologies used in the Hanford and SRS TBDs to determine medical, occupational, environmental, internal, and external doses. This comparison focused on the methodologies and assumptions associated with dose reconstruction and resultant values used to obtain a POC. A detailed analysis is provided in Attachment 6. Although SC&A is aware that there was a recent Rev. 01 to ORAUT-TKBS-0006-4 in April 2005, this Rev. 01 has not been evaluated in this report. The data provided in Attachment 6, Table A.6.1 are based on the Rev. 00 (Scalsky, 2003)

Information provided in the conference call with NIOSH was evaluated against the preliminary findings to finalize the vertical issues[12] addressed in the audit report. 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 of these have been completed.

After the Advisory Board and NIOSH have an opportunity to review this draft, SC&A plans to request a meeting with Board members and NIOSH representatives to discuss the report.

[12] The term "vertical issues" refers to specific issues identified during our review, which were identified as requiring more in-depth analysis due to their potential to have a significant impact on dose reconstruction.

Following this meeting, we will revise this report and deliver the final version to the Board and to NIOSH. We anticipate that, in accord with the procedures followed during previous site profile reviews, the report will then be published on the NIOSH Web site and discussed at the next Board meeting. This last step in the review cycle completes SC&A's role in the review process, unless the 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 Board requests SC&A to review these documents.

Finally, it is important to note that SC&A's review of the six 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 claims can be processed in a timely manner. With this perspective in mind, we identified a number of strengths in the Hanford Site TBDs. These strengths are described in the following sections.

4.1 COMPLETENESS OF DATA SOURCES

The Hanford TBDs exhibited 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 the Hanford Site, NIOSH effectively compiled facility-specific information from facility descriptions in Ballinger 1991, Gerber 1992, and Marceau 2002. In addition, a review was performed of the Hanford Site internal dosimetry (Bihl 2004) and external dosimetry (Fix 2004) TBDs, as well as Carbaugh 2000, Carbaugh 2003, Fix 1997a, and Fix 1997b. Facilities were divided into 58 categories, and 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 facility. SC&A considers this to be an important strength of the report. In developing the site profile, NIOSH drew upon information contained in 229 reports cited in the reference sections of five of the six TBDs.

(2) In compiling the atmospheric source terms for deriving outdoor occupational exposures to unmonitored workers, NIOSH made a concerted effort to compile the source term data needed to reconstruct the doses to these categories of workers. ORAUT-OTIB-0007 (Strome 2003) identifies sites where elevated ambient levels of external radiation might be expected, and presents some examples of elevated mrem/hour readings in specific plants. This TIB specifically includes Hanford in its discussion. Notwithstanding this effort, there are opportunities for improvement in the methods used to reconstruct the doses, especially for workers in the early period, i.e., prior to 1968.

(3) For the purpose of developing data needed to reconstruct internal doses based on historical operations, NIOSH compiled a significant amount of data identifying the radioactive materials at the various 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 internal exposures.

4.2 TECHNICAL ACCURACY/CLAIMANT FAVORABILITY

The Hanford TBDs exhibited the following strengths in terms of their technical accuracy and claimant favorability:

(1) NIOSH made a concerted effort to determine the minimum detectable levels (MDLs) that were associated with the various types of dosimeters used over the life of the facility, for the different types of operations, and for the different types of external exposures (Fix 2004, Table 6-30, page 47). However, SC&A noted a discrepancy in Table 6-30. The MDLs listed in Table 6-30 are half those in Table 6-31. The MDLs listed in Table 6-31 should be those used in Table 6-30.

(2) ORAUT-OTIB-0002 (Rollins 2004) provides a method for processing claims "which involve cancer to an organ with little or no reported internal dose from internally deposited radionuclides." This method is recommended for evaluating "Category 2" claims, as described in Section 3.1.5 of the present report. Workers at sites with reactors, such as Hanford, who fall into these categories, are to be assigned intakes of 28 radionuclides that would result in body burdens equal to 10% of the maximum permissible body burden (MPBB) for each nuclide. The list comprises "the most plausible radionuclides for all the [DOE] sites, even though it is implausible that any one worker had intakes of all the radionuclides." Twelve nuclides are common to all DOE sites, and an additional 16 nuclides, consisting of fission and activation products, are specific to sites with reactors. These guidelines apply to workers at Hanford who were hired after 1952. They can also be applied to workers hired prior to 1953, if the dose reconstructor can establish that the doses calculated according to the TIB would overestimate the actual doses. The recommended methodology is claimant favorable for applicable cases, in that the doses would be overestimated if the directions for applying this methodology were followed by the dose reconstructor.

(3) Medical x-ray and photofluorography procedures have been investigated thoroughly to determine the radiographic techniques used at the Hanford Site. The Hanford occupational medical TBD (Scalsky 2003) provides methods for calculating doses from lateral chest x-rays in Section 3.3.2 (pp. 18–19), and from photofluorography calculation methods in Section 3.3.4 (p 14). Explicit consideration of photofluoroscopic examinations is especially important because of their potential for relatively large exposures, as compared to conventional radiographs recorded on photographic film.

(4) 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 Hanford TBDs is more user friendly than previous TBDs, such as those for SRS. The use of examples in the text, especially in the environmental TBD, makes the process more clearly understood. These examples are beneficial to the dose reconstructor as they summarize the process to be followed.

(5) NIOSH published a number of TIBs that provide further direction to the dose reconstructor. These documents were beneficial in understanding the application of the six Hanford TBDs to the dose reconstruction process.

(6) Hanford has utilized a superior puff advection model (RATCHET) as compared to the SRS conventional annual average Gaussian model to estimate dose from atmospheric releases to workers outdoors at the Hanford site, but it did not capitalize on its superior methodology to calculate doses resulting from daily episodic releases.

4.3 ADEQUACY OF DATA

The TBDs benefited from having access to information and data that were compiled as a part of the Hanford programs:

(1) Hanford's radiological control program has undergone continuous improvement since the inception of site operations. Radiological control personnel have implemented improved procedures and technology over time to reduce radiation dose to workers and have improved personnel monitoring programs.

(2) The radiation monitoring program utilized timekeeping, pocket ionization chambers (PICs), and air sampling to monitor worker exposure prior to the use of film badges and bioassay sampling. The two-element film badge was used as early as October 1944 (Fix 2004). The site continued to use timekeeping and PICs to monitor real-time personnel dose after the implementation of the film badge.

(3) Hanford initially used boron-lined PICs to measure slow neutron exposure. Starting in 1950, Hanford implemented a program using Eastman Kodak Nuclear Track Film Type A (NTA) film, primarily for plutonium areas (Fix 2004).

(4) The external dosimetry staff at Hanford has studied the uncertainties and technology shortfalls associated with the dosimeters and made necessary adjustments to the program. For example, the under-response of the original two-element dosimeter in plutonium fields was identified and adjustments were made to account for the under-response.

(5) Whole-body and chest counting were initiated in 1960 and 1967, respectively. Thyroid counting was used for high-risk workers as early as 1956. Hanford has also performed special counts, such as head, wound, liver, and lymph node. In vivo analysis started with a routine plutonium urinalysis program in 1946 (Bihl 2004). Throughout operations, additional in vivo techniques were added to the internal monitoring program as they became available and/or there was a need for them.

(6) Hanford implemented environmental monitoring early in its operational history.

(7) Hanford used multiple methods for documenting incidents and unusual occurrences.

Although Hanford has significant quantities of personnel monitoring data as well as field radiological control data, there are gaps in the information. There are problems with the adequacy of data particularly with regard to neutron dosimetry, early period doses, internal dosimetry, and doses to workers at the Tank Farms and waste disposal sites.

4.4 CONSISTENCY AMONG THE SITE PROFILES

Although Hanford and SRS had similar missions, there are some differences in the facility processes, design of facilities, and radiological practices. In some cases, these differences require site-specific assumptions in dose determinations. For example, due to the design of the REDOX facility at Hanford, there were substantial particulate releases of ruthenium that had to be considered. In the case of SRS, the facilities were built later, and this was not an issue. 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 Hanford and SRS TBDs are consistent in many cases, although there is room for improvement in some areas. This consistency was especially apparent in the medical occupational exposure sections, as seen in Table A.6-1 in Attachment 6.

4.5 REGULATORY COMPLIANCE

The TBD's use of personnel monitoring data and environmental monitoring data to determine dose is consistent with the requirements outlined in 42 CFR Part 82, as follows:

- Where in vivo and in vitro analyses are available, this information is provided for use in determination of internal dose.
- Where routine beta/gamma and neutron dosimeters are available and adequate, this information is provided for use in determination of external exposure.
- Where environmental measurements are available, these data are used as the basis for environmental dose.

NIOSH has effectively complied with the hierarchy of data required under 42 CFR Part 82 and its implementation guides for monitored workers.

5.0 VERTICAL ISSUES

SC&A has developed a list of key issues regarding the Hanford Site Profile. These issues relate to each of the five objectives defined in SC&A 2004. Some issues are related to a particular objective, while others cover several objectives. A matrix relating the particular objectives and the relative importance of each issue is provided in Section 6.0 of the present report. Many of the issues raised below 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 ISSUE 1: NEUTRON DOSIMETRY AND EXPOSURE

Neutron dosimetry is considerably more complex and difficult to assess than beta/photon dosimetry. The principal difficulties in assessing neutron doses relate to the technical capabilities of past dosimeters used at Hanford. Initially, Hanford used boron-lined pocket ionization chambers to measure slow neutron exposure. NTA film was introduced at Hanford in 1950 to measure neutron exposures, and was used for neutron monitoring until 1971. Starting in 1972, Hanford introduced the Hanford Multipurpose TLD (HMPD), which was designed to assess exposure to (1) photons, (2) high-energy betas, and (3) thermal and fast neutrons. The HMPD was first accredited by the DOE Laboratory Accreditation Program (DOELAP) for neutron categories in 1989 and reaccredited thereafter on a biannual basis.

NTA Neutron Monitoring: Several studies that evaluated and compared NTA film with the HMPD and a tissue-equivalent proportional counter (TEPC) demonstrated limitations of NTA film for use in monitoring neutron doses in the workplace. "Fundamentally, the NTA dosimeter is capable of an accurate dose estimate for higher energy neutron radiation greater than about 1 MeV because the NTA has a lower energy threshold of about 700 keV" (Fix 2004, page 19). Due to this limitation, the HMPD was substituted for NTA film for monitoring neutron doses starting in 1972. For the period prior to 1971, NIOSH concluded that, for dose reconstruction, the under-response of the NTA film could be accounted for by facility-specific neutron-to-photon exposure. NIOSH considered the application of facility-specific neutron-to-photon ratios to be a suitable surrogate for neutron dosimetry data.

To arrive at neutron-to-photon ratios for pre-1972 Hanford facility operations, the TBD identified three discreet areas with potential neutron exposures. The method for deriving a neutron-to-photon ratio for each of the three operating areas was based on very different measurements and assumptions. Presented below is a summary of the methods employed by NIOSH for assigning neutron-to-photon ratios and an evaluation of their scientific validity and claimant favorability for use in neutron dose reconstruction.

5.1.1 Neutron Exposure to Reactor Workers Not Adequately Characterized

The TBDs do not sufficiently address neutron dosimetry issues at the reactors. For example, a retrospective assessment of personnel neutron dosimetry for workers at the Hanford site concluded that (Fix et al. 1977a):

> *Based on current knowledge, there is under-recorded neutron dose, primarily prior to the use of the Hanford TLD in 1972, and specifically for the relatively few*

> *Hanford workers involved with plutonium-handling operations. This occurred because of significant neutron radiation in the workplace less than the energy threshold of about 1 MeV (based on a minimum of 4 grains to be counted as a track) for the Hanford NTA film dosimetry.*

In the same retrospective study, it was also concluded that:

> *The neutron dose was under-recorded during January 1980 through January 1984 when the four-element Hanford TLD was used. A study conducted at that time showed an under-recorded whole body dose of about 25%.*

In addition, guidance for reconstruction of neutron exposures in Revision 01 of the Hanford Site Profile, relative to Hanford's reactors, is based largely on the same retrospective assessment noted above. Fix et al. (1997a, page 7) concluded that "relatively little personnel neutron dose is expected from Hanford production reactor facilities. Personnel neutron dose was limited by large quantities of shielding, high temperatures near the reactor core, and the general practice to limit personnel access to areas during reactor operation."[13]

A preliminary review of historical records suggests that this assumption may have overlooked reactor shielding problems that persisted at Hanford for many years. Interviews with some of the Hanford site experts, summarized in Attachment 5, provide some interesting perspectives on early reactor operations.

Purpose of Reactor Shielding at Hanford

Reactor shielding at the Hanford reactor serves the following purposes:

(1) Protecting workers from harmful exposure to photons and neutrons
(2) Protecting monitoring and control instrumentation from radiation damage
(3) Preventing neutron activation and damage to reactor components, such as piping
(4) Protecting workers and reactor components from the high heat generated by the reactor

In general, Hanford's reactors were shielded by the following components:

- **Reflector –** Its primary purpose is to reflect, or turn back, most of the neutrons escaping from the reactor core.
- **Thermal Shield –** The thermal shield is a dense material usually placed next to the reflector to reduce heat and radiation that escape from the reactor core and reflector.
- **Biological Shield –** The biological shield is made of thick layers of materials located outside of the thermal shields that are designed to absorb neutrons and photons escaping from the core, the reflector, and the thermal shields, as well as providing support for the thermal shields and other reactor components. Reactor biological shields "are enveloped

[13] Fix, J. J.,, R.H. Wilson, W.V. Baumgartner, *Retrospective Assessment of Personnel Dosimetry for Workers at the Hanford Site*, Pacific North West National Laboratory, PNNL 11196, February 1997. page 7.2.

by a graphite reflector, a cast iron thermal shield and a biological shield consisting of either laminations of iron and Masonite or of high density concrete."[14]

Reactor shielding was supposed to provide adequate protection to personnel working in and near the following:

- Front or inlet face of the reactor
- Laboratory area (experimental levels)
- Rear or outlet face
- Inner and outer rod rooms
- Top of the reactor
- Bottom of the reactor

Worker Exposure Issues and Problems at the Reactors

Unlike modern nuclear power plants, Hanford's reactors required a significant amount of "hands-on work involving thousands of entryways through the biological shield to permit charge and discharge of the fuel elements,"[15] as well as maintenance, repair and experimental activities on all sides of the reactor. Attachment 7 contains engineering drawings that provide some details regarding shielding design and the extent of penetrations in the bioshield to allow for hands-on worker operations and maintenance. As reactor power and heat levels were increased beyond original design capacities, problems such as graphite distortion, structural stress of the reflectors, and deterioration of biological shielding material led to potentially excessive exposures of workers to neutrons and photons. Leakage of photons and neutrons from fuel tube penetrations and other inlets was chronic. These problems, in turn, led to an increase in radiation exposures of workers from operations, maintenance and repair, and experiments to address ongoing problems. Attachment 7 also provides a partial chronology of major reactor upgrades and repairs.

The following are some examples of these problems:

- **1949 –** Expansion of graphite at the B, D, and F reactors had proceeded to the extent that the supporting beams for the shielding and the matching interlocks at pile corner were displaced. In the D and F Areas, pile shielding at the corner beneath the top and far side had separated to the extent that there was a gap through the shielding at this point. Such a gap through the pile shielding allowed a high-intensity beam of radiation to leak out, rendering the adjacent area unsafe:

  > *…The escape of stray neutron beams from the process unit makes it difficult for the Health Instruments technicians to prescribe danger areas accurately. The beams are usually narrow but extend for long distances and are highly dangerous to operating personnel* (GE 1949).

[14] Peterson E. S., and W.L. Bunch, *The Effect of Fringe Poison on Heat Generation in the Shield Complex*, March 19, 1962, HW-73064.

[15] Bunch, W., *Shielding Effectiveness of the Tapor Bore Process Tube Entry*, February 12, 1958, HW-54955. page 2.

- **1952 –** "Several persons received neutron/gamma exposures in an unmonitored zone at 105 building… A significant radiation field existed in the zero level far side region of the pile building from scatter of a radiation leak at B-hole on the experimental level above… Neutron badges and pencils were not worn, however."[16]

- **1955 –** It was reported that high temperatures were causing deterioration of the iron-Masonite bioshields, which increased radiation leakage rates beyond that expected for increased power levels.[17]

- **1955 –** "In the original reactors [B, D, and F], it was necessary to evacuate personnel from the offices underlying the rod rooms when a rod is removed. Dose rates up to 1 R/hr … may be encountered at 10 feet from a horizontal rod, twelve hours after reactor shutdown, when it is removed for burial."[18] While this event deals with exposure to photons, it highlights an inadequacy in shielding design.

- **1956 –** Fast neutron and gamma dose rates during a ball valve discharge operation at the C reactor were found to be as high as 13 rem/hour and 5.8 rem/hour respectively. "The high dose rates indicate the need for additional rear face shielding or an evacuation of all affected areas prior to extensive use of this discharge method."[19]

- **1956 –** "Higher shield temperatures, which are expected to result from increased power levels, fringe enrichment, and higher graphite temperatures, will markedly increase the rate at which the Masonite will burn out. The laminated iron Masonite biological shield will lose, as a result of burnout, the hydrogen and oxygen necessary to attenuate and moderate neutrons. It appears likely that radiation leakage will severely limit access to the experimental levels and top of the unit of the B, D, F, DR, and H piles within one or two years of operation, unless preventative action is taken."[20]

- **1958 –** "Distortion of the graphite stacks continues to be a problem of real concern at all reactor s… During the early [years] operation of the older reactors at low temperatures distorted the entire graphite stack… Graphite distortion results in distortion or damage to reactor components, such as bending of the process tube, flattening of the tube gun barrel and fracturing of the graphite blocks at the entry of the gun barrel … B, D, and F reactors suffer considerable 'crater' distortion of the stack."[21]

- **1958 –** During an outage, the "lower left hand window [in the C reactor observation room] broke, allowing the zinc bromide solution to flood out, emptying the window and leaving an unshielded radiation path through the reactor shield wall… Should the break

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[16] *Radiological Sciences Department Investigation*, Class I, No.205, April 15, 1952, HW-24270.

[17] L.A. Wilson, *Hanford Shield Masonite Deterioration Studies*, August 8, 1955, HW-35202.

[18] Memorandum, Shielding Effectiveness at the KE Reactor, November 7, 1955, HW-39286.

[19] Letter to R.L. Junkins From Carl M. Unruh, *Reactor Building Dose Rates Associated with Ball Valve Operation*, July 19, 1956, HW-43814.

[20] W.L. Bunch, *The Effect of Masonite Burnout on Shield Attenuation Properties*, May 23, 1956, HW-38418.

[21] D.H. Curtiss, G.C. Fulmer and W.D. Gilbert, *Reactor Components Study*, H.W.-57222, August 25, 1958.

occur while the reactor was operating and while a person was in the observation room, the person would be exposed to intense radiation." The cause of the window breakage was not exactly determined but was assumed to be due to mechanical or thermal shock.[22]

- **1958–1959 –** "It was discovered after three years of operation of the K-reactors that top center was settling, control rod and process tube channels were becoming distorted, the side reflectors were moving out toward the thermal shields at the bottom, and gross voids were forming inside the reflector."[23]

- **1959 –** After an unexpected reactivity event at the F reactor, "some evidence was detected of the smoldering of the Masonite in the [biological] shield."[24]

- **1959 –** Unmeasured neutron exposure to reactor workers was first indicated after initial deployment of the Hanford whole body counter detected ^{24}Na, an activation product, in reactor area employees.[25]

- **1960 –** Neutron flux leakage rates through the iron-Masonite bioshields for Hanford's two original reactors were estimated to have increased by 57% in 2 years at the F reactor and 35% in 2.5 years at the H reactor. It was noted that dose rates from neutrons leaking from vertical safety rod openings at the top of the H reactor averaged 250 mrem/h. Neutron leakage measurements taken from the H reactor far-side shield indicated that the dose from fast neutrons was 6 times greater than thermal and "intermediate" neutrons.[26]

- **1960 –** "Dose rates up to about 30 mrem/hr from neutrons plus about 25 mrem from gamma now exist on the charge elevator. . . in the front face of the B reactor."[27]

- **1960 –** Hanford's Whole Body Counter detected ^{24}Na in 18 out of 59 reactor area employees (31%). Initial detection of ^{24}Na in four reactor employees ranged from 2 to 11 nanocuries. "Because of the relatively short half-life of ^{24}Na (15 hours), it is generally observed in greater quantities in subjects examined during the afternoon who have come to the Whole Body Counter directly from work in the reactor areas. ^{24}Na has been observed only in reactor area employees. Fourteen of the 59 were assigned to reactor areas farthest upstream and, therefore, were not regularly exposed to drinking water supplies which had been used as reactor coolant."[28]

- **1961 –** Around 1961**,** the operating dose rates on the charge elevators at the older reactors became a problem. (Supplementary control techniques in support of more efficient

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[22] B.B. Brendan, *105-C Shielding Window Breakage*, April 16, 1958, HW-55770.

[23] W.K. Alexander, *Reactor Moderator Distortion Study*, May 15, 1964, DUN-3424 RD.

[24] R.E. Kleinknecht, *Effect of High Temperature on Masonite*, April 14, 1960, HW-64868.

[25] Staff Members of Internal Dosimetry, Hanford Whole Body Counter Activities for 1961-1963, HW-82409, May 1964, p.18.

[26] W.L. Smalley, *Attenuation Effectiveness of the Hanford Reactor Iron Masonite Shield*, April 27, 1960, HW-64951.

[27] E. Peterson and W.L. Smalley, *Proposed Procedure for Reducing Reactor Front Face Dose Rates*, H.W. 66117, July 19, 1960.

[28] F. Swanberg Jr., Hanford Whole Counter Results for 1960, HW-67245. April 1961, pp. 19-20.

operation required considerable front elevator work during operation.) The dose rates increased due to increased radiation leakage through inlet shields, which in turn were caused by higher power levels, by fuel charges placed further upstream, by removal of the front nozzle studies, and by a combination of these factors.[29]

- **1961-1963 –** "Na-24 has been detected primarily in reactor area workers. Correlation of results to environmental parameters such as place of residence and work area was precluded by the relatively short half-life of the radionuclide…and by the fact that many of the employees examined had been away from their location long enough for a significant fraction of Na-24 to have decayed…. **Results for 1961-1963 indicate that 47% of reactor area employees examined had measurable body burdens [of sodium-24] ranging up to 3.8 nanocuries**."[30] [Emphasis added.] These measured values were approximately 10%–14% lower due to decay, because the estimated time it took a worker to arrive at the counting facility, sign in, and be measured was ~2 hours.[31]

- **1962 –** The two most highly exposed workers, from "slow" neutrons due to a criticality accident on April 7, 1962, at a plutonium waste recovery operation (RECUPLEX), were found, after whole-body counting, to have ^{24}Na levels lower than those measured in reactor workers in 1960. The two workers were found to have body burdens of 7.55 nCi and 4.2 nCi, respectively. Estimated doses were 3 rads from neutrons and 13 roentgens for the first worker, and 0.34 rads from neutrons and 1 roentgen for the second.[32] (In 1985, revised dose estimates indicate that three workers received approximately 110 rem, 43 rem and 19 rem respectively from the RECUPLEX criticality accident.)[33]

- **1965 –** Within a five-year period, radiation leakage rates at the B, D, F, and H reactors would likely reach 1.0 R/h or more, if their biological shields were allowed to reach and sustain 160° C.[34]

- **1967 – "**Extrapolated leakage rates . . .show that B and D leakage rates have improved whereas leakage rates for F and H reactors have increased in magnitude since the most recent previous report [April 15, 1965]… This is believed to be the result of having charged thoria in the fringe of B and D reactors which reduced the shield temperatures and so reduced the leakage rates, while at F reactor the shields were allowed to operate at slightly higher temperatures…"[35]

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[29] W.L. Smalley, R.L. Loundagin, *Inlet Shield Radiation Penetration Studies*, April 27, 1961, HW-69348.

[30] HW-82409, page 17.

[31] D.N. Brady, F.N. Swanberg, *The Hanford Mobile Whole Body Counter*, HW-SA-3559, June 1, 1964, page 6.

[32] Dosimetry Investigation of the RECUPLEX Criticality Accident, November 8, 1962, HW-75546. pp. 20,30.

[33] *Hanford Exposures Over Limit, Whole Body Penetrating Exposures at Hanford, and Non Operational Exposures at Hanford*, compiled January 1985 for period 1944-1984, Hanford Declassified Document Retrieval System (DDRS).

[34] P.D. Gross, *Hanford Shield Masonite Attenuation Effectiveness*, April 15, 1965, RL-REA-1031.

[35] D.E. Clark, *Iron-Masonite Shield Status Report No. 6*, June 30, 1967, DUN-2674.

To compound the problems at the reactors, as discussed above, reactor workers were not routinely monitored for neutron exposure (see Attachment 5). Health Instrument Division reports from the reactors indicate that numerous measurements were taken in the reactor areas indicating the potential for neutron exposure.

5.1.2 Adequacy of Data

Nuclear power operators indicated in interviews that although they were actively involved in operations potentially involving neutron exposure, they were not provided with neutron dosimetry. This is true of operators even into the 1970s and 1980s. Early reactor operation personnel indicated that some individuals were monitored and others were not. Even during the period of monitoring, dosimeters were not assigned on a continuous basis. An evaluation of the comprehensiveness of the neutron monitoring program and the potential neutron exposure should be completed to determine whether existing site profile methodologies bound neutron doses. This analysis should extend to other areas of Hanford where there was a potential for neutron exposure, such as "tail-end" portions of the separations process, work with plutonium-contaminated tanks and burial sites, and research and development facilities involved with the use of plutonium.

5.1.3 Completeness of Data

The TBD identifies the following facilities as having a potential for neutron exposure (Fix 2004, Section 6.3.4.5, pp. 31-32):

- 100 Area reactors (105-B, 105-C, 105-D, 105-DR, 105-F, 105-H, 105-KE, 105-KW, and 106-N reactors)
- Plutonium finishing process (224, 231-Z, 232-Z, 234-5Z, 236-Z, 242-Z, and 2736-Z)
- 300 Area facilities (308, 309, 324, 3745A, and 3745B)
- Fast Flux Test Facility (FFTF)

Table 6-22 of the TBD provides the neutron-to-photon ratios to be used by the dose reconstructors for calculating neutron doses (Fix 2004, page 42). Building 271 is listed in Table 6-22 as having a neutron-to-photon ratio; however, it is not listed under Section 6.3.4.5 as a facility with the potential for neutron exposure. Conversely, Buildings 224, 232-Z, 236-Z, 242-Z, 3745A, and 3745B and the FFTF are listed in Section 6.3.4.5 as facilities having a potential for neutron exposure; however, no neutron-to-photon ratios are provided in Table 6-22. This can be a source of confusion for the dose reconstructor.

The TBD is silent on other facilities that may have potential neutron exposure. The plutonium nitrate feed material was transferred from the "tail-end" of the canyon buildings to the Z-plant for plutonium finishing. With the existence of a large quantity of plutonium nitrate at the "tail-end" of the canyon, there may be a potential for neutron exposure. The 233-S Plutonium Reclamation Facility also handled significant quantities of plutonium solutions. Also of concern are the waste tanks and burial ground that supported these facilities. Further investigation of the potential for neutron exposure and the completeness of monitoring in these areas should be undertaken.

5.1.4 Neutron-to-Photon Ratios have Facility-Specific Deficiencies

All Hanford single-pass reactors (i.e., B, D, F, H, DR, C, KW, and KE reactors) were shut down prior to the introduction of the HMPD in 1972; for this reason, available measurements for potential use in estimating neutron-to-photon dose ratios were photon doses measured by conventional film dosimeters and neutron doses calculated from NTA film dosimeters assigned to seven workers between 1950 and 1961. These workers were primarily assigned to the Hanford reactors. Neutron doses for these seven workers were analyzed by five different methods that were ranked from maximally conservative to most technically accurate. For example, Method 1 used a gross track count (i.e., no background subtraction), which resulted in the highest neutron-to-photon ratios and was therefore considered maximally conservative. Least conservative (but regarded by NIOSH to be technically most accurate) was Method 5, which employed "background subtraction based on the standard deviation of the number of neutron tracks on the blank films." Table 1 summarizes the neutron-to-photon ratios for the seven workers based on the five methods employed.

Table 1. Neutron-to-Photon Ratios for Seven Hanford Reactor Workers
(Source: Fix 2004, Table 6-20)

Worker #	Method 1	Method 2	Method 3	Method 4	Method 5
2	0.39	0.14	0.11	0.06	0.06
3	0.14	0.06	0.05	0.03	0.03
6	0.64	0.34	0.31	0.24	0.24
7	0.25	0.10	0.08	0.04	0.04
11	0.13	0.04	0.03	0.01	0.02
12	0.75	0.29	0.21	0.11	0.12
14	0.72	0.26	0.20	0.11	0.12
0	0.431	0.176	0.141	0.086	0.090

NIOSH concluded the following:

> *. . . since it is not known which method is the most accurate, all of the data in Table[1] was used in this analysis. The data closely fit a lognormal distribution . . . with a geometric mean of **0.11,** a geometric standard deviation of **2.79**, and an upper 95th percentile of **0.62**.* [Emphasis added.]

These calculated neutron-to-photon ratios had to be corrected for the reduced sensitivity of the NTA film to neutrons with energies < 1MeV. As stated in the TBD:

> *Neutron spectra measurements of the single pass Hanford reactors **during operation** have **not** been located.* [Thus] *The under response of the NTA film due to neutron energy can only be estimated. Nichols et al. (1972) showed that for the single 100 KE* [reactor] *. . . positive measured NTA dose measurement, the NTA film recorded 28% of the **true** neutron dose as measured by the TEPC* [tissue equivalent proportional counter]*. . . To estimate the neutron-to-photon ratio, a factor of 28% was used to adjust the neutron- to-photon ratios from Table* [1] *to account for the unmeasured neutron dose. This resulted in a lognormal*

> *distribution . . . with a* [neutron-to-photon ratio having a] *geometric mean of 0.41, a geometric standard deviation of 2.79, and an upper 95th percentile of 2.22.* [Emphasis added.]

These values are cited in Tables 6-22 and 6E-5 of the TBD (Fix 2004) as the recommended values for deriving neutron doses to workers at the B, D, F, H, DR, C, KW, and KE reactors prior to 1972.

5.1.4.1 SC&A's Assessment of Ratio Values for 100 Area Reactors

The principal technical limitations for the above-described approach include the following:

- **The Use of NTA Film Data** – In Section 6.3.4.6 of the TBD (Fix 2004), NIOSH readily acknowledged the limitation of NTA film in monitoring workers for neutron exposure by stating:

  > *Considering the uncertainty in the neutron recorded dose at Hanford reactor and plutonium facilities prior to implementation of the HMPD in 1972 and the recommendations of the 1972 AEC review . . . of pre-1972 NTA neutron dose results in Hanford plutonium facilities, the recommended method to estimate dose to workers from neutron radiation is using a claimant-favorable neutron to photon ratio*.

  Thus, while DOE recognized the limitations of NTA dosimeters when used for personnel monitoring, NIOSH accepted the use of NTA film data for defining the neutron-to-photon ratio, which is to serve as the surrogate for neutron dosimetry data for monitored personnel. As described below, the 28% correction factor does not appear to solve the problem.

- **Use of All Five Methods of Evaluating Neutron Dose** – NIOSH described Method 5 (which assessed worker neutron exposure by subtracting the number of tracks on control badges from tracks observed on worker badges) as the most technically accurate. However, it is not clear how the background subtraction "based on the standard deviation of the number of neutron tracks on the blank films" was carried out. NIOSH should provide a more detailed explanation of this method, including the actual data used in the calculation. SC&A has a further concern about where these blank films were kept.

  A common past practice was to store control badges with security personnel at each reactor facility. Under these conditions, control badges would likely register low-level neutron exposures 24 hours per day during a given cycle. Thus, such films would be not be a valid basis for background subtraction.

  We thus agree that "it is not known which [of the five methods] is the most accurate…" Under such circumstances, NIOSH is obligated to use the most claimant-favorable assumption, which in this case is Method 1. Using Method 1, the mean 0.43 neutron-to-photon ratio, listed in Table 1, when corrected for the 28% efficiency factor, would correspond to a mean neutron-to-photon ratio of 1.54.

- **Limited NTA Data** – For the eight single pass production reactors, the neutron-to-photon ratio data are based on only seven workers who, moreover, were described as having primarily worked at the Hanford reactors.

- **Absence of Neutron Spectra for Hanford Reactors** – The TBD (Fix 2004) states that ". . . Neutron spectra measurements of the single pass Hanford reactors during operation have **not** been located." As stated in the TBD, NTA film has a low-energy threshold of 700 - 800 keV, and is capable of accurately measuring doses only from neutrons with energies above 1 MeV. In the absence of spectral data, it is not possible to adjust the measurements made with the NTA film to account for the lower-energy part of the neutron spectrum.

- **Limited Data for Correcting NTA Deficiencies –** Linked to the absence of reactor spectral data is the acknowledged under-response of NTA film to neutrons below 1 MeV. NIOSH's correction factor that involves the 28% neutron detection efficiency is of limited validity, since this value is based on a single NTA dosimeter measurement that was compared to the dose rate measured by the TEPC.

- **Questionable Accuracy of Recorded NTA Data** – Aside from the limited dosimetry data involving only seven workers, the accuracy of NTA film for determining neutron doses is limited. According to Hine and Brownell (1956) "the dosimetry of neutrons by photographic methods, as by all other methods, is still in a rather unsatisfactory state. Except under specialized conditions of neutron energy distribution, and preferably absence of accompanying γ-radiation, accuracy of measurement is often low." A major factor contributing to inaccuracy is the dosimeter's angular dependence of inducing optically discernable/measurable tracks. For example, for the high-energy polonium-beryllium neutron source, the number of visible tracts at 90º incidence is only 50% of that at 0º incidence. For rotational exposure, the efficiency is reduced to 65%.

  Lastly, the neutron energy threshold for inducing discernable tracks may be even higher than the 1 MeV assumed by NIOSH. According to these authors:

  > *A serious limitation on fast-neutron dosimetry by track analysis is its unsuitability at lower energies. Tracks of length less than perhaps 3 μ (proton energy of about 0.3 MeV) are difficult to distinguish from the chance alignment of fog grains. Since three-tenths of the proton recoils from 1-MeV neutrons are of energy less than 0.3 MeV, the error introduced by neglecting to count the shorter tracks is large at neutron energies this low.*

In summary, measurable tracks from neutron interactions with NTA film are defined by the energy distribution of recoil protons (i.e., hydrogen ions). As noted above, the threshold proton recoil energy of 0.3 MeV is not defined by a fixed threshold of neutron energy (e.g., >700 keV). Instead, only the probability of recoil protons with energy in excess of 0.3 MeV increases with neutron energy above 700 keV.

5.1.4.2 Neutron-to-Photon Ratio for the N Reactor

The design of the Hanford N reactor is different than the design of the single pass production reactors. Since this is after the evaluation of neutron dose rates on the face of the B reactor (Peterson and Smalley 1960), which included an estimate of the reduction in the neutron flux that would be achieved by additional shielding, NIOSH assumed that such shielding would have been incorporated into the design of the N reactor. Table 6-18 in the TBD (reproduced in Table 2 of the present report) shows the reduction factor for the neutron-to-photon dose rate ratio for six of the single-pass reactors. Next, "…each of these factors were applied to the adjusted NTA film measurements and a combined data set was evaluated." Based on an analysis of these data, NIOSH assigned the following neutron-to-photon ratio values: geometric mean – 0.06, geometric standard deviation – 3.00, and upper 95th percentile – 0.37.

SC&A's Assessment of Assigned Ratio Values for N Reactor

A comparison of neutron-to-photon geometric mean value of 0.41 for the single pass reactors to the assigned value of 0.06 for the N-Reactor implies a reduction of about seven-fold. Discussed below is the "technical basis" and assumptions used by NIOSH to arrive at the assigned neutron-to-photon ratio values for the N-Reactor.

The seven-fold reduction in neutron-to-photon ratio for the N-Reactor is based on data presented in Table 6-18 of the TBD, which is reproduced herein as Table 2.

Table 2. Estimated Neutron and Photon Dose Rates for Hanford Reactor Front Face

Reactor	Prior to 1961			After adding external shielding			Ratio reduction factor
	Neutron (mrem/hr)	Photon (mrem/hr)	Ratio	Neutron (mrem/hr)	Photon (mrem/hr)	Ratio	
B	25	25	1.00	2	20	0.10	10.0
C	30	25	1.20	2	20	0.10	12.0
D	5	25	0.20	1	20	0.05	4.0
DR	10	20	0.50	1	16	0.06	8.3
F	15	25	0.60	2	20	0.10	6.0
H	5	25	0.20	1	20	0.05	4.0
KE	3	7	0.43	–	–	–	–
KW	3	3	1.00	–	–	–	–

Table 2 identifies values that "**imply**" a four-fold to twelve-fold reduction of neutron-to-photon ratios that were assumedly due to the addition of post-1961 shielding. A careful review of the wording contained in the TBD reveals the fact that the cited reductions in neutron-to-photon ratios do **not** represent empirical data but only projected calculations, as must be concluded from the following statements:

> *Worker exposure to neutron (and photon) radiation beams associated with instrument and test penetrations into the reactor core with the Hanford single-pass reactors that began operation in 1945 did occur. A report by Wilson (1956) summarizes the potential for significant neutron and photon dose rates for these beams and the concern for significant neutron dose to the eyes of workers conducting instrument measurements of the reactor core. In 1960,* ***Peterson and***

> ***Smalley (1960) evaluated the neutron dose rates on the face of the B reactor at Hanford.*** *The purpose of this evaluation was to develop a shielding method to reduce the neutron dose rate resulting from leakage through empty fuel tubes. As part of this analysis, they reported existing neutron and photon dose rates for the various Hanford reactors and the* ***estimated dose rates after adding external shielding.*** *This information is summarizes in Table 6-18* [Table 2 herein]. [Emphasis added.]

From data presented in Table 2 above as well as statements quoted from the TBD, SC&A concludes that the assigned geometric means of 0.06 for the neutron-to-photon ratio for the N-reactor incorporates an estimated seven-fold **average** shielding reduction factor **projected** for the B, C, D, DR, F, and H reactors, as given in Table 2 above. In truth then, Peterson and Smalley never **measured** the neutron-to-photon ratios (for shielding that may or may not have been added) but merely projected these estimates based on calculational methods.

In summary, the geometric mean neutron-to-photon ratio of 0.06 for the **N-reactor** is based on the 0.41 geometric mean derived for reactors B, C, D, DR, F, H, KE, and KW (as described in Section 1.2 above) and reduced by the seven-fold "shielding factor" **calculated** by Peterson and Smalley in 1960.

The technical credibility of the neutron-to-photon ratios assigned by NIOSH to the N-reactor, therefore, suffers not only from limitations identified previously for the B, C, D, DR, F, H, KE, and KW reactors, but also from the **uncertainty** that surrounds the seven-fold shielding reduction factor. This uncertainty corresponds to (1) **calculational** shielding values that were averaged in behalf of six single pass production reactors, (2) **assumption** of applicability of this shielding factor to the 105 N closed-loop graphite production reactor, and (3) the **unconfirmed** assumption that the N-reactor was provided this added shielding.

Regarding the **unconfirmed** assumption that the N-reactor was properly shielded (which would support the use of the seven-fold shielding factor used to derive the 0.06 neutron-to-photon ratio), NIOSH openly acknowledged that this assumption is at best a **speculation**, as provided in the following statement on page 40 of the TBD:

> *The purpose of the Peterson and Smalley report was to evaluate different shielding methods to reduce the neutron exposure on the face of the graphite reactors. It is not clear as to when the additional shielding recommendations identified in this report were implemented. Since the report was issued in July of 1960, and the first of the Hanford reactors were shutdown starting in 1964 with the last single pass reactor being shutdown in 1971,* ***it is possible that the additional shielding was only installed in some reactors*** *(later running reactors)* ***and not installed in others.*** *Until the installation time frame is identified the claimant favorable assumption that additional shielding was not installed prior to the reactor being shutdown is made.* [Emphasis added.]

SC&A concludes that NIOSH's **assumption** of enhanced shielding for the N-reactor (which would justify the seven-fold reduction in the neutron-to-photon ratio) is unconfirmed and therefore, pure speculation.

SC&A would also like to point to data in Table 2 above that further question the credibility of assigned neutron-to-photon ratios by NIOSH. Of interest are neutron-to-photon values reported by Peterson and Smalley (1960), for reactors B, C, DR, F, KE and KW. Peterson and Smalley cite neutron-to-photon ratios that range from 0.43 to 1.20. These values are described in the TBD "reported existing neutron dose rates" by Peterson and Smalley, which SC&A interprets as **measured** values. These measured values clearly exceed the "derived" geometric mean value of 0.41 assigned by NIOSH to these reactors, as summarized in Section 1.2 above.

5.1.4.3 Neutron-to-Photon Ratios for the 200 to 300 Area Plutonium Facility

Starting in 1944, Hanford began separating plutonium at T Plant. Initial plutonium finishing operations on plutonium nitrate began in 1945 in the Hanford 200 Area in the 231-Z Building. The plutonium finishing operations expanded with the construction of the Plutonium Finishing Plant in 1949. Over the years, the manual process expanded to the 300 Area and became more mechanized.[36]

Neutron dose is associated with all of these plutonium production processes, with PuF_4 as the most significant historical source for worker exposures. For the 200 and 300 Area plutonium facilities, neutrons are principally generated from (1) spontaneous fission of plutonium and (2) by the α, η reaction in low-Z elements, such as beryllium, oxygen, and fluorine.

An assessment of potential neutron spectra to which workers may have been exposed has in more recent times included bare PuF_4 and sources attenuated with various thicknesses of acrylic shielding. In general, these spectra showed a continuum of neutron energies from < 1 eV to > 20 MeV, with a peak at about 1 MeV (see Figures 6-11 and 6-12 of the TBD [Fix 2004]).

Since the introduction of the HMPD personnel dosimeter in 1972, neutron exposure to workers in the 200 and 300 Areas have been monitored and recorded.

To develop a neutron-to-photon ratio for dose reconstruction for workers exposed to neutrons prior to 1972, NIOSH reviewed post-1972 neutron and photon doses of workers assigned to the 200 and 300 Areas, who were monitored with the HMPD. Selected for analysis were the dosimetry records of 15 long-term workers between 1972 and 1991, which included a total of 186 matched dosimeter readings in which both the recorded photon and neutron exposures were $20 mrem. In brief, post-1972 recorded neutron and photon doses were used to establish a neutron-to-photon ratio that was to be used for calculating neutron doses prior to 1972.

The results of the 186 matched dosimeter readings were evaluated in terms of their neutron-to-photon ratios and yielded a distribution that is depicted in Figure 1. The data show a wide range in values with maximum neutron-to-photon ratios of up to 5.0.

[36] DOE, 2002, *The Hanford Site Historic District*, DOE/RL-97-1047, U.S. Department of Energy, Richland, WA.

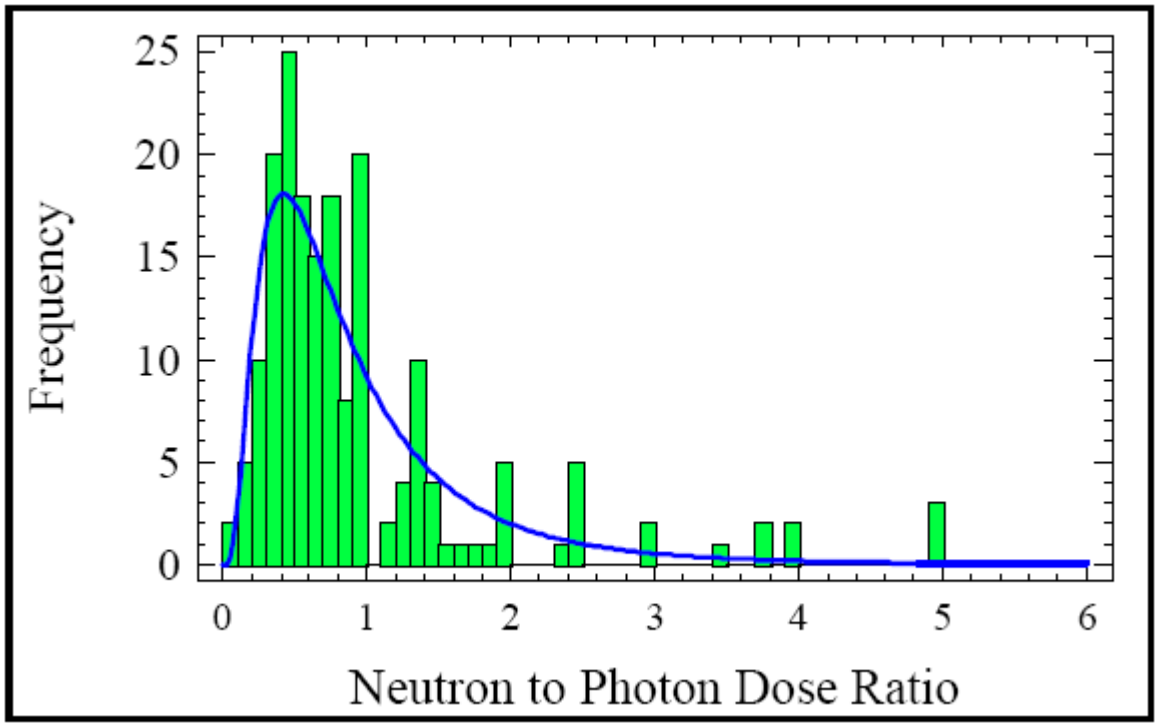

Figure 1. Neutron/Photon Dose Ratio Distribution of Hanford Plutonium Facility Workers

From this data set, NIOSH calculated a neutron-to-photon ratio with a geometric mean of 0.73, a geometric standard deviation of 2.10, and a 95th percentile value of 2.47.

SC&A's Assessment of Assigned Ratio Values for the Hanford 200 and 300 Areas

Among the three Hanford sites for which NIOSH derived/assigned neutron-to-photon ratios, SC&A regards the values for the 200 and 300 Areas as having the highest technical validity. Nevertheless, even these values are likely to suffer from technical shortcomings and uncertainties, as explained below.

Data Selection Bias: Determination of the neutron-to-photon ratio for the 200 and 300 Areas was based on 186 paired HMPD dosimeter readings in which both the recorded photon and neutron exposures were $ 20 mrem.

In reviewing Tables 6E-6 and 6E-7 of ORAUT-TKBS-0006-6 (Fix 2004), the following LOD (or MDL) values are cited for the period 1972–1994 for the HMPD:

	MDL (mrem)
Photon Deep Dose $_{(Hp(10))}$	20
Neutron	50

The potential impact of selecting recorded neutron dosimeter readings $ 20 mrem, which is less than one-half the stated MDL value of 50 mrem, cannot be assessed in the absence of the original raw dosimeter data. Nevertheless, there is the potential that this selection criterion for recorded neutron doses may have understated the neutron-to-photon ratios derived from these data.

Uncertainties Due to Extrapolation. Neutron-to-photon dose ratios for the 200 and 300 Areas are based on measured HMPD dosimeter data taken between 1972 and 1991. The extrapolation of data from this time period to the period of 1945–1972 would be valid if one could reasonably conclude that all source materials, production processes, engineering controls, and work practices remained constant between 1945 and 1991. Clearly, this assumption is not supported by facts. The TBD (Fix 2004) acknowledged the many changes in the plutonium facility, which undoubtedly also affected the ratio of neutron-to-photon dose rates. For example, the TBD describes the initial 1945 facility as follows:

> *The initial 234-5Z plutonium finishing equipment was termed the "Rubber Glove (RG)" line because it depended on personnel working with a series of 28* ***stainless-steel*** *gloveboxes, 55 meters long, to move the plutonium mixtures manually through the finishing process.* [Emphasis added.]

Over time, these manual processes were increasingly replaced by automated/remote-controlled processes, as described in the following TBD passage (pp. 34-35):

> *On March 18, 1952, a Remote Mechanical A (RMA) Line began operation. The RMA Line performed all the process steps in Pu metal production and fabrication except Task 1 (feed make-up and purification), which continued in the 231-Z facility. The RMA Line was in six rooms at 234-5Z. In mid-1957, the RMA Line was modified for a continuous calcination and hydrofluorination process that essentially handled the Task 1 activities previously done at 231-Z (i.e., all processing tasks). Many projects were undertaken at PFP 234-5Z from 1957 to 1961 to accommodate the significant increase in throughput. The most significant of these were the construction of the RMC Button Line and the RMC Fabrication Line. Both of these began operation in the mid-1960s. The RMC Line (button and fabrication components) consisted of a* ***completely self-contained, remotely operated series*** *of glove boxes similar to the RMA Line areas.* [Emphasis added.]

A reasonable (if not inevitable) question that must, therefore, be raised is the following: How did the steady conversion of manual operations to fully automated, remote-controlled, and/or self-contained processes, as well as added shielding, affect the neutron-to-photon dose rate ratios experienced by workers over time? Attachment 9 to the present report is a 1972 letter from R. P. Corlew to the U.S. Atomic Energy Commission (AEC) that includes the following observations, which confirm some of SC&A's concerns:

(1) The introduction of the HMPD dosimeter in 1972 demonstrated the fact that neutron exposures were well above previous levels measured by NTA dosimeters.

(2) This acknowledgement of higher neutron levels triggered the installation of added shielding in the 200 and 300 Areas, which must reasonably be assumed to have affected subsequent neutron-to-photon ratios.

(3) Table 1 in Appendix A of Corlew's letter also provides insight into neutron-to-photon ratios prior to 1972 (i.e., subtracting values cited in Column #2 from Column #3 and dividing this net value by Column #2 yields a neutron-to-photon ratio). For most of the

twenty workers assigned to the 234-5-Z Building on the RMC Button Line, a neutron-to-photon ratio > 1.0 is calculated.

In summary, the extrapolation of post-1972 neutron-to-photon data for the reconstruction of pre-1972 neutron doses at the 200 and 300 Areas fails to address the impacts of the many changes that occurred over time. In combination, changes to production processes, engineering controls, and work practices undoubtedly altered neutron-to-photon ratios and thereby the magnitude of uncertainty of pre-1972 reconstructed neutron doses.

A quantitative assessment of these impacts on neutron-to-photon ratios is beyond the scope of this review; however, based on the limited evidence uncovered in this review, SC&A concludes that pre-1972 workers may have been exposed to radiation fields with neutron-to-photon ratios well in excess of those cited in the TBD.

5.1.4.4 Summary Conclusions Regarding Reconstruction of Neutron Doses at Hanford

For many workers, neutron exposure contributed a large fraction of the total dose from external radiation. In fact, when neutron doses are adjusted to account for the current International Council of Radiation Protection (ICRP) neutron-weighting factor, neutron doses for workers at the 200 and 300 Area plutonium facilities dominate the external dose. For this reason, SC&A critically reviewed the technical basis by which NIOSH assigned neutron-to-photon ratios, as summarized in Table 3 below.

Table 3. NIOSH-Derived Neutron-to-Photon Dose Ratios for Use in Pre-1972 Neutron Dose Reconstruction

Facility	Neutron-to-Photon Ratio Values		
	Geometric Mean	Geometric Standard Deviation	95th Percentile
B, D, F, H, DR, C, KW, KE Reactors	0.41	2.79	2.23
N Reactor	0.06	3.00	0.37
200 and 300 Areas	0.73	2.10	2.47

Source: From Table 6-22 of TBD (Fix 2004).

SC&A's review identified various issues that include: (1) the use of questionable data and/or the selection of limited data that were not claimant favorable, (2) the use of incomplete/insufficient data, (3) the use of unconfirmed assumptions, and (4) the failure to account for critical variables, which limit the use of extrapolated data over time.

While SC&A fully recognizes the difficulties of dose reconstruction in instances of missing or inadequate neutron monitoring data, SC&A's findings suggest that NIOSH filled these gaps by methods and assumptions that were not necessarily conservative and claimant unfavorable. An example of this is NIOSH's use of the unconfirmed assumption about the shielding design of the N Reactor and the assignment of a geometric mean neutron-to-photon ratio of 0.06. This shielding assumption was based on the timeframe of the construction of the N Reactor and the calculated reduction in the neutron-to-photon ratios for single-pass production reactors, were additional shielding to be installed. This and other assumptions are not conservative or claimant favorable.

In summary, SC&A's review of available data involving worker neutron exposures has identified a number of potential uses. With the exception of the energy-dependence of the NTA film dosimeter, the TBD has not adequately addressed other issues associated with neutron doses, as discussed above.

5.2 ISSUE 2: EARLY WORKER RADIOLOGICAL MONITORING

The early Hanford radiation protection program was in the process of development during the initial years of operation at the site. The lack of bioassay data during the period 1943 - 1946 makes it difficult to properly quantify worker internal dose, and is especially a problem because of the potentially high exposures that often occurred during that period. Plutonium bioassay began in September 1946 (Bihl 2004, page 13). Other bioassay data were either not taken or are unreliable until later dates. Uranium urinalysis data are not reliable prior to 1948 (Bihl 2004, page 24). Even so, there are gaps in uranium bioassay data in regard to recycled uranium trace constituents and ^{233}U (see below). Fission product urinalysis did not begin until 1947, but data prior to 1948 are deemed unreliable (Bihl 2004, page 27). The potential for gaps in missed dose, therefore, is a significant dose reconstruction issue. This absence of bioassay monitoring in the 1940s is particularly important for isotopes such as curium, polonium, and americium, since bioassay monitoring for these radionuclides was not available until the late 1960s or early 1970s. Uncertainties in the actual bioassay techniques and instruments used to quantify internal dose and the MDAs established needs to be more thoroughly evaluated. In vivo counting was not introduced at Hanford until mid-1959. In addition, there is a lack of internal monitoring data on which to base worst-case dose reconstructions. A potential for missed tritium dose exists in facilities handling tritium.

Despite the availability of both in-vivo data and in-vitro/bioassay data from 1960 onward, the TBD has not used them together to attempt to validate earlier bioassay data.

5.2.1 Derivation of Intakes in the Years Prior to Implementation of Routine Bioassay Programs

In the Hanford internal dose TBD (Bihl 2004, Section 5.1, page 7), NIOSH states the following:

> *Because there is little information concerning intakes in the years prior to implementation of routine bioassay programs, the following default assumptions should be made unless there is better information in the worker's file. All of the values below should be considered the modes of triangular distributions with zero minimums and maximums of twice the modes. Although it is unlikely that all workers were actually exposed to air concentrations at just under the respiratory protection required levels for 40 hours per week (basis for the mode), it is also possible that workers were exposed to greater concentrations for short periods of time. The maximum values account for:*
>
> - *short term exposure to higher air concentrations*
> - *the possibility that the respiratory protection was not always effective*
> - *uncertainty in the date the respiratory protection requirement was implemented.*

In the internal dose TBD, NIOSH states the following assumptions for intakes by workers who wore dosimeters but had no bioassay (pp. 49 and 50):

> *For the years following the periods cited in section 5.1, the DR should base the intake assessments on the daily intake values presented in Table 5.7.1….*
>
> *For the years 1947 though 1952, the daily intakes were based on the assumption that exposure to airborne concentrations occurred at just under the respiratory protection required value for 4 hours each week or 10% of the respiratory protection required value for 40 hours each week. This assumes that these workers either 1) occasionally entered areas with airborne contamination just below requirements for posting and respiratory protection or 2) were located full time in an area with low but arguably not zero airborne contamination…*
>
> *For 1953 through 1988, the daily intakes were based on the assumption that exposure to airborne contamination at 10% of the limiting air concentration occurred for 4 hours each week. For particulate sampling, the limiting air concentration used to interpret total beta counts on air filters was based on ^{90}Sr, and the limiting air concentration for total alpha counts was based on ^{239}Pu…*

NIOSH uses the respiratory protection limits to derive the worker doses, assuming that the air-monitoring program reflected the worker exposures in that area. NIOSH assumes that the air monitoring program in the area would prevent the workers from being exposed to concentrations above the limiting air concentration and that otherwise the workers would wear masks. NIOSH does not discuss the fact that air monitoring, especially from static samplers, cannot easily be associated with the real intake by the workers.

NIOSH does not specify the technical basis for the following assumptions:

- A triangular distribution for the earlier years (reactor workers for the years 1944- 1948, separation plants for 1944–1946, 231-Z Building for 1945-1946, 300 Area uranium fabrication buildings for 1944–1947, and laundry for 1944–1946)

- The maximum of the triangular distribution equal to twice the modes based on air concentrations at just under the respiratory-protection-required levels for 40 hours per week

NIOSH does not specify which statistical distribution the dose reconstructor should use after the period cited in Section 5 (Bihl 2004). NIOSH does not explain the technical basis for the following assumptions:

- For the period 1947–1952, daily intakes based on the assumption that exposure to airborne concentrations occurred at just under the respiratory-protection-required value for 4 hours each week or 10% of the respiratory-protection-required value for 40 hours each week.

- For the period 1953–1988, daily intakes based on the assumption that exposure to airborne contamination at 10% of the limiting air concentration occurred for 4 hours each week.

In the internal dose TBD, on pp. 48 and 49, it is stated that:

> *Air sampling was performed in facilities from the earliest days of operation … Historical criteria for placing workers on routine bioassay schedules have not been discovered for all times or all facilities and probably were facility-by-facility specific. However, all facilities had broad scoped radiation protection programs that were designed to limit intakes of radioactive material and monitor workers at risk for intakes. Workers who had dosimeters but no routine bioassay would have been those who were only occasionally in areas with the potential for airborne contamination or with measured but low airborne contamination or had work locations distant enough from airborne contamination that the air was considered uncontaminated. For the period 1953 though 1988 entry into areas with airborne concentrations exceeding the maximum permissible concentrations (MPC) required respiratory protection….*

SC&A feels that it is questionable to assume that the air sampling was done in all areas, from all facilities, at all times since 1947, and that sample data are complete and can be easily used to infer the intakes of airborne activities by workers. The assumption that areas with measured low airborne contamination had air samplers located at the breathing zone of the workers may not be claimant favorable, and NIOSH should document evidence that this is true before conclusions are made regarding the intake of workers.

The ICRP has pointed out in various publications that the interpretation of the results of measurements from air sampling may be misleading. In Publication 78, *Individual Monitoring for Internal Exposure of Workers* (ICRP 1997), the ICRP specifies that static air samplers can underestimate concentrations in air in the breathing zone of a worker, typically by a factor of up to 10.

The ICRP supporting Guidance 3, *Guide for the Practical Application of the ICRP Human Respiratory Tract Model* (ICRP 2002), specifies that the use of static samplers does not ensure a representative measurement of exposure of the worker and cites that, for localized releases of activity such as a hole in a glove, aerosol concentrations can vary by several orders of magnitude within a meter of the point of release. It also points out the problem of sampling high specific activity particles, such as ^{238}Pu, when air concentrations at protection levels may correspond to low particle number concentrations. It points out that the issue of specific activity is important in relation to flow rates and sampling durations.

NIOSH does not appear to cross compare air sample data from different areas and facilities, nor does it document the number of air samplers used to determine general or individual dose from airborne particulates. Uncertainties on the air sampling results due to the positioning of air samplers in relation to the worker, to flow rates of the different air samplers, and sampling duration are not considered by NIOSH.

NIOSH does not mention if personal air samplers were used and when. Instead it uses unexplained values that are discussed on pages 47 and 49 in the internal dosimetry TBD (Bihl 2004), as follows:

- For the earlier years, NIOSH uses a triangular distribution with maximum equal to twice the mode based on air concentrations at just under the respiratory-protection-required levels for 40 hours per week.

- For 1947–1952, NIOSH instructs the dose reconstructor to use 10% of the "respiratory-protection-required value" for 40 hours work per week.

- For 1953–1988, NIOSH instructs the dose reconstructor to use 1% of the respiratory-protection-required value for 40 hours of work per week.

SC&A believes that these unexplained values are not claimant favorable, since aerosol concentrations can vary by several orders of magnitude within a meter from the point of release (ICRP 2002).

5.2.2 Lack of Internal Monitoring Data

In the Hanford occupational internal dose TBD (Bihl 2004), NIOSH states on page 6 that:

> *When the first reactor was started on the Hanford site, there were no programs to monitor an employee for internal dose, with the exception of measuring particles in the air. The site was operating three reactors, a fuel manufacturing facility and four processing plants from 1943 to 1946 before a bioassay program was in place….*
>
> *….Air sample data from the reactors were almost never listed in these reports; the radiation protection emphasis at the reactors seemed to be external dose and effluents in the water. The records show that high air concentrations at the other facilities prompted use of respiratory protection.*

Air sample data were not routinely reported in the health instrumentation section of monthly reports for the reactors. Table 5.1-1 of the TBD (Bihl 2004), page 6, shows limited data for maximum air concentrations (uCi/cm^3) and "Most sampler" concentrations (uCi/cm^3). A review of the table reveals that there are many gaps in the data, particularly for the D reactor, F reactor and the T Plant from 1943 to 1946. This leaves these reactors without any air sampling data on which to base dose reconstructions for reactor workers.

5.2.3 Adequacy of Data

The lack of bioassay data during 1943–1946 makes it difficult to adequately determine worst-case dose, especially for workers in high exposure areas and operations during that period. Lack of reliable data for uranium and fission products extends to 1948. As for trace impurities in recycled uranium, there appear to be no data for production workers even after 1983, when alpha

spectrometry began to be used, because bioassay for uranium production workers continued to rely on the method of determining elemental uranium rather than an isotope-specific analysis.

A more thorough evaluation of the uncertainties in the actual bioassay techniques, and instruments used to quantify internal dose and the MDAs, is needed.

5.2.4 Regulatory Compliance

The dose reconstruction process must comply with the requirements of 42 CFR Part 82. As a method of effectively implementing these requirements, NIOSH has written technical guidance documents on external and internal dosimetry, and its contractor has committed to the use of these guidance documents in its quality assurance program plan.

NIOSH uses the respiratory protection limits to derive the worker doses, assuming that the air monitoring program reflected the worker exposures in the areas monitored and that the radiation protection program prevented the worker from being exposed to concentrations above the limiting air concentration. NIOSH does not discuss the considerable uncertainties related to air sampling monitoring results. NIOSH does not address the considerations given in NIOSH 2002b, Section 5.2, page 18, where it states the following:

> *In the absence of breathing zone samples, general area air samples can be used, but consideration must be given to any factors that could create a difference between general area and breathing zone concentrations. Some of the factors that should be considered are: the amount and direction of ventilation, the location of the airborne sources in relation to the individual and the air sampler, and whether the individual is mobile or stationary in the course of the work.*

5.3 ISSUE 3: EXTERNAL BETA/GAMMA DOSE ADJUSTMENTS AND UNCERTAINTY FACTORS

The Hanford external dosimetry TBD (Fix 2004) was evaluated to determine if adequate guidance was provided to perform a dose reconstruction of the beta/photon dose. The TBD was assessed with the following criteria:

(1) Does the TBD provide adequate background information?

(2) Does the TBD provide guidance for adjusting recorded photon dose measurements with regard to dosimeter type and tissue type?

(3) Does the TBD provide guidance for addressing missed photon dose for monitored workers and unmonitored workers?

SC&A determined that the external dosimetry TBD provides detailed background information, but the guidance has some deficiencies with regard to uncertainty, nonpenetrating beta doses, exposure geometry, and missed dose. In addition, the guidance is incomplete, often confusing, and in some cases appears to contradict other guidance provided in other NIOSH documents.

5.3.1 Use of Correction Factors and Uncertainty and Bias Factors for TLDs and Film Badge Readings is Not Clear in the TBD

The Hanford external dosimetry TBD (Fix 2004) is presented in two general sections: the main body of the report and Attachment 6E. The main body contains detailed descriptions of Hanford's history, dosimetry, and dosimeter response testing, as well as some discussion of missed dose and uncertainty. Attachment 6E contains the guidance and specific instructions for performance of dose reconstructions. However, it appears that some of the information pertaining to photon dosimeter uncertainty discussed in the main body of the report is not included in the Attachment 6E guidance.

In Tables 6-11 and 6-12, and later in Tables 6-32 and 6-33, the TBD describes the uncertainty in beta/photon Hp(10) values by presenting "bias factors" for each of the dosimeters used over time and for different facilities at Hanford. The footnote to Table 6-11 describes the bias factors as a "ratio of Hp(10) to recorded whole-body photon dose." The TBD devotes some time to these bias factors, but there is no mention of them at all in the Attachment 6E guidance. Page 78 of Attachment 6E of the external dosimetry TBD has the following instructions for calculating uncertainty for a dose reconstruction:

> *Parameter 2 is the standard deviation of the normal distribution for the organ dose. The individual dose result for each dosimeter exchange period will be available to calculate the mean and standard deviation for each year. If it is not available, the adjusted organ dose can be used for each year and a default standard deviation value used for parameter 2…. Uncertainty in recorded dose is an important consideration in claimant-favorable analysis. The overall uncertainty depends on (1) administrative practices, (2) dosimetry technology, (3) calibration, and (4) workplace radiation fields. The potential for each of these parameters on the recorded dose is described in previous sections.*

It appears that the guidance described in Attachment 6E would take precedence over information presented in the main body of the report. However, it is unclear why these bias factors are not included in the guidance and how, if at all, the dose reconstructor is to use these factors in Tables 6-32 and 6-33 to calculate uncertainty and Parameter 2 of the IREP code. In Table 6-12, page 28, and again in Table 6-32, page 51, the external dosimetry TBD provides a summary of overall bias magnitude for the various Hanford dosimetry systems. Footnote a to Table 6-12 states:

> *Based on the distribution of energy levels and geometry judged most likely, divide recorded dose by the table's bias value to calculate deep dose. Note that this use of bias factor does not apply to plutonium facilities.*

It is not clear if this is simply part of the discussion or if it is represents specific instructions to the dose reconstructor.

In addition, other guidance documents (i.e., ORAUT-OTIB-0008 and ORAUT-OTIB-0010) recommend applying a standard correction factor of 2 to the recorded dose, which eliminates the need for an uncertainty factor. This value would then be entered as a constant for Parameter 1 of the IREP without any distribution. The TBD does not indicate if this option is available to the dose reconstructors evaluating Hanford photon doses.

The basis for and guidance to dose reconstructors in the TBD for application of correction factors between TLDs and film badge readings is not clear, particularly for some external dose statistical issues and adjustment factor issues. For example, the external dosimetry TBD (Fix 2004) states on page 21:

> *For energies less than 100 keV, the two-element dosimeter will underestimate the photon dose without some method of adjustment such as a fraction of the dosimeter open window or silver shielded response.*

In order to address this concern, the external dosimetry TBD, page 17, recommends a correction factor to apply for the two-element film badge for plutonium workers as follows:

> *The ratio of the OW to the filtered film badge response was routinely used in dose evaluation (Larson and Roesch 1954) and there is reference to using a fraction (0.2) of the OW response to add to the penetrating dose in facilities with low-energy photons and no beta radiation (i.e., plutonium facilities) (Fix, Wilson, and Baumgartner 1997). However, it has not been validated that this was actually done. An analysis of the bias in the nonpenetrating and penetrating dose is presented in Table 6.3. The Hanford recorded skin dose is calculated as the sum of the open window (OW) and the silver (S) filtered film response. The recorded Hanford whole body (WB) dose is calculated using 20% of the OW film response in addition to the measured S film response using the historical Hanford testing data in Appendix A of Wilson et al. (1990)…. As such, a claimant favorable recommendation, for plutonium workers only, is to apply the calculation of the WB dose using 20% of the OW dose in addition to the measured S dose pending confirmation that the historical WB dose does indeed include the 20% of the OW dose* (Fix 2004).

If one assumes a measured dose of 100 mrem, the Hp(10) deep dose would vary as shown in Table 6-12, page 28 (Fix 2004), as follows:

	Mean Bias $H_p(10)$	High End of Bias $H_p(10)$
Two element film (1944-1046)	100 mrem/1.27 = 78.7 mrem	100 mrem/1.60 = 62.5
Multi-element film (1957-1971)	100 mrem/1.02 = 98.0 mrem	100 mrem/1.12 = 89.3
HMPD (1972-1983)	100 mrem/1.12 = 89.3 mrem	100 mrem/1.16 = 86.2
HMPD (1984-1993)	100 mrem/1.01 = 99.0 mrem	100 mrem/1.05 = 95.2

It is not clear from this table just how dose reconstructors are to apply the systematic and random uncertainties provided in Table 6-12. In addition, Table 6-10, page 26, in discussing Hanford workplace photon dosimeter $H_p(10)$ performance, indicated that under the exposure geometry parameter for energies greater than 100 keV, the two-element film dosimeter has a bias of approximately +200%, others, +/-25%. For energies less than 100 keV, the bias is likely too low. Footnote a to the table states (Fix 2004, Table 6-10, page 27):

> *Bias represented as percent of the recorded dose compared to Hp(10) based on the judgment from laboratory and field measurements.*

The external dosimetry TBD on page 27 provides the following guidance on assessment of uncertainties:

> *The assessment at Hanford was 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.*

Table 6-6, page 21, of Fix 2004 shows that, with a 120 keV beam energy, the $H_p(10)$ estimate for the two-element film badge, if one assumes a recorded dose of 100 mrem, would be (100/1.6 = 62.5 mrem $H_p(10)$). The multi-element film badge for the same recorded 100-mrem dose would give an $H_p(10)$ dose of (100/0.64 = 156 mrem). On the other hand, the $H_p(10)$ estimate, if one assumes a recorded dose of 100 mrem for the TLD at the beam energy of 120 keV, would be (100/0.87 = 114.9 mrem $H_p(10)$). Thus, in this example, the two-element film badge once corrected would result in an $H_p(10)$ dose that is 54.4% of that obtained by the TLD (i.e., 62.5/114.9 = 0.544).

If one, however, looks at the 59 keV beam energy, it is noted that the two-element dosimeter with a recorded dose of 100 mrem would give an $H_p(10)$ dose of (100/0.5 = 200 mrem), while the multi-element film badge for the same 100 mrem recorded dose would give an $H_p(10)$ dose of 100/1.1 = 91 mrem. In this case, the two-element film badge would give an overestimate of $H_p(10)$. This is the same conclusion reached by the site expert in Attachment 3, page 114, where it was concluded that:

> *It should be noted that the Hanford study dosimeter results for energies greater than 100 keV are consistent with the IARC results (which only includes > 100 keV irradiations), showing an overestimate of $H_p(10)$ for the two element dosimeter.*

The TBD (Fix 2004) instructions do not seem to clearly explain the need to review such differences or to provide any correction factor to resolve this difference. In fact, on page 22 of the external dosimetry TBD, it is stated that:

> *It is apparent from Figure 6.4 that the penetrating dose compares reasonably well between the Hanford multielement film and the HMPD [Hanford Multipurpose TLD] for all facilities although there appears from this data potential bias in multi-element film.*

During SC&A's site expert interview process, as shown in Attachment 3, Table 2 (a table provided by Nichols et al. 1972), it was noted that this is not the case for B Plant workers. Whereas the film badge recorded a workplace measured penetrating collective dose of 2,250 mrem, the TLD recorded a collective dose of 4,560. A Cutie Pie (CP) verified that the collective dose agreed closely with the TLD measured collective dose (i.e., 4,929 mrem). Thus, there is a 45.6% under response of the film badge to that recorded by the TLD (i.e., 2,250/4,929 = 0.456).

The external dosimetry TBD (Fix 2004), page 22, explains why B Plant was so different:

> *The most significant difference in penetrating dose occurred in B-Plant. This is likely associated with the relatively high nonpenetrating radiation dose indicative of beta and lower-energy photons, and the penetrating dose response of the HMPD to higher energy beta radiation as noted in Fix et al. (1982) and Wilson et al. (1990). The HMPD records a penetrating dose for higher energy beta radiation such as $^{90}Sr/^{90}Y$ when there should be none, because there is only 380 mg/cm^2 density thickness in the aluminum filter over the HMPD chip used to calculate the deep dose.*

In addition, the external dosimetry TBD (Fix 2004), page 21, points out the following:

> *The non-penetrating response of the film dosimeter was routinely calibrated with a uranium slab source, whereas a $^{90}Sr/^{90}Y$ source was used to calibrate the HMPD nonpenetrating response. There is an approximate factor of 2 difference in dosimeter response between these two sources and this is shown in the table [Table 6.7, page 22] i.e., for $^{90}Sr/^{90}Y$ source irradiation, 690 mrem for film versus 315 mrem for TLD.)*

This difference of a factor of two between film badges and TLDs, as determined when using a $^{90}Sr/^{90}Y$ calibration source, is not mentioned in Attachment 6E, which gives guidance to dose reconstructors on adjustments to photon dose. This could, therefore, be easily missed by the dose reconstructions and should be addressed somewhere in Section 6E.4.2.

5.3.2 Missed Photon Dose

Attachment 6E of the TBD (Section 6E.4.2.5) provides specific instructions for addressing missed photon dose:

> *Missed photon dose for Hanford workers can occur where (1) there is no recorded dose because workers were not monitored or the dose is otherwise unavailable, and (2) a zero dose is recorded for the dosimeter systems for any dosimeter response less than the MDL. Estimates of the missed dose can be made using dose results for coworkers or using the recorded dose before and after the period of missed dose…NIOSH (2002a) describes options to calculate missed dose. One option is to estimate a claimant-favorable maximum potential missed dose where MDL/2 is multiplied by the number of zero dose results.*

The TBD presents options for determining missed photon dose, but these options could result in inconsistencies in dose reconstruction from one case to the next. The TBD refers the dose reconstructor to the External Dose Reconstruction Guidelines (NIOSH 2002a) which presents different methods for determining missed dose for likely compensable versus not likely compensable cases. For example, for cases which are likely compensable, NIOSH 2002a recommends using the n*MDL value as a point estimate, with no uncertainty factor needed. For cases which are not likely compensable, NIOSH 2002a recommends n*MDL/2 plus uncertainty

for an assumed lognormal distribution. The Hanford TBD does not make a distinction between compensable and non-compensable cases.

In addition, the Technical Information Bulletins ORAU-OTIB-0008 and ORAU-OTIB-0010 also recommend multiplying the number of cycles by the MDL (n*MDL), which is a maximizing assumption, thereby eliminating the need for an uncertainty factor. The Hanford TBD does not indicate this option.

5.3.3 The TBD Does Not Consider Acute Beta Doses and Routine Nonpenetrating Doses

The TBD appears to indicate unspecified issues with the estimation of non-penetrating doses (Fix 2004, page 8):

> *The information in this section pertains to analyzing these records and does not address parameters regarding skin, testicular, or breast radiation dose that could result from acute beta (electron) radiation exposure under short-term accidental or incident nonroutine workplace exposure profiles. Nonpenetrating radiation during routine operations is also not addressed in this section.*

It is unclear whether this passage means that the record for non-penetrating dose should be considered inadequate. SC&A notes that Fix (2004) does not direct the dose reconstructor to Addendum F of ORAUT-PROC-0006 (Merwin 2003), which provides an external dose reconstruction procedure for shallow dose calculations for complex-wide cases. In that procedure, NIOSH states:

> *Under certain limitations, it has been determined that the calculated doses for skin cancers can be assured of being claimant-favorable if a factor of 2 is applied to deep doses reported by the sites post-1980 (and, for 1970 and later cases if an organ DCF is also applied), and when shallow doses are calculated according to the instructions provided below. Therefore, external doses of complex-wide skin, breast, and testicular cancers can be calculated according to this procedure*.

OCAS-IG-001 (NIOSH 2002a), pages 8 and 9, states:

> *1.2.3 Electron (Beta Particle) Exposures*
>
> *Generally, external electron exposures are only important for surface tissue such as skin. Thus, for skin cancer, a dose reconstruction from exposure to electrons is required. The exposure to skin can originate from either a strong unshielded electron source such as Sr-90 or uranium daughters, or from skin contamination with beta/gamma emitters. The other two organs for which external electron exposure from high-energy electrons (> 1 MeV) might be significant are the testes and the breast. For breast and testicular cancer, an evaluation of the maximum electron energy exposure should be conducted. Generally, if the electron energy is less than 1 MeV, the dose conversion factor (DCF) will be zero since the electron does not have sufficient energy to penetrate the outer layer of skin (ICRP 74, 1996.) In these cases a dose reconstruction is not necessary.*

OCAS-IG-001 does not state that dose reconstruction for skin, testicular, or breast cancer is excluded for acute/accidental exposure, yet it is specifically excluded in ORAUT-TKBS-0006-6. (Fix 2004). NIOSH should explain why this exclusion has been made and provide the technical basis for this decision.

5.3.4 Exposure Geometry

During the discussions with Hanford site experts (Attachment 3), it was also learned that geometry, angle of incidence, potential shielding, partial body, and badge location were usually considered and that for most routine operations these factors tend to average out over time. This is most likely correct, provided that the badges were calibrated on a phantom and exposed to radiation from different directions. This observation may not apply to accident or incident situations or special operations, where the source may be held close to an organ, when working in glove boxes or exposed to beam streaming from reactors.

NIOSH has not adequately characterized the impacts on exposure measurements due to exposure geometry, potential shielding, and partial body exposure and badge location. In one accident situation workers ran away from the source and it was clearly posterior-anterior (PA) geometry.

All of these concerns could be categorized under the issue of "geometry" concerning the location of the dosimeter in relationship to the source, either for calibration or on the wearer. The Hanford occupational external dosimetry TBD (Fix 2004) does mention geometry on pages 19, 51, 72, and 78. In Table 6-4 on page 19, it states that the angular response may be 50% low. The footnotes to Table 6-32 on page 51 contain references to geometry, and on page 72 it states that no adjustments are recommended. On page 78 it makes reference to uncertainties but does not explicitly mention geometry.

NIOSH needs to be more explicit on how the several geometry factors (angular response, potential shielding, partial body exposure, badge location, etc.) encountered in exposure and calibration are factored into its calculations and uncertainties. This could be especially important in accident/acute situations where the dose could be underestimated and not claimant favorable because of geometry factors.

5.3.3 Consistency Within Site Profile

The TBD does not use the same method to calculate missed photon and neutron dose throughout the document. Tables 6-30 and 6-31 of the TBD (pages 27 and 48) address the calculation of missed photon and neutron dose. However, the dosimeter MDLs, for the same dosimeter and period, are about half the value in Table 6-30 (used for photon calculations), as compared to MDLs in Table 5-31 (used for neutron dose calculations using the neutron/photon ratios in Table 6-22). As part of the SC&A site expert interviews, NIOSH explained in Attachment 3 that:

> *There appears to be an error in the preparation of Table 6-31, highlighted above, that appears in the published TBD version. The values in Table 6-31 should be the same as those shown in Table 6-30. It should be noted that the estimated missed neutron dose in response to the next comment is calculated using a neutron-to-photon dose ratio of 2 and the listed neutron missed dose in the last*

column of Table 6-31 used by the dose reconstructors is unchanged based on this method of estimating the missed dose.

This inconsistency should be corrected and the relative impact, if any, on previous dose reconstructions determined.

5.4 ISSUE 4: INTERNAL DOSE ASSUMPTIONS

In this section, detailed descriptions are provided of a number of internal dosimetry issues that should have been addressed or addressed in greater detail in the TBD. The significance of these issues is that dose reconstructors are provided incomplete guidance and information that may result in missed internal doses. In summary, the issues include the following;

(1) The historic bioassay program is inadequately described in terms of when bioassay was prescribed by radiation protection policy, how it addressed acute episodic exposures, and how uncertainties were addressed.

(2) The Hanford internal dose TBD (Bihl 2004) indicates that fission product analysis data prior to 1948 should be considered unreliable. Therefore, for the years prior to implementation of whole-body counting, default intakes based on air sampling are recommended. However, there is some question regarding whether air-sampling data can be used to reliably predict radionuclide intake rates given the high variability of airborne radionuclide concentrations in space and time, and the potential large differences between the airborne radionuclide concentrations as determined by general air samplers and breathing zone samplers. It is also noted in the internal dose TBD in Section 5.1, page 5, that additional data on assumed intakes have been added as part of that revision. There are no references or discussion of how the assumed inhalation intakes provided for the different type of facilities were derived, or references to support these assumed intakes. NIOSH should provide clarification on the derivation of these recommended intakes as well as the derivation of the limiting air concentrations and daily intakes (pCi/d) that are recommended for the various time periods in Table 5.7.1, page 57, with more discussion of how they were derived from NCRP 1959.

(3) Prior to 1986, or even until 1993, uncertainty in the results of whole-body counting appears to be inadequately addressed in the TBD. This can result in an incomplete characterization of the internal doses.

(4) The TBD states that the uranium urinalysis program prior to 1948 was not reliable, and it was not until 1983 that alpha spectrometry was introduced. In addition, even after 1983, only elemental uranium analysis was performed for production workers (page 25). This creates a significant issue in relation to workers who were exposed to recycled uranium, particularly in the 300 Area operations.

(5) During the period of when ^{232}Th target material was used to produce ^{233}U, with its concurrent production of ^{232}U, Hanford production workers were monitored only for elemental uranium. Dose reconsturtors are presumably expected to use the data in Table 5.2.4-1 to estimate the isotopic composition of the uranium intakes. However, this table gives no guidance that would apply to the intake of ^{233}U and its coproduct, ^{232}U. ^{233}U has

almost twice the specific activity of ^{233}U and slightly higher effective dose coefficients for both inhalation and ingestion. Due to its short half-life of 68.9 y, ^{232}U has a far higher specific activity than the other common uranium isotopes. The effective dose coefficients for inhalation and ingestion are up to an order of magnitude greater than those for ^{234}U. At a concentration of 50 ppm (see Section 5.7.1 of the present report), the contribution of ^{232}U could double the dose that would be calculated based on ^{234}U. Dose reconstructors should be given specific guidance for dealing with this issue.

(6) In reviewing Hanford individual dose reconstruction cases, SC&A notes that NIOSH is making use of ORAUT-OTIB-0002, *Technical Information Bulletin – Maximum Internal Dose Estimates for Certain DOE Complex Claims* (Rollins 2004).[37] The internal dose TBD, ORAUT-TKBS-00006-5, Revision 01, published on November 24, 2004, does not mention, nor does it direct dose reconstructors to, OTIB-0002 even though OTIB-0002 was published earlier, on May 7, 2004. Although dose reconstructors may be aware of OTIB-0002, it is possible that it could be overlooked and not used if the dose reconstructor is following the guidelines in the internal dose TBD.

(7) OTIB-0002 provides guidance for deriving the doses for individuals who may have been exposed to internal emitters, but are anticipated to have relatively low (non-compensable) exposures. The guidance lists a total of 28 radionuclides and default intake assumptions that are designed to bound the intakes for these individuals. There is some question whether the list of radionuclides and default assumptions regarding assumed intakes are adequate to address exposures to radionuclides associated with many of the activities described below that took place at Hanford, including recycled uranium operations, waste management, special tank farm activities, special campaigns involving ^{233}U and thorium processing, and decontamination and decommissioning activities. These activities are of particular concern, because they either not addressed or inadequately addressed in the TBD.

5.4.1 Inadequate Description of the Historic Bioassay Program

The TBD does not comment on the rapid and sharp decline in annual bioassays for Hanford workers after 1958. This deserves further investigation by NIOSH, since it may imply that there is a significant potential for missed internal doses. The discussion below first covers the historical monitoring issues, followed by an evaluation of the analysis of the minimum detectable limits for bioassay and in-vivo counting in the TBD (Bihl 2004).

In the late 1940s through the late 1950s it was reported that:

> *… everyone was monitored in the early days. Later it was left up to the decision of the field monitoring people who were more familiar with work assignments and various work locations… Later beginning in 1965 when several organizations*

[37] Rollins, G., *Technical Information Bulletin – Maximum Internal Dose Estimates for Certain DOE Complex Claims*, ORAUT-OTIB-0002, Rev. 01 PC-2 dated May 7, 2004, page 4.

> *assumed responsibility for operating parts of the Hanford plant, each contractor developed their own guidelines.*[38]

It appears that NIOSH has not reconciled this important policy change in measuring internal exposures at Hanford with respect to the impacts of such a significant cutback in bioassays and the subsequent fragmentation and decentralization of the Hanford bioassay program after 1965. As a result of this policy change, annual bioassays at Hanford dropped from approximately 79% of the workforce in 1958 to about 50.9% in 1959. After that, annual bioassays further declined by 1962 to about 20% and less through 1998 (see Figure 2).[39]

It was reported in 1986 that approximately half of the 15,000 Hanford workers were in the monitoring program. Thus, monitoring involved 10,000 whole-body counts, 1,500 lung counts, and about 3,000 urinalyses, reflecting the shift from urinalysis to in-vivo methods. The site contractors selected which of their employees will be monitored.[40]

It is worth noting that the objective of the monitoring program was to identify internal exposures at 5% of the DOE's protection standard. However, results for insoluble plutonium and uranium were above that stated standard, and this was noted as a problem that they were working to address in 1986.[41]

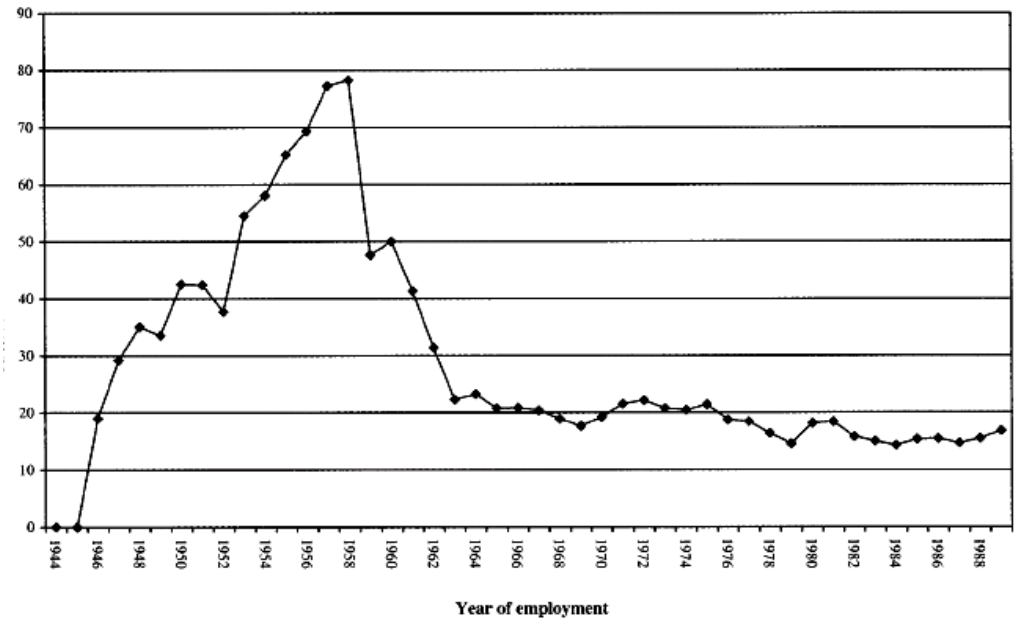

Figure 2. Annual Bioassay Monitoring of Workers at the U.S. DOE Hanford Site (%), 1944–1989

Source: Wing 2004: Figure 1, page 155

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[38] K.R. Heid, Manager Personal Dosimetry, Battelle Pacific Northwest Laboratory, Letter to R.H. Mole, Medical Research Council, United Kingdom, June 22, 1979.

[39] Steve Wing, David Richardson, Susanne Wolf, and Gary Milhan, *Plutonium-Related Work and Cause-Specific Mortality*, American Journal of Industrial Medicine, 45: 153-164, 2004.

[40] Sula, M.J., Carbaugh, E.H., Bihl, D.E. *Technical Basis for Internal Dosimetry at Hanford*. Richland, WA: Pacific Northwest Laboratory, 1989.

[41] PNL-SA-13657/DE86.

According to an assessment of internal dosimetry performed at the University of North Carolina:

> *The minimum detectable dose for alpha emitters such as plutonium was generally higher when using in vivo methods (relative to the MPBB) than when using bioassay methods of detection, particularly for the years prior to improvements in in vivo detection methods for plutonium (with the incorporation of the phoswhich detector in the late 1960s).*
>
> *The minimum level of detection of the Hanford plutonium monitoring program can be evaluated by contrasting internal doses to the magnitude of typical external ionizing radiation doses. Even though urinalysis is more sensitive than in vivo counting, the dose corresponding to the detection limit is typically still relatively large.*[42]

For example, in 1986, Sula reported the following:

> *…sensitivity of measurement for plutonium-239 urinalysis at Hanford was 0.02 dpm/sample.*[43, 44]

This is also the minimum detectable activity (MDA) reported after 1985 in the TBD (Bihl 2004, page 20). The corresponding effective dose equivalent (Q = 20) for soluble plutonium is 70-130 mrem in the first year, and for insoluble plutonium it is 280-560 mrem in the first year. The total dose would be larger than the first year dose, and some organ doses would be larger than the effective dose equivalent.

In earlier years, 1949–1964, the detection limit and corresponding doses were initially an order of magnitude higher.[45] The first-year lung dose (Q = 10) corresponding to the detection limit for a chest count is in the range of 600 to 15,000 mrem.[46] The large dose range for a single measurement result reflects the dependence of the measurement on the in-growth of ^{241}Am and indicates a significant source of uncertainty in converting chest-count results to dose.

The routine use of in-vivo counting starting in about 1960 (Bihl 2004, page 33) could create significant problems for dose reconstruction, notably for workers who worked with plutonium whose internal dose record consists primarily of in-vivo lung counts. The TBD acknowledges the problem that the MDA for lung counts of plutonium is so large as to make it unusable (Bihl 2004, page 37):

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[42] Mani, K.V., *Assessment of Oak Ridge National Laboratory Internal Dosimetry Data for DOE Health and Mortality Study*. In: Department of Epidemiology. Chapel Hill, N.C.: University of North Carolina at Chapel Hill, 1983;88.

[43] PNL-SA-13657/DE86

[44] Sula, M.J., Carbaugh, E.H., Bihl, D.E. *Technical Basis for Internal Dosimetry at Hanford*. Richland, WA: Pacific Northwest Laboratory, 1989.

[45] K.R. Heid, Manager Personal Dosimetry, Battelle Pacific Northwest Laboratory, Letter to R.H. Mole, Medical Research Council, United Kingdom, June 22, 1979.

[46] PNL-SA-13657/DE86, Table 2.

> *MDAs for direct measurement of ^{238}Pu and ^{239}Pu using the 17 keV x-rays were calculated at times, but the values were extremely large relative to the Maximum Permissible Lung Burden so primary reliance was placed on measuring ^{241}Am and applying a plutonium to americium ratio.*

However, the use of this ratio can also create large errors, especially at Hanford. The ratio of alpha radiation due to plutonium isotopes to that due to ^{241}Am depends of the degree build-up of ^{241}Am from decay of ^{241}Pu. That, in turn, depends on the isotopic composition of the plutonium and on the age of the plutonium in question. Both of these factors are important at Hanford, since a variety of grades of plutonium with varying proportions of ^{241}Pu were produced in the N-reactor. The large implications were apparently recognized by NIOSH, but the recommendation for dose reconstruction of this central problem is given in passing and without analysis or references (Bihl 2004, page 46):

> *Because the uncertainty in the ^{241}Am to Pu alpha ratio can vary nearly two orders of magnitude, plutonium intakes should not be determined solely by chest counting data if possible. If no information concerning the isotopic mixture of an intake is available and a default mixture is assumed, then a GSD of 5 uncertainty should be associated with an intake determined by chest counting.*

This specification for dose reconstruction does not substantively address or analyze the major issues associated with estimates of internal dose due to plutonium using in-vivo counts. Given that Hanford produced various isotopic mixtures of plutonium, there is no persuasive technical reason to use a default mixture, unless it can be shown to be consistently claimant favorable. This would be difficult, since freshly separated plutonium would evidently contain almost no ^{241}Am. It is plausible that the alpha particle emission ratios of plutonium to ^{241}Am could be even greater than 100. Further, the choice of a GSD of 5 applied to a intake that is based on a default assumption about a mixture would appear to compound the problem of estimation that is not well founded on scientific and statistical assumptions that actually relate the radiological conditions and measurement techniques.

The problem of in-vivo measurements also extends to some extent to fission products. Only ^{90}Sr bioassay continued on until 1964.

> *After whole body counting came into routine use, regular use of fission product urinalysis continued for many workers at facilities such as B Plant and Semi-works where the intakes of pure ^{90}Sr were possible. So it was apparently being used as a ^{90}Sr bioassay. The records show fission product analysis being used this way until early 1964. The same workers show actual ^{90}Sr analysis results starting in 1965, probably starting with a new contract with U.S. Testing* (Bihl 2004, page 28).

The internal dose TBD, pages 41 and 42, discusses fission and activation products urinalysis.

> *Fission and activation product mixtures up through 1987 (when N Reactor shut down) were much more complex and variable. The fission product urinalysis procedure measured beta activity from any radionuclides of strontium, yttrium,*

> *barium, lanthanum, cerium, europium, and promethium…there is no straightforward way to determine the true makeup of an intake that resulted in a high fission product urinalysis result or to estimate how to apply the MDAs for missed dose calculations* (Bihl 2004, pages. 41 and 42).

The TBD, therefore, recommended against using the fission product urinalysis results to determine intakes of fission or activation products. Therefore, for the years prior to implementation of whole-body counting, the TBD recommends default intakes based on air sampling. While that is the formal recommendation in the TBD, it also recommends specific default intakes for the years 1948–1952 and 1953–1959 (Bihl 2004, page 42) when in-vivo fission product counting was begun. These intake rates do not correspond to the maximum air concentration guideline or the maximum regulatory value of air concentrations for the two periods. For instance, the suggested value for the 1948–1952 period is 6,600 pCi/day, while the intake calculated from the level at which respiratory protection would be required would be 9,600 pCi/day (at a breathing rate of 1.2 m3/h, setting aside the issue of heavy work in this context).

The TBD does not adequately document why air sampling would be superior to fission or activation product urinalysis in terms of its accuracy of claimant favorability or why default values are recommended for dose reconstruction instead. SC&A believes that during the special campaign years of the 1950s, the lack of such fission and activation product urinalysis data brings into question the adequacy of the data available to do worst-case dose reconstructions or dose reconstruction for the period prior to the start of fission product in vivo counting in a manner that would systematically give the claimant the benefit of the doubt in the face of uncertainty when potential exposure to fission products was involved.

5.4.2 Uncertainties in Bioassay Measurements

The Hanford internal dose TBD indicates that fission product analysis data prior to 1948 should be considered unreliable. The ferrous hydroxide precipitation used in 1947 for routine fission product urinalysis provided erratic results with occasional breakthrough of ^{40}K. The TBD also explains that the fission product urinalysis, although effective in quantifying many counted radioisotopes of alkaline earths and rare earths, did not account for radioisotopes of ruthenium, cesium, zinc, cobalt, manganese, niobium, or zirconium. Workers in 1947, when fission product analysis data are suspect and who had the potential for exposure to the latter unaccounted radionuclides, may have missed doses that may need to be quantified.

The TBD (Bihl 2004, page 45) also notes those uncertainties for the bioassay measurements that were included in the database starting in late 1981 for excreta measurements. The assumption of two weeks uncertainty in the date of a urine collection may lead to errors that can vary from one to three orders of magnitude in the calculation of the intake for type F compounds for known episodic intakes. Errors of one order of magnitude might be possible even for type M. There would be little change in chronic intakes during this time period.

Uncertainties for bioassay measurements prior to 1981, therefore, are not available for worst-case dose reconstructions, and such bioassay data may not be claimant favorable unless NIOSH develops some means of determining these uncertainties.

5.4.3 Uncertainties in Whole-Body Counting Results

The Hanford occupational internal dose TBD (Bihl 2004, page 46) reports the following:

> *For in vivo results, uncertainties were not reported until 1986 for detected radionuclides and 1993 for the default set of radionuclides. These were one sigma counting errors until 1995. Total propagated error was determined and submitted to the records since then. The propagated uncertainty includes counting uncertainty, calibration uncertainty, and a generic 5% positioning error (for both whole body and lung). The calibration uncertainty includes the uncertainty in source activity, counting error, decay correction, and interpolation using the calibration curve. Uncertainty associated with reproducibly positioning a person to get the same result was studied at Hanford and found to be 5%. All calibrations were made using phantoms, and there is considerable uncertainty associated with the representativeness of phantoms versus humans. Just recently a study was done for whole body counting at Hanford using a 95th percentile reference man phantom. There was a low bias of about 20% for the coaxial HPGe detector system for 662 and 1332 keV gamma rays. A similar value of uncertainty (+/- 20% can reasonably be assumed for the other whole body detectors (1-meter arc, shadow shield, and standup counters)).*

The lack of the above uncertainty corrections for whole-body counting prior to 1986 or even until 1993 for default radionuclides means that there can be significant whole-body counting error that could lead to an underestimate of worker dose prior to this timeframe. NIOSH needs to develop a means to address these errors, especially when determining worst-case doses.

5.4.4 Recycled Uranium

The Hanford internal dose TBD (Bihl 2004), Section 5.2.5, is incomplete relative to exposure to recycled uranium. NIOSH should establish a more conservative, claimant-favorable approach that addresses the magnitude of the Hanford recycling operations and the lack of reliable data on workplace exposure, contaminants in recycled uranium, and personnel dosimetry.

SC&A is concerned that NIOSH may not be considering the potential radiation exposure and toxic effects of ^{237}Np and other trace contaminants. The main reason for this concern is that trace radionuclide contaminants were not measured in the routine bioassay program for uranium workers, including those who were exposed to recycled uranium. NIOSH needs to develop dose reconstructor flow sheets that help them address dose related to trace radionuclides in recycled uranium, including ^{237}Np, plutonium isotopes, and fission products, including ^{99}Tc, in light of the following discussion.

Like other sites, Hanford did not measure, control, or protect workers from potentially hazardous contaminants in recycled uranium, such as ^{237}Np. The amounts of unmeasured transuranics in recycled uranium handled by workers probably varied depending on the changes in reactor burnup of uranium, which ranged from 400 MWD/ton to 1,600 MWD/ton; and different recovery efficiencies of several chemical separations plants separating previously irradiated uranium.

Managers at the Paducah Gaseous Diffusion Plant raised concerns over the hazards of 237Np with Hanford officials in 1960. In 1961 and 1962, Hanford researchers reported disturbing evidence of harm from ^{237}Np to laboratory animals. However, no efforts were made to limit this contaminant in recycled uranium or to limit worker exposures. It was only in 1984 that Hanford began to report amounts of ^{237}Np in recycled uranium to receiving sites.

Procedures issued in the 1950s and 1960s for measuring radionuclides in areas where recycled uranium was handled did not require measurements for transuranics. Nor is it clear that workers handling recycled uranium received bioassays or in-vivo measurements for transuranics and fission products especially during the period of peak production between the 1950s and early 1970s.

Beginning in 1952 and for some 35 to 40 years thereafter, Hanford generated and shipped approximately 109,792 metric tons of previously irradiated uranium for recycle.[47] This constitutes nearly 85% of the total estimated 130,000 metric tons of all recycled uranium produced at DOE's four production sites.[48] Recycled uranium is so called because it is recovered from reprocessing plants after it has already been through a reactor one or more times. This creates uranium that contains radioisotopes that are not found in uranium that has not been used in a reactor; virgin uranium contains only ^{238}U, ^{235}U, and ^{234}U. Recycled uranium contains all three of these as well as other isotopes of uranium, notably ^{236}U, and traces of certain fission products and transuranic radionuclides.

Recycled uranium produced at Hanford was sent to several sites, including the Feed Material Production Center in Ohio, the gaseous diffusion plants in Oak Ridge, Tennessee and Paducah, Kentucky, which received about 75,000 metric tons.[49]

According to a 2000 DOE assessment of recycled uranium at Hanford (see Figure 3):

> *The facilities that were identified to have involved the handling of recycled uranium included (1) the 313-314 building complex in the 300 area which was primarily involved in the fabrication of aluminum-clad reactor fuels; (2) the 333 building complex in the 300 Area, which was involved in the fabrication of zirconium-clad reactor fuels;(3) the REDOX chemical separations plant in the 200 West Area where reactor fuels were dissolved for plutonium and uranium (including transfer of the UNH to the $U0_3$ Plant for calcinations), (4) the U plant in the 200 West Area where pre-1952 tank wastes were processed for uranium*

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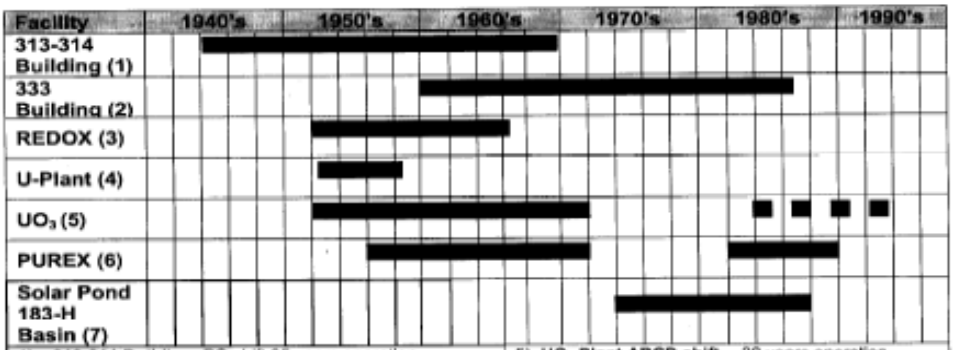
[47] U.S. Department of Energy, *A Preliminary Review of the Flow and Characteristics of Recycled Uranium throughout the DOE Complex 1952-1999,* Project Overview and Field Site Reports, F-001-001, March 2001, Hanford Uranium Mass Balance Project, DOE/RL-2000-43, page 28.

[48] Ibid, page 8

[49] DOE/RL-2000-43, Section 3. page 28. Other sites that received the remaining balance of recycled uranium from Hanford include: Simonds Saw and Steel in New York; Mallinckrodt Chemical Works and Weldon Spring in Missouri; New Brunswick Laboratory in New Jersey; National Lead, Linde Works, Tonowanda, and Sylvania Corning in New York; Battelle Columbus Laboratory, the Portsmouth Gaseous Diffusion Plant, and Harshaw Chemical Company in Ohio; the Y-12 Plant, Oak Ridge National Laboratory, Nuclear Fuel Services, United Nuclear, and Kerr McGee Oil in Tennessee; the Bureau of Mines in Oregon; Bridgeport Brass in Connecticut; Los Alamos Laboratory in New Mexico; the Savannah River Plant in South Carolina; and Gulf United Nuclear Fuels.

recovery, (5) the UO_3 plant in the 200 West Area, where uranium was recovered as UNH was received, concentrated, calcined and packaged for shipment and recycle, (6) the PUREX plant in the 200 East Area where irradiated fuels were dissolved for the recovery of uranium and plutonium (with the UNH shipped to the UO_3 plant by truck for calcination), and (7) the 183 H solar basin in the 100 H-Area that was used for a ten-year interval to evaporate dilute liquid wastes generated at the 300-Area fuel fabrication plants.[50]



1) **313-314 Building** PQ shift 25 years operation
25 operators/shift
10 crafts/shift
average time at facility - 5 years
(25 + 10) x 2 x 25/5 = 350 personnel

2) **333 Building** PQ shift – 25 years operation
20 operators/shift
10 crafts/shift
average time at facility – 5 years
(20 + 10) x 2 x 25/5 = 300 personnel

3) **REDOX ABCD shift** – 15 years operation
Assume 3 employees/shift worked with recycled uranium
average time at facility – 5 years
3 x 4 x 15/5 = 36 personnel

4) **U Plant** ABCD shift – 4 years operation
Assume 3 employees/shift worked with recycled uranium
average time at facility – 4 years
3x 4 x 4/4 = 12 personnel

5) **UO_3 Plant ABCD shift** – 20 years operation
12 operators/shift
8 crafts/shift
average time at facility - 5 years
(12 + 8) x 4 x 20/5 = 320 personnel
Calciner Development ABCD shift – 1 year operation
4 techs/shift
2 crafts/shift
average time at facility – 1 year
(4 + 2) x 4 = 24 personnel

6) **PUREX ABCD shift** – 20 years operation
Assume 4 employees/shift worked with recycled uranium
average time at facility - 5 years
4 x 4 x 20/5 = 64 personnel

7) **183-H Basin** ABCD shift – 10 years operation
Assume 1 employee/shift worked this task
average time at facility – 2 years
1 x 4 x 10/2 – 20 personnel

Note: Shift definitions = ABCD is three 8 hour shifts/day, 7 days/week
PQ is two 8 hour shifts/day, 5 days/week

Source: DOE/RL-2000-43, Section 6. Table 6.1, p.8.

Figure 3. Estimate of Personnel Having Work Potentially Involving Exposure to Recycled Uranium at Specific Facilities

Data collected by DOE in 2000 indicate that between 1,126 to approximately 4,200 Hanford employees are estimated to have been involved in the recycle of uranium.[51]

By 1953, a limit of 10 parts per billion (ppb) for plutonium was formally established at Hanford as a contaminant level for plutonium in recycled uranium.[52] Product specifications for ^{232}U, ^{233}U, and ^{237}Np were discussed in 1953 and later in 1962, but were not adopted.[53] There are at

[50] Ibid, Section 6, page 8.
[51] Ibid, Section 6. page 9.
[52] Ibid, Section 4, page 2.
[53] Ibid.

least five instances where Hanford shipped recycled uranium in excess of the 10-ppb limit.[54] For instance, in 1953 a shipment containing 30 ppb was sent to the Harshaw facility in Ohio.[55]

Despite Hanford's establishment of a contaminant limit for plutonium, a 1985 DOE review concluded:

> *…a formal, technically sound, understood and accepted specification for maximum transuranic and fission product contaminants in uranium recycle material has probably never existed either within or between sites.*[56]

Concern over neptunium and ^{238}Pu contaminants in recycled uranium was expressed as early as 1948 and 1949:

> *If neptunium remains in the uranium during the recycling process, the content of plutonium-238 in product will become important… present standards can be met by the removal of neptunium from uranium during the recycle process. . .* [57]

By February 1949 it was stipulated at Hanford that "recycled uranium be substantially free of neptunium if exposed at 400 MWD/ton."[58] At the time, concern over neptunium was due to its likely conversion to ^{238}Pu during repeated irradiation campaigns.

In the early 1950s exposure levels were increased to 600 MWD/ton and by 1964 went as high as high as 1,600 MWD/ton.[59] Since Hanford was engaged in different production campaigns, such as high-burnup of uranium to produce ^{240}Pu, the amounts of transuranic contaminants not being controlled in recycled uranium probably varied and could have been present in greater amounts depending on reactor burnup and recovery inefficiencies at the reprocessing plants.

Concerns were raised at Hanford in 1956 regarding the impacts of fabrication of recycled uranium in the 300 Area:

> *The question has been raised about the allowable concentrations of plutonium and fission products in uranium salts and metals recovered from the Hanford storage tanks. The answer to the problem lies not in the direct calculation of the hazard involved, but in a balancing of operating procedures to be used in producing the metal and machining the metal after it is produced . . . A concentration of one thousandth of a percent by weight would double the hazard due to inhalation … If the new value for the tolerance concentration is adopted here (as it has been at Argonne National Laboratories and Oak Ridge National Laboratories) it will mean that methods of handling natural uranium will have to be improved … attention should be given to the probable reduction by a factor of*

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[54] Ibid, Section 4, page 5.

[55] Ibid, Section 4, page 6.

[56] U.S. Department of Energy, *The Report of the Joint Task Force on Uranium Recycle Materials Processing, Oak Ridge Operations*, Report No. DOE/OR-859, September 1985, page xi.

[57] Memo to File, P.F. Gast, *Plutonium isotopes in recycled uranium*, HW-12166, January 17, 1949.

[58] Memo to P.F. Gast, From: R.B. Richards, *Plutonium Isotopes in Recycled Uranium*, February 21, 1949.

[59] D.L. DeNeal, *Historical Events, Reactors and Fuel Fabrication*, DUN-3232, November 1, 1967.

> *perhaps 100 fold, in the permissible concentration of both plutonium and uranium. If the feasible purity limit [for plutonium] is one part per million, design changes in metalworking would be necessary… These considerations apply only* **[original emphasis]** *to the Hanford problem. Other locations processing Hanford metal may have worse health hazard problems that would define the preferred approach.*[60]

That same year, corrosion of calcining equipment in the 224 UO_3 facility was enough that, "air contamination had been gradually increasing, and this, together with a lowering of mask working limits has required personnel to wear assault masks continuously."[61]

Between 1960 and 1962, the Atomic Energy Commission (AEC) and contractors operating uranium recycling operations at Hanford and the Paducah, Kentucky gaseous diffusion plant became concerned about the potential hazards of ^{237}Np. In 1957, measurements were initiated at Paducah on a monthly composite sample basis; "The average concentration of ^{237}Np in uranium oxide received from Hanford and Savannah River prior to 1967 was 240 ppb, with a range of 10 to 600 ppb."[62]

Dust samples from processing recycled uranium collected at the Paducah plant were sent to the Biological Laboratory at Hanford in 1960 to determine body retention, organ concentration, and toxicity of ^{237}Np in rats and sheep.[63] By 1962 it was reported that ^{237}Np lodged in the lungs, kidneys, and bone and that the exposed animals showed evidence of kidney, liver, neurological brain damage, as well as unexpected deaths.[64, 65]

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[60] C.N. Gross, Plutonium and fission product contamination in Uranium, HW-10515, May 26, 1956

[61] W.P. Ingalls, K.L. Sanborn, *Reduction of airborne contamination UO_3 plant*, HW 43099, May 11, 1956.

[62] DOE/RL-2000-43, Section 4, page 21.

[63] U.S. Atomic Energy Commission, Letter from [illegible], Manager, Oak Ridge Operations to C.E. Center, Vice President, Union Carbide Corp, July 28, 1960. The dust contained ^{237}Np - 320 ug/g, ^{99}Tc - 194 ug/g, ^{137}Cs - < 1 x 10^4 dpm/g, ^{106}Ru - < 10^4 dpm/g, UX-1* - 2.05 x 10^5 dpm/g, UX-2* - 2.05 x 10^5 dpm/g, ^{90}Sr - < 5 x 10^3 dpm/g, ^{239}Pu - 8.8 x 10^3 dpm/g, Gross beta - 4.1 x 10^6 dpm/g, Gross gamma - 1.9 x 10^5 dpm/g, and * unknown radioisotope.

[64] W.J. Bair, A.C. Case, *Preliminary Studies of Inhaled Dust Containing Neptunium-237*, HW-70940, October 1961 The study found that following inhalation exposure for 45 minutes to submicron particles that the distribution of Np-237 occurred primarily in the lung sand gastrointestinal tract, with lesser amounts translocating in the kidney, liver, muscle, and bone. *Evidence of kidney damage was seen. Kidneys from all rats showed various degrees of change*, the authors reported. According to the study's conclusion, *prolonged or repeated exposures to aerosol concentrations on the order of 1 ug/cc or above10 ug/cc could cause injury in some cases.*

[65] Bair, W. J, Ballou, J. E. and Case, A. C., 1962, *Studies with Neptunium in Rat*, HW-SA-2614, May 25, 1962. *From one to five percent of the material* [Paducah dust] *originally deposited was retained in the lung. Of that absorbed from the lung, a major fraction was deposited in the bone . . . we also fed plutonium nitrate under conditions identical to those employed with neptunium. The measured plutonium absorption was 20 fold lower than observed for neptunium… We feel that this is rather convincing evidence that the absorption of neptunium is substantially greater than that of plutonium.* And, *…the absorption of neptunium from the intestinal tract is substantially greater than has previously assumed – and substantially greater than that exhibited by plutonium . If this observation of higher than previously recognized absorption has the effect of increasing the hazards of ingestion, the results of inhalation studies would suggest the hazard of inhalation is less than previously considered.*

Tentative efforts to establish a bioassay technique for ^{237}Np began in 1960,[66] underscoring the fact that internal dosimetry for ^{237}Np and ^{99}Tc for uranium recycling workers is sparse, if not nonexistent.

According to a 2000 DOE study of recycled uranium at Hanford:

> *Hanford did not routinely analyze $U0_3$ product for neptunium and technetium before 1980.*[67]*… From the earliest records of uranium production at the UO_3 facility, plutonium analyses were required, however the same was not true for ^{237}neptunium.*[68] ^{237}Np measurements were not routinely reported for offsite shipments until the mid-1980s.[69]

A 1964 UO_3 product specification analysis upholds this statement with respect to analytical techniques:

> *Plutonium content in UO_3 powder is determined by extraction and alpha counting of plutonium … Americium, curium and neptunium are quantitatively non-extractable.*[70]

In 1963, ^{237}Np at Hanford was recovered on a "semi-continuous basis," in which the concentration of ^{237}Np in recovered uranium "varied by an order of magnitude," ranging from 40 ppb to 490 ppb.[71]

As the following table suggests, such trace contamination from ^{237}neptunium has been shown to have the potential of very significant radiation doses, if the concentrations are high enough.[72]

Table 4. Estimated Bone Surface Doses from Recycled Uranium to Workers at the Paducah Gaseous Diffusion Plant

(Committed Dose Equivalent – CDE in a Year)

Average Air Concentrations	Maximum Air Concentrations
48.06 -- 188 rem	599.24 -- 2,238 rem

Source: Footnote 72.

The REDOX processing plant had widely varying decontamination factors (DFs), which resulted in some instances in significantly greater contaminants in the feed that was processed at the $U0_3$ plant. This increased exposure hazards to the workers. In 1953 and 1954, the DFs, which

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[66] R.W. Perkins, *A Tentative Bioassay Procedure for the Measurement of Np-237*, HW-67067, October 10, 1960.

[67] DOE/RL-2000-43, Section 6, page 3.

[68] DOE/RL-2000-43, Section 4, page 18.

[69] Ibid.

[70] C.L Downey, W.H. Zimmer, Uranium Trioxide Specification Analysis, HW-65402, May 5, 1964, page 7.

[71] DOE/RL-2000-43, Section 4

[72] U.S. Department of Energy, Office of Environment, Safety and Health, *Exposure Assessment Project at the Paducah Gaseous Diffusion Plant,* December 2000, page 77

controlled the amounts of radionuclide contaminants in the UNH feed from the REDOX plant, ranged from 1.6 to 16.[73] As a result, from 1954 to 1955 the calciner "pot exposure (without shields) has increased from about 30 mr/hr to 1,500 to 2,000 mr/hr."[74]

Certain activities, particularly bag-filter changes at the $U0_3$ plant posed greater risks of exposure to recycled uranium than the calcining pots. According to a 1957 exposure assessment of bag house workers, "The work is necessarily done on hands and knees and is accompanied with a noticeable cloud of UO_3 dust,"[75] Dust samples collected after a radiation incident in the bag house indicated that: ". . . the weighted average exposure for a 35-minute period in the X-3 bag house was 6.8 x 10^{-7} uCi/cc, which is more than 1000 times the permissible limit."[76]

Impurities such as neptunium, plutonium and fission products present in recycled uranium have been identified by NIOSH in Table 5.2.5-2 of the Occupational Internal Dose TBD (Bihl 2004), page 24. The TBD on page 23 states:

> *Recycle[d] uranium also had impurities build up and track with the uranium over time. Impurities can be approached in two ways, representative levels based on averages of several measurements at different times and upper limits based on tolerance specifications (e.g., not to exceed). Both of these approached are given in Table 5.2.5-2.*

However, this table is incomplete when compared with radionuclides of concern recommended by a special task force on recycled uranium convened in 1985 by the U.S. Department of Energy, which include ^{239}Pu, ^{237}Np, ^{99}Tc; ^{103}Ru, ^{106}Rh, ^{125}Sb, ^{95}Zr, ^{95}Nb, ^{232}U, ^{233}U, ^{236}U, and ^{237}U.[77] Moreover, despite the availability of some early data on trace element content of recycled uranium, the TBD cites 1988 and 1993 data to indicate that neptunium and plutonium content of Hanford recycled uranium was low (Bihl 2004, Table 5.2.5-2, page 24). This is misleading and inappropriate in that data on recycled uranium, including that cited in this review, indicate that out-of-specification trace element content was more likely to have occurred in the years and decades prior to the measurements cited in the TBD. Hence, the use of maximum allowable concentrations for trace contaminant content of RU is not a claimant-favorable approach for estimating RU dose.

The potential for high doses resulting from work around recycled uranium would appear to be significant, particularly in the S-224 Uranium Trioxide facility and 300 Area operations. The TBD states that the uranium urinalysis program prior to 1948 was not reliable. It wasn't until 1983 that alpha spectrometry and the PNNL isotopic analysis was introduced and began to be used. The TBD does not provide adequate data to properly determine individual or group dose from recycled uranium. The TBD points out that in approximately 1995, mass spectrometry was used as an investigational tool to discriminate between natural background uranium and recycled

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[73] R.B. Abrams, L.E. Brums, Radioactive Decontamination Across the UO3 plant, HW-34808, January 27 1955.

[74] Ibid.

[75] W.E. Gill, *UO_3 Exposures During Bag Filter Changes at the Uranium Reduction Plant, 224U Building*, HW-48505, February 15, 1957.

[76] Ibid.

[77] DOE/OR-859, page 68.

uranium through measurement of ^{236}U. The presence of ^{236}U confirms occupational intake of recycled uranium.

5.4.5 300 Area Facilities Missed Dose

Section 5.7 of Bihl (2004) provides direction to dose reconstructors on the assignment of internal dose to unmonitored workers. Internal doses for those who wore dosimeters, but did not have bioassays, are based on facility assignment and time period, where this information is available. Bihl (2004, pp. 51–52) lists recommended intake assumptions for the following 300 Area facilities.

> ***300 Area Test Reactors, 1944–1978:*** *Use particulate beta/gamma and tritium intakes. See Table 5.7-2 to determine the radionuclide that maximizes the dose to the organ of concern….*
>
> ***300 Area Fuel Fabrication Facilities (303, 306, 313, 314, 333), 1944–1988:*** *Use insoluble uranium intake. Input as ^{234}U (M or S).*
>
> ***308 Plutonium Fuels Pilot Plant, 1960–1990:*** *Use particulate beta and particulate alpha intakes. Use Table 5.7-2 for beta/gamma intake (reactor) and ^{239}Pu (M or S) for alpha intake.*
>
> ***324 Building, 1966– 2001:*** *Use particulate beta/gamma and particulate alpha intakes. Use ^{137}Cs (F) or ^{90}Sr (F) for beta intake and ^{239}Pu (M or S) for alpha intake.*
>
> ***325 Building, 1953–present:*** *Use particulate beta and particulate alpha intakes. Use Table 5.7-2 for beta/gamma intake (waste management) and ^{239}Pu (M or S) for alpha intake.*
>
> ***327 Building, 1953–1987:*** *Use particulate beta/gamma and particulate alpha intakes. Use Table 5.7-2 for beta intake (reactor). Input the alpha intake as either ^{234}U (F) or ^{239}Pu (M or S).*

Gerber (1992) compiled historical information from the 300 Area activities and facilities since the inception of operations. Bihl (2004) assumed that the primary radionuclide of interest in the 300 Area was uranium, with the exception of the 300 Area Test Reactors, 308 Building, 324 Building, 325 Building, and 327 Building (Bihl 2004, pp. 51–52). Although uranium is common in the 300 Area, it certainly should not be considered the only radionuclide of interest outside of the exceptions noted above. Table 5 provides examples of buildings where internal exposures are not adequately addressed in the TBD.

Table 5. Select 300 Area Buildings and Potential Sources of Internal Exposure
(Gerber, 1992b)

Building	Potential Sources of Internal Exposure
306, Metal Fabrication Development Building	Uranium and thorium sludge in sewer lines (page 17).
3708, Radiation Measurement Building	This building was converted to a fuel fabrication facility and began producing neptunium oxide fuel targets. In the early 1970s, a portion of the building was used for experimental canning of americium oxide and curium oxide fuel blends (page 41).
3722, Area Shops	From 1968-1970, palletized thorium oxide fuel targets for ^{233}U production were fabricated. The building also housed a furnace for recycling of depleted thorium after it was processed in PUREX. During the period of operation, ruthenium boiled off from the thorium oxide sintering furnace (page 51).
3732, Process Equipment Development Laboratory	From 1965–1967, powdered thorium oxide fuel targets for ^{233}U production were fabricated. This spread fines and particulate contamination throughout the building (page 55).
321 Separation Building	This facility was heavily involved in wartime efforts to improve the separations process. Pilot tests for the REDOX process were run in the 321 Building. Other operations included extraction of high-heat isotopes from waste (^{90}Sr, ^{137}Cs, ^{144}Ce, ^{147}Pm, and ^{237}Np) (pp. 73-78).
3706, Technical Building	The original mission was to perform small- and large-scale experiments with both low- and high-activity radioactive material. From 1947 to 1953, much of the radiochemical development of the REDOX, PUREX, and RECUPLEX processes took place. Wastes and contamination include plutonium, uranium, and thorium (pages 83-86).

The TBD (Bihl 2004) has clearly not addressed all of the radiological hazards associated with operations in the 300 Area. In some cases, buildings are absent from the TBD. Additional evaluations of the operations and radiological constituents, both past and present, should be completed. It is pertinent to note that operations in the 300 Area should include the burial grounds and trenches that are currently under remediation.

There are also discrepancies in the radionuclides given in the site description for Buildings 303, 305-B, and 309. The 303 facilities were used for metal storage, including fresh uranium, chemicals, uranium scraps, and plutonium. PNNL used the 303-C Building for storage of plutonium and americium. There was a documented spread of plutonium contamination that shut down the building (Selby 2004, page 14). Dose assignments for the 303 facilities only considered ^{234}U. An accident occurred at the 305-B Physical Constants Test Reactor that could be a source of minor internal exposure to plutonium during cleanup (Selby 2004, page 9). The Plutonium Recycle Test Reactor (309 Building) tested various mixed oxide fuels, including a variety of powdered and pelletized fuels using plutonium oxide blended with uranium oxide and other metallic oxides (Selby 2004, page 9). These facilities are considered 300 Area Test Reactors, resulting in an assignment of tritium and particulate beta/gamma dose with no alpha dose assignment. Each section of the site profile should be consistent. If radionuclides mentioned in the site description are to be excluded from internal dose consideration, clarification should be provided to the dose reconstructors.

5.4.6 Completeness of Data

Bihl (2004) uses ORIGEN 2 code calculations to provide information on the most predominant radionuclides of interest at the facilities. Attachment D, Table D.2.1.1-1, provides the relative abundance of fission and activation products in Hanford fuel at the time of dissolution. The footnote to the table indicates that all the activity data were obtained from ORIGEN 2 calculations performed for the Hanford environmental dose reconstruction. The Hanford site description TBD (Selby 2004), Appendix A, pages 30 through 37, discusses the years of operation and radionuclides of concern (Tables A1-1 and A1-2, respectively). Table A1-4, page 31, provides a list of radionuclides of concern for all research and test reactors. Table A2-2, page 32, provides the radionuclides of concern for all reprocessing facilities. Table A3-2, page 34, provides the radionuclides of concern for fuel fabrication and storage facilities. Impurities in recycled uranium are identified in Table A3-4, page 35, and in this case the footnote indicates that the source is from Thompson (1986).[78] Table A4-2, page 37, provides radionuclides for plutonium finishing facilities. In all but one case, these tables do not provide references for the origin of the radionuclides of concern. It is difficult, therefore, to ensure that these tables are, in fact, complete with respect to the types of radionuclides present at each of the facility categories.

The TBD does not document and discuss the application of other data, such as field characterization data, to better indicate the radionuclides encountered by workers and the relative ratios of isotopes.

NIOSH needs to provide more complete guidance to dose reconstructors to ensure complete dose estimation for workers in reactor areas exposed to activation products and potential leaks from fuel rods, especially in fuel storage pools. Fuel failures, irradiation of lithium targets, and beam streaming all resulted in many off-normal high radiation doses in reactor areas. The TBD does not address such cases and does not provide guidance to dose reconstructors on these kinds of exposures. Even the exposure to activation products was uneven and depended on which reactor component a worker had recently worked on.

In determining worst-case chronic intakes, Bihl (2004, page 47) indicates that fission and activation products were ignored if they constituted less than 1% of an MPBB, and only a few fission and activation products were thought to contribute to dose:

> *For whole body counts, it is unreasonable to assume that a worker was exposed to all the radionuclides potentially reportable simply because an MDA was determined; on the other hand, for years Hanford only reported ^{24}Na, ^{65}Zn, ^{40}K, and ^{137}Cs of which only ^{137}Cs is of concern to dose reconstruction, while other fission and activation products were ignored if less than 1% of a MPBB. A recommended approach would be to use an indicator radionuclide to determine intake, then add only whole body count radionuclides in the mixture, as discussed in Section 5.4. Since 1987 or 1988 the only whole body count radionuclides of potential exposure have been ^{137}Cs and ^{154}Eu (maybe ^{60}Co into the early 1990s).*

This leaves many radionuclides unaccounted for, especially in the early days prior to startup of whole-body counting in 1959 or 1960. Although each may not contribute a significant dose,

[78] Thompson, 1986. UO_3 Plant operating specifications, OSD-U-185-0001.

considered together they may represent a potential for missed dose that could be important when reconstructing worst-case doses. A report listing the radionuclides detected in workers at the whole-body counter facility in 1961 included ^{24}Na, ^{60}Co, ^{65}Zn, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{99}Tc, ^{103}Ru, ^{106}Ru, ^{131}I, ^{137}Cs, and ^{144}Ce (Henle 1962). A similar report summarizing 1961-1963 results added ^{46}Sc, ^{51}Cr, and ^{59}Fe to the list.

It is noted in the Hanford site description TBD (Selby 2004), Attachment A, that radionuclides used for internal dose calculations and the radionuclides of concern listed in the site description are different for the same facilities. For example, Table A.1-2, Attachment A, page 31, lists the radionuclides of concern for all production reactors as ^{76}As, ^{41}Ca, ^{58}Co, ^{60}Co, ^{51}Cr, ^{64}Cu, ^{59}Fe, ^{72}Ga, ^{133}I, ^{135}I, ^{54}Mn, ^{56}Mn, ^{24}Na, ^{97}Nb, ^{239}Np, ^{32}P, ^{31}Si, ^{91}Sr, ^{92}Sr, ^{238}U, ^{90}Y, ^{92}Y, ^{93}Y, ^{65}Zn, and ^{69}Zn. Bihl (2004, page 43) lists ^{54}Mn, ^{58}Co, ^{60}Co, ^{59}Fe, ^{51}Cr, ^{46}Sc, and ^{99}Mo, ^{154}Eu, and ^{155}Eu are also mentioned for N reactor. NIOSH should explain how it defines "radionuclides of concern" and should resolve the differences between the two TBDs.

5.4.7 Consistency Among Site Profiles

SC&A performed a comparison between the methodologies used in the Hanford and SRS TBDs to determine internal dose. This comparison focuses on the methodologies and assumptions associated with dose determination and the values used to obtain a POC. The detailed comparison of assumptions is provided in Attachment 6. NIOSH should ensure that consistent procedures are used, particularly when dealing with two DOE facilities with many similar operations and similar radionuclides.

The Hanford TBD includes a discussion on recycled uranium and the impurities associated with that operation at Hanford (Bihl 2004, page 24). Hanford's treatment of recycled uranium, for instance, attempts to treat dose potential from this source, whereas this was not addressed well in the SRS Site Profile. As pointed out in Section 5.4.4 of this report, however, even though Hanford made an attempt to address exposures from recycled uranium, there are still many other areas at Hanford where the magnitude of the Hanford recycling operations, the lack of reliable data on workplace exposure, and contaminants in recycled uranium need to be further developed in order to facilitate claimant-favorable worst-case dose estimates for workers exposed to recycled uranium. The complete neglect of the consideration of recycled uranium in the SRS TBD produces an inconsistency between sites. A consistent criterion for inclusion of impurities in organ dose assessments should be developed and applied for DOE and AWE facilities. A formalized complex-wide policy for impurities in recycled uranium was not in effect until the later years of processing. As a result, careful consideration should be given to limits established by individual sites and their adherence to these limits during receiving and shipping of recycled uranium.

5.4.8 Adequacy and Interpretation of Monitoring Data

The conversion from Hanford's solubility studies to ICRP absorption types was conducted in the 1980s. There is no guarantee that the uranium compounds used prior to the 1980s possessed the same solubility as those tested in the 1980s. Processes may have changed and some plants were shut down before the solubility tests were conducted. Thus, the use of the lung absorption types recommended in Table 5.2.5-3 (Bihl 2004, page 23) and in note "b" to that table may not be claimant favorable for all periods

^{233}U was handled at the 231-Z Building in the mid-1960s as a special project, possibly extending into the early 1970s (Bihl 2004, page 25). There is no instruction to dose reconstructors on how to deal with potential exposures to ^{233}U that are not documented in the claimant's records.

Regarding excretion of natural uranium from a worker's diet, Bihl (2004, page 27) states, "only urinary excretion values greater than 0.2 μg/d, which converts to 0.15 dpm/d for ^{234}U and ^{238}U and 0.007 for ^{235}U, are considered indicative of a potential occupational source." This is much higher than the value usually employed for the natural excretion of uranium, less than 40 ng/d (Stradling 2002). Thus, the use of 0.2 μg/d as an implicit background level is not claimant favorable.

The guidelines to determine intakes from fission product urinalysis results at the separation plants are provided by Bihl (2004, pp. 43–44). NIOSH should provide an explanation as to how the recommended intake values were calculated.

NIOSH instructs the dose reconstructor to use Table 5.7-2 to determine the radionuclide that maximizes the dose to the organ of concern. SC&A finds that some of the values in Table 5.7.2 are not consistent with those provided in the ICRP CD-ROM on the ICRP database of dose coefficients, *Workers and Members of the Public* (ICRP 2001). For example, ^{60}Co type S should be the radionuclide of concern for 50-year committed doses to the breast and esophagus (not ^{106}Ru type F, as specified in Table 5.7-2). For some organs, the time between exposure and the calculation of doses is important in relation to the choice of the radionuclide of concern. Thus, it is not considered claimant favorable to use the radionuclides in Table 5.7-2 for all intervals of time between the exposure and the cancer appearance.

Bihl (2004, page 49) states the following:

> *Assumptions on intakes for workers who wore a dosimeter but did not have any bioassay . . . depend on the time period of exposure and the facility or general category of facility they accessed…. For the years following the periods cited in Section 5.1, the DR should base the intake assessments on the daily intake values presented in Table 5.7-1.*

Table 5.7-1 specifies values to be used from "Start to 1946," with a footnote to see Section 5.1 for facility-specific dates, 1947-1952, 1953-1967, 1968-1988, and 1989-present. NIOSH should specify more clearly which data to use for reactor workers in 1947 and 1948, and for the 300 Area uranium fabrication buildings in 1947.

5.4.9 Contamination Spreads Create the Potential for Missed Dose

Bihl (2004) provides no specific information regarding spreading of contamination in the reactor building, 231-Z Plutonium Isolation Facility, concentrator buildings, or uranium metal fabrication shops during the period from 1943–1946. NIOSH should identify and review such records, which might show additional dose to workers. A discussion should be included as to why other radionuclides such as ^{137}Cs or ^{14}C have been excluded from the list of radionuclides. NIOSH should also consider if intakes of ^{90}Sr may have been higher than 200 nCi per year.

5.5 ISSUE 5: ENVIRONMENTAL DOSE

Atmospheric releases were often episodic, although the internal dose TBD handles them as chronic. Since different isotopes were released from the reactors at different times, the meteorological conditions at the time of the releases could have been substantially different. Therefore, the screening process may be flawed for the development of worst-case doses.

5.5.1 Hanford's RATCHET Puff Advection Model Could Be Put to Greater Use in Reconstructing Dose from Acute Daily Episodic Releases

The Hanford occupational environmental dose TBD (Savignac 2003), describes the use of the RATCHET puff advection model. However, the model appears to have been used to evaluate doses corresponding to continuous, uniform, annual releases. More claimant-favorable environmental doses might have resulted if RATCHET were used for dose estimation of daily intakes from episodic releases.

Table 4.3.1-1 (page 21) of the TBD (Savignac 2003) presents a thorough compendium of the external dose rates as a function of year and locations at the site, based on data collected in quarterly environmental monitoring reports beginning in 1945. This is a useful compendium of data for reconstructing outdoor onsite doses to workers due to external radiation related to airborne effluents from the facilities. The tabulated dose rates appear to be claimant favorable because natural background radiation is not subtracted from the values.

Airborne sampling of particles was not used as the basis for deriving inhalation exposures. Instead, atmospheric transport modeling was utilized. A legitimate question can be raised regarding the contribution to dose from resuspended particles. Page 26 of the TBD states that the inhalation dose from resuspension of deposited radionuclides was not explicitly considered in the TBD. Some discussion is needed justifying the exclusion of this exposure pathway.

NIOSH used the atmospheric transport model RATCHET to model exposures to unmonitored outdoor workers resulting from airborne emissions from the various facilities at Hanford. RATCHET provides a scientifically robust method for deriving doses to workers from atmospheric releases. The use of the RATCHET code represents a significant improvement over a Gaussian model, which uses average annual meteorology, such as that employed in the SRS.

Using meteorological data, which can be obtained from multiple meteorological towers, RATCHET can create a wind field that changes as the meteorological conditions (wind speed, direction, and stability class) change. Hourly emissions from individual release points can be used as input to the code, and the code can track each 1-hour puff moving as a Gaussian puff within a changing wind field. The advantage of this type of puff advection model over a model that employs conventional average annual Gaussian models is that it can take into consideration the wind speed, direction, and stability class at the time of an episodic release, as opposed to averaging the releases over the course of a given year and using average annual meteorology. As such, RATCHET is a preferred code for modeling not only chronic, but also episodic releases.

It is not apparent, however, whether the code was actually employed in this manner to perform the calculations described in the TBD. For example, there apparently were a number of

relatively large, short-term, ground-level and elevated atmospheric releases at Hanford. However, it is not apparent whether the doses from these releases at the various receptor locations were modeled as hourly releases or whether they were simply input as part of a continuous, uniform, annual release. Tables A-1 through A-21 of the TBD (Savignac 2003) indicate that all releases are expressed in terms of Ci/year from the various facilities. In addition, Table A-10 presents atmospheric dispersion factors in units of Bq – (sec/m^3) per Bq released at the 38 receptor locations for releases from each of 15 point sources at the site. It would seem that, based on inspection of these tables, all releases were treated as annual releases and all atmospheric dispersion factors were treated as average annual values. Therefore, it is not clear that NIOSH took full advantage of the power of RATCHET to model episodic releases.

Annual internal exposures from inhalation are discussed in Section 4.2.3 of the TBD and are derived for discrete locations based on: (1) annual radionuclide emissions, (2) annual average atmospheric dispersion coefficients, and (3) breathing volumes.

While annual emissions and the assumed annual inhalation volume of 2400 m^3 are reasonable, what may not be reasonable are the assumed atmospheric dispersion factors provided in Table A-10. For example, the lowest dilution values for ^{131}I (released from T Plant and B Plant) at the closest receptor locations (Nodes 1 and 2) are given as 1.18E-07 Bq-sec/m^3 and 1.35E-07 Bq-sec/m^3, respectively. These receptor locations are estimated to be no more than 500 meters from the point of atmospheric discharge. For other onsite locations, values of 10^{-8} to 10^{-10} Bq-sec/m^3 are common.

One of the critical parameters that affect the dispersion factor is the effective release height. The TBD provides no site-specific information regarding the exhaust stacks for the T Plant and the B Plant. In Section 3.2.1 (page 3-10) of Till 2002, however, reference is made to an assumed "physical height of release stacks" of 61 meters.

Accordingly, additional information and data are required to verify NIOSH's assumptions and methods used to derive the annual average atmospheric dispersion factors given in Table A-10. For ground or near-ground releases, χ/Q values that are several orders of magnitude greater than those given in Table A-10 of the TBD could easily apply. For example, Table 6 provides generic average annual χ/Q values as a function of distance and height of release that are significantly larger than those cited in the TBD. The values were derived using the recommendations in NCRP Report 123 (NCRP 1996) for stack releases that are 2.5 times higher than the height of nearby buildings to avoid building wake effects. SC&A used the recommended default windspeed of 2 m/sec and assumed that the wind blows toward the receptor 25% of the time. (It is our understanding that the prevailing winds at the Hanford Site are west to east.) The results reveal that the χ/Q values used in the Hanford Site Profiles are appropriate only if the releases were relatively uniform over the course of a given year and that the releases were from a stack with a height of about 100 meters. If any of the releases were, in fact, closer to ground level or highly episodic, the doses are likely to have been underestimated by over one order of magnitude.

Table 6. Generic Average Annual Dispersion Factors

	Average Annual χ/Q (sec/m^3) Release Height (m)			
Distance (m)	**0**	**10**	**50**	**100**
500	2.50E-05	2.50E-05	3.13E-06	5.00E-07
1000	7.50E-06	7.50E-06	3.13E-06	5.00E-07
2000	2.50E-06	2.50E-06	1.25E-06	5.00E-07
3000	1.13E-06	1.13E-06	7.50E-07	4.38E-07
4000	7.50E-07	7.50E-07	5.00E-07	3.75E-07
5000	5.00E-07	5.00E-07	3.75E-07	2.50E-07

5.5.2 Episodic Releases May Represent a Significant Potential for Missed Dose

The atmospheric source terms used by NIOSH to reconstruct outdoor exposures to onsite workers were taken from reports by Till et al. 2002 and Heeb 1994. Section 4.2.1 of ORAUT-TKBS-0006-4 (Savignac 2003) cites a study by Heeb (1994), which provides estimates of the releases of ^{41}Ar, ^{14}C, and ^{3}H from reactors based on historical measurements. The TBD states that screening analyses eliminated ^{3}H and ^{14}C from further consideration because the dose from ^{41}Ar dominates. NIOSH made this determination by considering the quantity of the radionuclides released and the screening factors recommended in NCRP 1996. A review of NCRP 1996 confirmed the screening factors and the validity of eliminating ^{3}H and ^{14}C from further consideration. However, a question that might arise related to this screening process is whether the releases of the three radionuclides occurred at the same time. If the releases from each reactor occurred at the same time, the screening decisions are unquestionably valid. However, if the releases were episodic, and the different isotopes were released from the reactors at different times, the meteorological conditions at the time of the releases could have been substantially different. If so, then the screening process may be flawed for the reasons discussed below regarding atmospheric modeling of episodic releases.

Page 10 of the TBD (Savignac 2003) discusses routine releases from the T and B reprocessing plants. Apparently, the releases were associated with corroding ductwork in the ventilation system. Given that the ductwork was corroding, it is possible that some of the releases were unmonitored and occurred at ground level, as opposed to being released from the stack. Some discussion of unmonitored, episodic, and/or ground-level releases is needed because, if such releases occurred and were not accounted for, they could represent an important contributor to onsite exposures, even if the quantity of the material released in this manner was a small fraction of the quantity of material released via the stack.

Of particular concern are the releases of plutonium from the T and B reprocessing plant from 1945–1951. Neither the TBD nor the dose reconstruction performed by Till et al. (2002) have reconciled estimates of very large doses from inhaled plutonium and radiocerium developed by site health physicists at the time of these early releases. This

problem was first brought to light in 1989 in a majority staff report of the U.S. Senate Committee on Governmental Affairs.[79] According to excerpts from that report:

> *The problem was first discovered in September 1947 when highly radioactive particles containing radiocerium and plutonium were found on the ground near two plutonium chemical separations facilities… The leak occurred as a result of corroded fan-duct work in the stacks of the two plants and was estimated to have started about six months before it was discovered. A March 1948 report prepared by Dr. Herbert M. Parker, Director of the Hanford Health Instruments Department of the General Electric Company…. stated that contamination 'has been increasing quite rapidly prior to its discovery.' Urgent measures, stated Parker, were taken to 'avoid the risk of corrosion to the point of collapse.' [Parker, H.M., Review of the Stack Discharge Active Particle Contamination Problem, March 22, 1948, HW-9529, p .4]*
>
> *Data collected on the particles in October 1947 indicated that the activity ranged from 0.06 to 1.2 microcuries of total beta activity and from 0.072 to 2.6 nanocuries of total alpha activity per particle. The average activity per particle was 0.77 microcuries total beta activity and 0.61 nanocuries total alpha activity. About 50 percent of the beta activity was determined to be from radiocerium and 77 percent was determined to be from plutonium. [Mickelson, M.L., Preliminary Report on Existing Particle Hazard- 200 Areas, October 22, 1947 HW-7862] A March 1948 report prepared by Dr. Herbert M. Parker stated that, 'specks showing alpha particles have a plutonium to uranium ratio of 20:1' [HW-9529, p. 21]. At the time Parker estimated that an average of about 7.4 billion particles were being released from the two plants per month. [Atomic Energy Commission Advisory Committee on Biology and Medicine, Minutes of the Twelfth Meeting, Richland WA, October 8, 9, 1948]*
>
> *According to Parker, 'The critical hazard is to the inhalation and lung retention of particles' and that 'the available active particles can produce radiation damage.' [HW-9529, p.4] At the time, it was estimated that a worker could be inhaling about 16 particles per month. Based on particle data collected in 1947, this suggests that over a year, a worker could have possibly inhaled about 90 nanocuries of plutonium – an amount which is twice the current official life-time lung burden allowed for DOE workers which was set in 1960.*
>
> *Unfortunately, health and safety officials discovered that repair of the defective duct work did not prevent the discharge of small long-lived particles. According to an internal Hanford memo prepared in 1948, the corroded duct work was responsible for 'only a fraction of the total long-lived emissions from the stacks.'[HW-9529, p .4]. After replacing the fan ducts, Parker reported 'another serious problem' in the form of 'ultrafine' particles that were leaking out of the T and B stacks since 1945. 'Removal of the offending duct work eliminated the large*

[79] U.S. Senate Committee on Governmental Affairs, Majority Staff Report, *Early Health Problems of the U.S. Nuclear Weapons Program and their Implications for Today*, December 1989.

particles but had relatively little influence on the emission of smaller particles.[HW-95295, p. 15]

Moreover, Parker reported that contamination extended to "proposed sites for future expansion." According to Dr. Parker's 1948 report, "it will be noted that depositions in Areas D ,E, F, G greatly exceed the previously reported surface concentration originally defined in zone 3 { the outer boundary area for onsite contamination}. 'There is some evidence that the apparent increase of concentration with increasing distance from the stack is real' [HW-95295, p 13]. According to Parker, " if the data {in the outer edge of contamination} are nearly correct there is a backlog of tens or hundreds of millions of particles unaccounted for in the plant vicinity. [HW-95295, p .14.

By 1948, the 'particle problem' was brought to the attention of the Atomic Energy Commission's (AEC) Advisory Committee on Biology and Medicine (ACBM)… At the meeting a consultant suggested that radioactive particles would give lung doses ranging between 0.5 reps and 10 reps per day. [Atomic Energy Commission Advisory Committee on Biology and Medicine, Minutes of the Twelfth Meeting, Richland WA, October 8, 9, 1948]. An earlier dose estimate by Parker done in March 1948 suggested 'an order of magnitude' estimate of doses ranging from 130 reps per day to 2,750 reps per day. [HW-95295, p. 32]

Page 13 of the TBD (Savignac 2003) addresses nonroutine releases from the REDOX plant. This section specifically addresses ammonium nitrate flakes up to 10 cm in diameter that were released from the stack of the REDOX plant from 1952−1954, and which could produce surface contact doses of 20 rad/hr due to the presence of ^{103}Ru and ^{106}Ru in the flakes. Later in Section 4.3.3 of the TBD, additional discussion is provided of the release of ammonium sulfate flakes also containing ^{103}Ru and ^{106}Ru from the T and B reprocessing plants. It appears that the TBD limits the evaluation of the doses from these flakes to external doses to skin from the flakes falling on nearby workers. Two questions arise with respect to these exposures. First, could these flakes have fractured into smaller particles and have been inhaled or inadvertently ingested? Second, in deriving the skin dose, NIOSH applied a usage factor of 0.1 to account for the fact that most of the flakes would fall on a person's clothing, and the clothing would shield the skin from the beta emissions. Attachment 8, which presents an evaluation of the beta emissions from ^{106}Ru, reveals that the progeny of ^{106}Ru, (i.e., ^{106}Rh) decays by a relatively strong beta emitter that may not be shielded by clothing. This issue should be addressed.

The Hanford occupational internal dose TBD (Bihl 2004) treats ^{106}Ru/Rh releases as chronic, when in fact only a smaller fraction of the releases were chronic and large episodic releases occurred over a period of hours to one or two days. Treating these brief episodic airborne releases as chronic releases may result in large underestimates of the doses.

The Hanford occupational environmental TBD, (Savignac 2003, page 19) states the following:

The net gamma exposure rates are mainly due to the ^{131}I and ^{106}Ru release.

Attachment 1 of the TBD (Savignac 2003) provides annual release quantities in Table A-7 (page 9) for 1953 of 176 and 353 Ci, respectively. The TBD-prescribed methodology for the

inhalation exposure to ^{106}Ru/Rh, therefore, would use (1) the annual release rate to derive an average hourly release rate, (2) the average annual atmospheric dispersion factor, and (3) the total annual inhalation volume of 2400 m^3 to derive inhalation quantity.

A review of HW-34882 (Irish 1955) identifies the fact that only a smaller fraction of ^{106}Ru/Rh releases were chronic and that large episodic releases occurred over a period of hours to one or two days. As further stated in Issue 4 of Attachment 1:

> *In brief, the TBD's treatment of these brief episodic airborne releases as chronic releases results in low estimates for the following reasons:*
>
> - *When treated as chronic releases, the exposure is reduced by a factor of 4.38 (i.e., 8760 hours per year release/2000 hour-year exposure).*
>
> - *When treated as chronic releases, the use of the average annual atmospheric dilution factor will severely underestimate actual air concentrations that existed in downwind sectors during the brief episodic emissions. For an episodic release, exposures may have resulted under conditions that yielded atmospheric dispersion values that were several orders of magnitude lower.*

Furthermore, as discussed above, fallout resuspension and inhalation/ingestion of ^{106}Ru/Rh were not addressed in the TBD (Savignac 2003) and may have resulted in large doses if they involved hot particles as described in the TBD.

A status report done on a ground contamination problem at REDOX in 1954 (Parker 1954) indicated that:

> *Since January 1953 there have been several emissions of radio-ruthenium particulates from the Redox stack that have collectively established a ground contamination problem pattern more severe than any that has been observed at this site in previous years…. As a basis of point, a particle reading 100 mrads per hour on a routine survey has anticipated linear dimensions of about 40 microns, contains about 1.1 uCi $Ru^{103,106}$ and can give a skin contact dose of about 160,000 rads in 24 hours. It is estimated that this particulate irradiation would be more nearly equivalent to 200-2000 rads (perhaps ~ 300 rads) as conventionally given over a small extended area (a few cm^2). Inhalation and ingestion hazards are also considered; with some reservations these are likely to be critical under existing conditions or contamination.*

This same report provides a status of several emissions of radio-ruthenium particulates from the REDOX stack documents for which the ground contamination pattern was becoming more severe than in previous years (Parker 1954). This report elaborates on these ground contamination problems and illustrates the episodic nature of the releases:

> *(1) March 8, 1952 failure of caustic scrubber. local contamination*
> *(2) April 3, 1952 high active particle (40 rad/hr) found on survey instrument*

(3) April 29, 1952	*widespread contamination in Redox area readings up to 800 mrad/hr program of examining deposition on glass fiber mats began*
(4) June 24, 1952	*large flaky particles of ammonium nitrate carrying Ru contamination*
(5) September 1952	*30-fold increase in particle deposition*
(6) August 14, 1953	*large fragments of ammonium nitrate. up to several inches length and width and ¾ inch thick. curvature compatible with origin in stack liner. dose rates up to 15 rads/hr*
(7) Sept. 5,6, 1953	*approximately 115 curies of ruthenium released. no specific deposition problem reported*
(8) Jan. 2, 1954	*approximately 250 curies released. narrow band of contamination northeast, detectable as far as Spokane. general ground contamination up to 7.5 rads/hr in vicinity of Redox plant. Access controls applied.*
(9) Jan. 5, 1954	*approximately 70 curies. possibly caused by stack flushing. contamination local to Redox area.*

Other examples of large episodic environmental releases include fuel element failures in the 100 Area; waste site releases; the Green Run, T&B Plant stack releases from 1945-1951; REDOX particle releases from 1952–1954; ^{131}I releases, tritium releases (purposeful or accidental, accounted for in the calculation of environmental dose for workers in the vicinity of these releases. The TBD does not make clear how annual average stack releases are being used to assess environmental exposure. This could result in significant underestimates of exposures for early workers and those workers present during episodes like the Green Run. NIOSH needs to elaborate more on how environmental exposures from episodic releases (whether accidental, process, or deliberate) were taken into account.

On pages 14-15 of ORAUT-TKBS-0006-4, NIOSH recommends using the RATCHET model and rejects using the Hcalc dose reconstruction model developed by Till et al. "because Hcalc calculates 'worst case' or maximum hypothetical doses as opposed to a realistic dose." RATCHET is a superior model for episodic releases involving small particles less than 1 micron and gasses. However RATCHET was initially designed to address offsite doses to radioiodine and was not designed to accommodate large particles, from acute plutonium and radioruthenium releases at the separations plants, which in several instances were visible to the eye. Therefore, NIOSH should use an appropriate method to as effectively address particles greater than 1 micron to reconstruct onsite doses.

Lastly, HW-34882 (Irish 1955) makes reference to problems that may have affected exposures to workers indoors, as given in the following statement:

> *…the other deleterious aspect of the ruthenium problem, gross contamination of the canyon and crane, continues to be urgent and extremely difficult to control. An estimated 6.5 per cent loss in on-stream efficiency is attributed to this difficulty. Primarily because of this crane contamination problem, it is considered prudent to discontinue the permangenate head-end process until a satisfactory alternate is available for use.* [Emphasis added.]

NIOSH should develop a better roadmap for the calculations, other than the two references.

5.5.3 Consistency Among Site Profiles

Although Hanford and SRS had similar missions, there are some differences in the facility processes, design of the facilities, and radiological practices. In some cases, these differences require site-specific assumptions in dose determinations. For example, due to the design of the REDOX facility at Hanford, there were substantial particulate releases of ruthenium that had to be considered. In the case of SRS, the facilities were built later and this was not an issue. NIOSH made a concerted effort to recognize and address these differences in the TBDs.

The methodology used to determine environmental dose is not consistent between the Hanford TBDs and the SRS TBDs (Scalsky 2004bb). At SRS, a Gaussian air dispersion model was used. In the Hanford TBD (Savignac 2003), the RATCHET model (a Puff advection model) and an Excel spreadsheet were used to calculate intakes from airborne radionuclides. The RATCHET model used for Hanford is more powerful than the Gaussian air dispersion model used for SRS because it is capable of handling episodic releases. Nevertheless, NIOSH does not appear to use RATCHET for dose estimation for daily intakes from episodic releases. SC&A considers the RATCHET model a far better methodology for estimating dose in situations of daily episodic releases and would recommend that NIOSH consider its use for this purpose.

The Hanford TBD included evaluation of episodic releases, such as the release of ruthenium particles from the REDOX facility. However, these were modeled by developing atmospheric dispersion (χ/Q) values, which may not be as claimant favorable as the RATCHET daily episodic dose estimates. These data were converted to χ/Q tables for elevated and ground releases. Although it is not clearly defined in the SRS TBD, the environmental dose appears to include estimates from inhalation of radionuclides in air, direct external exposure to plumes, and exposure from resuspension of soil. It is not evident from the SRS TBD whether episodic releases were included in the calculations.

In summary, the methodologies employed to calculate onsite ambient environmental dose to workers may not be appropriate and are not necessarily claimant favorable. The analyses of environmental dose were based on only a few radionuclides and did not include many of the radionuclides the site has documented as being released from its facilities. The components of the environmental dose and the methodologies adopted for calculation of this dose are inconsistent between DOE sites.

The Hanford TBD includes a discussion on recycled uranium and the impurities associated with that at Hanford (Bihl 2004, page 24). Hanford's treatment of recycled uranium, for instance, attempts to treat dose potential from this source, whereas this was not addressed well in the SRS Site Profile. As pointed out in Section 5.4.4 of this report, however, even though Hanford made an attempt to address exposures from recycled uranium, there are still many other areas at Hanford where the magnitude of the Hanford recycling operations, the lack of reliable data on workplace exposure, and contaminants in recycled uranium need to be further developed in order to facilitate claimant-favorable worst-case dose estimates for workers exposed to recycled uranium. Neglecting recycled uranium in the SRS TBD (Scalsky 2004bb) produces an inconsistency between sites. A consistent criterion for inclusion of impurities in organ dose should be developed and applied for DOE and AWE facilities. A formalized complex-wide

policy for impurities in recycled uranium was not in affect until the later years of processing. As a result, careful consideration should be given to limits established by individual sites and their adherence to these limits during receiving and shipping of recycled uranium.

5.6 TANK FARMS AND WASTE DISPOSAL

The Tank Farm characterization in the TBD (Bihl 2004) is inadequate for dose reconstruction guidance in several respects. The following are the categories under which the TBD guidance needs to be more specific and complete:

- Radionuclide lists and waste management operations are incomplete for both internal and external radiation.
- Current environmental restoration and waste management projects included in the TBD are incomplete.
- Completeness and adequacy of Tank Farm' data used in the TBD are in question.

In this section, the term "Tank Farm workers" refers to all personnel who performed work in the 200W, 200E, and 300 Areas.

5.6.1 Radionuclide List and Waste Management Operations

The list of radionuclides used in some of the dose reconstructions appears to be incomplete, resulting in the potential for missed dose. The site profile relies predominantly on ORIGEN calculations to provide information on the most predominant radionuclides of interest in facilities where field characterization data may better indicate the radionuclides encountered by the workers and the relative ratios of isotopes (see Attachment 2). Attachment D, Table D.2.1.1-1 (Bihl 2004) provides the relative abundance of fission and activation products in Hanford fuel at the time of dissolution. The footnote indicates that all the activity data was obtained from ORIGEN 2 calculations performed for the Hanford Environmental Dose Reconstruction.

In the context of Hanford's high-level radioactive waste tanks, the TBD provides guidance that is neither accurate nor complete relative to radiation exposures to workers handing these wastes. According to the TBD:

> *^{144}Ce, ^{106}Ru, ^{137}Cs and ^{241}Am are usually associated with the supernatant and generally more available as contaminants. Plutonium, ^{90}Sr, ^{147}Pm are associated with the sludge. Lacking any other information, assume an intake of one radionuclide of the first group (supernatant or general contamination) exists in a mixture of the following: equal activities of ^{144}Ce, ^{106}Ru, ^{137}Cs, 0.1 ^{99}Tc, 0.1 ^{90}Sr, and 0.001 ^{241}Am. Assume an intake of one radionuclide of the second group (sludge) exists in a mixture of the following: equal activities of ^{90}Sr and ^{147}Pm, 0.1 ^{137}Cs, 0.001 ^{239}Pu and 0.001 ^{241}Am [Carbaugh 1995].* [80]

[80] ORAU-TKBS-0006-5, pages 44-45.

Between 1944 and 1988, Hanford generated high-level wastes containing more than 830 million curies of radioactivity.[81] The radionuclides ^{106}Ru and ^{144}Ce have half-lives of 284.3 days and 368.2 days, respectively, and are germane in assessing exposures during the production period at Hanford. In addition, ^{147}Pm, with a half-life of 2.6 years, is germane. However, ^{144}Ce, and ^{147}Pm are not accounted for in current HLW tank inventory data. Also, ^{106}Ru and ^{144}Ce do not represent significant risks when compared to radionuclides that are germane to recent efforts to process Hanford's HLW.

As indicated in Table 7, several additional radioisotopes should be considered when performing dose reconstruction. Risks of exposure to the radionuclides listed in this table increase upon the retrieval and processing of Hanford's HLW:

Table 7. Radionuclides in Soluble and Insoluble HLW Wastes Hanford's 177 Tanks (a) (curies)

Radionuclide	Soluble	Insoluble	Total
^{137}Cs (b)	82,708,000	7,192,000	89,900,000
^{90}Sr (b)	5,030,000	95,570,000	100,600,000
Transuranics	28,320	325,680	354,000
^{99}Tc	20,803	8,497	29,300
^{79}Se	134	--	134
^{14}C	3,010	--	3,010
^{129}I	48	--	48
^{3}H	10,400		10,400
^{126}Sn	--	600	600
Uranium (c)	59	929	988
^{106}Ru	8,300	---	8,300
^{152,154,155}Eu	---	180,610	180,610
^{60}Co	---	8,080	8,080
Total	87,809,074	103,286,396	191,095,470 (d)

a. Based on ratios contained in WHC-SD-WM-TI-699, Rev. 2, and Table 3-1.
b. Includes ^{137}Cs decay product--^{137m}Ba--and ^{90}Sr decay product--^{90}Y.
c. Includes ^{232}U, ^{233}U, ^{234}U, ^{236}U, and ^{238}U.
d. Other radionuclides not included constitute approximately 2.9 million curies.
Data Source: DOE, Tank Waste Inventory Network System Best Estimate, September 2003.

Moreover, exposure to transuranics from wastes in Hanford tanks could be quite significant. For instance:

- There are three 1-million-gallon capacity tanks that contain significant amounts of soluble transuranics – more than half of all soluble transuranics in Hanford tanks. In

[81] Westinghouse Hanford Corporation, *Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks*, WHC-SD-WM-TI-699, Fig. 3-2 (corrected by the author for ^{137}Cs and ^{90}Sr decay products).

these three tanks, transuranics are bound in organic complexants.[82] The interaction of radioactivity with water and organics in these tanks generate gases that are likely to contain significant amounts of soluble transuranics.

- In 2003, DOE identified 12 tanks with very high quantities of transuranic wastes. The wastes from these tanks are to be removed and processed at Hanford.[83]

- Between 1996 and 2003, the estimated amount of transuranics in Hanford HLW radioactive waste tanks went from 131,000 curies[84] to 354,000 curies – an increase of 270%.[85]

Between 2002 and 2004, some 90 workers reported illnesses and injuries to site medical professionals, claiming that they were caused by exposure to tank vapors.[86] This has resulted in investigations by the State of Washington, the U.S. Congress, NIOSH,[87] and the DOE Office of the Inspector General.[88, 89] NIOSH reported in 2004 that, "A summary of work stoppages from 2002 to 2003 was provided by management…. Reasons for the work stoppages were radiation contamination concerns (21), personal safety concerns (14), equipment concerns (14), procedural concerns (13), beryllium concerns (3), and only one vapor-emission concern."[90] NIOSH concluded that:

> *The Hanford Tank Farm working environment is complex in that both radiation and chemical hazards are present and the respective hazard levels are subject to change during waste disturbing activities.*[91]

5.6.2 Environmental Restoration and Waste Management Projects

The TBD (Savignac 2003) also does not include a complete listing of DOE's current environmental restoration and waste management projects, which constitute the primary mission of the site for the next several decades. For instance, there appear to be no references in the TBD

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[82] WC-SD-WM-TI-699 Rev.2 (1996) 4-14. "An analysis of DST radionuclide removal requirements on a tank-by-tank basis has determined that only three DSTs (241-AN-102, 241-AN-107, and 241 AY-101 require some removal of soluble TRU from supernatants to meet NRC Class C requirement of less than 100 nCi TRU/g glass. These three tanks are classified as [complexant concentrate] wastes and contain an estimated 0.0054 MCi (56 percent) of the total soluble TRU."

[83] U.S. Department of Energy, Office of River Protection, *Integrated Mission Acceleration Plan* (IMAP) Rev. 0.

[84] WHC-SD-WM-TI-699, Rev. 2 (1996).

[85]. U.S. Department of Energy, *Tank Waste Inventory System, Best Basis Estimate*, September 2003.

[86]Clare Gilbert and Tom Carpenter, Knowing Endangerment: Worker Exposure to Toxic Vapors at the Hanford Tank Farms, Government Accountability Project, September 2003, Washington, D.C.

[87] U.S. Centers for Disease Control, National Institute for Occupational Safety and Health (NIOSH), *NIOSH Health Hazard Evaluation Report*, HETA-2204-0145-2941, July 2004.

[88] Matthew L. Wald and Sara Kershaw, *Wider Investigation Sought at Nuclear Site*, New York Times, February 26, 2004.

[89] Blaine Harden, "Waste Cleanup May Have Human *Price*," *Washington Post*, March 6, 2004.

[90] NIOSH, HETA-2204-0145-2941, page 15.

[91] NIOSH, HETA-2204-0145-2941, page 4.

for several waste management facilities that have been identified as having potentially significant risks of radiation exposure.

Hanford is subject to compliance under the Tri-Party Agreement with the State of Washington and the U.S. Environmental Protection Agency. The Agreement provides a framework for meeting environmental restoration milestones that currently include over 60 treatment, storage, and disposal groups.[92] Each group contains several units. For purposes of illustration, facilities that should be referenced in the TBD include, but are not limited to:

- **The Hanford Solid Waste Operations Complex** – This operation includes eight burial grounds and covers approximately 1,400 acres. These burial grounds contain more than 38,000 drums and boxes of transuranic wastes buried with soil. From 1998–2001, DOE and its contractor removed more than 1,400 contact-handled transuranic drums stored on pads in the burial grounds. Several containers of remote-handled transuranic barrels await removal. Of current concern are drums in burial ground 4C, which contain significant amounts of ^{238}Pu in oxide form. Centerline temperatures in these storage containers are estimated to reach as high as 682° F. The Defense Nuclear Facilities Safety Board estimated that a single drum conflagration accident could deliver a dose of 260 rem to an exposed worker.[93]

- **The K-Basins Spent Fuel Project** – This project began in 1995 with the objective of removing 2,800 MTU of spent fuel (more than 90% of all DOE spent reactor fuel) that were stored in two basins at the KE and KW reactors. Removal of the spent fuel began in 2000 and approximately 90% has been removed, processed, and stored in silos in the 200 Area. Removal of sludge and debris from the K-East Basin, however, is proving to be a difficult task. The K-Basin sludge is highly radioactive and contains as much as 878,000 curies of radioactive materials, in approximately 52 m^3.[94]

- **Processing and Disposal of High-Level Radioactive Wastes** – The DOE announced plans in 2002 to terminate its environmental mission at Hanford and all other DOE sites over the next 30 years. During this time, DOE intends to dispose of approximately 90% of Hanford's HLW onsite, process the remainder into glass for geological disposal, and permanently close 177 large tanks and related infrastructure.

5.6.3 Adequacy and Completeness of Data

The list of radionuclides used in some of the dose reconstructions appears to be incomplete, thus resulting in the potential for missed dose. The TBD (Savignac 2003) does not include a complete listing of DOE's current environmental restoration and waste management projects, which constitute the primary mission of the site for the next several decades. For instance, there appear to be no references in the TBD for several waste management facilities that have been identified as having potentially significant risks of exposures to radionuclides. In the context of

[92] U.S. Department of Energy, *Hanford Facility Dangerous Waste Permit Application*, RL-91-28, http://www.hanford.gov/docs/rl-91-28/rl91-28app_2d.htm.

[93] Defense Nuclear Facilities Safety Board, Staff Issue Report, *Transuranic Waste Retrieval, Hanford Site*, October 13, 2003.

[94] Ibid.

Hanford's HLW tanks, the TBD provides guidance that is not accurate or complete relative to radionuclide exposures to workers handling these wastes. The TBD acknowledges that ^{144}Ce, ^{106}Ru, ^{137}Cs, and ^{241}Am are usually associated with supernatant and generally more available as contaminants. In addition, ^{90}Sr and ^{147}Pm are associated with sludge. Table 7 in Section 5.6.1 above indicates several additional radioisotopes that should be considered when performing reconstructions. Risks of exposure to these radionuclides in Table 7 increase upon the retrieval and processing of Hanford's HLW. Also, NIOSH has not addressed exposure to transuranics from waste in Hanford tanks that can be quite significant.

5.7 SPECIAL CAMPAIGNS

Hanford was involved in both minor and major special campaigns. Larger campaigns included irradiation of thorium and production of polonium and ^{233}U. The Irradiation Testing Group did experimental testing in the reactor areas. The radionuclides involved in these operations included ^{60}Co, ^{14}C, and isotopes of tantalum and yttrium, among others. These experimental tests supported both onsite and offsite groups (see Attachment 5 to this report).

NIOSH needs to provide a detailed revision in the Hanford occupational internal dose and external dosimetry TBDs (Bihl 2004 and Fix 2004) to properly account for dose from processes carried out in the 100, 200, and 300 Areas, particularly during peak production of ^{233}U in the 1960s through the early 1970s. For workers exposed to thorium in the 1950s and 1960s, NIOSH needs to confirm such thorium exposures by later urinalysis data for individual claimant thorium workers, and dose reconstructors need to carefully review potential dose in the 1960s and 1970s from irradiated thorium.

5.7.1 Uranium-233 Production and Thorium Processing Campaigns

NIOSH has not fully addressed the intrinsic linkage between the use of thorium at Hanford and the production of significant amounts of ^{233}U, which occurred over a 25-year period at the site. The radiation hazards associated with the processing of thorium and production of ^{233}U are quite significant, and proved to be a key obstacle that led to the ultimate termination of this program in the early 1980s. From 1945 to the early 1980s, a considerable effort involving several sites in the Federal nuclear complex was made by the AEC and its successor agencies to produce ^{233}U and to develop military and civilian applications for this fissile material. Currently, DOE has approximately 2 metric tons of excess ^{233}U in storage.[95]

Overall, the Hanford TBD (Bihl 2004) does not adequately address the production of ^{233}U. Production at Hanford was extensive and involved laboratory facilities, fuel fabrication facilities, several reactors, chemical separations operations, irradiated thorium recycling, and waste management activities.

A preliminary review indicates that Hanford produced more than 800 kg of ^{233}U.[96] Given the magnitude of production at Hanford, the possibility that Hanford produced the

[95] C.W. Forsberg, L.C. Lewis, *Uses for U-233: What should be kept for future needs?* ORNL-6952, September 1999, page xv.

[96] D. Dunning, Hanford Uranium-233 Production History, Columbia River Component of the River Corridor Baseline Risk Assessment, Second Workshop, Attachment 1 Production Campaigns, January 24,25, 2005.

preponderance of DOE's current excess ^{233}U inventory of some 2 metric tons should not be ruled out.

From the mid-1960s to the early 1970s during peak production, hundreds of kilograms of ^{233}U were made into uranyl nitrate solutions at a concentration of approximately 300 grams of ^{233}U per liter. ^{233}Uranyl nitrate solutions were then sent to the Oak Ridge site, at a rate of 20 to 40 kilograms per shipment.[97] Oak Ridge subsequently shipped ^{233}Uranyl nitrate to the Rocky Flats plant[98] and possibly other facilities for processing.

Of particular concern is exposure to ^{232}U , which is coproduced with ^{233}U during the neutron irradiation of ^{232}Th. The mechanism of production of ^{233}U is neutron capture by ^{232}Th, which produces the short-lived isotope ^{233}Th. ^{233}Th undergoes β^- decay to ^{233}Pa, which in turn β decays to ^{233}U. Simultaneously, side reactions of ^{232}Th with neutrons above thermal energies enhance the production of ^{232}U. The chemical separation of uranium from thorium produces a mixture of 5–50 ppm of ^{232}U, which is typical of ^{233}U currently stored at DOE sites,[99] the balance being ^{233}U.

This ^{232}U component has profound radiological significance. ^{232}U and ^{233}U are both α emitters—the only other decay mode is the very infrequent spontaneous fission—and thus produce no significant external exposure. However, ^{232}U decays to ^{228}Th, which has a half-life of 1.9 years. Thus, a few years after the uranium was separated from the ^{232}Th, ^{228}Th will have fully grown in. ^{228}Th, in turn, undergoes a series of rapid radioactive decays, producing, among other progeny, ^{208}Tl, which is a powerful γ emitter.

The year following the initiation of large-scale production of ^{233}U at Hanford, the Rocky Flats plant processed ^{233}U in uranyl nitrate and then fabricated ^{233}U metal components in kilogram quantities. According to a DOE contractor assessment performed in 2000:

> *The first processing operations at the Rocky Flats Plant involving uranium-233 (U-233) occurred in 1965. … The material also contained approximately 50 parts per million (ppm) contaminant … A 50 ppm U-232 content equates to approximately 13R/hr at 1 foot and with extrapolation, a 5 to 10 ppm content would emit approximately 5R/hr.* [100]

Over time, as ^{228}Th grows in, dose rates from external exposure increase significantly. In 1967, Hanford radiation readings were taken of bottles containing ^{233}U nitrate solutions. After 75 days, the external dose rate from the decay of ^{232}U progeny increased seven

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[97] Hanford Chemical Processing Division, *Hanford Contribution to Oak Ridge Fuel Processing Stu*dy, ARH-602RD, May 20, 1968, pp. 5.4-5.5.

[98] U.S. Department of Energy, *Disposition of Uranium-233 (U-233) in Plutonium Metal and Oxide at the Rocky Flats Environmental Technology Site*, RFP-5332, Rev. 1, March 2000, pp. 3-4.

[99] U.S. Department of Energy, Highly-Enriched Uranium Working Group Report on Environmental, Safety and Health Vulnerabilities Associated with the Department's Storage of Highly-Enriched Uranium, Volume I, DOE/EH-025, page 4.

[100] C.J.Frieboth, F. Gibbs, Disposition of U-233 in Plutonium Metal and Oxide at the Rocky Flats Environmental Technology Site, IT Corporation, RFP-5332, Rev. 1, March 1, 2000, pp. 4-5.

fold.[101] In conjunction with another study, SC&A performed an MCNP calculation of the dose rate from a 16-kg thin-walled spherical shell of uranium containing 5 ppm of ^{232}U in full secular equilibrium with its progeny. At a distance of 1 m, the effective dose rate to an anthropomorphic phantom in the anterioposterior orientation was 1.7 rem/h. This would increase to 17 rem/h for a ^{232}U concentration of 50 ppm.

Available information regarding accurate measurements of radioactive contaminants in ^{233}U at Hanford leaves much to be desired. Quantities of ^{232}U, and other contaminants in ^{233}U reported at Hanford do not appear to be based on direct measurements. Rather they appear to be derived from reactor burn-up calculations, and default approximations of radiological contaminants in the final product of ^{233}U nitrate solutions.

The approximations were apparently necessitated by the high gamma dose rates from irradiated thorium, which exceeded the maximum range of the radiation monitors at the PUREX plant.[102] This also led to a failure to accurately control the amount of thorium contamination of ^{233}U solutions, which varied by more than an order of magnitude.[103]

Laboratory testing at Hanford for thorium's potential use in reducing pile reactivity began shortly after Hanford Atomic Products Operations (HAPO) received a small amount of the material in 1945.[104] Research and development of thoria fuel proceeded at Hanford; and in March 1954, production of ^{233}U began through the H reactor fringe irradiation of thorium. This was followed by irradiation of thorium in the C reactor. This initial program was completed in October 1955.[105]

Early in 1964, Hanford's ^{233}U production program began on a large scale, with the arrival of virgin thorium from the National Lead Company of Ohio, and with the initial test irradiation of six tons of thoria targets in D reactor. After favorable results, four more tons were tested in F reactor in September 1964. A core loading at F reactor containing two tons of thoria followed shortly thereafter.[106] By February 1965, thoria "blankets" were being irradiated in the B, C, D, KE, and KW reactors, which continued through the early 1970s.[107]

It appears that fuel processing/fabrication of thorium occurred in the 306, 321, 313, and 314 Buildings. Exposure to other radionuclides present in virgin and recycled thorium should not be

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[101] ISOCHEM Inc., Purex Process Subsection, Purex Plant Thorium Process Operation Report, ISO-419, December 1967.

[102] "The accuracy and precision of the PUREX Analytical Laboratory gamma scans during the thorium campaign were relatively poor. [Emphasis added.] The decontamination factors listed for individual cycles are therefore approximations. This was the result of the presence of many gamma emitting isotopes not encountered during normal uranium-plutonium production. …The relatively high detection limits for thorium in the process streams precluded determination of accurate thorium decontamination factors in the U-233 cycles…The overall separation factor for thorium from the product stream was. 6×10^6 and for U-233 from the thorium product was 2.9×10^2 with individual batches as high as 5×10^3." Ibid.

[103] Ibid, page VII-24.

[104] S.P. Gydesen, *Hanford Experience with Thorium,* HW-31222, March 26, 1954, page 7.

[105] D.L. DeNeal, *Historical Events –Reactors and Fuel Fabrication*, DUN-3232, June 1967, page 11.

[106] E.C. Bowles, J.S. Schmidt, *Thoria Target Element Failures*, DUN-1010, April 16, 1966.

[107] DeNeal 1967.

ruled out for workers in the 300 Area and NIOSH should determine the extent to which recycling of thorium might have taken place relative to fuel fabrication.

In April 1967, a review of the experience of thorium fuel fabrication and irradiation at Hanford reported: "The handling of thoria presents radiological hazards because of gamma emission and the tendency for accumulation of inhaled or ingested thorium in the bones."[108] Between 1964 and 1967, workers in the 300-Area processed and fabricated about 180,000 thorium reactor fuel elements[109] in open faced hoods without the use of respirators.[110]

The potential dose, as demonstrated above, can be significant. NIOSH should examine in greater detail the radiological impacts on workers from the production of ^{233}U at all Hanford facilities. These operations represent a significant work commitment at Hanford that should be adequately characterized. More needs to be done to assess what appears to be a significant source of missed dose.

The NIOSH occupational internal dose TBD (Bihl 2004, page 32) discusses irradiated thorium extraction campaigns conducted at PUREX:

> *PUREX ran thorium campaigns in the 1960s and early 1970s. In terms of grams or curies, the thorium campaigns were small compared to the normal separation of plutonium. ^{232}Th was irradiated to produce ^{233}Th, which decays to ^{233}U. Although called a thorium campaign, it was the ^{233}U that was separated at PUREX and transported to 231-Z for experiments. Hence, the most likely source of intake was the ^{233}U during the loading out and transportation. Thorium exposure was more likely at the 3732 Building where the powdered thorium fuel targets were fabricated, which apparently contaminated the building with thorium "fines." Some work was also done with ^{232}Th slurries in the 3720 Building in the mid 1990s. The plan was to collect baseline urine samples on the few workers involved, and then collect special bioassay samples if air samples exceeded a cumulative exposure of 40 DAC-hrs. The urinalysis MDA was stated to be 0.1 dpm/sample.*

NIOSH points out in the Hanford occupational internal dose TBD (Bihl 2004, page 25) that:

> *^{233}U was handled at the 231Z Building in the mid 1960s as a special project, maybe extending into the early 1970s. This project involved thorium campaigns at PUREX, separation of the ^{233}U and shipment to the 231Z Building. No details about this work have been uncovered yet, such as isotopic purity… the specific bioassay used for the ^{233}U project has not been discovered.*

In the absence of assay data on ^{233}U produced at Hanford, as mentioned previously, NIOSH should make the claimant-favorable assumption that the ^{232}U levels in ^{233}U are the same as those in ^{233}U processed at the Rocky Flats plant, in 1965, a year after peak production at Hanford

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[108] H.G. Powers, G. A. Huff, *Thoria Target fabrication and Irradiation Experience*, DUN-2409, April 7, 1967, page 3.

[109] Powers and Huff 1967, page 10.

[110] Ibid., page 3.

began. The measured ^{232}U concentration was at Rocky Flats was 50 ppm, which would produce potentially significant dose rates from external exposure.[111]

NIOSH needs to provide a detailed revision of the Hanford occupational internal dose TBD (Bihl 2004) to properly account for dose from processes carried out in the 231-Z plutonium isolation building in the mid-1960s to 1970. As discussed above, specific ^{233}U bioassay techniques had yet to be developed. NIOSH has provided dose reconstructors with guidance on calculating ingestion dose in OCAS-TIB-009 (OCAS 2004).

For workers with no confirmed intakes, NIOSH has concluded in the occupational internal dose TBD (Bihl 2004, page 46):

> *Because plutonium, americium, and thorium stay in the body for a very long time, and because the urine excreta curve (activity per day excreted versus days after intake) has a small slope beyond the first year, an intake of plutonium or thorium that might have been missed in the 1950s or 60s because of poor detection capability, missed samples, or poor sampling after a potential intake, can still be confirmed or otherwise by urinalysis obtained years later. This is especially true for type S materials, but even applicable to type M.*

For workers exposed to virgin thorium from 1945 and through the 1950s, NIOSH needs to confirm such thorium exposures by later urinalysis data for individual claimant thorium workers, and dose reconstructors need to carefully review potential dose from irradiated thorium in the 1960s and 1970s, most likely received in the 3732 Building, where the powdered fuel targets were fabricated, and in the mid-1990s in the 3720 Building with ^{232}Th slurries.

This situation also applies to 300 Area reactor fuel production facilities, "special precautions for U and Th as radioactive substances were not taken…solutions, scraps, and other substances…were handled and disposed as ordinary process wastes."[112] Multiple fires and leaks in barrels and waste "load luggers" containing scraps of thorium occurred throughout the building histories. Some uranium and thorium entered the sanitary sewer system from personnel who contacted these substances.[113]

A 1979 HEDL radiological engineering report makes note of the 306 Building, involved in the production of thorium target elements:

> *All sewer lines leading from this building are suspect. The lime pit… contain[s] uranium and thorium sludge. Surface and near surface contamination around this building is to be expected.*[114]

Another report admits that there is pervasive uranium, thorium, and chemical contamination in the soil beneath and near 321 Building, extending to ground water.[115]

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[111] Frieboth and Gibbs 2000, page 3.
[112] http://www.hanford.gov/history/300area/300-1st.htm#300-1-0.
[113] Ibid.
[114] www.hanford.gov/history/300areas/300-1st.htm.
[115] www.hanford.gov/history/300area/300-4th.htm.

Twenty-six target element failures had occurred as of April 1966.[116] Most, if not all, were caused by water entry through closure welds into the thoria wafers; monitoring procedures adopted in March and April 1965 greatly decreased the failure rate. Of those failures, 14 resulted in unscheduled shutdowns in the C, KE, or KW reactors.[117] NIOSH should determine if this large number of failures led to radiation exposures of reactor area workers that are not covered in the TBD.

By June 1967, at least eight test-scale recycling campaigns of irradiated thoria were conducted at the 224 U.S. UO_3 facility.[118] Again, NIOSH should determine if the conversion of uranyl nitrate hexahydrate (UNH) solutions of irradiated thorium to UO_3 in the 224 U.S. facility led to radiation exposures not addressed in the TBD.

By August 1968:

> *….the K reactors are currently involved in producing approximately 460 kg of ^{233}U for [the AEC]…*[119] *Atlantic-Richfield Hanford Company reported in August 1967 that it had approximately 200 tons of thorium on-site, in the form of thoria nitrate solutions resulting from the chemical processing of previously irradiated thoria elements in production operations.* [120]

Hanford's use of thorium came to an end in the early 1970s. An online history of Hanford, for example, notes "shortly after [1970]…thorium oxide fuel was ruled out for large scale development."[121]

5.7.2 Polonium Campaigns

It is likely that ^{210}Po was made at Hanford during World War II and into the early 1950s (Marceau et al. 2002). Early ^{210}Po production seems to have been omitted and no bioassay data are mentioned in the TBD (Bihl 2004) to quantify worker dose to ^{210}Po. Bihl 2004, page 30, states the following:

> *There is an indication of work with pure ^{210}Po in the 308 Building in 1968 and again in 1975. Whether the work in the 308 Building was continuous through that period or just in those two years was not determined. Inference can be made that there was work somewhat prior to 1968 based on a handwritten note documenting a telephone conversation in November 1967 in which it was stated that the ^{210}Po starts in the process in the soluble form but is converted to the insoluble form. However, U.S. Testing was asked to develop a bioassay procedure in March 1968 and did so shortly thereafter, so apparently concern for possible intakes became*

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[116] DeNeal 1967, page 39.

[117] Bowles and Schmidt 1966.

[118] 200-Area Monthly Report, June 1967, HAN-96601, page 5.

[119] J.P. Schmidt, *Production Test Authorization 149 Large-Scale Thoria Irradiation*, August 15, 1968DUN-4462, August 15, 1968, page 3.

[120] Ibid.

[121] www.hanford.gov/history/mr0437/mr0437.htm.

> *important in early 1968. There also was work with ^{210}Po in the 325 Building that started in June 1972 and was slated 'to run for 2-3 years.'*

Thus, exposure to ^{210}Po may have been more than intermittent. NIOSH should ascertain if workers worked continuously in 308 Building during those years and, if so, assess the radiological impact of the resulting exposures. Table D.3.1-8 of Attachment D to Bihl 2004, does list, values for chronic intakes of ^{210}Po, assuming that the 24-hour urine specimens were below the MDA.

Directions are given to dose reconstructors for subtracting the natural background levels of ^{210}Po in feces from the levels measured in fecal samples from Hanford workers. The subtraction is to be based on levels reported in ICRP 23. These values, however, are based on only seven subjects and exhibit a range of 1.7 to 6.4 pCi/day. Given this level of uncertainty, the conservative and claimant favorable procedure should be to perform no background subtraction on the results of the fecal analyses, unless baseline levels had been determined for the individual workers. The lack MDA values for most fecal analyses for ^{210}Po during the period 1968 – 1985 leaves the dose reconstructor without guidance of how to assess analyses below the MDA. This issue should be addressed by NIOSH.

This issue was discussed during SC&A's site expert interviews, as illustrated in Attachment 3 to this report. It was noted by a site expert interviewee that the dose reconstructors use the normal procedure for assigning intakes if the energy employee's file shows bioassay for ^{210}Po.

NIOSH should better characterize dose from thorium and polonium. Thorium seems important since ^{233}U production occurred for a considerable amount of time in the 1950s and 1960s. Over 800 kg of the material was processed at Hanford. As a result of this process, thorium was recycled. Bihl (2004, page30) states that there was no bioassay for ^{210}Po prior to 1968, when some work with ^{210}Po was probable. NIOSH needs to reevaluate the cases in which doses from thorium and polonium may have been missed. The lack of ^{233}U bioassay throughout the mid-1960s is yet another example of cases in which lack of good bioassay data makes it difficult for dose reconstructors to develop worst-case dose reconstructions.

5.8 DECONTAMINATION, DECOMMISSIONING, AND DEACTIVATION

The occupational environmental dose TBD (Savignac 2003) is incomplete with respect to remediation and disposal sites.

Although NIOSH has included descriptions of key production and storage facilities, it has not addressed the numerous environmental waste streams and cribs that are and have been under remediation in the past (e.g., 300-FF-1, 216-Z-1). There are also Hanford-owned disposal sites (e.g., ERDF). These areas pose a radiological risk to those workers involved in the remediation and disposal process. Also, as these areas continue to age, the radionuclides of concern may be different from those in the original operations (e.g., longer lived fission and activation products may become predominant). Dose reconstructors need to take into account the risks associated with these areas at the Hanford Site and the variability in radionuclide concentrations.

5.8.1 Decontamination and Decommissioning

The characterization of decontamination and decommissioning (D & D) workers is absent from the TBD. NIOSH has chosen to address the unique exposure situations encountered for construction workers; however, it has not considered exposure of the workers involved in the D&D process. The following are the categories under which the TBD guidance needs to be more specific and complete:

- Radionuclide constituents and chemical forms
- Unknowns associated with D & D
- Lack of engineering controls in deactivated facilities and outdoor areas

In this section, the term "D & D workers" refers to all personnel, historical or current, who have been involved in decontamination, decommissioning, and deactivation of facilities.

5.8.2 Radionuclide Constituents and Chemical Forms

Hanford contains numerous environmental waste streams, cribs, and burial grounds. With the active D & D program at Hanford, individuals are likely to be exposed to a different mixture of radionuclides than their predecessors who operated the Hanford facilities.

In the course of D & D work, the environmental restoration contractor (ERC) encounters a wide range of radiological conditions, from essentially pure plutonium to extremely aged fission and activation products. These radioactive materials are spread over the site in a variety of forms ranging from traditional surface contamination on inoperative processing equipment to matrixed material in soil and slurries. While radionuclides with short half-lives often predominate during periods of operation, long-lived fission and activation products, transuranics, and uranium become the predominant radionuclides during environmental restoration.

As part of the D&D and environmental restoration process, areas are characterized to determine the activity, concentration, and identity of the contaminants. Based on these radiochemical analyses, the Radiological Control Organization determines the required radiological control and release processes. Radionuclides identified by Bechtel Hanford, Inc., in aging facilities from radiochemical analysis have included ^{60}Co, ^{137}Cs, ^{152}Eu, ^{154}Eu, ^{90}Sr/^{90}Y, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{79}Se, ^{99}Tc, ^{107}Pd, and ^{155}Eu. The mix of radionuclides is highly variable based on the facility and its original mission. In general, ^{152}Eu, ^{60}Co, ^{137}Cs, and ^{90}Sr are the radionuclides of concern. Inside the reactor building, americium, plutonium, ^{14}C, and ^{63}Ni are of concern. In the 200 Area, ^{137}Cs, ^{90}Sr/^{90}Y, and plutonium are of concern. With respect to the 300 Area, uranium, including uranyl nitrate, plutonium from fuel separations, and multiple chemicals are of concern. In addition to the conventional plutonium-production processes, the site processed a number of "exotic" materials, including neptunium and thorium, as well as smaller amounts of research materials. These materials exist in various concentrations at different locations throughout the site. Even within the same building, different radionuclides can be predominant in different areas. For example, in the reactor portion of the 100 Area reactor building, ^{60}Co,, ^{90}Sr/^{90}Y, , ^{137}Cs, ^{152}Eu, and ^{154}Eu,predominate. In the inner rod room of the same building, ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, and ^{99}Tc predominate (see Attachment 5).

In general, some of the facilities have been shut down for so many years that radionuclides with intermediate to long half-lives and daughter products become a larger proportion of the radionuclide mixture. Radionuclide mixes from operating periods are not appropriate for the assessment of a facility has been shut down for many years. With respect to research and development activities, one cannot assume that the traditional ratios of radionuclides exist. As time passes, the chemical forms of the radionuclides also change (e.g., plutonium nitrate converts to plutonium oxide). All these factors must be taken into account when reconstructing doses to workers involved in D&D activities.

5.8.3 Unknowns Associated with Decontamination and Decommissioning

For many years, the solution to disposing of highly radioactive materials at Hanford was to bury them. As a result, Hanford has been involved in extensive D & D activities. Some of these activities include tearing down buildings, removal of contaminated equipment and building structures (e.g., piping, ventilation systems), draining and filling reactor basins, excavating burial sites and trenches, and performing surveillance and maintenance of deactivated facilities. Although D&D has become a larger portion of today's missions at DOE sites, it has occurred throughout the years of operation at Hanford as facilities became obsolete (see Attachment 5). D&D presents different radiological challenges than those typically encountered during the years of operation. During the operation of a facility, the radionuclides of concern and exposure rates are known. Individuals involved in the operation of a facility are readily available to consult when uncertainties exist. During the process of decommissioning a facility or outside area, operations and radiological control often have to deal with unknowns, and there is a heavy reliance on historical documents. These historical documents are only as good as the recordkeeping practices of the time. As stated by Campbell and Powers (2005) with respect to 100B/C Area remediation:

> *But the listing in the subcontract is only as good as the characterization sampling and the record keeping for the time frame of 1943 through 1968 – the length of B Reactor's production run. There are also undocumented constituents listed in the subcontract as "unknown media and waste forms," which might include radionuclide and other hazardous materials encountered in the bits and pieces being excavated and removed.*

To quote a site expert actively involved in environmental restoration:

> *If you don't find a surprise in the burial grounds, that is the only surprise you get.*

Some of the "surprises" encountered by D&D workers have included unlabeled containers of material, mixed waste, high-activity radioactive sources, leaking canisters, damaged fuel elements, and laboratory waste material (see Attachment 5).

As Hanford's mission changed from production of plutonium to environmental restoration, the site made use of deactivated facilities. For example, the pipe galleries of the separations facilities became storage areas for radioactive equipment and materials. Storage in deactivated facilities has been used in lieu of disposal of material. Little documentation exists on what materials were placed in these deactivated facilities for storage. When dealing with the removal

of items, crews have to be cognizant of warning signs (e.g., lead shielding around an item) to prevent overexposure.

Although current workers are monitored for both internal and external exposure, many of the existing internal monitoring requirements are based on the expected mixes of radionuclides. The unknowns have created a situation in which internal monitoring occurs after the fact based on field indicators. The problem is further compounded by the turnaround time of analytical results on these field samples. It is likely that chronic intakes below the bioassay detection limits and intakes from unexpected radionuclides may be missed. For example, with the implementation of the DAC-hour tracking program for the Environmental Restoration Contractor, radiological engineers have noted that it is not uncommon to see one-half DAC-hour per day. Special bioassay techniques are not required, according to the Hanford TBD, until an individual receives 10 DAC-hours during a particular job evolution (see Attachment 5).

Lack of Engineering Control in Deactivated Facilities and Outdoor Areas

Deactivation of facilities and remediation of the environment result in special challenges to radiological control staff in implementing engineering controls. Some of the field radiological conditions limiting the use of traditional engineering controls include:

- Absence of utilities such as electricity and water
- Use of large equipment that cannot easily be contained
- Destruction of buildings that cannot be contained
- Outdoor remediation areas are more rugged than buildings
- Early shutdown of facilities involved the removal of ventilation systems
- No gloveboxes or fume hoods, as their removal is part of the deactivation
- Burial sites are in a different location than expected

As a result of the limitations on engineering controls, D&D contractors have resorted to the implementation of administrative control limits and the use of personnel protective equipment (PPE). Shielding is also used to reduce external exposure. Again, when dealing with unknowns, it is difficult to determine whether the selected PPE is appropriate for the job, or what kinds of radiation fields will be encountered.

Former site profiles have reserved a section to specifically address construction workers. These individuals are considered a special class of workers due to their mobility throughout the site, the increased likelihood of hands-on work, and difficulty in obtaining records. The TBDs reviewed to date (i.e., Bethlehem Steel, Mallinckrodt Chemical Works, SRS, and Iowa Army Ammunition Plant) have not addressed workers heavily involved in D & D, who encounter radiological constituents different from operational facilities and unknown radiological conditions during the course of their work, and who have to deal with issues such as no electricity, interim storage of waste in deactivated facilities, and no ventilation. Although NIOSH has indicated that D & D workers are routinely monitored, early D & D workers may not have been adequately monitored, and, with unknown conditions, monitoring has to be completed after the fact. NIOSH should investigate the differences between operational and D & D workers to identify whether specific situations encountered in D & D are bounded by current modeling used for operational workers.

5.9 INCIDENTS AND ACCIDENTS

The method of locating, evaluating, and integrating incident data into the dose reconstruction is not clear in the Hanford TBDs. NIOSH needs to investigate if accident and incident records can be found to provide additional dose information for accidents and incidents.

NIOSH is relying on the claimant interviews and personal dosimetry records to identify incidents in which individuals may have been involved. Other sources of data exist (although they may be somewhat biased) that can provide valuable data on incidents, such as occurrence reports, employee concern programs, DTPA administration lists, REX database incident listings, and former or current employees. Especially for the cases involving survivor claims, these data may be important for the proper evaluation of claimants' doses. Note that minor incidents may not be documented in an employee's personal dosimetry file and these are the types of incidents that have occurred and do routinely occur (e.g., skin and clothing contamination, changes in radiological conditions which lead to higher posting levels, etc.).

For example, a review of several hundred Radiation Occurrence reports from the Hanford reactor area, railroad operations, Tank Farm, and Burial Grounds between 1955 and 1987 found numerous instances of excessive exposures to process operators and construction workers.[122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 132, 133, 134, 135, 136, 137, 138] The radiation incident reports reviewed are

[122] Manufacturing Department, Radiation Hazards Incident Class II, No. 50, May 14, 1953.

[123] Radiological Sciences Department Investigation Radiation Incident, Class II, No. 69, HW-30783, February 15, 1954.

[124] Chemical Processing Division, 204 S Area, Radiation Occurrence Reports, January 10, 1957, August 8, 1957, October 1, 1957, September 12, 1958, January 15, 1980, December 21, 1967, January 10, 1968, July 17, 1968, March 2, 1973, July 11, 1978, November 9, 1978, January 16, 1979, July 19, 1979, February 20, 1980, October 11, 1980, September 3, 1981, November 23, 1981,February 11, 1982, April 19, 1982, December 12, 1984, July 9, 1987,

[125] Chemical Processing Division, 241 AX Area, Radiation Occurrence Reports, April 21, 1976, August 24, 1976, September 30, 1976, December 3, 1976, August 25, 1977, October 16, 1977, July 18, 1978, July 27, 1978, September 12, 1978, February 27, 1979,

[126] Chemical Processing Division, 241AN Tank Farms, Radiation Occurrence Reports, June 13, 1979, April 17, 1982, June 16, 1982, February 23, 1982, February 27, 1984, June 13, 1984, July 26, 1984, October 15, 1984, September 25, 1985, October 29, 1985, January 17, 1986, February 10, 1987, February 27, 1987.

[127] Chemical Processing Division, 241 AX Tank Farms, Radiation Occurrence Reports, January 30, 1968, January 2, 1970, April 5, 1972, July 10, 1972, October 16, 1972, November 6, 1972, November 7, 1972, February 27, 1974, June 5, 1974, October 9, 1974, October 14, 1974, November 5, 1974, November 22, 1974, February 7, 1975, March 24, 1975, September 2, 1975, October 31, 1975, November 14, 1975, December 8, 1975.

[128] Chemical Processing Division, 241 A Tank Farm, Radiation Occurrence Reports, May 1, 1980, October 10, 1980, December 10, 1980, November 19 and 22, 1982, July 8, 1983, December 12, 1983, December 17, 1984, September 12, 1985, October 17, 1985, March 27, 1986, March 14, 1987.

[129] Chemical Processing Division, 241 B Tank Farm, Radiation Occurrence Reports, October 27, 1957, January 4, 1968, January 8, 1968, January 12, 1968, March 7, 1968, June 14, 1968, October 14, 1969.

[130] Chemical Processing Division, 241 BY Tank Farm, Radiation Occurrence Reports, June 2, 1970, January 10, 1972, May 12, 1972, August 26, 1974, February 12, 1975, April 29, 1975, May 7, 1976, October 5, 1976, December 12, 1978, June 28, 1978, November 29, 1978, February 26, 1980, April 4, 1980, May 17, 1980.

[131] Chemical Processing Division, 241 CR Tank Farm, Radiation Occurrence Reports, July 1, 1957, September 6, 1957, September 30, 1960, July 24, 1967, July 31, 1967, October 15, 1969, October 17,18, 1969, November 20, 1970, August 31, 1973, April 25,1974, September 27, 1974, October 31, 1974, July 22, 1975,

probably a small fraction of radiation incident records kept by DOE and its contractors from the beginning of operations. Some of the more significant occurrences found in our limited review include:

- **May 12, 1953 –** Six reactor operators were overexposed unknowingly when they handled three highly radioactive spacers that fell out of the reactor face onto the floor in the charge elevator. It was estimated that the gamma dose rate at one foot from the spacers was approximately 1,000 R/hr. The beta/gamma contact dose rate was estimated at 500,000 rep/hr or 430,000 rad/hr.

- **February 15, 1954 –** Excessive contamination of railroad locomotives and cask cars resulted in contamination of nine employees. One employee had unknowingly worn a contaminated sweater for about 70 hours. The sweater was subsequently estimated to have a dose rate of 750 mrad/hr, resulting in a maximum estimated dose of about 50 rad.

- **May 6, 1959** – An instrument technician working on the 241 SX Tank Farm was sprayed with radioactive mist from a pipe connection. The dose rate resulting from contamination on the worker's coveralls was measured to be 15 rads/hr.

February 10, 1976, November 12, 1976, October 2, 1978, November 14, 1979, March 5, 1981, October 22, 1981, September 18, 1984, April 20, 1985, July 7, 1986.

[132] Chemical Processing Division, 241 SX Tank Farm, Radiation Occurrence Reports, February 3, 1956, August 19, 1957, June 12, 1958, July 22, 1958, July 31, 1958, August 18, 1958, August 25, 1958, January 27, 1959, March 16, 1959, May 6, 1959, May 23, 1960, July 21, 1961, December 1, 1961, August 13, 1962, November 18, 1965, September 12, 1966, August 26, 1968, January 16, 1969, June 14, 1971, February 8, 1972, April 21, 1972, October 10, 1972, March 21, 1973, August 14, 1974.

[133] Chemical Processing Division, 242 T facility, Radiation Occurrence Reports, August 19, 1966, January 26, 1967, April 3, 1967, March 10, 1968, November 3, 1968, July 11, 1969, November 10, 1969, April 13, 1971, May 4, 1971, April 19, 1972, April 23, 1972, December 19, 1972, March 5, 1973, September 7, 1973, December 1, 1975, March 1, 1976, June 8, 1976, October 4, 1976, May 5, 1977, May 24, 1977, October 19, 1977, February 14, 1978, December 22, 1978, October 25, 1979, February 26, 1980, July 20, 1980, July 25, 1980, March 16, 1982.

[134] Chemical Processing Division, 241 T Tank Farm, Radiation Occurrence Reports, June 21, 1956, February 20, 1963, May 4, 1966, February 7, 1967, March 25, 1967, May 24, 1967, August 31, 1967, June 18, 1969, November 14, 1969, September 24, 1980.

[135] Chemical Processing Division, 241 U Tank Farm, Radiation Occurrence Reports, July 6, 1956, July 9, 1958, September 24, 1958, October 17, 1958, June 13, 1958, May 5, 1969, February 2, 1973, January 24, 1974, January 2, 1977, August 29, 1978, September 25, 1979, November 1, 1979, December 12, 1979, February 20, 1980, August 25, 1980, June 16, 1982, June 26, 1984.

[136] Chemical Processing Division, TX Tank Farm, *Radiation Occurrence Reports*, December 12, 1965, March 1, 1966, July 25, 1966, April 4, 1967, April 11, 1967, August 26 to September 28, 1967, September 9, 1967, October 3, 1968, November 26, 1968, February 13, 1968, October 28, 1970, January 7, 1971, May 8, 1975, February 5, 1977, April 15, 1977, July 22, 1977, August 24, 1977, January 10, 1978, August 8, 9, 10, 21, 28, and September 7, 1979, June 5, 1980, July 31, 1980, March 27, 1981, January 12, 1983.

[137] Chemical Processing Division, 244 AR Area, Off-Normal/Radiation Occurrence Reports, January 27, 1968, March 28, 1972, May 19, 1974, July 22, 1976, December 14, 1977, March 2, 1978, March 28, 1978, March 14, 1980, May 8, 1980, August 17, 1983, January 10, 1985, January 11, 1985, October 1, 1986.

[138] Chemical Processing Division, 200-Area Industrial Burial Ground, Radiation Occurrence Reports, December 9, 1955, January 27, 1956, January 30, 1956, January 31, 1956, July 7, 1956, October 2, 1957, October 24, 1957, October 31, 1957, November 6, 1957, November 11, 1957, January 8, 1959, June 17, 1960, July 2, 1965, September 21, 1966,April 11, 1968, March 4, 1980, July 23, 1980, August 15, 1980, March 13, 1981, October 13, 1981, December 13, 1985.

- **November 24, 1969 –** A nuclear chemical operator working in the 200-East Area Tank Farms received significant hand and whole-body radiation exposure when he pulled a cable out of a HLW. Estimated doses were 10-30 rems to the hands and 3–5 rem to the whole body.

- **October 16, 1972 –** Personnel working an 8-hour day shift in the 244 AR Vault and the 102 AX Tank work area were found to have been exposed to significant contamination levels resulting in doses as high as 3,000 mrad/hr. Contamination was found in the homes of two workers, including contamination of a child (400 cpm on his hands and 2,000 cpm in the child's car seat) and bedding (10,000 cpm). Bioassays of a worker and his family were taken to detect possible internal exposures to ^{90}Sr.

- **February 12, 1974 –** Two crane riggers at the 241 AX Tank Farm were contaminated. Measured doses in the workplace were as high as 5,000 mrads/hr. The workers did not wear respiratory protection. One worker had face, neck, and hair contamination levels of 50,000 cpm, 40,000 cpm, and 40,000 cpm, respectively.

- **September 12, 1978 –** Two chemical operators were contaminated while working on a diversion box at 241 A Tank Farm. The workers did not have adequate protective clothing or respiratory protection. After extensive decontamination, one employee had 9,995 cpm remaining in his nostril.

- **May 17, 1983 –** Several workers involved in removing a dip tube from the 111 B tank were exposed to dose rates that exceeded the measurement range of the dose meter (5 r/hr). A dose rate of 120 rads/hr was measured at 12 inches from the dip tube.

- **July 8, 1973 –** An employee working in the East Tank Farms received a 1,560 mrem dose, in excess of the quarterly limit of 1,250 mrem.

- **June 13, 1984 –** Two construction workers working near a process transfer line in the 241A Tank Farm were exposed unknowingly to dose rates of 150 mr/hr instead of the presumed 3 mr/hr.

- **October 29, 1985 –** A leaking jumper in a valve pit in the 242 AW Tank Farm resulted in four employees having positive nasal smears and skin/hair contamination as high as 5,000 cpm beta/gamma.

NIOSH needs to investigate whether records can be found to provide additional information on doses from accidents and incidents. It would be important, if such records still exist, to determine if there could have been any underreported or underestimated doses resulting from accidents and incidents. Hanford site experts (Attachment 3 to this report) stated that records are no longer retrievable; thus, the only way to reconstruct doses from accidents and incidents is from information in the individual worker files. SC&A is concerned that the default assumptions for determining missed dose from these kinds of incidents and high-risk jobs may not include the necessary instructions to dose reconstructors that are needed to develop claimant-favorable doses. Such instructions are not evident in the internal dose TBD (Bihl 2004). It is important that the dose reconstructors be alerted to likely situations and watch for the need for dose

reconstruction from incidents and high-risk jobs. This is particularly important in claims from survivors in which the survivor's scant knowledge makes it difficult to include enough detailed data in their CATI and where such information is not included in the individual's worker file. ORAUT-OTIB-0002 (Rollins 2004), not mentioned in the internal dose TBD, does not adequately address these incident and high-risk job dose potentials, which may require different assumptions than those listed. As can be seem from the discussion in the section above, there are many work assignments in which significant reported doses from incidents and accidents have been documented for Hanford workers.

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

The procedures used by SC&A call for both a "vertical" assessment of a site profile 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. 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.

6.1 SATISFYING THE FIVE OBJECTIVES

The SC&A review procedures, as approved by the Advisory Board, require 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 Hanford Site Profile finds that the profile generally satisfies these objectives, although 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 analysis of data, or incomplete reflection of operational or dosimetric history. Key issues are summarized below and in Table 8, which provides a matrix representation of the identified issues sorted according to the SC&A findings. Detailed evaluation 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 Hanford 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.

Table 8. Issue Matrix for the Hanford Site TBDs

Descriptions	Issue Classification	Obj. 1: Completeness Of Data	Obj. 2: Technical Accuracy	Obj. 3: Adequacy Of Data	Obj. 4: Site Profile Consistency	Obj. 5: Regulatory Compliance
Issue 1: Neutron Dosimetry and Exposure	Finding	X	X	X		
Issue 2: Early Worker Radiological Monitoring	Finding			X		X
Issue 3: External Beta/Gamma Dose Adjustment and Uncertainty Factors	Finding				X	
Issue 4: Internal Dose Assumptions	Finding	X		X	X	
Issue 5: Environmental Dose	Finding		X		X	
Issue 6: Tank Farms and Waste Disposal	Finding	X		X		
Issue 7: Special Campaigns	Finding			X		
Issue 8: Decontamination, Decommissioning, and Deactivation	Finding	X				
Issue 9: Incidents and Accidents	Finding			X		

6.1.1 Objective 1: Completeness of Data Sources

The breadth of data sources used as a basis for the Hanford Site Profile is evident in the 229 reports cited as references, including a number of authoritative historical documents dating back to the start of operations in the early 1940s. Based on a review of the Hanford site description TBD (Selby 2004), it is evident that NIOSH effectively compiled and characterized activities and operations at 58 building or facility types. In fact, this review cites the breadth of operational data provided (e.g., the Attachment D.3 tables on the internal dose TBD (Bihl 2004) and atmospheric source terms in the occupational environmental TBD (Savignac 2003), as a strength. Also noteworthy is the radionuclide-specific data (i.e., types and relative importance of various radionuclides) regarding contributions to internal and external exposures that are provided in Attachment A of the site description TBD (Selby 2004).

However, SC&A also found a lack of characterization of potential worker exposures at the Hanford Tank Farm 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. Although extensive descriptions of key production and storage facilities are included, the numerous environmental waste streams and cribs that existed at Hanford, but that have been since cleaned up, are not considered in the TBDs, although they certainly may have contributed significantly to worker exposure both before and during cleanup.

It is also clear that neutron exposures to reactor workers should have been better characterized for a number of Hanford facilities, including the "tail-ends" of the canyon operations, the 233-S Plutonium Reclamation Facility, and, again, the waste tanks and burial ground. Likewise, the implication of radiation shielding that was added over time to various reactors should have been addressed, given its potential significance to characterizing potential worker exposure.

6.1.2 Objective 2: Technical Accuracy

The derivation of neutron-to-photon dose ratios provided in the Hanford external dose TBD (Fix 2004), incorporates uncertainties that are not properly addressed; nor are the ratios tailored for the three distinct types of operational areas where neutron exposure was most prevalent. Site expert interviewees have provided testimony in Attachments 3 - 5 to this report that there were other areas where the potential for neutron exposure has not been sufficiently characterized by NIOSH.

The technical accuracy of the approach suggested for estimating doses for workers exposed to plutonium whose intakes 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 at Hanford. Further, the suggested use of a default mixture and geometric standard deviation of 5 for estimating plutonium burdens from in vivo data may not be consistently claimant favorable. There are also technical questions associated with the accuracy of fission product internal dose estimates that would be estimated from default air concentrations.

In addition, the puff advection capabilities of the atmospheric transport code RATCHET were not fully utilized. Specifically, the TBD did not evaluate the doses to workers outdoors associated with episodic releases to the atmosphere.

6.1.3 Objective 3: Adequacy of Data

Questions regarding data adequacy, where they arise in the SC&A evaluation, have largely focused on the adequacy of available dosimetry data for the early monitoring program where unmonitored workers are more likely to have had missed dose due to inconsistent monitoring in the field. SC&A also found that the TBDs did not fully explore and develop procedures and guidelines to dose reconstructors that would lead them to focus on gaps, primarily in internal dose, that could lead to a significant underestimate of worker dose.

Input from site experts interviewed indicates that there were situations where reactor workers were not provided neutron dosimeters or were not monitored on a continual basis. There is evidence to suggest that this is true for nuclear power operators even into the 1970s and 1980s. These data cannot be considered adequate unless an evaluation is conducted of the comprehensiveness of the neutron-monitoring program and to what extent existing dose estimation assumptions and methodologies address this potential missed dose.

The lack of bioassay data during the 1943–1946 period represents an important area in which adequacy of data is of concern. In addition, bioassay data for fission products and uranium are not reliable until 1948 – this is acknowledged in the TBD. 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 early period.

There is an inadequacy of data for tritium dose to Hanford workers prior to 1949. Potential tritium dose was not adequately documented until tritium urinalysis commenced in 1949. No detection level of tritium existed until good liquid scintillation tritium analysis was implemented in 1958. Hanford worker doses due to tritium in these early years pose a potential for missed internal dose in the Tritium Extraction Facility 108B (P-10 Plant) in the late 1940s to the mid-1950s.

As noted earlier, the source term list of radionuclides provided for the U.S. Department of Energy's (DOE) current and past environmental restoration and waste management projects is incomplete and may contribute to missed dose. Risks of exposure to radionuclides that workers have encountered when retrieving and processing Hanford's high-level waste have not been adequately addressed. Also, NIOSH has not addressed exposure to transuranics from wastes in Hanford tanks, which can be significant.

6.1.4 Objective 4: Consistency Among Site Profiles

While Hanford and the Savannah River Site (SRS) had similar missions, 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 two sites, while mirroring those assumptions and approaches where justified. The SRS profile predates the Hanford Site profile; therefore, NIOSH benefited greatly from the early efforts at SRS. Hanford was therefore able to remedy many of the apparent inconsistencies in the SRS TBD.

An extensive comparison was performed by SC&A to compare and contrast the methodologies used in the 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 probability of causation for individual claimants. A detailed analysis is provided in Table A.6.2 of Attachment 6 to this report. This table demonstrates, in detail, where the 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 improvements that were introduced in the Hanford TBD by NIOSH based on experience with SRS and earlier reviews.

Attachment 6 provides, in tabular form, an evaluation and comparison of the default assumptions for each element of exposure (i.e., occupational medical dose, internal dose, external dose, and environmental dose). 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 SRS and Hanford. Hanford's treatment of recycled uranium, for instance, more adequately treats dose potential from this source, whereas this was not addressed well in the SRS site profile. But even then, there are still many areas at Hanford for which dose to workers handling recycled uranium is not adequately addressed and characterized.

6.1.5 Objective 5: Regulatory Compliance

NIOSH has effectively complied with the hierarchy of data required under 42 CFR Part 82 and its implementation guides.

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 Hanford 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 probability of causation 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.

Hanford did a more effective job than SRS in performing annual chronic intake assessments and daily intake rates based on unit MDA urinalysis on the last day of the period for the significant radionuclides. Those data are provided in the tables in Attachment D of the internal dose TBD (Bihl 2004).

The Hanford internal dose TBD (Bihl 2004) does not mention the use of ORAUT-OTIB-0011 (Siebert 2004) for calculated doses from tritium and estimates missed dose from this nuclide.

This may be useful, especially for the early period at Hanford. Dose reconstructors should be alerted to the use of this technical information bulletin.

Hanford did not address the opportunity to use surrogate radionuclides when data are not available for a less commonly encountered radionuclide and thus the 95th percentile cannot be applied in estimating the upper bounds of a like dose.

The Hanford internal dose TBD does not mention the use of an approach recommended for other similar DOE facilities when determining maximum dose. Attachment A to ORAUT OTIB-0002 (Rollins 2004) however, was developed for specific use in determining maximum dose estimates for Hanford workers from 1953 through 1969. In reviewing some of NIOSH's individual dose reconstructions for Hanford claimants, SC&A noted that NIOSH used this TIB to provide a worst-case, most claimant-favorable dose.

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. Almost to a fault, NIOSH provides guidance to dose reconstructors on how to navigate through the complex mix of radionuclides required to reconstruct historical internal exposures to workers. Notwithstanding this achievement, there are opportunities for improvement in the data sets and instructions to the dose reconstructors with respect to reconstructing internal exposures.

6.3 UNRESOLVED POLICY OR GENERIC TECHNICAL ISSUES

A number of issues were identified that are common to both the 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 (e.g., oro-nasal breathing), 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 these more generic issues. The following represents those issues identified in the Hanford Site Profile review that in SC&A's view represent transcendent issues that need to be considered by NIOSH as unresolved policy or generic technical issues.

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

(2) Mobility of work force between different areas of the site should be addressed. Site expert testimony that many workers moved from one plant to the next is a complicating factor. Establishment of an accurate worker history is crucial in such cases. This will be especially difficult for family member claimants.

(3) Statistical techniques used in the application of the data to individual workers should be considered.

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

(5) Assumptions on solubility, oro-nasal breathing, and ingestion should be addressed.

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

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

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

(9) 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.

7.0 REFERENCES

42 CFR Part 82, Title 42, *Public Health*, Part 82, "Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program Act of 2000," *Code of Federal Regulations*, May 2, 2002.

Alexander, W.K., 1964, *Reactor Moderator Distortion Study*, DUN-3424 RD, May 15, 1964.

Bair, W. J, Ballou, J. E. and Case, A. C., 1962, *Studies with Neptunium in Rats*, HW-SA-2614, May 25, 1962.

Ballinger, M.Y., and. Hall, R.B, 1991, *A History of Major Hanford Facilities Involving Radioactive Material*, PNL-6964 HEDR, March 1991.

Bihl, D.E., 2004, *Technical Basis Document for the Hanford Site – Occupational Internal Dose*, ORAUT-TKBS-0006-5, Revision 01, Oak Ridge Associated Universities, Oak Ridge, TN, November 14, 2004.

Biles, M.B., 1972, Memorandum to M.B. Biles from T.A. Nemzek, *Fast Neutron Dosimetry – 234-5 Building*, November 2, 1972.

Bunch, W., 1958, *Shielding Effectiveness of the Tapor Bore Process Tube Entry*, February 12, 1958.

Brendan, B.B., 1958, *105-C Shield Window Breakage*, HW-55770, April 16, 1958.

Bunch, W., 1958, *Shielding Effectiveness of the Tapor Bore Process Tube Entry*, HW-54955, February 12, 1958, p. 2.

Bunch, W.L., 1956, *The Effectiveness of Masonite Burnout on Shield Attenuation Properties*, HW-38418, May 23, 1956.

Buschbom, R L. and Gilbert, E. S., 1993, *Summary of Recorded External Radiation Doses for Hanford Workers 1944-1989*, PNL-8909, Pacific Northwest Laboratory, Richland, WA, on Web at http://www/pnl.gov/eshs/cap/dss/epub.html.

Campbell, L. and Powers, K. (2005), "Like a Box of Chocolates? At the Hanford Burial Grounds, You Never Know What You're Gonna Get," Radwaste Solutions, March/April 2005.

Carbaugh, E.H., 1995, Letter to distribution, *Bioassay for Spent Fuel and Waste Handling, Storage, and Analysis Facilities*, June 1, 1995.

Carbaugh, E.H., 2000, *Hanford Internal Dosimetry Project Manual*, PNL-MA-552, Rev. 3, 1997, available on the Web at http://www.pnl.gov/eshs/cap/dss/epub.html.

Carbaugh, E.H., 2003 (or current version), *Methods and Models of the Hanford Internal Dosimetry Program,* PNL-MA-860, available on the Web at http://www.pnl.gov/eshs/cap/dss/epub.html.

Clark, D.E., 1967, *Iron Masonite Shield Status Report No. 6*, DUN-2674, June 30, 1967.
Curtiss, D.H., Fulmer, G.C., and Gilbert, W.D., 1958, *Reactor Components Study*, HW-57222, August 25, 1958.

DeNeal, D.L. 1967, *Historical Events, Reactors and Fuel Fabrication*, DUN-3232, November 1, 1967.

Department of Energy (DOE), 1948, *Radioactive Contamination in the Columbia River & in the Air & Radiation Levels Measured in the Air at HW and Vicinity for 1945, 1946, 1947 and Early 1948*, HW-9871, May 24, 1948.

DOE, U.S. Office of Environmental Management, 1997, *Linking Legacies: Connecting the Cold War Nuclear Weapons Production Processes to Their Environmental Consequences*, January 11997, on Web at http://legacystoryapps.em.doe.gov.

DOE, 2000, *Recycled Uranium, The Flow and Characteristics of Recycled Uranium Throughout the DOE Complex, 1952-1999,* Vol. II, Part 1 "Hanford Site," DOE/EH-0617 or DOE/RL-2000-43, DOE Richland Operations Office, Richland, WA, 2000.

Eckerman, K.F., et al., 1993, *Limiting Values of Radionuclide Intake and Air Concentrations and Dose Conversion Factors for Inhalation, Submersion, and Ingestion – Federal Guidance Report No. 1,* report prepared by Oak Ridge National Laboratory for the U.S. Environmental Protection Agency, EPA-520/1-88-020, 1988.

Eckerman, K.F. and Ryman, J.C., 1993, *External Exposure to Radionuclides in Air, Water, and Soil, Exposure-to-Dose Coefficients for General Application – Federal Guidance Report No. 12,* report prepared by Oak Ridge National Laboratory for the U.S. Environmental Protection Agency, 1993.

Figgins, P.E. and Kirby, H.W, 1962, *Survey of Sources of Ionium (Thorium-230),* AEC Research and Development Report*;* MLM-1349, 1962.

Fleishman-Hilliard, Inc., 1967, *Fuel for the Atomic Age, Completion report on St. Louis area uranium processing operations*, prepared for Mallinckrodt Chemical Works to document performance under AEC Contract No. W-14-108-Eng-8, September 30, 1967.

Fix, J.J., Wilson, R.H., and Baumgartner, W.V., 1997a, *Retrospective Assessment of Personnel Neutron Dosimetry for Workers at the Hanford Site,* PNNL-11196, Pacific Northwest National Laboratory, Richland, WA, February 1997.

Fix, J.J., Gilbert, E S., and Baumgartner, W.V., 1997b, *An Assessment of Bias and Uncertainty in Recorded dose from External Sources of Radiation for Workers at the Hanford Site*, PNNL-11106, Pacific Northwest National Laboratory, Richland, WA, 1997.

Fix, J.J., 2004, *Technical Basis Document for the Hanford Site – Occupational External Dose*, ORAUT-TKBS-0006-6, Rev. 01, January 9, 2004, Oak Ridge Associated Universities, Oak Ridge, TN, January 9, 2004.

Gast, P.F., 1949, Memo to File, *Plutonium isotopes in recycled uranium*, HW-12166, January 17, 1949.

General Electric (GE) Report, 1949, *Preliminary Project Proposal Temporary Pile Shielding Repairs*, HW-13243, May 13, 1949.

Gerber, M.S., 1992, *Legend and Legacy: Fifty Years of Defense Production at the Hanford Site.* WHC-MR-0293, Rev. 2, Hanford Reading Room Number 10879, September 1992.

Gerber, M.S., 1992b, *Past Practices Technical Characterization Study – 300 Area – Hanford Site,* WHC-MR-0388, Westinghouse Hanford Company, December 22, 1992.

Gross, C.N., 1956, *Plutonium and fission product contamination in Uranium*, HW-10515, May 26, 1956.

Gross, P.D., 1965, *Hanford Shield Masonite Attenuation Effectiveness*, RL-REA-1031, April 15, 1965.

Hanford Internal Dosimetry Project Manual, PNNL-MA-552, Richland, WA, on Web at http://www.pnl.gov/eshs/pub/pnnl552.html.

Harmon, M.K., 1955, *Technical Appraisal of Redox Ruthenium Problems and Their Resolution,* HW-35496, Addendum to HO-34882, Hanford Atomic Products Operation, Richland, WA, February 24, 1955.

Hauff, T. W., 1955, Memorandum to H.H. Schipper, Chief Engineering Branch, U.S AEC, Hanford Operations Office, *Shielding Effectiveness at the KE Reactor*, HW-39286, November 7, 1955.

Healy, J.W., 1948, *Bioassay at Hanford*, HW-10522, Hanford Works, Richland, WA, 1948.

Heeb, C.M., 1994, *Radionuclide Releases to the Atmosphere from Hanford Operations, 1944 – 1972,* IPNWD-2222 HEDR, Appendix A, Pacific Northwest Laboratories, Richland, WA, 1994.

Henle, R.C., 1962, Letter to F. Swanberg, Jr., *1961 Whole Body Counter Results*, dated January 9, 1962.

Hine, G.J. and Brownell, G.L., 1956, *Radiation Dosimetry,* Academic Press, Inc., New York, NY, 1956.

Irish, E.R., 1955, *Technical Appraisal of Redox Ruthenium Problems and Their Resolution,* HW-34882, Hanford Atomic Products Operation, Richland, WA, January 21, 1955.

Ingalls, W.P., and Sanborn, K.L., 1956, *Reduction of airborne contamination UO_3 plant*, HW 43099, May 11, 1956.

International Commission on Radiological Protection (ICRP), 1959, *Report of Committee II on Permissible Dose for Internal Radiation,* ICRP Publication 2, Pergamon Press, London, 1959.
International Commission on Radiological Protection (ICRP), 1975, *Reference Man, Anatomical, Physiological and Metabolic Characteristics*, ICRP Publication 23, Pergamon Press, Oxford, England, 1975.

International Commission on Radiological Protection (ICRP), 1982a, *Protection of the Patient in Diagnostic Radiology,* ICRP Publication 34, Annals of the ICRP, Vol. 9/2, 1982.

International Commission on Radiological Protection (ICRP), 1983, *Protection of the Patient in Diagnostic Radiology*, ICRP Publication 34, Pergamon Press, Oxford, England. 1983.

International Commission on Radiological Protection (ICRP), 1990, *Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Annals of the ICRP, Vol. 21, Pergamon Press, Oxford, England.

International Commission on Radiological Protection (ICRP), 1994, *Human Respiratory Tract Model for Radiological Protection,* ICRP Publication 66, Pergamon Press, Oxford, England 1994.

International Commission on Radiological Protection (ICRP), 1996, *Conversion Coefficients for Use in Radiological Protection Against External Radiation,* ICRP Publication 74, Pergamon Press, Oxford, England, 1996.

International Commission on Radiological Protection (ICRP), 2002, Supporting Guidance 3, Guide for the Practical Application of the ICRP Human Respiratory Tract Model, Vol. 32/1-2, Pergamon Press, Oxford, England. 2002.

Kathren, R.L., Shockley, V.E., Thomas, E.M., and Taulbee, T.D., 2003, *Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures*, ORAUT-TIB-0006, Oak Ridge Associated Universities, Oak Ridge, TN, 2003.

Kiel, G. R., 1954, *Review of Redox Plant Head End Treatment*, HW-32164, Hanford Atomic Products Operation, Richland, WA, June 17, 1954.

Kleinknecht, R.E., 1960, *Effect of High Temperature on Masonite*, HW-64868, April 14, 1960.

Makhijani, A., 1988, *Release Estimates of Radioactive and Non-Radioactive Toxic Materials to the Environment by the Feed Materials Production Center 1951-85*, Institute for Energy and Environmental Research, Takoma Park, MD, July 7, 1988.

Mani K.V., 1983, *Assessment of Oak Ridge National Laboratory Internal Dosimetry Data for DOE Health and Mortality Study*. In: Department of Epidemiology. Chapel Hill, N.C.: University of North Carolina at Chapel Hill, 1983;88.

Marceau, T. E. et al., *History of the Plutonium Production Facility at the Hanford Site Historic District, 1943 through 1990*, DOE/RL/97/1047, U.S. Department of Energy, Richland Washington, June 2002.

McConiga, M. W., Soldat, J. K., 1957, *Radioactive Contamination in the Hanford Environs for the Period April, May and June 1957*, HW-52803, Richland, WA, October 28, 1957.

Merwin, S.E., 2003, *Dose Reconstruction Project for NIOSH – External Dose Reconstruction*, ORAUT-PROC-0006, Rev. 00, PC-2, December 11, 2003.

National Bureau of Standards (NBS), 1953, *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, NBS Handbook 52, Superintendent of Documents, Washington, DC, 1953.

National Bureau of Standards (NBS), 1959, *Maximum Permissible Body Burdens and Maximum Permissible Concentrations in Air and Water for Occupational Exposure*, NBS Handbook 69, Superintendent of Documents, Washington, DC. 1959.

National Council on Radiation Protection and Measurements (NCRP), 1989, *Medical X-Ray, Electron Beam, and Gamma-Ray Protection for Energies Up to 50 MeV (Equipment, Design, Performance, and Use),* NCRP Report 102, 1989.

National Council on Radiation Protection and Measurements (NCRP), 1996, *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground,* NCRP Report 123, Vol. 1, Bethesda, MD, May 9, 1996.

National Council on Radiation Protection and Measurements (NCRP), 1999, *Recommended Screening Limits for Contaminated Surface Soil and Review of Factors Relevant to Site-Specific Studies*, NCRP Report 129, ISBN 0-929600-61-4, 1999.

National Institutes of Occupational Safety and Health (NIOSH), 2002a, *External Dose Reconstruction Implementation Guideline*, OCAS-IG-001, Rev. 1, August 2002.

National Institutes of Occupational Safety and Health (NIOSH), 2002b, *Internal Dose Reconstruction Implementation Guideline,* OCAS-IG-002, Rev. 0, August 2002.

National Institutes of Occupational Safety and Health (NIOSH), 2002c, *NIOSH-InteractiveRadioEpidemiological Program (NIOSH-IREP) Technical Documentation, Final Report,* NIOSH, Office of Compensation Analysis and Support, Cincinnati, OH, June 2002.

Neton, J.W., 2004, *Estimation of Ingestion Intakes*, OCAS-TIB-009, Rev. 00, Office of Compensation Analysis and Support, April 13, 2004.

Nichols, L.L., Endres, G. W.R., Shipler, D.B. , Oscarson, E.E. , and Crass, L.L., 1972, *Hanford Multipurpose TL Dosimeter Field Tests and Evaluation, BNWL-B-127*, Pacific Northwest National Laboratory, Richland, WA, 1972.

Oak Ridge Associated Universities (ORAU), 2003, *ORAU NIOSH Dose Reconstruction Project Quality Assurance Program Plan*, ORAUT-PLAN-001, Oak Ridge, TN, January 30, 2003.

Parker, H.M., 1954, *Status of Ground Contamination Problem*, HW-33068, September 15, 1954.
Patterson, C.M., 1949, Letter to L.S. Taylor, National Bureau of Standards, *General Operating Tolerances*, HW-12710, Hanford Works, Richland, WA, March 11, 1949.

Perkins, R.W., 1960, *A Tentative Bioassay Procedure for the Measurement of Np-237*, HW-67067, October 10, 1960.

Peterson, E. and Smalley, W.L., 1960, *Proposed Procedure for Reducing Reactor Front Face Dose Rates*, H.W.-66117, July, 19, 1960.

Peterson, E.S. and Bunch, W.L., 1962, *The Effect of Fringe Poison on Heat Generation in the Shield Complex*, HW-73064, March 10, 1962.

PNL-SA-13657/DE86 – To be provided.

Richards, R.B., 1949, Memo to P.F. Gast, *Plutonium Isotopes in Recycled Uranium*, February 21, 1949.

Richardson, D., Wing, S., Watson, J., and Wolf, S., 1999, *Missing Annual External Radiation Dosimetry Data among Hanford Workers,* J Expo Anal Environ Epidemiol, 9(6): 575-85, Nov-Dec 1999.

Richland Operations Office, 2000, *Review of Generation and Flow of Recycled Uranium at Hanford,* DOE/RL-2000-43, June 30, 2000.

Rollins, G., 2004, *Technical Information Bulletin – Maximum Internal Dose Estimates for Certain DOE Complex Claims*, ORAUT-OTIB-0002, Rev. 01, PC-2, Oak Ridge Associated Universities, Oak Ridge, TN, May 7, 2004.

Savignac, N., 2003, *Technical Basis Document for the Hanford Site – Occupational Environmental Dose*, ORAUT-TKBS-0006-4, Rev. 00, Oak Ridge Associated Universities, Oak Ridge, TN, October 15, 2003.

S. Cohen & Associates (SC&A), 2004, *SC&A Standard Operating Procedure for Performing Site Profile Reviews*, Rev. 0, March 18, 2004.

SC&A, 2004, *Review of NIOSH Site Profile Review of NIOSH Site Profile for Bethlehem Steel Plant, Lackawanna, NY*, Task 1, Document No. SCA-TR-Task1-0001, September 22, 2004.

Scalsky E. D, 2003, *Technical Basis Document for the Hanford Site – Occupational Medical Dose*, ORAUT-TKBS-0006-3, Rev. 00, September 16, 2003, Oak Ridge Associated Universities, Oak Ridge, TN.

Scalsky E.D, 2004a, *Technical Basis Document for the Hanford Site – Introduction*, ORAUT-TKBS-0006-1, Rev. 01, Oak Ridge Associated Universities, Oak Ridge, TN, January 9, 2004.

Scalsky, E.D., 2004b, *Technical Basis Document for the Savannah River Site To be Used for EEOICPA Dose Reconstructions*, ORAUT-TKBS-0003, Rev. 02, Oak Ridge Associated Universities, Oak Ridge, TN, October 29, 2004.

Selby, J., 2004, *Technical Basis Document for the Hanford Site – Site Description*, ORAUT-TKBS-0006-2, Rev. 00 PC-1, Oak Ridge Associated Universities, Oak Ridge, TN, December 29, 2004.

Siebert, S.R., 2004, *Technical Information Bulletin – Tritium Calculated and Missed Dose Estimates*, ORAUT-OTIB-0011, Rev. 00, Oak Ridge Associated Universities, Oak Ridge, TN, June 29, 2004.

Smalley, W.L., and Loundagin, R.L., 1961, *Inlet Shield Radiation Penetration Studies*, HW-69348, April 27, 1961.

Spiers, F.W., 1946, *Effective Atomic Number and Energy Absorption in Tissues*, British Journal of Radiology, Vol. 9:52, 1946.

Stradling, N., Hodgson, A., Ansoborlot, E., Berad, P., Etherington, G., Fell, T., Rance, E., Le Guen, B., 2002, *Industrial Uranium Compounds: Exposure Limits, Assessment of Intake and Toxicity after Inhalation*, NRPB-W22, Chilton, Didcot, ISBN 0 85951 493 5, 2002.

Strome, D., 2003, Technical Information Bulletin – *Occupational Dose from Elevated Ambient Levels of External Radiation*, ORAUT-OTIB-0007, Rev. 00, Oak Ridge Associated Universities, Oak Ridge, TN, October 12, 2003.

Sula M.J., Carbaugh E.H., Bihl D.E., 1989, *Technical Basis for Internal Dosimetry at Hanford*. Richland, WA: Pacific Northwest Laboratory, 1989.

Swanberg Jr., F., 1961, *Hanford Whole Body Counter Results for 1960*, HW-67245, April 1961.

Taylor, G.A., Crase, K.W., La Bone, T.R., and Wilkie, W.H., 1995, *A History of Personnel Radiation Dosimetry at the Savannah River Site,* WSRC-RP-95-234, Westinghouse Savannah River Company, Aiken, SC, 1995.

Taylor, G. A., 2000, *The Evolution of Dosimetry Bioassay Methods at the Savannah River Site*, WSRS-MS-2000-00290, Westinghouse Savannah River Company, Aiken, S.C.

Thompson, 1986, *UO_3 Plant operating specifications*, OSD-U-185-0001, 1986.

Till, J., Voilleque, P.G., Killough, G.G., Rope, S.K, 2002, *Final Report, Methods for Estimating Radiation Doses from Short-lived Gaseous Radionuclides and Radioactive Particles Released to the Atmosphere During Early Hanford Operations*, Centers for Disease Control and Prevention, Department of Health and Human Services, November 2002, see on Web at http://www.cdc.gov/nceh/radiation/hanford/HanfordParticles.pdf .

Unruh, C.M., 1956, Letter to R.L., Junkins, *Reactor Building Dose Rates Associated with Ball Valve Operation*, HW-43814, July 19, 1956.

U.S. Atomic Energy Commission, Letter from [illegible], Manager, Oak Ridge Operations to C.E. Center, Vice President, Union Carbide Corp, July 28, 1960.

U.S. Department of Energy, *The Report of the Joint Task Force on Uranium Recycle Materials Processing, Oak Ridge Operations*, Report No. DOE/OR-859, September 1985.

U.S. Department of Energy, Office of Environment, Safety and Health, *Exposure Assessment Project at the Paducah Gaseous Diffusion Plant,* December 2000.

U.S. Senate Committee on Governmental Affairs, Majority Staff Report, *Early Health Problems of the U.S. Nuclear Weapons Program and their Implications for Today*, December 1989.

Watson, E.C., 1959, *A Review of the NTA (Fast Neutron) Film Program,* HW-61008, Hanford Atomic Products Operation, Richland, WA, 1959.

Watson, J.E., Jr., Wood, J.L., Tankersley, W.G., and West, C.M., 1994, *Estimation of Radiation Dose for Workers Without Monitoring Data for Retrospective Epidemiologic Studies*, *Health Physics*, C. M. 67(4): 402-405, 1994.

Wilson, L.A., 1955, *Hanford Shield Masonite Deterioration Studies*, HW-35202, August 8, 1955.

Wilson, R. H., 1987, *Historical Review of Personnel Dosimetry Development and Its Use in Radiation Protection Programs at Hanford, 1944 to the 1980s*, PNL-6125/UC-41, Pacific Northwest Laboratory, Richland, WA, February 1987.

Wilson, R.H., Fix, J.J., Baumgartner, V., and Nichols, L.L., 1990, *Description and Evaluation of the Hanford Personnel Dosimeter Program From 1944 Through 1989*, PNL-747, Pacific Northwest Laboratory, Richland, WA, 1990.

Wing S., Richardson, D., Wolf, S., and Mihlan, G., 2004, *Plutonium-Related Work and Cause Specific Mortality at the United States Department of Energy Hanford Site*, American Journal of Industrial Medicine 45:153-164, 2004.

ATTACHMENT 1: HANFORD SITE QUESTIONS SUBMITTED TO NIOSH

General Questions:

1. ORAU is relying on the claimant interviews and personal dosimetry records to identify incidents in which individuals may have been involved. Other sources of data exist (although they may be somewhat biased) that can provide valuable data on incidents such as occurrence reports, employee concern programs, DTPA administration lists, REX incident listings, and former or current employees. SC&A requests information on how the following issues have been addressed:

   - An incident may not be identified in the interview process and may not be available in the personnel radiation exposure file (i.e., it is stored elsewhere). What alternate mechanism does ORAU use to identify incidents and occurrences, which could impact dose reconstruction?

   - How are incidents such as spills, clothing and personnel contamination, and area contamination spreads that could result in internal exposure accounted for in dose calculations?

2. The list of radionuclides used in dose reconstructions appears to be incomplete:

   - Early ^{210}Po production (WWII to early 1950s) seems to have been omitted. Has this been taken into account in worker doses? Are there bioassays relating to it. How was the neutron radiation from Po-Be and Ra-Be sources taken into account in dose reconstruction?
   - Areas in the reactor building had airborne beta contamination on the order of 10^{-6} µCi/liter. Some of this activity resulted from release of fission gases in the work environment. Did the ORAU team consider the possible impact of internal and submersion dose from these gases?
   - Impurities such as neptunium, plutonium and fission products are present in recycled uranium. The composition of the RU samples was based on samples evaluated by ORAU for a few lots in 1988 and 1993. What is the technical basis behind applying data from samples in 1988 and 1993 to early operations with recycled uranium at Hanford?
   - The TBD uses the ORIGEN calculations to provide information on the most predominant radionuclides of interest at the facilities. Did ORAU consider other data such as field characterization data which may better indicate the radionuclides encountered by the workers and the relative ratios of isotopes?

3. How are large episodic environmental releases (e.g., fuel element failures in the 100 Area, waste site releases, the Green Run, T&B stack releases from 1945-51, Redox particle releases from 1952-54, ^{131}I releases, tritium releases), purposeful or accidental, accounted for in the calculation of environmental dose for workers in the vicinity of these releases? Is ORAU using an annual average stack release to assess environmental

exposure? If so, will this not result in significant underestimates of exposures for early workers and those workers present during episodes like the Green Run? How are environmental exposures from episodic releases (whether accidental, process, or deliberate) being taken into account?

4. Hanford has undergone significant deactivation and decommissioning operations since the shutdown of facilities began in the 1960s. Deactivation and decommissioning activities had included demolition of buildings, removal of equipment, and environmental remediation of waste sites, cribs, trenches and other areas. The TBD does not address the method for assigning dose to these workers. How does ORAU deal with the unique hazards associated with these sites? How are alpha exposures and activation product exposures being estimated? How are the issues of exposure geometry being handled? How is ORAU determining which D&D workers were monitored and which were not? How is the issue of transient workers and workers who are moved from one job to another being handled, especially if they are not monitored for at least some of the time?

5. There are several potentially significant issues regarding external dose measurement technology that have resulted in the questions listed below. SC&A requests information on how the following issues were handled:

   - A preliminary look at historical data for chest photofluorographic exams shows an ESE value in excess of the 1.53 R. Hine and Brownell (1956) identified an ESE value of 3.1 R for lateral projections. Did ORAU take into account literature from sources other than those generated at Hanford with regard to organ doses from medical exposures?
   - According to Spiers (1946) the photoelectric absorption by bone from low-energy photon x-rays is very large and the main effect in reducing deep dose. He found that in exposure conditions similar to those at Hanford, more energy is delivered to the bone layer at a depth of 6 to 9 cm than is delivered to the skin of the patient. How has ORAU compensated for this?
   - How was the potential underestimation of dosimeters with respect to energy limitations handled (i.e., neutrons and low-energy photons)?
   - It is well known that the response of personnel dosimeters is highly dependent on the angle of incidence of impinging radiation. This has been documented for both beta/photon and neutron dosimeters. What correction factors were applied with respect to angle of incidence?
   - Partial body exposures could result in significantly higher dose to an organ than what is measured on the film badge. Could ORAU explain how these situations are determined and how the organ dose is calculated in the case of partial body exposures?
   - The data presented in the TBD (0006-6) indicate a systematic underestimation of doses by film badges relative to TLD readings by perhaps 30%. But ORAU says that the two agree "reasonably well." Could ORAU explain how the data in Figure 6-4 of 0006-6 were used to arrive at this conclusion, and what adjustments if any are being made for the film badge/TLD discrepancies in practice?

- It appears to SC&A that 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. ORAU appears not to have taken this into account in the TBDs. Was this issue evaluated and determined not to be significant? If so, please provide us with the technical basis for this conclusion. Were these issues considered with neutron dosimetry?
- The 225 waste processing does not list ^{90}Sr brehmstrahlung radiation? Did ORAU decide that this is not important? If so, please give us the technical reasoning behind this assumption.

6. There are several potential issues regarding internal dosimetry that have resulted in the questions listed. SC&A requests information on how the following issues were handled:

- Early in Hanford history, there were individuals referred to as "Rovers." These individuals were not necessarily assigned to a particular area or building but tended to move throughout the site. The dose reconstruction process with respect to these individuals is not clearly defined in the technical basis document. What is the approach to reconstructing the doses of these workers?
- How does ORAU properly quantify the uncertainties for whole body counting prior to 1993, since there was no default set of radionuclides until that timeframe?
- Have uncertainties been developed (particularly for the 1945-1946 time period) to account for variations in instrument response during thyroid scans, especially when considering ^{131}I thyroid burdens that exceed 2μCi?
- How was ^{131}I dose accounted for from 1946 until the 1960's? Since monitoring for ^{131}I was performed for only some high risk workers, how did you determine which workers were monitored and the uncertainties associated with this monitoring?
- Since worker doses can be high in many waste handling operations, are NIOSH and ORAU working on TBD supplements that explain the types of radionuclides and absorption types for waste management facilities (i.e., tank farms, evaporations, transfer lines, etc.)?
- Have NIOSH and ORAU considered further refining inhalation classes for Hanford uranium compounds, providing more comprehensive information on solubility and making relevant associated studies available to the dose reconstructors?
- Have procedures been established and has appropriate raw data been made available to provide claimant-favorable dose estimates for workers exposed to contamination spreads in reactor buildings, the 231-Z plutonium isolation facility, the concentrator building, and the uranium metal fabrication shops from 1943-1946?
- How does the fact that the site profile discusses specific buildings and their radionuclides of concern, yet the internal TBD (0006-5) only provides general area information that may have an affect on dose reconstruction?
- Explain why the assumptions used to derive the tables in Attachment D are claimant favorable.

(Note: The amount excreted in urine varies with the time when sample was taken. For ^{239}Pu for example, type S, 5μm chronic inhalation during 2 years, 5 days a week working conditions. Assuming a given sample result, if the sample was taken just after the 2 years work, the expected intake rate is 12 times smaller than the expected intake rate from a urine sample taken 30 days after the beginning of the work and 50 times lower than if the sample was taken 1 day after the beginning of the work. Actually the sample taken just after the termination of the work is the one that gives the lower intake rate, from all samples taken after termination of work and before termination of work. Thus this is the less conservative approach in terms of dose calculation.)

7. SC&A is seeking additional information on four specific issues/questions pertaining to the Hanford Site Profile, TBD ORAUT-TKBS-006-4, as cited below.

- **Issue #1: 4.3.1 Ambient Radiation**

Section 4.3.1 of the TBD deals with ambient external gamma doses at Hanford for the years 1945 through 2001 and cites HW-9871 and Till et al. 2002 as the primary references. The TBD states that, for the early years, ambient dose rates (from localized fallout) were principally due to the release of I-131 and to a much lesser extent Ru-106 (see Figure 4.3.1-3 of TBD) from the chemical separation area (200 East).

Yearly external doses from ambient doses are given in Table 4.3.1-1 of the TBD for various locations, which includes the 200 East area. However, no roadmap is provided regarding their derivation other than the two references.

Review of reference HW-9871 identified that two sets of Victoreen ionization chambers were used: chamber "C" and chambers "M" and "S." The description of these chambers is enclosed as Exhibit #1 and identifies the fact that they were essentially identical except that the sides of the "C" chambers were protected by semi-waterproof cardboard. (Note: the 1 mil Al corresponds to a wall thickness of 68 mg/cm^2).

Locations for the two sets of chambers (relative to 200 East) are provided by Till et al. 2002 and identified in Exhibit #2.

Actual dose rate measurements (mrep per 24 hours) for "C" and "M & S" chambers are contained in HW-9871 and enclosed as Exhibits 3A and 3B. It should be noted that chambers "C" and "M and S" did not share common locations. For illustration, I have normalized the data as average daily doses for 1945 and for the following locations:

- C chambers for 1945 in 200 E Area: 0.938 mrep/day
- M & S chambers (within 5 miles of 200 East at the Rt. 45 mile 6 or map location 23) for 1945: 6.92 mrep/day
- M & S chambers (within 5 miles of 200 East at the "Meteorology" or map location #26) for 1945: 5.56 mrep/day

At these discrete locations, the 2000 hr/y dose would translate to the following yearly doses:

Chamber C at 200 East: (0.938)(2000/24) = 78 mrep
Chambers M & S at Rt. 45 mile 6: (6.93)(2000/24) = 577 mrep
Chambers M & S at Meteorology: (5.56)(2000/24) = 463 mrep

Table 4.3.1-1 of the TBD identifies a single value of 194 mrem/y for the 200-E area for 1945, which cannot be matched with any of the above (or other monitored) locations.

Questions:

1. How was this value (i.e., 194 mrem/yr) derived?
2. Is this an average value for all C and M & S chambers located in the 200-E area out to 10 miles?

- **Issue #2: Intake of Airborne Radionuclides**

Annual internal exposures from inhalation are discussed in Section 4.2.3 of the TBD and are derived for discrete locations based on (1) annual radionuclide emissions, (2) annual average atmospheric dispersion coefficients, and (3) breathing volumes.

While annual emission and the assumed annual inhalation volume of 2400 m^3 are verifiable and/or reasonable, what may not be reasonable are the assumed atmospheric dispersion factors provided in Table A-10. For example, the lowest dilution values for I-131 (released from T Plant and B Plant) at the closest receptor locations (Nodes 1 and 2) are given as 1.18E-07 s/m^3 and 1.35E-07 s/m^3, respectively. Without definitive knowledge, these receptor locations are likely no more than 500 meters from the point of atmospheric discharge. For other onsite locations, values of 10^{-8} to 10^{-10} s/m^3 are common.

Chief among the critical parameters that affect the dispersion factor is the effective release height. The TBD provides no site-specific information regarding the exhaust stacks for the T Plant and the B Plant. In Section 3.2.1 (page 3-10) of the Till et al. 2002 report, however, reference is made to an assumed "physical height of release stacks" of 61 meters. The authors of the TBD furthermore appear to have rejected Till's Hcalc as described in Section 4.3.2 of the TBD, which states the following:

> *The computer program RATCHET and an Excel® spreadsheet were used to calculate claimant-favorable but realistic intakes from airborne radionuclides as opposed to the computer program Hcalc (Till et al. 2002) because Hcalc calculates "worst case" or maximum hypothetical dose as opposed to a realistic dose. Examples of the conservative overbiasing in the Hcalc program include:*
>
> - *The program neglects plume rise at the point of release of radionuclides to the atmosphere. Without plume rise the program*

overestimates air concentrations and deposition on vegetation and soil near the point of release.

- *The program neglects plume depletion caused by wet and dry deposition. Without plume depletion this program overestimates air concentrations and deposition on vegetation and soil.*

- *The program overestimates the radionuclides concentrations in the shallow surface soil and the external gamma dose because the program neglects leaching.*

- *The program assumes a 12-year, as opposed to the NCRP (1996) default value of 30-year, buildup of radionuclides in the root zone of soil. The smaller buildup maximizes root uptake for the food ingestion pathway.*

For many of the radionuclides released and considered in the report, plume rise, plume depletion, and leaching were not likely to have had any impact. For example, close-in receptor locations are not significantly affected by depletion, and for short-lived nuclides (e.g., I-131), leaching may reasonably be neglected.

In brief, additional information and data are required that would verify NIOSH's assumptions and methods used to derive the cited annual average atmospheric dispersion factors, as given in Table A-10. For ground or near-ground releases, X/Q values that are several orders of magnitude greater than those given in Table A-10 of the TBD could easily apply. For example, the following table of generic average annual X/Q values as a function of distance and height of release, potentially reflect dispersion factors (X/Q values) that are significantly larger than those cited in the TBD. The values were derived using the recommendations in NCRP Report No. 123 for stack releases that are 2.5 times higher than the height of nearby buildings to avoid building wake effects. If the releases are affected by building wake, dispersion would be greater in the near field and the closer in X/Q values will be smaller. We used the recommended default wind speed of 2 m/sec and assumed the wind blows toward the receptor 25% of the time. (It is our understanding that the prevailing winds at the Hanford site are west to east.) The results reveal that the X/Q values used in the Hanford site profiles are appropriate only if the releases were relatively uniform over the course of a given year, and that the releases were from a stack with a height of about 100 meters. If any of the releases were, in fact, closer to ground level or highly episodic, the doses are likely to have been underestimated by over 1 order of magnitude.

Generic Average Annual Dispersion Factors

	Average Annual X/Q (sec/m^3)			
	Release Height (m)			
Distance (m)	**0**	**10**	**50**	**100**
500	2.50E-05	2.50E-05	3.13E-06	5.00E-07
1000	7.50E-06	7.50E-06	3.13E-06	5.00E-07
2000	2.50E-06	2.50E-06	1.25E-06	5.00E-07
3000	1.13E-06	1.13E-06	7.50E-07	4.38E-07
4000	7.50E-07	7.50E-07	5.00E-07	3.75E-07
5000	5.00E-07	5.00E-07	3.75E-07	2.50E-07

C/Q = fP/u, if H>2.5x height of building
f = 0.25
u = 2 m/s

- Issue #3: Internal Exposures from Resuspension and Inhalation of Local Fallout

For select years, local fallout was dominated by the release of I-131. As stated in the TBD, ". . . up to 90,000 curies of I-131 were released each month [in 1945] . . ." However, the TBD makes no reference to or provides data for internal exposure that involves the resuspension/inhalation pathway for I-131 (or any other radionuclide).

The potential for thyroid exposures from this pathway are significant and may exceed the plume inhalation pathway, as explained below.

Method for Estimating Potential Thyroid Doses from Resuspension

As provided in reference HW-9871 (or enclosed Exhibits 3A and 3B), average annual "ambient dose" rates, as measured by the M & S chambers can be used to infer the freshly deposited quantity of I-131, which must reasonably be assumed to represent the overwhelming source term for these empirical measurements. Thus, for 1945, at the Rt. 45 mile 6 location, the average 24-hour time-integrated dose was 6.93 mrad/day.

The following assumptions may be used to derive resuspension/inhalation thyroid doses at the Rt. 45 mile 6 location:

- The 6.93 mrad/24-hr corresponds to 0.289 mrad/hr dose rate.
- From Federal Guidance Report 12, Table III.3, we derive the dose coefficient for exposure to I-131 contaminated ground surfaces.

- Using the thyroid dose coefficient (which corresponds to approximately the 1000 mg/cm^2 deep dose equivalent), the coefficient of 3.71E-16 Sv/sec per Bq/m^2 is converted to 4.94E-3 mrem/hr per μCi/m^2. By simple inversion, this coefficient implies that, to get 1 mrem/hr dose rate from freshly deposited I-131, the contamination level of 202 μCi/m^2 is required.

- Thus, the average annual dose rate of 0.289 mrad/hr observed at the Rt. 45 mile 6 location translates to an average surface contamination of 58.5 $\mu Ci/m^2$.

- Till et al. (2002) identified resuspension factors for Hanford that corresponded to ". . . a log-triangular distribution with minimum, likeliest, and maximum values of 2×10^{-6}, 5×10^{-4}, and 1×10^{-3} m^{-1}."

- The TBD assumed an annual inhalation volume of 2400 m^3.

- For the "likeliest" resuspension factor of 5×10^{-4} m^{-1}, the annual intake would correspond to 70.2 $\mu Ci/y$:

$$70.2\ \mu Ci/y = (58.5\ \mu Ci/m^2)(5 \times 10^{-4}\ m^{-1})(2400\ m^3/y)$$

- Using EPA's Federal Guidance report thyroid DCF of 1.1 rad per μCi inhaled, this translates to 77.14 rads to thyroid for 1945 to a worker at this location.

It should further be noted that the Federal Guidance Report 12 surface contamination dose rate coefficient does not reflect "roughness," which can be assumed to reduce the observed dose rate by a factor of two. In turn, this would imply that an observed dose rate of 1 mrem/hr actually reflects a surface contamination that equals 2×58.5 $\mu Ci/m^2$ or 117 $\mu Ci/m^2$ with a corresponding resuspension/inhalation thyroid dose of about 154 rads/y.

- **Issue #4: Failure to Separate Episodic from Chronic Releases and Their Consequences to Dose Calculations**

  The TBD acknowledges the releases of Ru/Rh-106 in the text and provides annual release quantities in Table A-7, which for 1953 and 1954 corresponded to about 176 and 353 Ci, respectively.

  The TBD prescribed methodology for the inhalation exposure to Ru/Rh-106 would, therefore, use (1) the annual release rate to derive an average hourly release rate, (2) the average annual atmospheric dispersion factor, and (3) the total annual inhalation volume of 2400 $m^{`3}$ to derive inhalation quantity.

  Our review of HW-34882, however, identifies the fact the only a smaller fraction of Ru/Rh-106 releases were chronic and that large episodic releases occurred over a period of hours to one or two days, as given in the following statements:

  > *Not included* [in the monthly release data] *were 113 and 230 curies of ruthenium emitted in September 5 and 6, 1953 and January 2, 1954 respectively. These emissions were fundamentally the result of equipment failures as discussed HW-32164 . . .*
  >
  > *The two large emission or radio-ruthenium in September, 1953, and January, 1954, spread contamination throughout the Hanford Works area and beyond. The contamination level decreased until April, 1954, and*

> *held fairly constant until July. Since July, there has been a decided decrease in the radioactive intensity of the particles found, although the number of particles has remained relatively constant. Maintaining this present condition will thus, by attrition, furnish decontamination of the Hanford Works area to a point of negligible concern.*
>
> *The ground contamination in the immediate vicinity of Redox (within 3000 feet of the stack) has been more intense than in the outlying areas due to the deposition of radioactive particles and droplets emitted from the stack. This area has been maintained within reasonable radiation levels (approximately six mrads/hour at ankle level) by the use of water sprinkling, which carries the activity into the ground. During September 1954, certain areas outside the Redox exclusion area but within the 200-W Area were plowed to bury contamination and then planted in rye to prevent the top soil from blowing into outer areas.*

In brief, the TBD's treatment of these brief episodic airborne releases as chronic releases results in low estimates for the following reasons:

- When treated as chronic releases, the exposure is reduced by a factor of 4.38 (i.e., 8760 hours per year release/2000 hour-year exposure).

- When treated as chronic releases, the use of the <u>average annual</u> atmospheric dilution factor will severely underestimate actual air concentrations that existed in downwind sectors during the brief episodic emissions. For an episodic release, exposures may have resulted under conditions that yielded atmospheric dispersion values that were several orders of magnitude lower.

Furthermore, as discussed above, fallout resuspension and inhalation/<u>ingestion</u> of Ru/Rh-106 were not addressed in the TBD and may have resulted in large doses if such involved hot particles as described in the TBD.

Lastly, the HW-34882 document makes reference to problems that may have affected exposures to workers <u>indoors</u>, as given in the following statement:

> *. . . the other deleterious aspect of the ruthenium problem, <u>gross</u> contamination of the canyon and crane, continues to be <u>urgent</u> and <u>extremely difficult</u> to control An estimated 6.5 per cent loss in on-stream efficiency is attributed to this difficulty. Primarily because of this crane contamination problem, it is considered prudent to discontinue the permangenate head-end process until a satisfactory alternate is available for use.* [Emphasis added.]

All river water is analyzed by evaporating 500 ml. of the sample (some earlier samples were 100 ml.) and transferring the residue to a 1½ inch diameter stainless steel plate. The residue is counted directly for beta activity using a thin mica-window counter, and a standard alpha counter for the alpha activity.

All counting rates were corrected for counter geometry and sample decay, when known. The limit of sensitivity for the beta and gamma activity measurements in the river water for this type of analysis is approximately 5×10^{-5} uc/liter.

Activity is conventionally expressed in microcuries on the basis of one microcurie corresponding to 3.7×10^{4} disintegrations per second. Inasmuch as the relevant disintegration schemes are rarely known, the purist will note that the true microcurie content is seldom known. It is believed that the conventional microcurie statement is well understood in the biological field, and is more convenient than a statement of measured beta particles in the counters.

Radiation Levels Observed in Air

"C" Chambers and "M" and "S" Type Chambers

Tables 2-A, 2-B, 2-C, 2-D, and 2-E, summarize the radiation level as measured by the "C" type chamber for the period, September 14, 1945 to April, 1948, inclusive.

Tables 3-A, 3-B, 3-C, 3-D, 3-E, 3-F, 3-G and 3-H include the radiation level as measured by the "M" and "S" type chambers for the period July 13, 1945 to April, 1948, inclusive.

The "M" and "S" chambers are cylindrical ionization chambers with 50% of the wall composed of 1 mil aluminum foil. They are placed in pairs at the various locations, charged by a Victoreen mincmeter, and the rate of discharge measured at regular intervals using the same mincmeter. The "C" chamber is basically the same as the "M" and "S" chamber, except that the sides are constructed of heavy semi-waterproof cardboard. The "C" chamber walls will transmit approximately 25% of the beta radiation from 8-day radio-iodine.

Air Sampling (Filter) Program

Tables 4-A, 4-B, and 4-C, summarize the monthly average contamination levels detected in air using air filters for the period January, 1946, to April, 1948 inclusive.

The filter program consists of a continuous intake of air through a filter about 1-3/4 inches in diameter. The rate of air flow through the filter is about two cubic feet per minute. These filters are counted directly on thin mica-window counters. Corrections for rate of activity of these filters are made for geometry, collection efficiency and decay with the assumption that all beta activity comes from 8-day radio-iodine. Data which shows residual long-lived activity after the decay of I^{131} are not incorporated in this report.

W. Singlevich

W Singlevich/kz

HEALTH INSTRUMENT DIVISION

Exhibit 1 HW-9871 Description of Two Sets of Victoreen Ionization Chambers used at Hanford (chamber "C" and chambers "M" and "S")

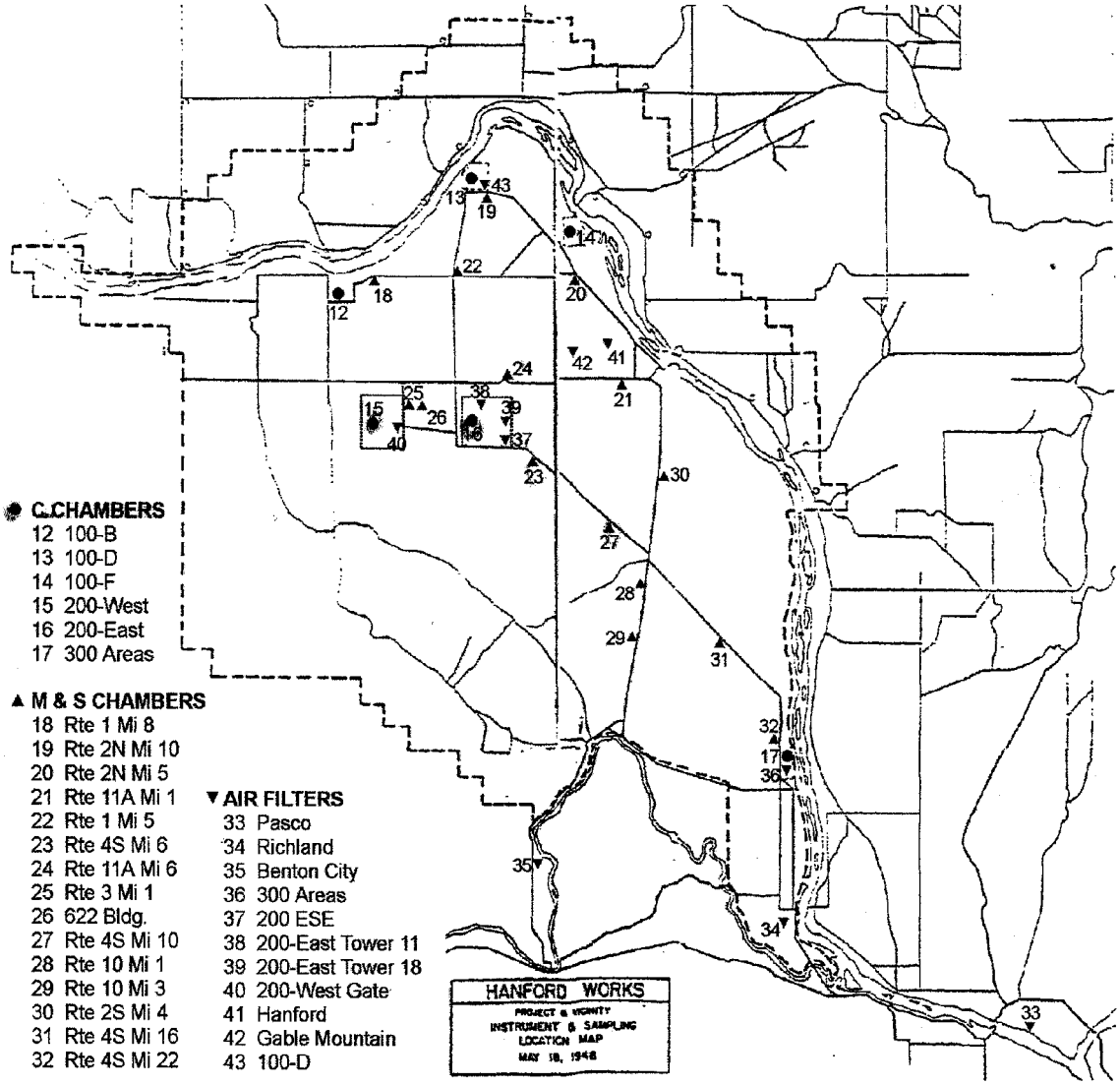

Figure 4-1. Locations for environmental monitoring of airborne radioactivity on the Hanford Works in May 1948. Legibility of the source map (HW-9871) was poor, so the legend and location symbols and numbers were redrawn. Also, river monitoring stations were removed from the original figure.

Exhibit 2 Map Showing Locations for Environmental Monitoring of Airborne Radioactivity

TABLE 2-A

Radiation Levels Observed in Air - Detachable Chambers

(Mrep per 24 hours)

"C" Chambers

-1945-

WEEK ENDING DATE	100-B Area	100-D Area	100-F Area	200-E Area	200-W Area	Outlying Areas
September 14, 1945	- - - -	- - - -	- - - -	1.0	0.4	- - - -
21, 1945	- - - -	- - - -	- - - -	1.7	0.3	- - - -
28, 1945	- - - -	- - - -	- - - -	- - - -	- - - -	- - - -
October 3, 1945	- - - -	- - - -	- - - -	- - - -	- - - -	- - - -
10, 1945	0.5	0.5	0.5	0.5	1.2	- - - -
17, 1945	0.5	0.5	0.6	1.0	0.6	- - - -
24, 1945	0.4	0.5	0.6	1.1	0.5	- - - -
31, 1945	0.4	0.5	0.6	1.0	0.4	- - - -
November 7, 1945	0.5	0.5	0.5	0.8	0.5	- - - -
14, 1945	<0.3	0.5	<0.3	0.8	0.8	- - - -
21, 1945	0.5	0.5	0.3	0.7	0.5	- - - -
28, 1945	0.4	0.4	0.4	0.7	0.5	- - - -
December 5, 1945	0.3	0.4	0.3	0.8	0.5	- - - -
12, 1945						
19, 1945	0.4	0.4	0.4	1.3	0.6	- - - -
26, 1945	0.4	0.4	0.4	0.8	0.6	- - - -

All the above values include the background measurements of the instruments which vary from 0.3 - 0.5 mrep per 24 hours.

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Exhibit 3A HW-9871 Radiation Levels Observed in Air – Detachable Chambers *Table 2-A*, "C" Chambers, 1945

TABLE 3-A

Radiation Levels Observed in Air - Detachable Chambers

"M" and "S" Chambers

(Mrep per 24 hours)

-1945-

WEEK ENDING DATE	July 13	July 20	July 27	Aug. 3	Aug. 10	Aug. 17	Aug. 24	Aug. 31	Sept. 7	Sept.14	Sept.21	Sept 28
100 Area & Environs												
Rt. 1 Mi. 8	- -	- -	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Rt. 2E 100-D	- -	- -	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.7
Rt. 2N 100-F	- -	- -	0.5	0.5	0.5	0.5	0.5	0.7	0.7	0.5	0.5	0.7
Rt. 11A Mi.1	- -	- -	- -	- -	- -	- -	- -	- -	- -	0.7	1.4	2.2
Rt. 1 4N	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	0.7	- -
Within 5 Mi. 200-E												
Rt. 4S Mi. 6	2.6	3.4	2.9	7.9	5.0	5.5	5.5	6.5	10.0	9.4	8.9	9.8
Rt. 11A Mi. 6	- -	- -	1.2	0.7	0.7	0.7	0.7	0.7	1.0	0.7	0.7	2.9
Meteorology	- -	5.3	7.7	6.3	3.8	2.9	2.9	2.9	5.3	3.1	4.3	11.6
Within 10 Mi. 200-E												
Rt. 10 Mi. 1	- -	0.6	1.0	1.0	1.0	1.4	1.9	1.2	1.9	1.9	2.9	1.9
Rt. 10 Mi. 3	- -	1.9	1.0	1.2	0.7	1.2	1.7	1.0	1.7	1.7	2.2	2.2
Rt. 2S Mi. 4	- -	- -	0.7	1.0	0.7	1.4	1.0	1.2	1.2	1.9	1.9	2.6
	2.6	2.8	2.2	3.0	1.98	2.18	2.28	2.25	3.51	3.11	3.48	5.76
Others												
Rt. 4S Mi. 14	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -
Rt. 4S Mi.10	- -	1.0	1.0	1.9	1.7	1.9	1.9	1.4	1.7	1.7	2.9	2.9
300 Area	0.48	0.48	0.5	0.5	0.7	0.7	1.0	0.7	1.2	0.7	1.2	1.7

All the above values includes the background measurements of the instruments which vary from 0.3 - 0.5 mrep per 24 hours.

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Exhibit 3B HW-9871 Radiation Levels Observed in Air – Detachable Chambers *Table 3-A*, "M" and "S" Chambers, 1945

TABLE 3-B

Radiation Levels Observed in Air - Detachable Chambers

"M" and "S" Chambers

(Mrep per 24 hours)

-1945-

WEEK ENDING DATE	Oct. 3	Oct. 10	Oct. 17	Oct. 24	Oct 31	Nov. 7	Nov. 14	Nov. 21	Nov. 28	Dec. 5	Dec.19	Dec. 26
100 Area & Environs												
Rt. 1 Mi. 8	1.0	0.7	0.7	1.0	0.7	1.0	1.0	1.0	0.7	1.0	1.0	1.2
Rt. 2N 100-D	1.0	0.7	1.0	1.2	0.7	1.0	1.0	1.2	0.7	0.7	4.9	0.7
Rt. 2N 100-F	1.0	0.7	1.0	1.0	1.7	1.2	1.0	0.7	0.7	0.7	1.0	1.0
Rt. 11A Mi. 1	1.7	1.7	1.2	1.4	1.2	1.4	1.7	1.4	1.9	1.0	3.8	1.4
Rt. 1 4N	1.0	0.7	0.7	1.0	0.7	1.0	1.0	1.0	0.7	1.0	1.0	1.2
Within 5 Mi. 200-E												
Rt. 4S Mi. 6	12.0	7.2	8.6	8.2	6.0	4.6	6.6	5.8	5.3	7.2	9.6	7.9
Rt. 11A Mi.6	2.4	1.0	1.0	1.4	1.7	2.4	2.9	1.2	4.8	3.1	7.9	9.1
Meteorology	10.3	6.5	5.6	4.8	4.1	7.2	9.4	4.8	5.8	2.9	6.0	4.8
Within 10 Mi.200-E												
Rt. 10 Mi. 1	5.8	6.6	2.9	6.0	3.4	1.9	2.2	2.2	1.4	1.9	2.4	3.8
Rt. 10 Mi. 3	2.4	2.2	7.2	4.6	2.9	1.7	1.4	1.9	2.6	2.2	2.4	4.6
Rt. 2S Mi. 4	2.4	1.9	1.9	2.2	1.9	1.9	2.4	1.4	1.0	1.0	1.4	1.7
	5.88	4.23	4.52	4.57	3.33	2.3	4.13	2.88	3.48	3.05	4.95	5.31
Others												
Rt. 4S Mi. 14	2.9	2.6	1.4	1.7	1.7	1.9	1.5	1.4	1.7	1.4	1.7	3.1
Rt. 4S Mi. 10	6.7	5.5	3.4	2.9	3.1	2.2	4.8	3.4	2.9	2.4	2.4	3.8
300 Area	1.4	1.7	1.4	1.4	1.9	1.4	1.4	1.2	1.2	1.2	1.7	2.9

All the above values include the background measurements of the instruments which vary from 0.3 - 0.5 mrep per 24 hours.

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Exhibit 3B HW-9871, Radiation Levels Observed in Air – Detachable Chambers *Table 3-B*, "M" and "S" Chambers, 1945

Hanford Site Technical Basis Document Questions

1. Is ORAU making the assumption that all tank farm workers have worn respiratory protection for all periods? If not, please specify the assumptions ORAU is using about respirator use in the tank farms.

2. Is the potential for americium intake taken into consideration at the reactors, 233-S, and other facilities outside of PFP, Redox, and Purex?

3. Why are the radionuclides used for internal dose calculations and the radionuclides of concern listed in the site description different for the same facilities? For example, Table A.1-2 in Attachment A to the Site Description TBD lists the radionuclides of concern for all production reactors as ^{76}As, ^{41}Ca, ^{58}Co, ^{60}Co, ^{51}Cr, ^{64}Cu, ^{59}Fe, ^{72}Ga, ^{133}I, ^{135}I, ^{54}Mn, ^{56}Mn, ^{24}Na, ^{97}Nb, ^{239}Np, ^{32}P, ^{31}Si, ^{91}Sr, ^{92}Sr, ^{238}U, ^{90}Y, ^{92}Y, ^{93}Y, ^{65}Zn, and ^{69}Zn. Section 5.4.1 [now Section 5.4.1.1 of the Rev. 01 TBD) of the internal dosimetry technical basis document lists ^{54}Mn, ^{58}Co, ^{60}Co, ^{59}Fe, ^{51}Cr, ^{46}Sc, and ^{99}Mo. ^{154}Eu and ^{155}Eu are also mentioned for N-reactor. How does ORAU define a "radionuclide of concern"?

4. For those buildings not outlined in the site profile, what assumptions are being made with respect to the radiological hazards present? For example, the auxiliary support facilities.

5. How did ORAU ensure that the monitoring practices corresponded to the actual workplace hazards?

6. Are area dosimeter results used to assign dose to unmonitored workers? If so, how?

7. How is missed dose assigned in multiple badge situations? Is there a missed dose assigned for each assigned badge?

8. How did ORAU make corrections for differences in calibration practices over time (e.g., change in calibration source, free air verses phantom, quality factor, etc.)?

9. How are individuals linked to particular environmental receptor locations?

10. Why is there a default to nearby or group dose in the case of missed dose when secondary data may be present (e.g., pocket ionization chamber readings, timekeeping)?

11. There seems to be some disagreement been the SRS and Hanford TBD on the energy range of neutron detection for NTA film. At SRS they claim that the minimum energy which can be detected by the NTA film is 500 keV (WSRC-RP-95-234). At Hanford, they claim the energy sensitivity cutoff was 1,000 keV (PNNL-11196). Why is there a difference?

12. Were the relative concentrations of transuranics in Mark IV Fuel investigated? If so, why is it not called out in the TBD?

13. Has a dose fraction adjustment process been developed, particularly in the plutonium facilities, to correct for the significant under-estimation of photon dose for the Hanford Beta/Photon two-element dosimeter for energies less than 100 keV?

14. How have NIOSH and ORAU dealt with the dose impact to workers from narrow beams of high energy neutron radiation from test ports into reactor cores where dose uncertainties are high?

15. How has NIOSH and ORAU quantified fission product dose from high energy beta radiation during sampling and maintenance during processing plant activities?

16. How has the lack of entry into the electronic database of urinalysis data prior to 1974 affected the ability to provide claimant-favorable dose estimates? Are raw data records being utilized in providing dose estimates prior to 1974?

17. How are NIOSH and ORAU properly accounting for dose from processes carried out in the 231-Z building in the mid 60s to 1970 since specific ^{233}U bioassay techniques had yet to be developed? Why was it assumed that the concentration was 8×10^{-13} $\mu Ci/cm^3$ at the Separation Plant 231-Z?

18. How has missed dose been calculated to account for the presence of ^{241}Pu and ^{241}Am since prior to 1983 the recorded alpha activity from plutonium did not account for the presence of ^{241}Pu and ^{241}Am? Is the Hanford methodology developed to account for the presence of ^{241}Pu and ^{241}Am robust enough?

19. Has potential dose been quantified for workers from ^{210}Po in the 308 building in 1968 and again in 1975? Have the ICRP 23 values for feces be re-evaluated (since the TBD explain they are not conservative) to ensure they are claimant favorable?

20. Has the potential dose from ^{232}Th as a result of fuel fabrication (e.g., 3720 Building fuel target fabrications) and production (e.g., reactor and separation runs for ^{233}U) been accounted for in the dose reconstruction? The site description seems incomplete in this regard.

21. Has the potential for dose been characterized for workers in reactor areas exposed to activation products and potential leaks from fuel rods, especially in fuel storage pools. Have the methods used to monitor reactor workers been carefully reviewed?

22. Is 1,500 pCi per day for 7 hours per day exposure to natural uranium in the 300 Area uranium fabrication buildings valid based on an assumed average air concentration of 2×10^{-10} $\mu Ci/ cm^3$ (Absorption type S) during the period 1944-1947? How many samples collected were above 2×10^{-10} $\mu Ci/ cm^3$?

23. Has NIOSH and ORAU reviewed the codes and radionuclides associated with bioassay at Hanford? Were the radiological characteristics constant during all years of operation?

24. Were biological tests for nephrotoxic effects of uranium done and if so were the tests positive for worker's uranium exposures at the UO_3 plant?

25. Have differences between a Monday morning bioassay sample and a Friday bioassay sample been adequately accounted for when determining an individual worker's internal dose?

26. Has NIOSH and ORAU recently developed the basis for the guidelines for determining intakes from fission product analysis results? The TBD states that the basis for the guidelines is provided in Attachment D 3.2 which is not yet inserted into the internal dose TBD.

27. Explain the reasons for calculating internal doses for 1944-49 in production reactor areas, assuming 200 nCi per year for ^{49}Sc, ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{60}Co, ^{90}Sr, and ^{137}Cs. Why was there a change to a 100 nCi intake per year and which years does this apply to? How are ^{90}Sr and ^{137}Cs treated following the reduction in intake?

28. What was the basis for determining who was put on a bioassay program and how are potential intakes prior to the initiation of bioassay assigned?

References:

Fix, J. J., Wilson, R. H., and Baumgartner, W.V. (1997). *Retrospective Assessment of Personnel Neutron Dosimetry for Workers at the Hanford Site*. PNNL-11196, Pacific Northwest National Laboratory, Richland, WA.

Hine, G.J. and Brownell, G.L. (1956). *Radiation Dosimetry*. Academic Press, Inc. New York, NY.

Spiers, F.W., 1946, *Effective Atomic Number and Energy Absorption in Tissues*. British Journal of Radiology, Vol. 9:52, 1946.

Taylor, G.A., Crase, K.W., La Bone, T.R., and Wilkie, W.H. (1995). *A History of Personnel Radiation Dosimetry at the Savannah River Site*. WSRC-RP-95-234, Westinghouse Savannah River Company, Aiken, South Carolina.

ATTACHMENT 2: CONFERENCE CALL WITH NIOSH AND SC&A

Date: September 17, 2004
Time: 10:00-12:00 pm

Individuals who participated included:

John Mauro, Bob Alvarez, Kathy Robertson-DeMers, Arjun Makhijani, Jim Neton, Dave Allen, Tim Taulbee, Don Bihl (ORAU team), Jack Fix (ORAU team), Ed Scalsky (ORAU team), Arjun Makhijani, Paul Ziemer (Advisory Board), Joe Fitzgerald

SC&A: On June 22, 2004, the SC&A team submitted a long list of questions with respect to the Hanford Site profile. On September 3, 2004, a two-page list of talking points was provided. We will concentrate on these talking points during this call. If we don't cover something you think we should please let us know.

The lead writer on the Hanford site profile report is Tom Bell with support from Bob Alvarez and Kathy Robertson-DeMers. Tom Bell is not on the call as he just had eye surgery.

NIOSH: We assume you will be preparing minutes on this conference call as before.

SC&A: That is correct. We will prepare minutes and provide them to NIOSH for review. To reemphasize, these calls are part of the fact finding process. SC&A realizes it is difficult to prepare written responses to questions provided and as a result the conference call minutes become the record responses to our questions. They become part of our records we maintain on the review process.

NIOSH: We agree that there are some gaps in the TBD as defined in Item #1 of the talking points. The site profile is not meant to be an all inclusive document of everything that happened on the Hanford Site.

SC&A: How does the dose reconstructor determine what is important in the gaps?

NIOSH: We make extremely favorable claimant assumptions.

SC&A: Some radionuclides ^{232}Th and ^{210}Po seem to have been missed.

NIOSH: We realize we need to do a better job with regard to polonium and thorium.

SC&A: How do you now what is important if there is no screening calculation?

NIOSH: We have relied on workers who have complete or fairly complete monitoring like whole body counts and we rely on that to construct missed doses in a completely claimant-favorable manner. We need to do a little better job in the area of thorium and polonium.

SC&A: Thorium seems important since ^{233}U production occurred for a considerable amount of time in the 1950's and 1960's. They processed approximately one ton of the material. As a result of this process thorium was recycled.

SC&A: Will you go back and reevaluate the cases where you may have missed Po and Th doses?

NIOSH: It is our standard practice to go back and re-do the cases when we think we may have missed something.

SC&A: Eventually they gave up on working with ^{233}U due to the extreme hazards associated with the material. Please remember to take into account decay products of ^{232}Th.

NIOSH: We are aware of the decay products issue.

SC&A: What instructions do you give your dose reconstructors regarding departing from site profile?

NIOSH: The worst solubility class is assumed. If necessary, cases will be reevaluated. The site profile is the road map to dose reconstruction and not only piece of information used in the dose reconstruction process. CATI interviews and dosimetry records are also used. Individuals will bring up special situations in their interviews. For example, sheep shearing of animals at animal farm is not covered in the TBD. So we need to look at what the sheep were injected with. Also, fish from retention ponds were not considered and we are evaluating the effect on dose reconstructions that have been done. For people who have say 30 rem based on monitoring records and missed dose of 30 rem and a PoC of 20% then you have to greatly increase dose to get over 50% PoC. Many of these situations may not meet this threshold.

SC&A: How do the dose reconstructors know they have encountered a special situation?

NIOSH: We rely on other information from the CATI interview. Unique circumstances are primarily defined in the CATI interview process.

Another example of a special situation is workers making small radioactive sources at the Hanford reactors. This is not covered within the TBD. Another example of a unique situation is a pilot at the NTS. This individual indicates that he wore cracked goggles when he flew over NTS and this caused his eye cancer.

SC&A: When the CATI is an important contributor, is there a special section on this. And this allows access to unique circumstances. This is a safety net for catching things that are not in the TBD.

NIOSH: Special situations are communicated to the dose reconstructors based on information obtained in CATI interviews so they can make use of them. Initially, this is done via e-mail. Eventually these situations are documented in Technical Information Bulletins if a sufficient number of claimants are affected. Even though a survivor may not know the details of an incident, if the individual was monitored the incident report is not necessary. And we assign these to a senior dose reconstructor. There is no specific guidance.

SC&A: Will you issue a TIB for any of the four gaps defined in the talking points provided? We think this is important for site profiles since survivors may not know the details.

SC&A: Pu urinalysis at Hanford dropped by 60% after 1958 and was replaced with in vivo counting.

NIOSH: The dose reconstructors will assign a missed dose to the claimants. The way we run our program, the less frequently a worker is sampled the greater the missed dose. If the person is monitored once a year the missed dose is higher than someone monitored more frequently.

SC&A: We are developing a system to create a linkage between the site profile, dose reconstruction procedures, and dose reconstruction reviews. We realize that overall strategies may accommodate some of our questions. The marriage of the three provides insight into the effectiveness of the TBDs and the dose reconstruction process. Currently, it is difficult for us to judge how the holes in the TBDs may affect individual dose reconstructions. The site profile reviewers and dose reconstruction reviewers will be holding an internal conference call on Hanford and SRS early next week.

Jim Neton leaves call.

SC&A: It is not clear in the Hanford TBD how the issue of multiple badge dose assignments is addressed. Could you elaborate on this process?

NIOSH: Assignment of dose from multiple badges assigned at one time is specifically addressed in the external dose implementation guide. There is a method for best estimate and maximum estimate.

SC&A: During the pre-1970 period, how did you adjust for the underestimation of the dose by film?

NIOSH: We need to go back and look at this and not prepared to comment it now. It could be significant and we agree with your concern. We may have to go back and re-do the dose reconstruction and issue a report on this issue.

ORAU: Irradiation studies of all types of dosimeters have been performed by IARC. This shows the relative response of film to exposure conditions.

SC&A: Is it in the database?

ORAU: Summary information only is available as the IARC work has not been published.

SC&A: Have you resolved the issue of dosimeter exposure geometry and the relative impact on the dose?

NIOSH: We need to look at this and don't know if there is an underestimate issue. We do not know if a TIB is needed. It is still up in the air and we can't answer that at this time. In one accident situation, workers ran away from the source and it was clearly PA geometry. This was evaluated and estimates were put into dose records.

SC&A: Then you agree that there is an underestimate?

NIOSH: We don't know if there is.

SC&A: How far back was multi-badging done?

NIOSH: The purpose of multi-badging is to evaluate geometry. We recognize the film badge might not be true dose to, say, the lower abdomen, or skin cancers on the hands, etc. These are handled on individual basis right now but we need to address them on a global basis. Multiple badges would have a limited impact on organ dose and there haven't been used.

SC&A: In vivo counters before 1993 may have undercounted dose. After 1958 there was a drop in bioassays because chest counting replaced traditional bioassay techniques. Uncertainties in chest counters were not ironed out until 1993. So if chest counters do not provide an accurate indication of dose, how is the dose reconstruction taking that into account?

NIOSH: Right now the TBDs are in good enough shape to allow us to do people with good monitoring data. We are still working on the situation where only chest counting was done and there were no bioassay samples.

SC&A: How do you account for the potential missed dose?

NIOSH: Bioassay data is used as the primary source of data and results are checked against in vivo data. If there was a big discrepancy between in vivo counting and bioassay, an individualized assessment would be completed.

SC&A: There are cases where workers had positive nasal smears and no chest counting was done. So there is a question of what was the threshold of actually doing a chest count and if you don't know if workers were given a bioassay.

NIOSH: Remember, monitoring doesn't mean exposure and exposure doesn't mean monitoring. We have to see where the worker stands on an individual case basis.

SC&A: As the site profiles are developed to cover general circumstances, there should be some process whereby a situation triggers investigation and issuance of a Technical Information Bulletin. Do you have a mechanism that triggers when you turn on a line of enquiry that leads to a TIB? Is it formalized or ad hoc?

NIOSH: The TIBs are developed more ad hoc. The judgment is made whether the issue affects an individual, or a number of cases. The TIBs develop out of the process.

SC&A: When audit reports are issued, further lines of inquiry will likely need to be investigated. This should trigger TIB development.

Paul Ziemer: We had indicated that part of the reason for an audit is exactly that. That is really part of the whole function of the audit.

SC&A: How will the process work? How will audit findings be incorporated into NIOSH work?

Paul Ziemer: SC&A will present report to the Board and presuming the Board accepts the report, it becomes the Board's report and it gets adopted. That will become our advice then to the Secretary of Health and Human Services and that will also inform NIOSH and their contractor of these issues. The site profile is not as straightforward. If you have issues, NIOSH says that that is a good point and we need to go back and flesh that out. This audit process can cause those changes to occur. So the real effort here is to ask: Is there something here that has been overlooked in regard to importance for compensability and this is crucial? We must remind ourselves that it is not the accuracy of the dose because we are overestimating dose. If it affects compensability then it is very important. If it refines the dose by a few decimal points then it is not important.

SC&A: We are not running IREP and not checking doses.

Paul Ziemer: It is your job to raise the issues and NIOSH should decide how to deal with those questions. You should raise the question, particularly if it not obvious that the issue is not addressed by NIOSH.

SC&A: ORAU met with the Hanford bargaining unit in April 2004. The union brought up the issue of lack of references in the internal dose TBD. To date this has not been updated.

NIOSH: This is fixed and it is in the process of being signed off on and should be on the web site in the next few weeks.

SC&A: Have you looked in the possibility of home bioassay samples being diluted by workers who were approaching the dose limits?

NIOSH: Are there any affidavits?

SC&A: No, there are no affidavits; however, home samples were being allowed.

NIOSH: We have not looked at it. Workers would have to be pretty sophisticated to tamper with internal samples because diluted urine results would jump around and it would be obvious that there was tampering. For example, Pu values that are high one day and low the next are questionable.

SC&A: How do you calculate a dose in the absence of dosimetry?

NIOSH: In the absence of dosimetry, we look for an investigation in the individual files.

SC&A: What about timekeeping? Are these records provided to you and used in dose reconstruction?

NIOSH: We don't receive many of these records. These may be there in cases where they were involved in incidents. Using stay-time and highest dose recorded by area

monitors is not as accurate as using film badge data. If they did not have a film badge then we would go to those reports.

SC&A: What triggers you to use timekeeping records?

NIOSH: If they did not have a film badge, timekeeping records will be used? Whenever there is a gap in the data. I can't think of a situation where we have had to do this.

SC&A: Time sheets may be a concern with a construction worker who is also an operator. Construction workers may have had doses estimated by time sheets.

NIOSH: This depends on where they worked. Coworker data is also an option. We are developing a method to assign coworker dose; however, we have not yet done any dose reconstruction for this type of worker. There will be an addendum to TBD for construction workers for the Hanford site. We are developing those procedures.

SC&A: How do you deal with construction workers who became operators? Is there an indication in the record of construction worker status?

NIOSH: There are records indicating construction worker status.

SC&A: What happens if time sheets indicate missed doses and the dose reconstruction has already been done? Do you revisit the claim? How does it affect a claimant who has been denied?

NIOSH: Are you asking what happens to the individual case?

SC&A: Suppose you find something that applies to many workers such as absence of time sheet information and there are claims that are denied. If you find this is a significant factor, how do you proceed?

NIOSH: When additional information is acquired, NIOSH or its contractor will issue a revision to the TBD, write Program Evaluation Report (PER), and/or reevaluate cases. The PERs are issued whether the additional information affects the cases or not.

SC&A: I have two program evaluation reports from SRS.

NIOSH: There is one more in the works.

SC&A: How were doses to Rovers handled?

NIOSH: We are only doing cases with good data. For Rovers, we assumed the worse case for that particular claim. Work areas were ascertained via PIC and dosimeter data. If we don't have information we would assume typically the most claimant favorable.

SC&A: In addition to radionuclides defined by the ORIGEN code, did you considered field characterization data when determining radionuclides of interest?

NIOSH: ORIGEN was originally used to define radionuclides that occur in large quantities. Dose reconstructors have indicated this hasn't worked well. Currently we are not using field characterization data to identify radionuclides present in different areas.

SC&A: How do you determine isotopic ratios?

SC&A: Where did 1500 pCi per week intake for 1944-47 uranium workers come from? There is no reference in the TBD.

NIOSH: The value of 15 pCi/g number came from air sample data. The data came from monthly reports of health protection department. That got rolled up into single summary report that went to president of General Electric. The high values were reported and others were mentioned. And the determination was based on the general air samples. Once in a while they talk about portable air samples. It is my understanding is that they were no lapel samplers.

SC&A: Is this data on the NIOSH Database?

NIOSH: Yes, there are monthly reports from the facilities.

SC&A: What type of air samplers?

NIOSH: We didn't distinguish the type of air sample.

SC&A: Have you used ICRP 35 guidance regarding general air samples for estimating individual doses?

NIOSH: No, we assumed a chronic intake. This justifies the use of fixed air heads. In an incident situation what a worker gets on his face you can have a sharp gradient. When you have a chronic situation what is on the wall is a good indication of being representative for the workplace also.

SC&A: When we see an issue, it is helpful to have your assessment.

NIOSH: We have revised that to 8-hour day and have changed that from 7-hour days. We had assumed 7-hour days to account for lunch breaks, etc. But 8-hour day is more claimant favorable.

SC&A: The environmental TBD for Hanford pulls information directly out of Till's dose reconstruction report. The John Till dose estimator was created to calculate onsite doses for construction and military workers. Till's approach to determining doses was claimant favorable. Till performed with work for the CDC. Rather than use these data, the Ratchet code was to calculate dispersion values. Ratchet does not allow for individual dose calculations. Furthermore, the Veterans program is currently using the Hanford calculator (i.e., Till's method of calculation.) The exposure from particles would be significantly higher than those determined with Ratchet. Organ doses could have been as high as several tens of rads for lung and GI organs. Why did you substitute Ratchet with the Hanford calculator developed by Till?

NIOSH: We don't have the expertise around this table to give you the answer. We will have to get back to you about that.

SC&A: What about elevated versus ground level exposures? If there is a potential for ground level release, this will increase the dose by an order of magnitude.

NIOSH: We know large number of major releases were from 100m stack.

SC&A: It is not clear to me that as a model designed to address offsite doses it would be able to handle larger particles [which would have settled out].

SC&A: Have you evaluated issues regarding medical x-ray exposure and microdosimetry discussed in previous conference calls?

NIOSH: We haven't looked into microdosimetry issues. We do know that photofluorography was used at Hanford. There are measurement data from Hanford x-ray units published in the American Journal of Roetgenology (Norwood 1958). We are quite confident in 1.53R for Hanford site. In other cases where we don't have as much information, we use 3R.

SC&A: Is it on the database?

NIOSH: If it is not, this can be sent to you.

SC&A: Do you take the solubility and particle size into account together to determine what is most claimant favorable?

NIOSH: We use defaults. There is only one default for particle size that is 5 microns. There are three for solubility and we make a claimant-favorable choice for that. They are not considered together.

SC&A: How do you address dose reconstruction when there is question about respiratory protection?

NIOSH: Coworker data is used prior to area dosimetry data for assignment of dose. Then we would go to area dosimeters. So far we have not done any dose reconstruction with co-workers.

SC&A: How is the change from gross alpha to isotopic urinalysis taken into account?

NIOSH: TBD considers aged weapons grade and 10 year fuel grade mix. The default fuel type used in dose calculations is ten year old fuel with 12% Pu-240.

SC&A: How did you evaluate dose prior to bioassay techniques being developed for a particular radionuclide? Take the SRS case where they did not have a technique to measure ^{241}Am in bioassay during the period when ^{241}Am-241 intakes were likely. There may have been some in vivo counting for a portion of this time.

NIOSH: This is a case by case thing for people with little or no monitoring. This is dependent on individual claim and when these situations arise. We use in vivo data, bioassay data, and source term ratios. If one of these elements is not available, we have delayed these cases.

SC&A: NIOSH has to build trust so We appreciate that NIOSH is in a difficult situation with respect to building trust so people will appreciate how difficult it is to do a dose reconstruction and how sophisticated work it takes.

NIOSH: We would be interested in your thoughts on how we can bridge the gap and help the claimants understand where they are coming from.

SC&A: At some point the claimants are going to have to trust you. How do you go about creating a situation that the claimants trust you?

Paul Ziemer: The audit process will help bridge this gap to some extent. NIOSH is opened to changing items based on the results of the audit reports.

ATTACHMENT 3: SUMMARY OF SITE EXPERT INTERVIEWS – DOSIMETRY AND DOSIMETRY RECORDS

An interview with Hanford Site dosimetry experts was performed on January 5, 2005. Several SC&A audit team members provided TBD-specific related questions, as well as general program questions. The dosimetry experts provided written responses to these questions. The TBD questions and answers are provided below. Information provided on general program questions is provided in Attachment 4.

External Dosimetry

ORAU General Comments:

The content of the Hanford TBD Chapter 6, External Dosimetry, was based on a general approach as follows:

1. An assessment of the response characteristics of the historical dosimetry **technology** used primarily to measure and record worker doses.

2. An assessment of the methods of **calibration** of the technology as used to measure dose since this is necessary to arrive at a consistent basis of evaluation throughout all years of Hanford operations.

3. An assessment of **workplace radiation field** radiation types, spectral and directional properties particularly in reference to #1 and #2.

4. An assessment of **administrative practices** to utilize other methods of workplace monitoring; to conduct investigations of incidents; the practice to assign, process, and exchange dosimeters and to calculate dose; and practices used to convert dosimeter measured doses into compliance dose quantities.

Significant background information is presented in the Hanford TBD prior to presenting dose reconstruction guidance.

SC&A: In TBD 0006-6 the issue of the two-element dosimeter used prior to April 1957 at Hanford underestimating the photon dose was addressed on Pages 16-18, 21, & 24 of the TBD by using the following equation: Adjusted penetrating dose = penetrating dose (i.e., S) + (0.2 * nonpenetrating dose (i.e., OW). How was the 0.2 factor derived and has it been verified to be claimant favorable?

ORAU: The issue in question concerns the performance of the two-element dosimeter in the Hanford plutonium facility only and the potential to under-estimate the primary 11-21 keV plutonium x-ray emission, although there are higher gamma ray energies as well (see DOE-STD-1128-98, *DOE Standard: Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities*). The precise origin of the factor of 0.2 is not fully known but is expected to have been the subject of significant effort to distinguish workers using the two-element dosimeter in Hanford facilities with mixed beta and photon fields and workers in plutonium facilities with essentially no significant beta radiation. Typically, to interpret dose, the an 80

kVp x-ray calibration (i.e., peak energy ~1/3 of 80 or 27 keV) in 1946 or later one of the numerous k-fluorescent x-ray calibrations would be used instead of a uranium source calibration to better estimate dose. There are numerous memos from 234-5 supervisory identifying workers for which the photon calibration must be used to revise the reported dose that was based on the uranium calibration. An assessment of the effects of the most operational pertinent radiation sources 16 keV, 59 keV, ^{137}Cs, ^{90}Sr/^{90}Y and U are presented in Tables 7.3 and 7.4 in the following:

Wilson, R. H., J. J. Fix, W. V. Baumgartner and L. L. Nichols. 1990. *Description and Evaluation of the Hanford Personnel Dosimeter Program From 1944 Through 1989.* PNL-7447, Pacific Northwest National Laboratory, Richland, WA.

The information in these tables, along with information in Appendix D of this report, shows that the given deep dose using a factor of 0.2 times the measured non-penetrating dose would result in a significant over-estimate of the actual deep dose.

SC&A: TBD 0006-6, Tables 6-30 and 6-31 on pages 47 and 48 address the calculation of missed photon and neutron doses. However, the dosimeter MDLs, for the same dosimeters and period, is about half the value in Table 6-30 (used for photon dose calculations) as compare to the MDLs in Table 6-31 (used for neutron dose calculations using the neutron/photon ratios in Table 6-22). Please explain these differences.

ORAU: The Hanford TBD tables in question are:

Table 6-30. Hanford Photon Dosimeter Period of Use, Type, MDL, Exchange Frequency, and Potential Annual Missed Dose

Period of use[a]	Dosimeter	MDL[b] (rem)	Exchange frequency	Max. annual missed dose (rem)[c]
Hanford beta/photon dosimeters				
Prior to October 1944	PIC	0.005	Daily[d] (n=250)	0.625
October 1944 - December 1950	Hanford two-element film	0.040	Weekly (n=52)	1.040
January 1951 - March 1957		0.040	Biweekly (n=26)	0.520
April 1957 - May 1957	Hanford multi-element film	0.040	Biweekly (n=26)	0.520
May 1957 - December 1971		0.040	Monthly (n=12)	0.240
January 1972 - December 1994	Hanford TLD	0.020	Monthly (n=12)	0.120
		0.020	Quarterly (n=4)	0.040
January 1995 to 2003 (ongoing)	Harshaw TLD	0.010	Monthly (n=12)	0.060
		0.010	Quarterly (n=4)	0.020

a. For many years, Hanford workers had a dosimeter assigned to each operating area where they worked.
b. Estimated MDLs for each dosimeter technology in the workplace. Dose values were recorded at levels less-than the MDL.
c. Maximum annual missed dose calculated using MDL/2 from OCAS-IG-001 (NIOSH 2000).
d. Not routinely exchanged.

Table 6-31. Hanford Neutron Dosimeter Period of Use, Type, MDL, Exchange Frequency, and Potential Annual Missed Dose

Period of use	Dosimeter	Exchange frequency	MDL (rem)[a]	Max. annual missed dose (rem)[b]
October 1944–December 1949	PICs with 10B enriched liners	Daily[c] (n=250)	0.010	1.300
January 1950–December 1950	NTA	Weekly (n=52)	0.080	2.100
January 1951–March 1957		Biweekly (n=26)	0.080	1.000
April 1957–May 1957		Biweekly (n=26)	0.080	1.000
May 1957–December 1971		Monthly (n=12)	0.080	0.500
TLD Dosimeter				
January 1972–June 1978	HMPD - 5 chips	Monthly	0.050	0.300
January 1984–December 1994	HMPD - 5 chips	Monthly	0.050	0.300
January 1995 - 2003	Harshaw TLD	Monthly	0.015	0.100

a. Estimated film dosimeter photon radiation detection levels before 1972 and neutron dosimeter MDLs after 1971.
b. Maximum annual missed neutron dose calculated using: Prior to 1972, neutron to photon ratio after combining the recorded and missed photon dose. The actual maximum annual missed dose will be the product of two lognormal distributions. After 1971, the lognormal distribution from the neutron dosimeter using a geometric mean of (n * MDL/2) and an upper 95% confidence interval of (n * MDL), where n is the number of missing dosimeter results.
c. Dosimeter not routinely assigned.

There appears to be an error in the preparation of Table 6-31, highlighted above, that appears in the published TBD version. The values in Table 6-31 should be the same as those shown in Table 6-30. It should be noted that the estimated missed neutron dose in response to the next comment is calculated using a neutron-to-photon dose ratio of 2 and the listed neutron missed dose in the last column of Table 6-31 used by the dose reconstructors is unchanged based on this method of estimating the missed dose.

SC&A: Additionally, the last column in Table 6-31 lists the "Max. annual missed dose (rem)," but appears to not include the neutron/proton dose ratio factors from Table 6-22 as listed footnote (b), and at the top of page 48, for data prior to 1972. Why is this?

ORAU: In accordance with the foregoing response to comment #2 acknowledging an error in this table, an improved revised Table 6-31, with changes highlighted, follows:

Revised Table 6-31. Hanford neutron dosimeter period of use, type, MDL, exchange frequency, and potential annual missed dose.

Period of use	Dosimeter	Exchange frequency	MDL (rem)[a]	Max. annual missed dose (rem)[b]
October 1944–December 1949	PICs with 10B enriched liners	Daily[c] (n=250)	0.005	1.3
January 1950–December 1950	NTA	Weekly (n=50)	0.040	2.1
January 1951–March 1957		Biweekly (n=25)	0.040	1.1
April 1957–May 1957		Biweekly (n=25)	0.040	1.1
May 1957–December 1971		Monthly (n=11.5)	0.040	0.5
TLD Dosimeter				
January 1972–June 1978	HMPD - 5 chips	Monthly	0.050	0.3
January 1984–December 1994	HMPD - 5 chips	Monthly	0.050	0.3
January 1995 - 2003	Harshaw TLD	Monthly	0.015	0.1

a. Estimated film dosimeter photon radiation detection levels before 1972 and neutron dosimeter MDLs after 1971.
b. Maximum annual missed neutron dose calculated using: Prior to 1972, neutron to photon ratio after combining the recorded and missed photon dose. The actual maximum annual missed dose will be the product of two lognormal distributions. After 1971, the lognormal distribution from the neutron dosimeter using a geometric mean of (n * MDL/2) and an upper 95% confidence interval of (n * MDL), where n is the number of missing dosimeter results.
c. Dosimeter not routinely assigned.

This table does incorporate the neutron-to-photon dose factor of 2.1 in plutonium facilities from Table 6-22. The method to be used to most reasonably estimate the actual missed dose is described in footnote b above but the entire topic of missed neutron dose is technically a challenging technical issue.

SC&A: It is well known that the response of personnel dosimeters is highly dependent on the angle of incidence of impinging radiation. This has been documented for both beta/photon and neutron dosimeters. The TBD presents data on this in Tables 6.6 and the footnote instructs the dose reconstructor to correct for angular response by dividing the recorded dose by the table value to estimate Hp(10).

- How were these correction factors derived?
- What determines when the reconstructor will apply these correction factors, and how does he/she know which one to use?

ORAU: The Hanford TBD table in question follows:

Table 6-6. Testing Results for Hanford Two-Element and Multi-Element Film Dosimeters for Energy and Angular Response[a,b]

Beam (energy, keV)	AP exposure			Rotational exposure		
	Film dosimeters		TLD 1972–present	Film dosimeters		TLD 1972–93
	Two-element 1944–56	Multi-element 1957–71		Two-element 1944–56	Multi-element 1957–71	
16[b]	0.1	0.9				
59[b]	0.5	1.1				
M150(70)	0.7	0.70	0.95	1.31	1.31	1.77
H150(120)	1.6	0.64	0.87	3.00	1.20	1.64
^{137}Cs(662)	1.0	1.0	1.0	1.46	1.46	1.46

a. Divide recorded dose by table value to estimate $H_p(10)$.
b. Based on Wilson et al. 1990.

This table was included in the Hanford TBD as part of a compilation of available information concerning Hanford beta/photon dosimeter response testing.

The methods used to arrive at the values in Table 6-6 are described in:

Fix, J. J, E. S. Gilbert, and W. V. Baumgartner, 1994, *An Assessment of Bias and Uncertainty in Recorded Dose from External Sources of Radiation for Workers at the Hanford Site*, PNL-10066, Pacific Northwest Laboratory, Richland, Washington.

Wilson, R. H., J. J. Fix, W. V. Baumgartner, and L. L. Nichols, 1990, *Description and Evaluation of the Hanford Personnel Dosimeter Program From 1944 Through 1989*, PNL-7447, Pacific Northwest Laboratory, Richland, Washington.

The objective of the laboratory studies listed in the Hanford TBD and performed at Hanford was to assess the accuracy of the dosimeter interpreted dose in a rotational geometry using selected beams of radiation. As noted in the Hanford TBD text, intercomparison testing of its historical film dosimeter designs using A-P (Wilson et al. 1990) and angular (Fix et al. 1994) irradiations on an Alderson Rando anthropomorphic phantom. The precise data analyzed to determine these factors in the original document is not known. However the tabled values are from Table 3.1 of reference #1 with the exception of those values identified as being from reference #2 for an A-P exposure only (i.e., rotational values for energies <100 keV are not available). Values for an A-P exposure in Table 6-6 from reference #2, Table 7-3 and Appendix D, Tables D.2 and D.3, can be calculated as follows:

Table 1. Calculation of A-P Parameters in Hanford TBD Table 6-6

Beam	Irradiation		Reported Dose		Ratio	
	Exposure (mR)	Deep Dose (mrem)	Two-element[(a)]	Multi-element	Two-element	Multi-element
16	40	15	3.5	14	0.23	0.93
16	80	30	3.5	27	0.12	0.90
16	160	61	1.5	56	0.02	0.92
Average					**0.12**	**0.92**
Hanford TBD					**0.1**	**0.9**
59	30	44	12	46	0.27	1.05
59	50	74	25	80	0.34	1.08
59	80	118	30	126	0.25	1.07
Average					**0.29**	**1.06**
Hanford TBD					**0.5**	**1.1**

(a) Average of 1944 and 1945 reported dose from Appendix D.

This table shows good comparison for all values in PNL-10066 except for the two-element response to 59 keV photons (i.e., response based on data in Table 2 is calculated to be about 0.3 instead of 0.5 shown in Table 6-6). It should be noted that the analyses of dose data used in these Hanford reports does not include the factor of 0.2 noted in comment #1 which would result in substantial overestimate of the given deep dose in Table 2 which is shown in the Hanford TBD in Table 6-3. The footnote to Table 6-6 was intended to be an accurate representation of Table 3-1 from reference #1 and not a recommendation to NIOSH dose reconstructors. Instructions to dose reconstructors occurs later in the TBD. The 59 keV and 16 keV dosimeter

response data are important technical considerations regarding dose reconstruction in Hanford plutonium facilities.

It should be noted that the Hanford study dosimeter results for energies greater than 100 keV are consistent with the IARC results (which only includes > 100 keV irradiations), showing an over-estimate of Hp(10) for the two-element dosimeter per the following:

Thierry-Chef, I., F. Pernicka, M. Marshall, E. Cardis, and P. Andreo, 2002, "Study of a Selection of 10 Historical Types of Dosimeter: Variation of the Response to Hp(10) with Photon Energy and Geometry of Exposure," *Radiat Prot Dos*, 102(2): 101-113.

SC&A: Partial body exposures could result in significantly higher dose to an organ than what is measured on the film badge. Could you explain how these situations would be determined by dose reconstructors and how the organ dose is calculated in the case of partial body exposures?

ORAU: There is no doubt that dose reconstruction for partial body exposures is highly uncertain particularly for some organ doses and considering the wide range of possible exposure scenarios. When I arrived at Hanford (1974), I understood that workers were instructed to place the dosimeter at the location of the "whole body" where the exposure rate was the highest such as the head, arms above the elbows and the legs above the knee. In this manner, the recorded dose is likely maximized. However, from a practical perspective, baring an accident that is normally the subject of an investigation, the majority of dose to most Hanford workers occurs from protracted exposures over a period of many months if not many years from photon radiation that even of low energy can penetrate large distances relative to the dimensions of a worker's body. In this typical scenario, exposure to workers becomes averaged over many worker orientations and radiation field spectral/directional characteristics. Likely assuming an exposure geometry of rotational or isotropic is the most reasonable option.

SC&A: The data presented in the TBD (0006-6) indicate a systematic underestimation of doses by film badges relative to TLD readings by perhaps 30%. But the 0006-6 TBD, however, states that the two agree "reasonably well" (middle of page 22). Could you explain how the data in Fig. 6-4 (page 23) of 0006-6 were used to arrive at this conclusion, and what adjustments if any were made for the film badge/TLD discrepancies in practice?

ORAU: The statement that the film and thermoluminescent dosimeter data agree "reasonably well" was based on the original document[139] that summarized parallel field measurements of Hanford multi-element film and thermoluminescent dosimeters. This report was re-examined to respond to this comment. The studies consisted of two primary tasks involving: (1) workers wearing both types of dosimeters, and (2) dosimeters (two of each type) placed on polyethylene carboys filled with water. The studies involved several Hanford facilities as described in the TBD. Results of dosimeters placed on the carboys at 49 work locations were considered by the authors [i.e., "most meaningful results were from the tests which use the set of six dosimeters (2 TLD, 2 - beta/gamma film and 2 NTA film) symmetrically attached to a jug. . ."] to represent the higher quality data and this is the information included in the TBD. In addition to dosimeter data, portable ionization chamber (i.e., Cutie Pies abbreviated as CP) measurements were

[139] Nichols, L. L., G. W. R. Endres, D. B. Shipler, E.E. Oscarson and L.L. Crass. 1972. *Hanford Multipurpose TL Dosimeter Field Tests and Evaluation.* BNWL-B-127, Battelle Northwest, Richland, Washington.

performed at the beginning and end of placing the carboys. The authors concluded that all penetrating dose comparisons were good except for B-Plant.

The high-level wastes generated by Hanford reprocessing facilities have been handled in the B plant, the 242-A Evaporator, and the waste tank farms. B plant was originally a reprocessing facility but was converted in 1968 to a waste fractionation plant and used for a time to remove ^{137}Cs and ^{90}Sr fission products. The west end of the building became the Hanford Waste Encapsulation and Storage Facility, where about 300 capsules of solidified strontium and cesium, containing almost 200 million curies of radioactivity, were in water-cooled storage.

The authors state that this issue was studied and concluded that the film was too black to permit an X-ray interpretation which should have been significant. The dosimeters on the carboys for 24 hours a day and 7 days per week likely accumulated higher doses than would be typical for a routine dosimeter exchange cycle. The customary procedure was to add 35% of the X-ray dose component to the penetrating dose and this could not be done. The authors also state that the thinner effective filters in the TLDs also tend to result in higher dose interpretations for intermediate energy photons and intermediate and high energy beta radiation. For example, based on Fix et al. 1982, the Hanford TLD will record a penetrating deep dose from $^{90}Sr/^{90}Y$ irradiations when there should be a zero deep dose equivalent to approximately 10% of the shallow dose. Also, when calibration of the Hanford TLD was changed to an on-phantom calibration the dose was reduced by approximately 10%.

Hanford TBD Table 6-7 summarizes the collective dose for the film, TLD and ionization chamber measurements for each workplace facility. A more detailed analysis of this data is shown in Table 2. The collective penetrating dose is presented along with the percent dosimeter measured collective dose relative to the CP measurements. It is evident in Table 2 that the primary problem was associated with B-Plant measurements as noted by the original authors.

Table 2. Workplace Measured Penetrating Collective Doses (Nichols et al. 1972)

Facility	Collective Dose, mrem			Percent	
	Film	TLD	CP	Film/CP	TLD/CP
B-Plant only	2,250	4,560	4,920	45.7	92.7
All locations	35,680	39,560	37,830	94.3	104.6
Minus B-Plant	33,430	35,000	32,910	101.6	106.4

SC&A: TBD 0006-6 mentions the term "geometry" on pages 19, 51, 72, and 78. On page 19 it states in Table 6-4 that the angular response may be 50% low. On page 51 the footnotes to Table 6-32 contain references to geometry. On page 72 it states that no adjustment is recommended… On page 78 it makes reference to uncertainties, but does not explicitly mention geometry.

- Could you explain how and when a geometry factor (angular response, potential shielding, partial body exposure, badge location, etc) is used by the dose reconstructor?
- How were these correction factors obtained?

- Where are the corrections to film badge data prior to 1970 documented for use by the dose reconstructor? For example, is there a document similar to TIB-0008 for early film badge programs?

This could be especially important in accident/acute situations where the dose could be underestimated and not claimant favorable because of geometry factors.

ORAU: The cited Hanford TBD text illustrates the complexity of assessing dosimeter performance under a wide range of conditions. The issue of geometry was well recognized historically as being important to the dosimeter response. Significant dose to a worker is typically accumulated over a working career involving many types of exposure conditions. The movement of the worker and typically the broad radiation fields in Hanford facilities implies that the personnel dosimeter would respond in photon fields, which contribute the vast majority of Hanford worker dose, similar to information for the rotational exposure geometry presented in Hanford TBD Tables 6-3, 6-4 and 6-5. Highly non-uniform exposure conditions are typically associated with specific work tasks. Perhaps the most significant routine situation concerns beta radiation with its characteristically limited range. Significant worker dose from an accident is subject to an investigation as to the extremity, skin and whole body dose to be recorded. It is likely that some reference to significant dose to a worker from an accident will be noted in the DOE dosimetry and incident information provided for each claimant.

SC&A: The 225 waste processing does not list ^{90}Sr brehmstrahlung radiation? Did you decide that this is not important? If so, please give us the technical reasoning behind this assumption.

ORAU: It is assumed that this comment is in reference to Table 6-9 as follows:

Table 6-9. Hanford Workplace Photon Spectra Measurements[a]

Facility	Description	Measurements	Results[b]	Reference
308 Bldg.	Room Background	Gamma	^{241}Am (100%)	Fix et al 1981
	Grinder Hood Bottom	Gamma	^{241}Am (100%)	
	Pellet Pressing Station	Gamma	^{241}Am (100%)	
327 Bldg.	Background A-Cell	Gamma	^{60}Co (85%), ^{137}Cs (8%), ^{54}Mn (8%)	
	Background G-Cell	Gamma	^{60}Co (79%), ^{137}Cs (9%), ^{54}Mn (12%)	
200W,2425	Evaporator Building, NE Corner	Gamma	^{137}Cs (100%)	
200W, Diversion Boxes	241-TX-302-C Catch Tank	Gamma	^{137}Cs (100%)	
	K2U	Gamma	^{137}Cs (100%)	
	Rigging Crew	TLD (Beta, gamma)	High energy, indicative of photon radiation	
B-Plant (225 Bldg)	A-Cell	Gamma	^{137}Cs (100%)	
	Between B-C Cells	Gamma	^{137}Cs (100%)	
	Between D-E Cells	Gamma	^{137}Cs (100%)	
	F-Cell	Gamma	^{137}Cs (100%)	
	Room Background	Gamma	^{137}Cs (100%)	
271B	Pipe Gallery –Cell 9	TLD (Beta, gamma)	Indicative of ^{90}Sr/^{90}Y	
324 Bldg.	A-Cell Gallery	Gamma	^{137}Cs (100%)	Fix et al 1982
	C-Cell Gallery	Gamma	^{137}Cs (100%)	
	Truck Dock	Gamma	^{137}Cs (100%)	
331 Bldg.	Office	Gamma	^{208}Tl (90%), ^{137}Cs(10%)	
	Change Room (SE)	Gamma	^{208}Tl (8%), ^{137}Cs(92%)	
	Change Room (Toilet)		^{208}Tl (64%), ^{137}Cs(36%)	
	Janitor's closet		^{208}Tl (46%), ^{137}Cs(54%)	
340 Bldg.	340-A Outside	Gamma	^{137}Cs (100%)	
	Control Room	Gamma	^{137}Cs (100%)	
	Decon Area	Gamma	^{137}Cs (100%)	
	Operations Office	Gamma	^{137}Cs (100%)	
3730 Bldg	Irradiation Room	Gamma	^{60}Co (100%)	
	Hallway	Gamma	^{60}Co (100%)	
234-5	Fluorinator Hood	Gamma	<200 keV (99+%) 17 keV (~50%)	Roberson and Cummings 1985

			Photon Energy, keV			
			< 200	200-2000	>2000	
234-5, Vault 4	Vault 4 Entrance	Gamma	13%	55%	33%	Roberson et al 1986
234-5, Vault 1	Phantom	Gamma	42%	55%	3%	
	floor	Gamma	50%	48%	2%	
	Entrance	Gamma	17%	61%	22%	
234-5, MT Room	At hoods near entrance	Gamma	0%	83%	17%	
234-5, C-Line, Room B	Toward neutron source	Gamma	92%	7%	1%	
	Toward room A	Gamma	0%	98%	2%	
	Near Entrance	Gamma	58%	28%	14%	

a. Only measurements that included photon spectra are listed.
b. Measured non-natural radionuclide significant to occupational exposure.

The measured field spectra show those photon energies of primary importance. It is not surprising that photon radiation associated with the relatively high energy ^{137}Cs gamma radiation is of most importance. Certainly, compton scattering of the ^{137}Cs gamma radiation is significant

as well as the bremsstrahlung radiation from ^{90}Sr and other sources of beta radiation. The personnel dosimeter should easily measure this radiation.

SC&A: It appears that the TBD does not address, in an obvious manner, the issue of exposures from incidents that may have occurred during missed-dose periods where the doses may have been assigned rather than actually measured; or if dosimetry was in place, if the dosimetry at the time of incidents accurately measured the doses during accident/acute situations. Has NIOSH addressed this issue, and how?

ORAU: The Hanford TBD addresses the typical situation for a worker who accumulates dose throughout their career. The Department of Labor form completed by DOE for each claimant does include a category for Incident Investigation Reports (i.e., categories on Request for Personnel Exposure Information Form are: External Dosimetry Records status, Internal Dosimetry Records status, Diagnostic Dose Records Status, Incident Investigation Reports Status and Other monitoring Results status. For each form, options are provided as follows: (1) "Provided", (2) "Not Readily Available" with the expectation that a date of availability will be specified, and (3) "Does Not Exist"). As such, the dose information for each claimant from a significant incident should be provided. To the best of my knowledge, Hanford practice has always required a formal method to assign or revise a worker's recorded dose. Significant dose from an incident would undoubtedly be the subject of an investigation and it is expected that this information will be provided by DOE for each claimant.

Internal Dosimetry

SC&A: Is it true that bioassays for Hanford workers dropped from around 80% in 1958 to about 40% in 1962? If this is the case, what is your opinion about the possible missed dose that may have resulted from using chest counts that were less reliable and had greater uncertainties up to the 1970s, especially for americium and plutonium?

ORAU: The total number of Pu bioassay did increase during the 1950s and seems to peak in the late 50s. We have no idea what % of total workers these numbers represent; however, if there is evidence of continued exposure to plutonium without bioassay, dose reconstructors (DRs) continue to apply missed dose, based on ½ the detection threshold for the bioassay, to the periods of exposure. Chest counting started in 1967, and there were only 52 chest counts in 1967 and 188 in 1968. So chest counts were not used in lieu of urinalyses to monitor for plutonium intakes.

Generally, DRs do not use chest counting to establish missed dose because of its inherent lack of sensitivity and greater uncertainty.

SC&A: According to Wing et al. (2004)[140] the percentage of annual bioassays of Hanford workers grew to nearly 80% from 1945 to 1959, then dropped to ~50% in 1960, and continued to decline to ~ 20% and less from 1962 to 1988. What is your opinion regarding this finding?

[140] Steve Wing, David Richardson, Susanne Wolf, and Gary Mihlan, Plutonium-Related Work and Cause-Specific Mortality at the United States Department of Energy Hanford Site, American Journal of Industrial Medicine 45: 153-164 (2004), p.155.

ORAU: No opinion. The history of the criteria for placing workers on bioassay programs has not been researched and compiled. At times the decision was up to individual facilities. Sometimes the decision was controlled at the level of the radiation work permit, some facilities had requirements that applied to any work in the facility, at other times some companies had general requirements for all employees (for instance, for many years PNNL required termination whole body counts for all employees, regardless of exposure potential). Most bioassay, except for tritium, uranium, and low levels of gamma-emitters in whole body counts, did not have detectable activity. See also the answer to question 19 in the second set of internal dosimetry questions for additional discussion about criteria for placing workers on bioassay programs.

Instructions for assigning intakes to unmonitored workers is provided in the TBD Section 5.7; the instructions are based on assumed exposure to 10% of the limiting air concentrations (alpha and beta, and for some facilities I-131 and tritium) for 4 hours per week. In addition, a study is underway to provide directions on how to assign intakes to unmonitored workers based on co-work data. This effort will apply to all sites; Hanford's data will be analyzed in January-February and the TBD will be revised accordingly by April. There is also a Technical Information Bulletin, TIB-014, that provides guidance on assigning intakes for unmonitored workers. Ultimately, the DR makes the final decision based on all information available pertinent to the energy employee on how to assign intakes during unmonitored periods.

SC&A: Are there additional steps that need to be added to the TBD to help dose reconstructors fill in the missed dose gap for ^{131}I uptakes from 1943 to 1946 when bioassays were not available, and uncertainties were greater for instruments, thyroid scans, MDAs and determination of ^{131}I results?

ORAU: New information has been obtained, but the existing guidance is apparently an overestimate. The TBD author has recently been made aware of a document that gives an excellent history of thyroid monitoring. It states that large scale, routine measurements started in June 1945, with the tolerance reading set at 5,000 cpm. Records show counts made as early as October 1944, but fewer in number. After additional studies, in December 1945, the tolerance reading was lowered to 750 cpm, which was based on improved calibration using a thyroid phantom with 1 μCi of I-131 (375 cpm/μCi). So for those 7 months in 1945, the allowable measurement was nearly 7 times higher than the later allowable measurement. However, the records also show that actual worker results were far below the tolerance levels. Out of nearly 1600 counts performed in 1944-46, 99% of the results were less than 22 cpm; the highest result was only 78 cpm, but this and the other four highest results were attributed to contamination on clothing. The report calculated thyroid burdens using the calibration factor determined in 1945 for the specific GM counters used for the counts and determined that the highest thyroid burden was 0.13 μCi with 99% of the burdens <0.06 μCi). So even allowing for errors in calibration or variation in measurement technique, the guidance in the TBD, based on a 2-μCi thyroid burden, is quite generous. [T. A. Ikenberry, 1991, *Evaluation of Thyroid Radioactivity Measurement Data from Hanford Workers, 1944-46*, PNL-7254-HEDR.]

SC&A: Have uncertainties been developed (particularly for the 1945-1946 time period) to account for variations in instrument response during thyroid scans, especially when considering ^{131}I thyroid burdens that exceed 2μCi?

ORAU: See response to question #3.

SC&A: How was ^{131}I dose accounted for from 1946 until the 1960's? Since monitoring for ^{131}I was performed for only some high risk workers, how would a dose reconstructor determine which workers were monitored and the uncertainties associated with this monitoring?

ORAU: The TBD author doesn't know for sure what records for I-131 monitoring will show in the energy employee's file. However, if there are no records, the TBD provides the following guidance in Section 5.7, page 49: "[For the period 1947-52,] use 0.1 times the vapor intake value used for the pre-1947 period, i.e., 7.5×10^4 pCi/d." This guidance was based on improved cleanup of radioiodines from the separations plants' ventilation systems and improved radiation protection in general. However, the guidance is consistent with the new information provided in the Ikenberry document discussed above.

SC&A: Have procedures been established and has appropriate raw data been made available to help dose reconstructors provide dose estimates for workers exposed to contamination spreads in reactor buildings, the 231-Z plutonium isolation facility, the concentrator building, and the uranium metal fabrication shops from 1943-1946?

ORAU: No. Air sample data were used to establish the default intakes for the 1943-46 period, plus an allowance was added for the possibility of ingestion. The air sample data were obtained from monthly reports that usually mentioned if the air concentrations resulted from contamination spreads, etc.

SC&A: There seems to be possible missed dose due the fact that prior to 1983 the recorded alpha activity from plutonium did not account for the presence of ^{241}Pu and ^{241}Am. What additional might be done in the TBD to account for such missed dose?

ORAU: The need to account for Pu-241 and Am-241 when determining intakes based on plutonium alpha or Pu-239 bioassay results is well known by the DRs, is covered in their procedures, and is built into the Hanford tool used by the DRs.

Dose from Pu-241 and Am-241 was also incorporated into Hanford site dose assignments for plutonium intakes from at least the mid 1970s and probably earlier.

SC&A: The TBD indicates that the intake of plutonium and thorium might have been missed in the 1950s because of poor detection capability, missed samples or poor sampling after a potential intake? The TBD reports that thorium exposure was more likely at the 3732 Building where the powdered thorium fuel targets were fabricated or in the 3720 Building where work with ^{232}Th slurries was done in the mid 1990s. Do dose reconstructors try to confirm such thorium exposures by finding later urinalysis data for individual claimants?

ORAU: Yes, DRs use all the available bioassay data for a given radionuclide and recognize that for radionuclides that are retained for long periods in the body later bioassay data may be as good or better indicator of intake than data taken immediately following the potential intake. This is part of the various training and information exchange sessions the DRs have had and is incorporated into their procedures.

SC&A: Reactor workers were principally exposed to activation products, but there seems to be little development in the TBD for possible intakes of activation products until 1960 except for air

sampling. The fission product analysis did not measure activation products. What in your opinion could be done to better characterize dose from such activation products?

ORAU: Very few air sample results have been found to date for the old reactors, so for activation products there was not much to go on. It is evident from the early Health Department monthly reports that if air samples were exceeding the limiting concentration for gross beta, it would have been mentioned in the reports (circa 1940s), and occasionally there was such a notice. Because the ratios of various activation products and fission products was not established, and probably varied considerably depending of the nature of the work or location in the reactor, Table 5.7-2 was developed so that the DR can assign all the potential intake (based on gross beta air concentration limits) to the radionuclide that produces the largest dose to the organ of concern. This ensures that the maximum dose is assigned regardless of what the mix of radionuclides might have actually been. Hence, it is assumed, for instance, that a reactor worker was chronically exposed to the limiting air concentration or, in some periods, some fraction of the limiting air concentration. This air concentration times the exposure period times the breathing rate results in a total intake of X pCi. Then Table 5.7-2 is used to determine both the radionuclide and absorption class that maximizes the dose to the organ with the cancer, say maybe Ru-106 type S. Then the organ dose is determined using an intake of X pCi of Ru-106 type S. The mix that was actually breathed by the worker is unknown but would have resulted in a smaller dose to the specific organ of concern.

SC&A: The Wilson history states that the uranium urinalysis program prior to 1948 was not reliable. This represents a potential for missed dose since there appears to be a significant potential for dose to workers working around recycled uranium in the 300 Area operations. This may also be the case for plutonium, neptunium and fission product hazards associated with recycled uranium processes in fuel manufacturing and at the Uranium Oxide Plant. Have you considered changes to the TBD that would more effectively determine individual or group dose from recycled uranium?

ORAU: Recycled uranium didn't occur at Hanford until February 1952. Contaminants in recycled uranium are assigned to uranium intakes from that time forward based on information provided in Table 5.2.5-2 (4th column). Because of the lack of reliability of the uranium urinalysis method, as you mention, for the period 1944-47, uranium intakes are not assigned from bioassay data. The TBD states to just assign intakes to all workers in the 300 Area Fuel fabrication plants based on an assumption of chronic exposure at 2E-10 μCi/ml (an estimate created from the Health Department monthly reports).

SC&A: Can you provide us more information on how dose reconstructors should properly quantify the uncertainties for whole body counting prior to 1993, since there were no default set of radionuclides until that timeframe?

ORAU: Actually there was a default set of radionuclides from 1960 through 1983, that being Zn-65, Na-24, Cs-137, and K-40. In 1985 the default set was expanded but Zn-65 was dropped. In 1987 the default set was reduced to just Co-60 and Cs-137. There have been other changes to the default set since 1992. Cs-137 and K-40 are the only radionuclides that have always been reported.

The lead for Hanford intake cases in the DR group provided the following response to this question:

> *For the situation where there are no positive whole body counts we follow this procedure. Missed dose is assigned by choosing the most claimant favorable nuclide (based on organ, exposure time, and detection limits). The mode is set at ½ the MDA; the MDA is set as the upper limit (of the triangular distribution). Also, the values from your table [5.3.1-1] that you said were really decision levels, we treat those as such and double them to call them MDA. Dose is assessed using IMBA.* [Although the table lists MDAs, it has since been determined that they were closer to decision levels prior to 1985, and the DRs were informed of that.]
>
> *For positive whole body counts:*
>
> *-assign the dose from any positives according to procedure*
>
> *-DRs often still run the assessment for missed dose on top of this (and it is usually claimant favorable since the missed dose is usually higher than the dose from assessing positive counts)*
>
> *-For Zn-65, if DRs see it was positive then they determine dose (basically using a constant chronic to exceed all data points). We run the thing for the entire period of employment at that level since there is basically no dose consequence.*

SC&A: Since worker doses can be high in many waste handling operations, are you considering any TBD supplements that explain the types of radionuclides and absorption types for waste management facilities (i.e., tank farms, evaporations, transfer lines, etc.)?

ORAU: Intakes by workers in waste handling operations haven't tended to be abnormal relative to the rest of Hanford facilities, except that the source terms are a mix, not pure Pu or U, for instance. In general, Chapter 2, Hanford Site Description and Processes, is where radionuclides and absorption types are presented relative to certain facilities or processes. Section 2.6 addresses waste handling facilities. Some additional source term and/or absorption type information has been included in section 5.4.1.3.

Generally, the TBD doesn't exactly specify the absorption classes, but rather gives the plausible range of absorption classes and DRs use the most claimant-favorable one. But more often, lately, the DRs have been using tools that pick the radionuclide that gives the highest dose to the organ of concern rather than trying to guess at a mixture.

SC&A: Have you considered further refining inhalation classes for Hanford uranium compounds, providing more comprehensive information on solubility and making relevant associated studies available to the dose reconstructors?

ORAU: Solubility studies were done on uranium contamination in several Hanford uranium facilities in the 1980s. This information is provided in Table 5.2.5-3.

SC&A: The claimant's personal file may refer to episodic events that represent a greater potential for dose. HW-34882 identifies the fact that only a smaller fraction of Ru/Rh-106 releases were chronic and that large episodic releases occurred over a period of hours to one or two days. Fallout resuspension and inhalation/ingestion of Ru/Rh-106 do not seem to be addressed in the TBD. Does this not represent an area of missed dose?

ORAU: This pathway would be considered environmental dose, and would need to be addressed in the environmental chapter, although, of course, intakes of Ru-106 by any pathway would have been detected by whole body counting after 1959. According to the recording practices in 1960 through 1984, body burdens of Ru-106 were not always recorded unless they exceeded a threshold value of 1% of the maximum permissible body burden (even if detected).

SC&A: The TBD states that there is indication of work with pure ^{210}Po in the 308 Building in 1958 and in 1975. The TBD does not seem to determine the potential dose impact to workers from ^{210}Po during these periods in the 308 Building. Have you been able to ascertain if such workers did work with ^{210}Po in the 308 building and if so what additional dose impact there would be from such continuous exposure?

ORAU: If the energy employee's file shows bioassay for Po-210, the DRs will use their normal procedure for assigning intakes and doses, provided below.

General Assessment of Bioassay Results

1. No positive bioassay.
   - Calculate annual dose assuming chronic intake; use critical level (or MDA/2 if CL is not available) at the end of the exposure period. For radionuclides with short effective half-lives, assume multiple exposure periods if the MDA changes over time. **Note**: The date of the last bioassay sample should be used for determining the intake rate. The total intake is determined by assuming the intake continued until the last date of exposure.
   - If there are multiple MDAs for the same time period, use the lower MDA (more sensitive analysis).
   - Triangular Distribution in IREP (NIOSH 2002c).
     - Min = 0
     - Mode = Annual dose
     - Max = Mode*2

2. All positive bioassay.
   - Determine best fit in IMBA . Can assume acute or chronic as best fits the data. No missed dose calculation; bioassay account for all dose.
   - Lognormal distribution in IREP.
     - Median = Annual dose
     - GSD = 3

3. Mix of positive and <MDA bioassay

   a. Ignore positive data and perform missed dose calculation as in item 1.
   b. Assess positive bioassay results using normal procedures.
      - Use all bioassay results starting at first positive value.
      - In IMBA, for results < MDA, Measurement result = MDA, Data type = "< LOD"

   c. Assigned annual dose for a given year is the maximum value of 3a or 3b. [For mixtures, e.g., Pu, the total doses from the mixture would be compared in this step.] Choose IREP annual dose distribution type based on missed or "positive" dose assignment:
      - For years in which the dose from 3a is larger than that from 3b, use the triangular distribution.
        - Min = 0
        - Mode = Annual dose
        - Max = Mode*2
      - For years in which the dose from 3b is equal to or larger than that from 3a, use the lognormal distribution with a GSD=3.

Radiological Records

SC&A: Have you been providing information from early DuPont records (e.g., clipped microfilm)? Have you provided visitor and temporary monitoring results to NIOSH?

ORAU: The Hanford Radiation Records Program (HRRP) has searched all our records systems for each EEOICP claimant. The search is no different for a visitor or an employee. Only after the search is the person's employment status identified. The jacketed cut microfilm records are mostly duplicates of the microfilm reels with the exception of a period in the '70s when no reels were created. All record sources were searched for each person.

SC&A: 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?

ORAU: HRRP has copied all available records for each NIOSH request. No records were omitted. There have been very few requests for follow up information other than when NIOSH changed the full file criteria. Summary dose information was first only requested for individuals with total recorded dose less than 100 mrem or over 30 rem. That was later changed to less than 500 mrem or over 10 rem. Any editing of the records sent to NIOSH is done after the record package leaves the HRRP.

ATTACHMENT 4: SITE EXPERT INTERVIEWS — FORMER AND CURRENT RADIATION PROTECTION STAFF

Radiological Control Organization

There has always been a Central Radiological Control group of some kind. This group has primarily been responsible for general radiological control procedures and regulatory implementing documents. The size of this Central Radiological Control organization has varied over time. There is evidence of its presence even in the early years through periodic reports (e.g., environmental monitoring reports.) Each facility (e.g., 100K, Tank Farms, etc.) had their own Radiological Control Manager. These individuals have either reported to the Radiation Control Director of the site or to the particular facility manager. Pacific Northwest National Laboratory (PNNL) and Bechtel Hanford, Inc. (BHI) have separate radiological control programs.

Initially, Du Pont was the only contractor for the Hanford Site. In about 1947, General Electric took control. When General Electric left, the contract was distributed among multiple contractors. Rockwell ran the Tank Farms, United Nuclear Corporation (UNC) was responsible for reactor operations and fuel fabrication, Westinghouse ran the laboratories, and PNNL provided support functions such as dosimetry, instrument calibration, and Research and Development. In the mid- to late-1980s Westinghouse won the contract for the entire site. In 1996, the Hanford contract was again split between multiple contractors. More than one contractor has worked in the same area of Hanford (e.g., 200 Area, 300 Area, etc.), while having different responsibilities. Currently, the major site contractors include:

Company	Responsibility
CH2M Hill Hanford Group, Inc. (CHG)	Tank Farms
Bechtel National, Inc. (BNI)	Vitrification Plant
Bechtel Hanford, Inc. (BHI)	Environmental Restoration
Fluor Hanford, Inc. (FHI)	Operating facilities and solid waste
Pacific Northwest National Laboratory (PNNL)	Dosimetry and calibration services, Analytical services, Research and Development

Each contractor has its own Radiation Protection Program Plan (RPP), and its own implementing procedures. Dosimetry and instrument calibration records from each contractor are maintained by PNNL for all contractors. Field radiological control records, incident reports, and audit assessments are maintained by the particular contractor and are not intermingled with information from the other contractors.

Although there may have been multiple contractors, they were all required to follow the same regulatory requirements (i.e., DOE orders, 10-CFR-835) and guidance documents (e.g., DOE Radiological Control Manual, implementation guides). Each subcontractor had their own interpretation of these regulations and guidelines, and developed their own implementing procedures based on their particular situation. The facilities had their own problems and solutions to these problems. For example, survey forms and methods for performing surveys may have differed between contractors. Some contractors generated more effective procedures

than others. Some contractors were very detailed in their procedures while others were fairly lax. When Westinghouse took over in 1986 – 1989, there was an effort to consolidate procedures.

In the early years of operation, there was a single contractor likely resulting in providing more consistency in the Radiation Protection program. As the single contractor gave way to multiple contractors, there was less consistency. Communication has not always been effective between various contractors, especially in the 1970s. Through time there was a movement to a more consistent program site wide.

Although the contractors currently function as separate entities, they established what is referred to as the Radiological Control (RadCon) Forum. The primary purpose is to facilitate communication between the different contractors. This lets one contractor know what the other contractors are doing. Any recommendations coming from the RadCon Forum are voluntary.

Work Permits

Historical Health Instrument Division reports indicate that Special Work Permits were used as early as the late 1940s. The details on when these permits were used, the content, and the requirements contained were unknown by site experts interviewed.

Work planning has been an integral part of performing work on the Hanford Site. There have been Radiation Work Permits (RWPs) at least since 1976. Initially, RWPs were prepared by supervisors and secretaries with little input from the technicians. RWPs were more of a guidance document in the earlier years. Field changes could be made on the spot in the field without consulting supervision. This was common as the technicians modified the RWPs based on actual field conditions they encountered as they performed a job. The contents of RWPs were variable by facility until site-wide procedures were implemented. Each facility maintained its own set of RWPs. Permits could be very general (i.e., one for all the labs) or job specific. Acknowledgement sheets were introduced with Westinghouse in about 1987. Some facilities used very short RWPs with only pertinent information, while others incorporated procedural steps into the RWP. Since Radiological Control initially had limited input on operating procedures, adding procedural steps to the RWP was one way of providing their input to a process. In the early years there was more exchange between the worker and the Radiological Control Technician (RCT) as things occurred in the field.

Radiological work is currently controlled by Radiation Work Permits (RWPs). The requirements are based on previous characterizations and surveys, chemical forms, and concentrations of radionuclides in the area. The levels of contamination present or expected determine the limiting conditions of the RWP. The External Dosimetry Group at PNNL establishes the generalized dosimetry requirements by facility. Currently, if there is a potential for 100 mrem in a year, an individual receives a beta/gamma thermoluminescent dosimeter (TLD). Also, if the individual has a potential for 100 mrem per year neutron exposure, a neutron dosimeter is assigned. The Internal Dosimetry Group at PNNL establishes similar requirements for bioassay. The recommendations made by PNNL for monitoring are based on facility-specific characterization data, which is updated periodically. The need for special dosimetry such as extremity dosimetry and multiple dosimetry, is determined by the field based on established procedures. Requirements for Personnel Protective Equipment (PPE) are also proceduralized. Some

contractors such as the Environmental Restoration Contractor (ERC) complete an internal and external dose estimate as a part of the work planning process to determine whether monitoring is required. It is also a default practice to monitor jobs for potential airborne exposure.

Around the early 1990s, the site implemented the use of void limits in the RWP. These void limits were based on the type of PPE worn and the expected radiological conditions during the job. If these limits were exceeded, the job was stopped and controls were reevaluated. When there are unknown situations such as the breach of a system, more stringent controls are typically included in the RWP.

Currently, changes to RWPs require a formal process of approval and review. Today RWPs are maintained by the particular facility (e.g., 100K, 222S Labs) and are 1-2 pages in length. Access Control Entry System (ACES) is used to log individuals into and out of an RWP. Active RWPs are maintained and are readily available. Out-of-date RWPs are retained for a period of time and are them submitted to Central Records. Compliance with radiation work permits is mandatory.

In the past, there was not a linkage between air samples, RWPs, and survey reports. The RWP was documented on the survey report. There was not a requirement to reference the RWP on work documents (i.e., surveys, air samples, etc.) Currently, there is linkage between the air sampling, surveys and RWPs.

In 1990, the Hanford Site implemented ACES. This allowed the field radiological control organizations to track entries under particular Radiation Work Permits (RWPs). They were also able to verify whether an individual was on the correct external and internal monitoring program for the particular job or task, and to ensure individuals had the appropriate training. Self Reading Dosimeter (SRD) results could also be entered into ACES to allow for real-time tracking of dose. Prior to this, SRD results were kept by hand.

Radiological Posting and Contamination Control

Radiological contamination controls varied by facility but included step off pads, personnel protective equipment, frisking, and use of hand and foot counters. As multiple contractors became involved in the Hanford mission, there were not universal radiological control practices between different contractors. Control of contamination has always been an issue at various facilities on the Hanford Site (e.g., 272AW, 234-5, and 222-S Analytical Laboratory). During certain operations (e.g., charging and discharging the reactors, collecting samples from the sample galleries), contamination was expected.

There were several occurrences of contamination in the reactor areas. Radioactive contamination was a routine issue on the top of the F-reactor. All of the reactors (e.g., 105B, 105D, 105F) had aluminum liners in the vertical safety rod channels. They were there to contain a liquid boron compound in case the first two systems, the vertical safety rods and the horizontal control rods, failed. When the reactor power levels were raised, the graphite also got hotter. The aluminum liners would melt. When they were removed at 105° F, some engineer had the vertical safety rod channel sealed with sodium silicate. After startup, the silicate immediately turned to fine powder and thereafter any job on top of F-reactor involved work in a highly contaminated area. Workers who went to the top of F-reactor thereafter for boroscope operations or other work frequently got contaminated. The 100K Area basins were contaminated by the addition of

N-basin fuel to the storage basins. In 100KE basin, the concrete is contaminated to a depth of ½" with ^{137}Cs.

Eating, drinking, and smoking were not allowed in the radiation zones. Coveralls were provided to workers. For some work the employees used two layers of personnel protective equipment. Assault masks, half-mask respirators, and fresh air were used during some jobs in the reactor areas. Rubber clothing was worn for wet work. Charging and discharging fuel from the reactors resulted in individuals getting wet. This was due to a high charging pressure. The workers did check themselves. If their person was contaminated with this water, they were required to take a shower.

Air samples and skin contamination (especially that occurring on the face) incidents are the first indicator of potential intake of radionuclides. In the more recent era, there have not been a lot of internal depositions at Hanford. There are also less skin contamination incidents. Precautions are effectively controlling the source of radiation.

Initially, areas were posted as Radiation Zones. There was no specific indicator of whether the zone was so posted due to contamination or radiation levels. Posting has become more specific over time as regulations have changed and consolidated posting practices were implemented. Today there are numerous postings such as contamination area, high contamination area, radiation area, high radiation area, very high radiation area, and airborne radioactivity area. Currently postings are much more understandable and provide useful information to individuals entering a radiological area.

Also, over time definitions of radiological areas have changed. For example, the definition of a radiation area went from 2 mr/hour at the boundary to 5 mr/hour at 30 cm from the source. When the site changed from ICRP 2 to ICRP 30, the basis for airborne areas switched from Maximum Permissible Concentrations (MPCs) to Allowable Limit on Intakes (ALIs).

Air Sampling

During Hanford operational years, there was a significant amount of air sampling completed. Fixed head air samplers were placed in numerous locations throughout the facilities and in each room at laboratory facilities. These locations included both high- and low-risk areas. Currently, air-sampling requirements for a particular facility are based on the air sampling technical basis document. The type and quantity of air samples taken are based on risk. For example, Plutonium Finishing Plant (PFP) has hundreds of fixed air monitors due to the high potential for internal exposure. When Hanford started basing their air sampling on risk, there was a decrease in the number of air samples taken site wide. Air sampling in low risk areas was eliminated. Job air sampling was used in lieu of fixed air-head samplers. Job air sampling was also a part of the air-sampling program when fixed air heads were routinely used.

Air samples were taken as a means of verifying the appropriateness of respiratory protection, to establish appropriate posting upon entry into an Airborne Radioactivity Area, and when there was a potential for generating airborne radioactivity (e.g., breaching a system). Air samples serve as the first indicator of a potential intake. Air sampling results are counted initially and after a period of decay. If the decay count results are above the detection limits, there are formal notification procedures. Each air sample was recorded in a log. Some technicians recorded air

sample results on their surveys while others did not. Once air-sampling procedures were developed, air samples were taken in accordance with established procedures. The air-sampling program has included fixed air-head samplers, environmental air samples, continuous air monitors, goosenecks, Staplex samplers, and lapel air sampling. Triton air monitors are used for tritium monitoring. These are flow-through ion chambers and are a relatively new system. They are very sensitive instruments.

The ERC has a routine DAC-hour tracking program and has used lapel sampling for several years. This program has indicated that there are low-level intakes of radioactive material that occur that are not detectable by bioassay techniques. Although the Personal Air Samplers do not indicate significant intakes, it is not uncommon to see ½ DAC-hours per day. This adds up over time. The use of lapel air sampling is a more recent addition to the air-sampling program for other Hanford contractors (e.g., FHI, CHG).

Historically, air sample data was rarely, if ever, used to assign internal dose. There have been a few cases in the last 10 years where air sample data were used to determine intakes when the magnitude of the intakes were too small and/or bioassay were obtained too late to provide sufficient sensitivity to show that intakes had not occurred. Hanford has guidance on when air sample data should be used to assign intakes based on the estimated DAC-hrs and the likelihood of confirming the intake using bioassay (PNL-MA-552, Section 5.3.3).

Bioassay versus air sampling: practical guidance and experience at Hanford (Carbaugh 2004) showed that air sample data typically overestimates intake and dose. Data were presented on 97 Hanford cases over a four-year period (CY-2000 to 2003) and compared the DAC-hours dose estimate with the bioassay-based dose assessment. Only 12 cases were assigned dose directly from the air sample data. In the other 85 cases the bioassay data were determined to be more representative of exposure, with 41 cases showing no confirmation of intake. In the 44 confirmed intakes, the DAC-hours dose estimates exceeded the bioassay-based estimates by a factor of 4 or more in 75% of the cases, and were within a factor of 3 for only 25%.

Radiological Instrumentation

With the inception of the radiation-monitoring program, there were few instruments available for radiation monitoring. Some of the many early radiological instrumentation included Poppies for alpha detection, Geiger-Mueller (GM) counters, and Cutie Pies (i.e., ionization chambers).

The Hanford radiation instrument program has remained fairly consistent over time. For the past 30 years or so, technicians have used Cutie Pies (CPs), Eberline GM Counters (phased out), Portable Alpha Monitors (PAM), and Ludlums with digital/analog counters. Their instruments are found to be reliable. The newest instrument in the field is the Snoopy at PFP for neutron dose measurement. Instruments used at Hanford have to withstand rugged use. The ERC moved away from standard Hanford instruments and started using Eberline E600, NE Electras, 2224 Ludlum Alpha/Beta counter hooked up to a scalar, and RO20 ionization chambers. They continue to update instruments as technology advances.

When a technician enters the field, they are able to identify whether alpha or beta/gamma contamination is present. They are not able to distinguish the particular radionuclides they are encountering. There is a lot of reliance on previous characterization reports. Most contractors

on site do not have the capability to do radionuclide characterization in the field. To determine radionuclide constituents on sampling media, samples have to be sent to an analytical laboratory for analysis. This is very costly and turnaround times can be as long as several weeks. There are also no current field instruments capable of differentiating between radon and transuranics.

Some contractors such as ERC have procured field spectrometers to allow them to characterize radionuclides in the field. These units are very expensive, so many contractors have not purchased these detectors.

External Dosimetry Monitoring

Individuals wore one dosimeter per area. For example, if they worked at two reactors, they had two dosimeters. Everyone had their own beta/gamma dosimeter in the reactor areas.

The Hanford security credential and dosimeter were combined from the beginning of operations until about 1995 into a single unit. The security credential was necessary to enter the respective Hanford 100, 200 and 300 operating areas. By attaching the dosimeter to the security credential this provided assurance that dosimeters were worn. Security credentials were also necessary to enter Hanford 600 and 700 support and administrative areas and subsequently the Fast Flux Test Facility 400 Area. However, not everyone entering a support or administrative area also wore a dosimeter. Hanford site practice was to assign dosimeters to all workers who entered one of Hanford's radiological controlled 100, 200 and 300 operating areas. In the early years, pocket ionization chamber (PICs) and personnel film dosimeters (i.e., meters at the time) were processed at the gatehouse into each of the respective operating areas. Later all site film dosimeter processing was done at a central laboratory. For many years, the security/dosimeter units were provided to workers using badge exchange racks located at the controlled gatehouse entrance to each operating area. There appears to be a strong practice of routinely assessing the overall environmental, occupational and industrial status of Hanford Operations as evidenced by the monthly Health Instrument Division monthly reports and the integration of this information into the Hanford Works monthly report as evidenced in the following:

> HW-10378, HM Parker, "Health Instrument Division Report June 1948."
>
> HW-19622, Compiled by Division Managers, "Hanford Works Monthly Report for November 1950," December 20, 1950.

Apparently these reports were prepared for each month. These reports include results of the routine and apparently frequent practice to use "Special Work Permits" to assess workplace hazards prior to performing work. There is also evidence of the type and number of personnel dosimeter calibrations. Likely the best reference of early Hanford radiation protection practices is presented in:

> Wilson, R. H., 1987, *Historical Review of Personnel Dosimetry Development and its Use in Radiation Protection Programs at Hanford: 1944 Through the 1980s*, PNL-6125, Pacific Northwest Laboratory, Richland, Washington.

Considering that the Hanford dosimeter was integrated with the security credential, and the security credential was expected to be worn in full view for viewing by security and other staff,

the use of dosimeters could easily be checked by other workers and by radiological control staff. Based on the content of the following reference, failure to wear personnel dosimeters (i.e., meters at the time) was considered to be a Class I radiation incident. There was a Hanford policy for workers to wear dosimeters as described in the following.

> Parker, H.M., 1955, letter., to all Hanford supervisors, "Wearing of Personnel Meters, June 13.

It is expected that Hanford routinely used temporary dosimeters to monitor worker exposure in addition to the routinely exchanged dosimeters. The precise use would be specified in the Special Work Permits along with specifications of other safety considerations. At one point in time at Hanford, temporary dosimeters were frequently used for visitors, workers who forgot their badge, etc. They were also used to measure dose to various parts of the body and for a time to measure dose to workers from off-site work.

The tally of reasons for dosimeters that could not be processed is apparently included in each of the monthly Health Instrument Division report. For example, in the report for June 1948, the following tally is presented.

Reason	**No. of dosimeters**
Badge lost in area	3
Badge dropped in liquid	2
Wet badge	1
Lost in processing	1
No packet in badge	1

The report also stated that "Investigation of the lost readings where required showed no possibility of an overexposure." This would seem to imply that all significant situations of worker exposure relevant to the radiation protection standards and Hanford practice were evaluated.

A summary of the PIC results is also apparently included in each of the early monthly Health Instrument Division reports. Two PICs were worn by each worker because of the variability of the PIC response and the significant tendency towards false high readings. For example, in the report for June 1948, a tally is presented for each operating area including the number of paired readings lost for the month and the total for the year. The assessment in the report stated the following:

> *No significant pencil result was confirmed by the badge result. Investigation of lost readings showed no possibility of an overexposure. Pencil results of 100-130 mr and badge result of 90 mr, for one day for an H.I. supervisor in a 100 area resulted when a planned exposure of 100 mr was calculated.*

Since its inception, PNNL has been responsible for providing dosimetry and instrument calibration services. Dosimeters (i.e., beta/gamma, neutron, extremity, and multiple dosimetry) are provided through PNNL. The contractor submits a form requesting dosimetry and PNNL provides this service. The field does not assign the dosimeter of record; however, they can issue

temporary dosimetry, SRDs, and extremity dosimetry. Subcontractors and visitors also receive their dosimetry through PNNL. Dosimeter exchange is performed by the cognizant manager. PNNL ships the new dosimeters to the manager, and the manager collects and returns used dosimeters. Issuance is now based on a potential exposure to exceed 100 mrem per year (i.e., beta/gamma and neutron). There has been some variance in the number of workers monitored at Hanford over time. In the 1970s and 1980s, individuals were less likely to be monitored as compared to earlier years. Since the 1980s, the number of TLDs issued has decreased.

There has not been a good process for monitoring shallow dose from beta exposure at Hanford. This is an issue as there are numerous beta emitters encountered (e.g., strontium, uranium). Hard-to-detect beta emitters have to be compensated for in the field procedures, as there are no instruments rugged enough for use in field conditions that can adequately detect them. Contractors other than Environmental Restoration Contractor (ERC) have decided not to address the detection issues associated with hard-to-detect beta emitters.

The Administrative Control Limit (ACL) has changed over time. Initially, ACLs were equivalent to the DOE regulatory limits of the time. There were no company-specific ACLs. There were also job specific limits for particular tasks (e.g., entry into the Ballroom at 100N Reactor). When an individual exceeded the limit of the time (i.e., 300 mr in a week or a particular job limit), they were removed from radiation work. As Low As Reasonably Achievable (ALARA) was eventually integrated into the radiation protection program. The DOE and companies eventually established ACLs below the regulatory limit. There was and is still a contractual agreement with the union that an individual will not received more than 300 mr in a seven day period.

Beta/Gamma Dosimeter Design and Response

Based on the brevity of the 234-5 memos from supervisors to the dosimetry group, it appears that the significant over-response of the early (1944–1957) Hanford two-element dosimeter open-window to low energy photons was a well-known situation Memos examined to date to utilize a low-energy photon calibration only referenced 234-5 workers. This workplace involved plutonium handling. Importantly, there was no beta exposure of these workers. Thus, a calibration correction could be applied (i.e., use a low-energy photon calibration instead of the usual uranium beta calibration). This adjustment would not be technically possible in the mixed beta and photon fields that were present in many Hanford workplaces, and thus, by implication it is expected that non-penetrating doses may be significantly over-estimated based on the primary use of a non-penetrating dose calibration based on uranium.

Although excluded in the current Hanford TBD, the specification of $H_p(0.07)$ at a density thickness of 7 mg/cm^2 (0.07 mm) for the skin represents a capability unattainable for most current personnel dosimeters and certainly for historical dosimeters. This depth in tissue is equivalent to shielding from about 5.6 cm of air. As such, the dosimeter design effectively places a lower energy threshold on the beta radiation that can reach the responsive dosimeter component. It is common to achieve accurate $H_p(0.07)$ results in laboratory testing to selected beta sources based on matching the calibration and testing spectra. However, in the workplace, straggling of beta radiation will degrade the spectra and could result in significant error in the measured dose. The preferred option operationally is to preclude worker exposure to low-energy beta radiation by use of protective clothing, glasses, etc., and certainly to handle significant beta

radiation sources at a distance using tongs, etc., such as in sample preparation. Operationally, Hanford dosimetry practice was to assign all recorded penetrating x-ray, gamm, and neutron dose to the skin in addition to all recorded non-penetrating beta and x-ray dose.

Neutron

The neutron radiation direct responding NTA neutron dosimeter response has been shown to be independent of whether the dosimeter was irradiated in-air or on-phantom as noted in:

> Budd, R.O., letter to file, "Single Collision Versus Multiple Collision Fast Neutron Dose Calibration of the Hanford Neutron Film Badge Dosimeter." June 2, 1963.

The albedo thermoluminescent neutron dosimeter (TLND) is very sensitive to the location from the surface of the body. Policies for radiation worker staff were to not use lanyards to hold the dosimeter and to affix the dosimeter at the location of a shirt pocket.

The Hanford practice to process NTA film only if the beta/photon dosimeter showed a photon dose greater than about 90 mrem, was apparently used only for non-plutonium facilities and began in the early 1960s based on information in:

> Watson, E. C., 1959, *A Review of the NTA (Fast Neutron) Film Program*, HW-61008, Hanford Atomic Products Operation, Richland, Washington.

For workers in Hanford plutonium facilities, all NTA film was processed regardless of the photon dose. It is not known if a blank or a zero dose was recorded, but undoubtedly there is an indication that a dosimeter was assigned.

The primary technological shortfall, as noted in the Hanford TBD, concerns the limitations of NTA film to reliably detect and accurately measure neutron dose in Hanford plutonium facilities. The first Atomic Energy Commission (AEC) workshop held 23–24 September 1969 included the stated concern: ". . .for intermediate energy . . .neutron sources, nuclear track emulsion, type A (NTA) personnel neutron dosimeters cannot be effectively used" as noted in the following:

> Vallario, E. J., Hankins, D. E., and Unruh, C. M. *AEC workshop on personnel neutron dosimeter*, Richland, WA; Pacific Northwest Laboratory; BNWL-1340; 1969.

Since this dosimetry method was essentially the only method available prior to the implementation of the TLND at Hanford on January 1, 1972, this leaves a concern for the accuracy of the recorded neutron dose, particularly in intermediate energy fields.

The primary source of neutron-to-photon dose ratios for Hanford is the 1972 AEC Headquarters evaluation of long-term Hanford plutonium facility workers. This evaluation considered neutron doses since the initiation of the Hanford plutonium facility operations, and was focused on the potential for a worker to have exceeded AEC radiation dose limit guidelines during any of the previous years.

Multiple Dosimeters

The chest dosimeter was primarily there to measure the dose the worker receives from general area exposure. There are situations where individuals receive higher levels of exposure on portions of the body other than the chest. For example, if an individual is standing above radioactive material, the feet receive more exposure than the chest. In some cases, there are detectable fields at the feet but little or no exposure is recorded on the routine dosimeter or hands. Other examples of non-uniform exposure include analytical laboratory operations, reactor operations and glovebox work. In non-uniform exposure conditions extremity dosimetry or multiple dosimetry are used. Multiple dosimetry packs are obtained from PNNL. The field radiological organizations do issue SRDs and finger rings when deemed necessary for particular operations.

Multiple dosimeters have likely always been used. Prior to the effective dose quantity that assigned weighting factors to combine dose to various parts of the body (ANSI/HPS N13.41) into an overall dose quantity, the dose was assigned to either the whole body or to the extremities based on the location of the dosimeter. For example, the dose would be assigned to the whole-body deep dose for a dosimeter located on the head, legs above the knees and arms above the elbow. The dose would be assigned to the extremities for a dosimeter located on the legs below the knees and arms below the elbow.

There have been situations were individuals were submerged in contaminated water. Underwater maintenance at the 100 N Reactor was done in the 1970s using several thermoluminescent dosimeters placed on selected locations of the worker's body. The dosimeters were sealed with plastic to minimize moisture penetrating the dosimeter.

Area Dosimeters

Personnel dosimeters are commonly used as area dosimeters along with other types of area or facility monitoring instrumentation. Measurements are typically used to assess the radiological levels or radiation types within a facility zone. This data is simply an adjunct to the extensive array of workplace instrumentation and contamination surveys routinely taken to assess the radiological status of the respective work areas. It is typically not a good candidate for dose reconstruction since occupancy of the worker, movement of the worker within the work area, and the time spent in the work area are not known.

Internal Dosimetry Monitoring

The bases for determining bioassay requirements varied over the years and a complete history has not been compiled. The Wilson history states that beginning sampling routines [circa 1946 for plutonium] were established for three classes of employees:

Class I – continuously, >50% in product work – every 2-3 months
Class II – occasionally, < 50% in product work – every 4-6 months
Class III – all other employees for control – every 12 months.

But for how long these criteria applied and exactly how people were assigned to Class III is not known. Other criteria include employees at certain plants, but not visitors unless hands on work

was being done, risk of intake at 1% of the body burden, risk of intake at 100 mrem CEDE. The 1982 dosimetry manual, for instance, states "All workers whose job assignments involve work with radioactive material having a potential for becoming airborne should be considered for inclusion in the bioassay program." However, ultimately, the decision to include a worker in the bioassay program belonged to the contractor organization. Usually judgment of risk has been subjectively determined by RadCon personnel in charge of writing radiation work permits. Since the early 1990s, guidance on who should participate in the bioassay program has been agreed upon through a committee with representation from all Hanford contractors, DOE-RL, and Internal Dosimetry staff. Such guidance is provided in Section 5 of PNL-MA-552, *Hanford Internal Dosimetry Project Manual*. Complete answers to these questions for the entire history of Hanford are not known.

Facility contractors have procedures for monitoring the workplace and using that information to decide who is placed on a bioassay program in concert with guidance provided by the Internal Dosimetry Program. Department of Energy Richland Operations (DOE-RL) requires that facility owners characterize their radionuclide source terms and sources of contamination. Field organizations currently do annual characterization surveys and submit these to the Central Radiological Control group. The contractor dosimetry representatives notify Internal Dosimetry if something new shows up. This happened very recently when a facility found a significant concentration of ^{151}Sm in the mix of their contamination. Consideration of the need for a special bioassay procedure for this material is underway. Contractor programs are audited regularly as is the Hanford Internal Dosimetry Program (IDP).

Since at least the 1980s, the contractor dosimetry representatives have been tasked with notifying Internal Dosimetry of upcoming potential exposure to radionuclides not covered by present bioassay methods, and bioassay methods have been developed prior to exposure (e.g., ^{227}Th). In a few cases with short-term exposure or where half-lives preclude adequate bioassay monitoring (e.g., exposure to pure ^{90}Y), air sampling has been used in lieu of bioassay. There have not been many instances of the need to develop new bioassay procedures in the last 20 years. There is anecdotal evident in the historical files that such a policy was applied historically (^{147}Pm, thorium), although present internal dosimetry staff cannot vouch for the entire history of Hanford.

Bioassay available at Hanford has included urinalysis, fecal analysis, lung counts and whole body counts. Currently, the primary bioassay technique used is WBC. Some high-risk groups such as Radiological Control Technicians submit one or more bioassay samples per year for key radionuclides. Bioassay type is dependent on the facility at which an individuals works. The key radionuclides are determined from area characterizations and input from the Internal Dosimetry. Since the site has switched to determining bioassay based on facility characterizations, the frequency of bioassay has decreased. During the Westinghouse era, in vitro bioassay monitoring was more widespread in an effort to prove individuals hadn't gotten an intake. This has resulted in a cost savings. For example, FFTF workers initially submitted plutonium urinalysis and received a whole-body count. Their workers now received only whole-body counts. Workers who are involved in radiological operations across the site may be required to conform to the bioassay requirements of each area.

In Vivo

The Minimum Detectable Activities (MDAs) for ^{235}U and ^{234}Th (assumed to be in equilibrium with ^{238}U) for the chest counters are given in Table 5.3.2-1 of the Hanford Internal dose TBD. It may be that the MDA values for the years prior to about 1987 were more like decision levels rather than MDAs; this is still being investigated. The "effectiveness" of detection depends on the solubility class, frequency of measurement, and the effectiveness of workplace monitoring to detect intakes leading to prompt in vivo counts. Uranium workers were also monitored by urinalysis.

Concerning technology shortfalls, prior to 1983, Sodium Iodine (NaI) detectors were used which had good detection efficiency but poor resolution and were more subject to interference from radon progeny, background uranium peaks, and low-energy noise. Beginning in 1983, HPGe detectors with thin windows have been used which have better resolution and hence less chance of interference. Improvements over the years have come from the use of more detectors or detectors with larger surface areas, better electronics, and elimination of low-level background peaks from material intrinsic to the detectors. In general, improvements have mostly affected reliability and usability, with modest improvements in detection levels.

The uncertainty associated with determining the activity of some radionuclides in the lung is quite large. Often, when ^{241}Am is measured as an indicator of the presence of plutonium, the largest uncertainty concerns the ratios of the plutonium isotopes to ^{241}Am. Other uncertainties arise from things such as:

- Differences between a reference man phantom and real humans
- The assumption of uniform versus the possibility of non-uniform deposition of material in the lungs
- The use of height/weight correction factors instead of ultrasound for determining chest-wall thickness prior to 1979
- Reproducibility in positioning of detectors
- Interference from activity in bone and liver (or for wounds, in axillary lymph nodes)
- More crude phantoms prior to the Livermore phantom. Prior to the Livermore phantom, Hanford used a masonite phantom and cross-comparisons were done when the Livermore phantom came into use. The agreement between the phantoms was not bad. Our technical specialist who worked at the in vivo counter at that time remembers the difference to be on the order of 15%.

The magnitudes of these uncertainties vary according to the energy of the photons being measured; for instance, the chest-wall thickness uncertainty can be quite significant for ^{241}Am photons and less so for the key photons from ^{234}Th and ^{235}U. The uncertainty specifically associated with cruder phantoms prior to use of the Livermore phantom has not been estimated.

The information on how often plutonium exceeded the detection limits is not readily available. The switch to HPGe detectors as chest counters did result in a noticeable increase in detection of ^{241}Am chest counts in the 1983-1984 time frame.

In Vitro

Minimum detectable doses have been officially documented since 1989. These are based on worst-case assumptions for routine monitoring. Very few intakes are actually discovered through routine monitoring, however. Most are discovered by workplace monitoring followed by bioassay taken shortly after the potential intake. Under most situations the minimum detectable dose using prompt bioassay is much smaller than the documented doses using worst-case assumptions. Present minimum detectable dose tables are found in PNNL-MA-552, Exhibits 5.1-8, Section 5 (Carbaugh 2000) or in each of the radionuclide chapters in PNNL-MA-860, *Methods and Models of the Hanford Internal Dosimetry Program* (Carbaugh 2003).

Present-day recommendations for bioassay frequencies are found in Section 5 of PNL-MA-552. A complete history of bioassay frequencies at Hanford has not been compiled. Since about the 1970s, plutonium workers have generally had annual urinalyses and many have had annual chest counts as well. Uranium workers have had different frequencies depending on the solubility of the uranium and risk of exposure, ranging from twice a month urinalyses to annual chest counts. Fission products are usually monitored by an annual whole-body count; ^{90}Sr monitoring has varied from annual to biennial urinalyses in conjunction with annual whole-body counts. Tritium monitoring has varied from daily to monthly.

The need for special bioassay samples has usually been a joint decision between facility RadCon or contractor dosimetry representatives and Internal Dosimetry Program staff. Guidelines for when to contact Internal Dosimetry staff have been around for years, although the complete history has not been determined.

There have been isotopes used as an indicator for the presence of other radionuclides. ^{137}Cs is often used as an indicator of intake for facilities where the source term may include lesser activities of ^{90}Sr or actinides, in which case positive indication of intake by in vivo counting is followed up with bioassay for the other radionuclides, or a radionuclide characterization of a sample of the specific source term (air sample, smear of the contamination) is used to account for the radionuclides not measured in the bioassay.

Co-worker data is often used to determine the mix of radionuclides or the solubility of radionuclides in an incident, based on the worker with the most robust bioassay data. However, each person in an incident is checked out personally and confirmation or not of an intake is based on that person's bioassay results. Intakes assigned from air-sample data can be assigned to a group of workers, but the total intake is specific to each person based on his/her individual exposure time. The only time remembered when co-worker data was used to assign dose was in the late 1980s, when all participants in a routine fecal program were given the same dose based on the statistics from the group's bioassay data.

Detectors and bioassay procedures are better today than they were in the 1950s, but that doesn't mean that the program in the 1950s had technology shortfalls. Until recently the idea of a technology shortfall was subjective and no doubt changed over time. At Hanford in at least the 1960s through mid-1980s, reporting requirements were based on long-term deposition in the body of 5%, then later 1%, of the Maximum Permissible Body Burden, with burdens less than these criteria being considered insignificant. When converted to the committed effective dose system used today, for Pu or U these criteria would appear to be quite significant. The DOE

Internal Dosimetry Implementation Guide uses the ability to detect an intake that would result in Committed Effective Dose Equivalent (CEDE) of 100 mrem as an acceptable program, and not achieving this is referred to as a technology shortfall. Under this definition, routine plutonium monitoring alone (i.e., without prompt bioassay due to a potential intake incident) constitutes a technology shortfall; so does uranium monitoring for class W or Y uranium compounds.

Dose Calculation

The default assumptions used when calculating internal dose (i.e., particle size, solubility class, date of intake, type of intake, etc.) are described below:

- Since 2001, the default particle-size has been 5 μm; prior to that it was 1 μm.

- Default solubility classes are generally decided per each incident or intake based on discussions with the facility RadCon organization. If no information is available, ICRP recommendations are used. Sometimes the solubility can be determined from the fit to the bioassay data. The internal dosimetrist uses all the information he/she can obtain and documents the final decision.

- If the date of intake is not known nor can be reasonably deduced, the midpoint of the period from the last appropriate bioassay is used.

- The default intake mode is acute inhalation unless there is strong evidence of another mode.

- Based on animal and solubility studies performed in England, it is assumed that plutonium slowly becomes more insoluble when exposed to air under ambient conditions, i.e., it oxides. Internal Dosimetry is aware of possible changes to uranium when disposed to the soil. Cesium and strontium are assumed to be class D under all conditions (no strontium titanate at Hanford). For other radionuclides, ICRP recommendations are used unless the bioassay data or specific facility information indicates otherwise. No generic changes have been made to the default assumptions because of material in the waste tanks. There have been no intakes from tank farms large enough to generate bioassay data that contradict the ICRP recommendations.

A particle-size determination was made years ago for europium radioisotopes at N Reactor (Weetman and DeHaven 1982). At one time, particle-size determinations based on nasal swabs were routinely made for significant intakes; however, nasal swabs do not contain particles that are representative of the air breathed and, hence, this information had limited use. Rarely, particle-size information is obtained on air samples from incidents. This information then becomes part of the documentation in the worker's dose assessment. Because intakes usually occur from specific incidents not chronic exposures, routine particle-size studies at various facilities have not been considered cost-effective.

The definition of "high-fired" varies among chemists, and some say the processes at Hanford do not meet the definition of "high-fired." Nevertheless, a step in the process at the PFP facility was to bake (calcine) the plutonium at about 400° C for an hour or two. The plutonium was not

left in this form, however. More recently, the plutonium in the plant has been "thermally stabilized," which has involved converting it to an oxide by heating. The solubility of the thermally stabilized material is not known; however, one intake involving this material has not, to date, shown it to be highly insoluble. There has been spotty evidence for other intakes of plutonium at Hanford that seemed to be more insoluble than normal class Y; however, this evidence was not officially accepted by the scientific community when submitted to a peer-reviewed journal. Some evidence for highly insoluble plutonium has been demonstrated at Rocky Flats and Savannah River as well as Hanford. This information was taken into account during the development of absorption type S material in the ICRP 66 lung model. Two case studies were published in *Radiation Protection Dosimetry* (Carbaugh and LaBone 2003).

There are no tritides presently at Hanford nor have there been in recent years. To date there is no evidence of use of tritides historically at Hanford.

Workers' intakes and doses are reevaluated when requested by the workers' employer (i.e., updated with the currently accepted methodology). These reevaluations and reassignments of internal doses are rare and usually occur for a single worker or perhaps a small group of workers. DOE Order 5480.11, which heralded in the ICRP 30 methodology in 1989, did not require historical intakes to be converted. Nevertheless, Hanford converted all prior intakes for workers active at that time that had potential for producing present (i.e., 1989) dose. Doses calculated and reported during the period 1989 to 1992 were annual doses, not committed doses; hence, prior intakes of elements like plutonium produced dose in 1989 and, therefore, were reevaluated using the ICRP 30 methodology. Workers no longer at Hanford in 1989 were not reevaluated.

Background Levels

Presently it appears that the ^{137}Cs background body burden is below detection (50^{th} percentile about 0.1 nCi and 99^{th} percentile about 0.5 nCi), unless the worker has had some unusual intakes, such as from significant consumption of wild game.

Background levels of uranium and thorium in feces have not been established. These bioassay are seldom obtained. Background uranium excretion in urine has been established and is documented in section 7.1.3 of PNL-MA-860. 0.2 μg/d is used as the upper boundary of background, although higher values have been observed in a few individuals with home water wells.

Recycled Uranium

Recycled uranium was handled at Hanford including in the 300 Area and at U-plant. When UNC served as a site contractor, they had control over the recycled uranium program for their tenure at the site.

An evaluation of the importance of radiological contaminants in recycled uranium was made in 1989, and the contribution to annual or committed effective dose from plutonium, neptunium, and technecium was found to be less than 10%. Uranium workers had bioassay for uranium, but were not sampled for the contaminants except when required because of other exposure; e.g., many PUREX workers were sampled for both uranium and plutonium due to exposure to both of the elements.

Radiological Records

Hanford radiation records were centralized in 1945. All records have remained at Hanford, except records for Du Pont personnel who left with the company in 1946. Other than those records, no records are known to have been removed from the systems. All bioassay records are included, except where dose was calculated directly from bioassay records. PIC data was retained in the centralized records through 1957, when the dosimetry reporting process was computerized. After 1957, records may exist in the field, but the Hanford records program has no responsibility for any records that do exist. Any non-routine calculations for dosimetry results were documented and placed in the individual's records. Any changes to doses in the database such as updated internal dose assessments are documented in the individual's radiation document file.

To the best of our knowledge, the records for Hanford workers who entered a security and radiological controlled area (i.e., 100 B, C, D, F, DR, H, KE, KW, N, 200E, 200W, 300, etc) are complete in that dosimeters were assigned to each worker, the dosimeters processed, and the results recorded. Doses for missing or destroyed dosimeters were estimated and recorded. An assessment of the integrity of Hanford worker doses used in the Hanford Health and Mortality Study is found in PNL-7439, *A Study of Detailed Dosimetry Records for a Selected Group of Workers Included in the Hanford Mortality Study.*

Hardcopy dose reports for early construction personnel were microfilmed in alphabetical order, but not indexed into the computer-aided recovery system. Recovery is done through manual searches. All measured doses are currently included in an electronic database including employees, subcontractor, and construction workers. This database is referred to as REX. There are numerous code names that may be encountered in the radiation dosimetry records. These codes are defined in the REX data dictionary. The radiological records group has also developed a code. Questions on a specific record may be directed to the Hanford Radiological Record Program (HRRP) manager.

In vitro bioassay results were recorded on cards that were copied into the worker's radiation exposure history file. These results are now available on microfilm and microfiche pertaining to that worker, and most of the information on the cards has been entered into the electronic database, REX. REX is not 100% complete however, as some gaps have been found; for instance, tritium bioassay results are not in REX until starting in the 1980s.

All incident and personnel contamination reports are included in the individual's radiation document file. There is no chronological or facility incident file. Personnel contamination reports are included in the individual's radiation document file. Contamination events without personnel contamination are not part of the HRRP records. Contamination surveys and documentation are the responsibility of the contractors. Field radiological records are the responsibility of the contractors. Specifics on these records are only available from the contractors.

On an annual basis, workers receive annual radiation exposure reports which include annual and cumulative dose. Doses of monitored visitors are reported as soon as their dosimeters are processed. Reports are sent to the individual and the employer. Employees receive annual reports. It is uncertain when reports to visitors were first required.

Historical dosimetry terms used with the Hanford radiological records system are confusing and should be interpreted in the context of the information being examined. One approach is to separate terms according to those used to record dosimeter-processing results and those used to estimate compliance dose quantities according to DOE and predecessor agency guidelines. There are terms such as non-penetrating and penetrating dose that have been used in more than one context. Basically, beta, gamma, x-ray, and neutron are generally used to record dosimeter processing dose results, whereas the non-penetrating, shallow, or skin are used to record the skin dose of the "whole-body" and the penetrating or deep, used to record the "whole body" dose. The fractions of the x-ray dose assigned to the skin and whole body dose were 65% and 35%, respectively.

The AEC required that the tritium dose be included in the compliance "whole body" penetrating dose beginning in about 1962, and this continued until DOE implemented the effective dose concept in 1989. The neutron dose is included in the compliance "whole body" penetrating dose for all years. There was a period when whole-body dose was calculated directly from tritium bioassay data and placed in the individual's dose record, but the bioassay data was not recorded.

Radiological records generated in the field were/are typically maintained by the responsible contractor. For example, if ERC documented a contamination incident, this record would be maintained by ERC and not by Hanford Central Records. Field radiological records are not maintained separately from the dosimetry records. These records include work permits, survey reports, air monitoring reports, instrument response checks, fixed instrument counting data, etc.

In more recent years, each radiological control group generates a Records Inventory and Disposition Schedule listing all quality records created and the required retention time. Field radiological control records are maintained onsite for three years, then forwarded to Seattle. Each survey, air sample, or counting record generated has a unique number. Logs are maintained of each survey and air sample taken. Radiological Control Technicians or their supervisors also maintain a daily log of activities including routine and non-routine situations. Field radiological records are usually kept onsite for three years then forwarded to the Seattle repository. These records are submitted by type and year making them difficult to retrieve. For example, boxes are submitted under general titles such as 1985 Dosimetry Records. Other contracts such as ERC maintain imaged copies of field records as well as the hardcopies.

Incident Reporting

There are several mechanisms for reporting incidents. There is the Occurrence Reporting System, which was established by the DOE, and has defined threshold levels for reporting. Each field organization has a different process for responding to incidents depending on the type and severity. Some facilities have a more formal process such as PFP, who establishes a command center. At other facilities the process is less formal and incidents are merely reported to management. When incidents are more common such as Continuous Air Monitor alarms at PFP, there may actually be a procedure developed for that type of incident.

Radiation Problem Reports (RPRs) were initiated in 1976 or 1977. These address situations such as cases of unexpected dose, spills of radioactive material, trending issues, and low threshold items that don't affect the environment or individuals. Incidents may be reported in the RCT logbook or documented on survey reports. Every RPR, occurrence report, or items from

assessment reports has some level of corrective action. There is also a corrective action management system which lists events and corrective actions taken. These items are tracked by a Corrective Action System number. The historical reporting system was not as extensive in the earlier years.

Incidents with potential for intake result in notification of Internal Dosimetry staff. Internal Dosimetry maintains a 24-hr, 365-day emergency response number for incidents that require immediate dosimetry assistance. Each facility RadCon organization has that number in their response procedures, as does the site medical organization and the site security and emergency response organization. This number had not changed for decades and is so ingrained in the RadCon community that the Internal Dosimetry group is often called before company management or the medical community. Guidelines for response to various potential intake situations, including guidance for when to call internal dosimetry and when to involve the medical community for consideration of treatment, are contained in the *On-Call Exposure Evaluator Manual*, PNL-MA-857. When prompt in vivo measurements are made, response involves the presence of the internal dosimetrist, the dosimetry representative from the involved contractor, and usually the worker's supervisor.

A case number for each worker in the incident is immediately established and case status is tracked through completion. A written evaluation is completed for each worker that includes a brief description of the incident and relevant workplace data, such as facial contamination, nasal swab results, and air sample results if available. If an intake is confirmed, the radionuclides and intake activities are included in the written report as well as doses, bioassay data, computer code printouts if used, etc. Basically, all the pertinent information needed to peer review the evaluation is included. Each evaluation is peer reviewed. The complete package is archived in the worker's radiation exposure history file. The field usually provides any information even remotely connected with the incident or in proximity; in such cases, most workers given prompt bioassay show no detection, whereas the few workers who were most at risk sometimes do show intakes. This gives everyone some confidence that all the exposed workers and many who weren't are being checked out.

Audits/Assessments

There are many reviews of the external dosimetry program, particularly in more recent years. Historically, there are the results of audit dosimeters processed with Hanford worker dosimeters. The audit dosimeters were exposed to a known radiation beam and exposure level. Hanford even assigned fictitious names (i.e., to represent a worker) to some audit dosimeters that were routinely processed. The details of the respective audit dosimeters evaluations are not easily available. However, a report published in 1967 to develop film dosimeter performance criteria used Hanford calibration dosimeter data developed during the previous 14 years per the following:

> Unruh, C.M., H. V. Larson, T. M. Beetle and A. R. Keene, 1967, "The Establishment and Utilization of Film Dosimeter Performance Criteria," BNWL-542, Battelle Northwest, Richland, Washington.

A particularly notable AEC review in 1972, as summarized in the Hanford TBD, concerns the evaluation of total photon and neutron dose for 18 long-term Hanford plutonium facility workers. Information for this review is available in:

> Biles, M. B., 1972, Letter to T. A. Nemzak (Ad Hoc Technical Committee Finds), Pacific Northwest National Laboratory, Richland, Washington, 23 November.
>
> Fix, J. J., R. H. Wilson, and W. V. Baumgartner, 1997b, Retrospective Assessment of Personnel Neutron Dosimetry for Workers at the Hanford Site, PNNL-11196, Pacific Northwest National Laboratory, Richland, Washington.

Another detailed examination of Hanford beta/photon film dosimeters was done in the mid- to late-1960s by the University of Pittsburgh as part of the early AEC health effect studies to include taking Hanford processed film to Pittsburgh for an independent evaluation. This is presented in the following:

> Mancuso, Thomas F., Barkev S. Sanders, and Allen Brodsky. "Feasibility Study of the Correlation of Lifetime Health and Mortality Experience of AEC and AEC Contractor Employees with Occupational Radiation Exposure. " NYO-3394-5, Progress Report No. 2, Department of Occupational Health, Graduate School of Public Health, University of Pittsburgh, Pittsburgh, Pennsylvania.

The internal dosimetry program is DOELAP accredited so it is audited at least once every three years. The Hanford contractors also perform audits about once every three years. Management and internal QA audits are also performed approximately annually.

Field Radiological Control also undergoes routine audits conducted by the Department of Energy. They are also required to self assess all elements of their program over a three-year period.

Unauthorized Practices

Unauthorized practices have been a part of Hanford history since the beginning. These practices have resulted in both positive and negative bias with regard to dose assignment.

There have been a number of practices that likely resulted in an underestimate of dose or potential intakes. In the early years at the laboratories, mouth pipetting was not uncommon, as many scientists came from universities where it was allowed. At times laboratory personnel would suck too hard and end up with mildly radioactive material in their mouth. What would be considered an unauthorized practice by todays standard, such as kicking fuel, was not uncommon in the early years.

In the interest of overtime, some individuals have tried to minimize their exposure to prevent burn out. For example, during fuel handling in Buildings 308 and 327, individuals would turn their extremity dosimetry outward or not wear it at all. Other individuals would wear their dosimeter on their back or in their pocket. There were situations where workers would remove their dosimeters on purpose to ensure they did not exceed the exposure limits.

The policy for ensuring workers wore their dosimeters into radiation zones was not as stringent in the early years as it is currently. If an individual entered an area without their dosimeter on, they were not immediately sent out of the area to retrieve it. There were times when individuals were in the area for an hour or more without dosimetry. Individuals were never directed not to wear their badge. In the 1970s, there was an improvement in enforcement of film badge wearing.

At times, RCTs would notice unexpected situations that didn't previously exist. No one would admit to causing the situation. Other practices have occasionally occurred even in the recent years. In some cases individuals have purposely done things such as chewing tobacco in the area, or taking contaminated tools home. In other cases, individuals have inadvertently done things such as rubbing their face or wiping it with their Personal Protective Equipment (PPE).

Conversely, workers have purposely exposed their badge, presumably to get out of work. Situations occurred in the late 1970s where workers purposely exposed their dosimeters to radiation. An investigation was completed including collection of evidence, arrangement for analytical services such as bioassay, medical examinations such as blood tests or chromosome aberration analyses, gamma spectroscopy, etc. In both instances, the workers were terminated from employment once it was certain from the collected evidence what the worker had done. It is expected that Hanford practice was to penalize any deliberate action to sabotage the integrity of the dosimetry system.

References

Biles, M. B., 1972, Letter to T. A. Nemzak (Ad Hoc Technical Committee Finds), Pacific Northwest National Laboratory, Richland, WA, 23 November.

Budd, R.O., letter to file, "Single Collision Versus Multiple Collision Fast Neutron Dose Calibration of the Hanford Neutron Film Badge Dosimeter." June 2, 1963.

Carbaugh, E.H., 2000, *Hanford Internal Dosimetry Project Manual*. PNL-MA-522, Rev. 3, http://www.pnl.gov/eshs/cap/dss/epub.html.

Carbaugh, E.H., 2003, Methods and Models of the Hanford Internal Dosimetry Program. PNL-MA-860, http://www.pnl.gov/eshs/cap/dss/epub.html.

Carbaugh, E.H., 2004, *Bioassay versus air sampling: practical guidance and experience at Hanford*. Health Physics Society Midyear Meeting, February 2004.

Carbaugh, E.H. and T. R. La Bone, 2003, "Two case studies of highly insoluble plutonium inhalation with implications for bioassay." *Radiation Protection Dosimetry*, Vol. 105, Nos. 1-4, pp. 133-138.

Fix, J. J., R. H. Wilson, and W. V. Baumgartner, 1997, *Retrospective Assessment of Personnel Neutron Dosimetry for Workers at the Hanford Site*, PNNL-11196, Pacific Northwest National Laboratory, Richland, WA.

Gilbert, E.S. 1990. *A Study of Detailed Dosimetry Records for a Selected Group of Workers Included in the Hanford Mortality Study.* PNL-7439, Pacific Northwest National Laboratory, Richland, WA.

HW 1950, (Compiled by Division Managers), *Hanford Works Monthly Report for November 1950*, HW-19622, December 20, 1950.

Mancuso, Thomas F., Barkev S. Sanders, and Allen Brodsky. "Feasibility Study of the Correlation of Lifetime Health and Mortality Experience of AEC and AEC Contractor Employees with Occupational Radiation Exposure. " NYO-3394-5, Progress Report No. 2, Department of Occupational Health, Graduate School of Public Health, University of Pittsburgh, Pittsburgh, PA.

Moore, R.L. (1992), *As I Knew Him: Recollections of a Shy Timid Soul Who Led a Quiet Life Spanning Some Interesting Times*. Published by Robert Lee Moore, Richland, WA, pp. 55-142.

Parker, H.M., 1948, *Health Instrument Division Report June 1948*. HW-10378.

Parker, H.M., 1955, letter, to all Hanford supervisors, "Wearing of Personnel Meters, June 13.

Unruh, C.M., H. V. Larson, T. M. Beetle and A. R. Keene, 1967, "The Establishment and Utilization of Film Dosimeter Performance Criteria," BNWL-542, Battelle Northwest, Richland, WA.

Vallario, E. J., Hankins, D. E., and Unruh, C. M. *AEC workshop on personnel neutron dosimeter*, Richland, WA; Pacific Northwest Laboratory; BNWL-1340; 1969.

Watson, E. C., 1959, *A Review of the NTA (Fast Neutron) Film Program*, HW-61008, Hanford Atomic Products Operation, Richland, WA.

Weetman, R. A. and H. W. DeHaven, 1982. *Characterization of Airborne Radionuclides in UNC Facilities*. UNI-2164, UNC Nuclear Industries, Richland, WA.

Wilson, R. H., 1956, Neutron Exposure at the 105-DR Reactor, to file, dated February 2.

Wilson, R. H., 1957, *Reproducibility of Personnel Monitoring Film Densities*, HW-51934, General Electric, Hanford Atomic Products Operation, Richland, WA.

Wilson, R. H., 1960, *Detection Level of the Film Badge System*, HW-67697, General Electric, Hanford Atomic Products Operation, Richland, WA.

Wilson, R. H., 1987, *Historical Review of Personnel Dosimetry Development and its Use in Radiation Protection Programs at Hanford: 1944 through the 1980s*, PNL-6125, Pacific Northwest Laboratory, Richland, WA.

Wilson, R. H., J. J. Fix, W. V. Baumgartner, and L. L. Nichols, 1990, *Description and Evaluation of the Hanford Personnel Dosimeter Program From 1944 Through 1989*, PNL-7447, Pacific Northwest Laboratory, Richland, WA.

Watson, E. C., 1959, *A Review of the NTA (Fast Neutron) Film Program*, HW-61008, Hanford Atomic Products Operation, Richland, WA.

ATTACHMENT 5: SITE EXPERT INTERVIEWS — PRODUCTION, ENVIRONMENTAL RESTORATION, AND CONSTRUCTION

Movement from the Site

Historically, much of the Hanford workforce was fairly stable. Staff in the 200 Area typically remained in the 200 Area; however, they may have moved between different 200 Area facilities. Personnel in other areas of the site were more stable. ROVERS primarily included Security, Health Physics, and Maintenance staff. Engineers also moved around. The maintenance personnel worked out of a particular shop and were deployed to areas of the site as needed. There was a linkage between 100 Area operators and 300 Area fuel fabrication operators. When the 100 Area operators reached their maximum allowable dose for the week, operators from the 300 Area would be called in to perform functions at the reactors.

In the early years of operation, J.A. Jones was one of the companies involved in the construction of the site. They had a standard crew that was located on the Hanford site and temporary crews that were brought in to provide support to various construction projects.

General Safety Environment

Even in the early years workers wanted to be safe and return to their families each night. Generally, workers are very safety conscious and cooperative. Hanford had numerous contractors over time. Each contractor had its own attitude towards health and safety. Many companies put production ahead of safety. This was demonstrated by the fact that operations had authority at the work sites and made the final decision on whether a task was to be performed. Other companies were primarily concerned with industrial safety and reportable safety events. For example, General Electric actively tried to prevent reportable safety incidents, yet they were more casual about radiation protection.

Some contractors were focused on radiation safety. Rockwell Hanford Operations was very politically sensitive when it came to radiation, as they had no previous experience with handling radioactive material. This ultimately affected the workers. Other subcontractors brought in professional support staff including Industrial Hygienists and Health Physicists and implemented a strong procedure program.

Currently, safety has been an important part of operations at Hanford. Although production activities have not taken precedence over safety, some folks will cut corners to expedite work completion. If these individuals are caught, they risk disciplinary actions. There are still significant worker concerns over chemical exposures. Companies such as United Nuclear Industries, Westinghouse, and Lockheed Martin were not concerned about chemical exposures.

Reactor Operations

B-reactor was the first reactor on the Hanford Site. It was built in 1944. The original reactors were referred to as piles. The “piles” were located next to the river so water could be withdrawn from the river, purified slightly, and then pumped through the reactors to remove heat from the fission reactors. The water was held up for a brief period of time and released back into the

river. The single pass reactors used both vertical and horizontal control rods. Boron balls were used as poisons to control reactivity. The B-reactor is about 40' x 40' x 40' in size. As a portion of their training, Reactor operators were required to spend one week at the 10 MW TRIGA Reactor in the 300 Area.

The reactors were staffed around the clock. For example, the N-Reactor worked on the ABCD schedule. There were four rotating shifts and a day shift. Approximately 45 individuals per shift worked the rotating shifts including operators and support staff. The day shift had a crew of 45 individuals with an additional 40 individuals for maintenance. This allowed the reactor to operate 24 hours per day.

This facility was used for fuel testing.

The reactor fuel was loaded into the process tubes on the front face of the reactor and discharged out the rear face by putting the new fuel into the reactor. This operation was done with the reactor shut down. The process for charging and discharging the fuel from N-Reactor involved three operators plus a supervisor on the front face of the reactor and two operators plus a supervisor on the back face of the reactor. The fuel was first shipped from the 300 Area. It was removed from the box and put into a monotube. The monotubes were loaded onto racks on the wall adjacent to the reactor. Blue-coded fuel indicated weapons grade fuel and red indicated all other types of fuel. At the rear face of the reactor, the team would remove the cap. The front face team would also remove the cap on their side. A charge head was put in place. This forced coolant water into the tube. The loading machine was used to load the reactors. The shield plugs on both sides of the reactor were pulled. The monotubes would be lined up in the charge head. Water pressure pushed the new fuel into the reactor and the old fuel out. The monotube was then removed from the reactor. Charge heads were removed and caps were replaced on both sides of the reactor. After discharge of fuel from the reactor, it was stored in the reactor basin for 60-120 days to allow for decay of short-lived radionuclides. Operations used 25-foot tongs to handle the fuel after discharge and during storage into the basin.

For charging and discharging the reactors, the operators originally wore whites with rain gear and a face shield. Workers on both the front and rear face of the reactor were sprayed with coolant water. Some complained of ingesting water. The Personal Protective Equipment (PPE) was eventually upgraded to include a power air-purifying respirator.

Reactor neutron flux was controlled with the use of samarium and boron balls. The balls were fed into ball hoppers, and retrieved as necessary. Because the fabricated boron balls were too large for the hoppers, the outer coating was removed to get them to fit. These balls would have to be cycled once a year and were highly radioactive when they were removed from the reactor. If there were issues with the retrieval of these balls, an operator could reach or exceed the administrative dose limit in less than one minute (i.e., burnout). At times the balls would become stuck in the reactor and create what operators refer to as a dead spot.

Some early single pass reactor workers such as those in the Irradiation Testing group were intermittently monitored for neutron exposure with NTA film. Those that wore NTA film badges were aware that they were not effective in the reactor areas. Early air sampling did occur in the reactor areas. Bioassay changed for reactor workers over time. For example, the Irradiation Testing group initially submitted a urine sample about every six months. They were

unaware of what analyses were being done on the samples. As time progressed, they were no longer required to submit these samples. In addition, in vivo counts were completed only when there was reason to believe an intake occurred.

N-reactor operators routinely wore beta/gamma dosimeters. A few operators were monitored with neutron dosimeters and "Death Chips" or Personnel Nuclear Accident Dosimetry. This continued to be the case into the 1980s. Exchange frequencies varied by the era but ranged from weekly to quarterly. Bioassay for reactor personnel primarily included annual whole-body counts once the counters were available. Chest counts were limited to those individuals involved in incidents until more recent times. Reactor operators do not recollect being placed on a routine urinalysis program even though there were incidents of personnel and area contamination. For example, as previously mentioned, reactor coolant water often shot out onto the operators during charging and discharging of the reactors. Timekeeping, Pocket Ionization Chambers, and pocket alarming dosimeters (later years) were also used at the reactor. Fuel basin divers wore multiple dosimetry.

The reactors did not always operate smoothly. Helium buildup in the graphite caused cracking. Uranium sometimes expanded under irradiation and burst or blocked the tubes. The reactors experienced graphite disease caused by helium buildup. This would result in growth, swelling, or even cracking of the graphite. There were cracks on the side of 100F and 100B reactors. Also, some areas of the reactor would burn out and were no longer usable.

Initially, there were some problems associated with the operation of D-reactor. As a result, DR-reactor (D replacement) was built to replace it. They later solved the issues with D-reactor and were able to operate both.

The bioshields of some reactors were fairly reliable; however, the masonite occasionally got crumbly. This was especially true for reactors that run at increased temperatures such as C-reactor. The K-reactors used high-density aggregate concrete and, therefore, did not suffer from this problem.

Most reactors have undergone or are in the process of decontamination and decommissioning. This includes demolition of ancillary buildings, cocooning the reactor, and soil remediation.

200 Area

There were a number of operations that occurred in the 200 Area including chemical separations, plutonium finishing, liquid and solid waste management, and special projects such as the Critical Assembly Room located in 209E Building. The chemical separations buildings (i.e., 221-T, 221-B, 221-U, REDOX, PUREX) were massive rectangular canyons constructed in the 200 Area to chemically separate plutonium from uranium and fission products. Operations were handled remotely to provide protection to the workers from the high radiation fields. T-Plant operations began in December 1944 and were shut down during August 1956, while B-Plant operations began during April 1945 and were shut down during October 1952. Some work at B-plant required the use of neutron accident dosimetry. Startup of U-Plant operations was delayed until March 1952 since the capacity was not needed, and at that time U-Plant was reconfigured for use in recovering uranium from process waste.

The separations process initially operated with the Bismuth Phosphate Process, which was designed to recover only plutonium. The process worked using plutonium precipitation and centrifuging. This process was repeated until desired purity was reached. With the startup of the Reduction Oxidation (REDOX) Plant, the separations facilities started recovering both uranium and plutonium. The uranium slug was dissolved in nitric acid, a salting agent such as ammonium nitrate or aluminum nitrate was added, the material was put in contact with Hexone, and the extracted plutonium and uranium were washed out of the organic layer. Many stages of extraction were required. To extract plutonium with the uranium, plutonium was oxidized to the plus six state with an oxidizing agent, such as dichromate. When the uranium and plutonium were pure, a reducing agent (e.g., ferrous iron) was added causing the plutonium to reduce to a plus three state and wash out of the organic away from the uranium. Fission products that caused radiological concerns during this process include ruthenium, cesium, iodine, zirconium, and niobium, especially ruthenium (Moore 1992).

The Metal Recovery Plant (UO_3 Plant) and the Plutonium-Uranium Recovery by Extraction (PUREX) process were used to recover the uranium disposed of in the original Bismuth Phosphate Process. The high-level liquid wastes from the B and T Plants, which were stored in single-shell tanks, contained large quantities of uranium. The uranium bearing sludge from the underground storage tanks was pumped to process vessels in U-Plant. Uranium was recovered from the waste via the PUREX process and shipped off-site for reuse. U-Plant operations were shut down during January 1958.

The separations facilities had sampling stations on the canyon decks that were used to obtain process control samples. Samples were taken by operations and analytical laboratory personnel. The samples were transported in pigs to laboratories in the back of a pickup truck.

In the early days, transportation of materials between the separations plants and the Plutonium Finishing Plant (PFP) was also done using pickup trucks. PFP converted plutonium nitrate from the tail end of the separations process to plutonium metal. Work at PFP required neutron dosimeters and Personal Neutron Accident Dosimeters.

The 233-S building was constructed in 1957 to contain the third and final concentration cycle at 202-S which is associated with the 202-A PUREX Facility (A Plant) that became operational in 1956 utilizing the plutonium-uranium extraction process to process irradiated fuel. The PUREX Plant was shut down in June 1972 and restarted in November 1983. During the shutdown a new process was added to convert plutonium nitrate to plutonium oxide. It was started up and shut down several times between 1983 and 1992, with final closure announced in December 1992.

Solid waste was disposed of in burial grounds. Those in the 200 Area contain transuranic waste. Historically, low-level waste was buried in cardboard boxes. Transuranic waste was buried in drums. Procedures for inspecting radioactive waste drums prior to shipment were written to ensure integrity of drums filled and buried. Initially, procedures did not contain all required integrity checks.

There were numerous other facilities in the 200 Area that supported the separation of plutonium, the management of waste, and analytical needs. The 222-S Laboratory and the Waste Sampling and Characterization Facility are two support facilities still in operation at the site. Current

radiological activities in this area are primarily associated with Surveillance and Maintenance or decommissioning of facilities no longer in operation.

300 Area

The 300 Area was involved in a wide variety of operations including fuel development, fuel fabrication, waste management, Research and Development, analytical support, and test facilities. At the North end of the 300 Area, fuel rods were manufactured and canned. The rest of the 300 Area was devoted to laboratory and support functions. In discussions with site experts, only a few of the many facilities were discussed.

Construction on the 306 building was completed in 1956 and it began operation in May 1957 as a pilot plant for reactor fuel production. The 306 building had a complete fuel element canning line with the exception of autoclaving. The co-extrusion process was developed in the 306-E building in the early 1960s in support of N-Reactor fuel needs. This facility was shut down in the early 1980s. It is expected that this is now 306E and 306W, both of which are being used for engineering fabrication and development by PNNL and COGEMA. At one time a few years ago, there were neutron sources kept in one of these buildings that were used for testing the moisture content of soils.

The N-Reactor fuel was produced in the 333 Building starting in 1961. The N-Reactor fuel elements used a tube-in-tube design consisting of slightly enriched uranium inner and outer cores and thin zircaloy-2 cladding. The production process included: cleaning of the uranium billets and Zirconium-2 cladding shells; billet assembly and preheating; extrusion; fuel element shaping and cleaning; and welding of the end cap. In 1965 –1967, outer driver fuel elements were produced in support of tritium production at N Reactor. The Li/Al inner target element was manufactured in the 3722 Area Shop. The 333 Building was placed on standby in 1988 when the manufacture of N-Reactor fuels ceased.

The High Temperature Lattice Test Reactor (HTLTR) was a 2 MW test reactor built in the 318 Building in 1967 with operation commencing in 1968. The reactor consisted of a graphite cube located in a large shielded room and was operated at elevated temperatures up to 1000^0 C. The mission was to advance reactor physics technology. It was shut down in 1971 after 3 years of operation, during which time they operated with six different cores (i.e., three - ^{233}U, one - ^{235}U, one - ^{238}U, and one - ^{239}Pu). Since the reactor was heavily shielded and operated remotely, external exposure was minimal. There were no contamination incidents so internal exposure was negligible. In the 1980s, this building was converted for use as the Hanford radiological calibrations laboratory to primarily calibrate dosimeters and portable radiation protection instrumentation. Numerous measurements of sources in this building have been done to ensure traceability with national standards laboratory. This is basically a low-dose facility to the workers within the building.

The 3706 Building was involved in handling plutonium. Analytical testing to improve fission product decontamination was conducted in this building. There was considerable work done with ruthenium, which was the most troublesome of the fission products in separations. In order to deal with the high dose rates, chemists had to build lead caves in the hoods and used extended tools. In the early days this building had stainless steel hoods that exhausted through a single Chemical Warfare System filter to a pipe through the roof. Figenshaw hoods were used where

gloveboxes would be used today. As they worked actual aliquots from the 200 Area, they began to have issues with skyshine and in some cases airborne contamination. Most hands-on work was done by technicians (usually women). In order to access the area, one had to sign the building sign-in logs (Moore 1992).

Radiochemists originally housed in the rickety 3706 Building moved to the High-Level Radiochemistry Facility (Building 325) in the mid-1950s. Operations in Building 325 involved production of purified isotopes including fission products and transuranium elements. Campaigns were run to separate ^{90}Sr, ^{137}Cs, ^{147}Pm, technicium, and ^{238}Pu. Radiochemists worked on a method to modify the PUREX process for the retrieval of ^{237}Np. Another mission of this laboratory was to address radioactive waste issues.

Special Campaigns

A number of the single pass reactors were used to irradiate lithium targets to produce tritium. The P-10 project, which involved tritium processing, ran out of the 108B Building. There were a few dozen individuals involved in the P-10 program. This operation had a significant potential for tritium exposure.

N-Reactor participated in a tritium production campaign in 1969. The test was completed to see if N-Reactor could be used for tritium production. There were campaigns involving the production of other isotopes; however, these were classified and the routine reactor operators were not allowed to be involved. The time period during which these tests were performed is sometimes referred to as the "Thermonuclear Period." There were often ruptures of the test fuel during these analyses that caused later issues with reactor operations.

Reactor operations involved not only the production of plutonium for use in atomic weapons, but also test irradiation that was performed by the Irradiation Testing group. This group was composed of approximately a dozen workers. They performed testing of a large variety of materials to support work at Oak Ridge, Argonne National Laboratory, Pacific National Laboratory, Lawrence Livermore National Laboratory, the military, and various other organizations. Radionuclide irradiation included ^{60}Co, ^{14}C, tantalum, yttrium, and numerous other radionuclides. The irradiation periods for these materials varied from minutes to years. In some cases, scientists came to Hanford to perform their own testing. For example, scientists from Los Alamos National Laboratory came to Hanford to perform a neutrino experiment. In this sense, the reactors served as a neutrino source. The irradiated material was shipped from the reactors to the 300 Area or to other DOE facilities for further testing.

Chemists in the 300 Area laboratories were involved in a number of special campaigns to produce various radionuclides for AEC use. The Savannah River Plant had construction and start-up issues initially, so neptunium was irradiated in one of the Hanford reactors and then sent to the 300 Area for processing. The resulting material was ^{238}Pu. The Fission Product Development group with support from the Chemical Processing Department purified and produced in significant quantities the following radionuclides.

- Technicium
- Rhodium
- Palladium

- Americium
- Curium
- ^{233}U

Many of these purified products were abstracted from waste material from other operations on the Hanford site. In addition to plutonium and uranium, neptunium was sometimes recovered from the separations process in a purified form (Moore 1992).

Radiological Hazards

With the diverse mission of the Hanford Site, radiological hazards varied by area and operations. The site is divided into the 100, 200, 300, 400, 600, 700 and 1100 Areas. The primary radiological hazard in the 100 Areas was from external exposure. In the 200 Area, the primary radiological hazard was from internal exposure. The 700 and 1100 Areas were located in Richland and generally did not handle radioactive material.

Some of the most hazardous exposure conditions on the Hanford Site include K-Basins, Zone 1 of the Reactor, the Plutonium Finishing Plant, Waste Encapsulation & Storage Facility (WESF), and the Solid Waste Burial Ground. K-basins contain decaying fuel and sludge. Zone 1 of the reactor during operation had high radiation areas. PFP was involved in the plutonium finishing process and handled large quantities of plutonium. The Solid Waste Burial grounds contain some high level waste and numerous radionuclides. Historically, the P-10 storage facilities around the reactors and the Semi Hot Works (Building 209) were also hazardous facilities.

Inappropriate shielding in areas resulted in irradiation to portions of the body. In these cases, the dosimeters were not representative of the exposure received. Inappropriate shielding also caused irradiation of personnel in other rooms. Skyshine was an issue at some facilities.

100 Area

The reactors had the highest potential for external exposure on the Hanford site as a result of high dose rates in areas of the reactor and when dealing with fuel. The fuel and other materials removed from the reactors were extremely radioactive. At times, the material would produce a blue glow in the basin. A five minute entry into the Ball room of the reactor resulted in approximately 100 mR. An operator could receive an exposure of 300 mR in less than one minute at certain locations in the reactor (e.g., 109 Pipe Gallery). The 100 Area pulled individuals from other areas of the site as their employees were reaching the radiological limits. In cases of high exposure jobs, Radiological Control used shielding and other engineering controls to reduce radiation exposure. The Radiation Control Technicians were not always required to provide continuous coverage in the reactor and thus did not do so.

There were parts of the reactor where neutron exposure was an issue (e.g., X-level, Rod Room). Neutron leakage was observed at the T-seams on the X-levels. Reactor workers were exposed to radiation beams from the reactor that may or may not have been effectively measured by their routine dosimeter. Neutron radiography was performed in the C Reactor. In order to do this, a 4-inch square hole was bored into the reactor. Appropriate neutron shielding was put in place such that the film could effectively measure the neutron exposure. The primary source of neutron exposure was from the Inner Rod Room of the reactor. To prevent neutron exposure,

individuals were not allowed access when the reactor was up and operating. Access into Zone 1 of the reactor was not allowed when the reactor was up and operating.

Entry into the axillary cell area of the 109 Building was allowed at N-reactor when the reactor was operating. This was a significant source of beta exposure to the workers. This area also contained streaming radiation sources that could result in a high difference between two adjacent workers' dosimeters. There was also a potential for partial body exposure to areas of the body that were not specifically monitored.

200 Area

The primary radionuclide of concern in the 200 Area was plutonium and its byproducts. There are trace amounts of neptunium and curium. Plutonium Finishing Plant was responsible for the plutonium finishing process. In addition to plutonium and plutonium byproducts, low energy photons and neutrons were also of concern in this facility. There was also a potential for neutron exposure at 233S and 271U due to involvement in plutonium production and storage of AmBe and PuBe neutron sources, respectively.

Liquid waste from separations and other operations were collected in tanks in the Tank Farms. Although the majority of the tanks are in the 200 Area, there are some in the 300 Area. The contents of the tanks were dependent on the facility that the particular tank serviced. Cesium and strontium were the primary radionuclides for many of the tanks. There are twelve tanks that have been designated specifically as transuranic tanks. Even in tanks containing primarily fission products, alpha contamination has been identified.

There have been a number of safety issues associated with the tanks. The tanks contain tritium that periodically off-gasses to the environment. Most recently, chemical releases from the tanks have become a concern.

300 Area

The 300 Area was involved in diverse activities. Some of the radiological hazards associated with the 300:

Table 1. Radionuclides of Concern from a Field Perspective in the 300 Area

Facility	Operation	Radiological Hazard
306	Nondestructive testing; Source type	Radiation Generating Devices (RGD), potential neutron
308	Manufactured fuel for FFTF	Pu, extremity exposure
309	Housed the PRTR, a heavy water reactor	Tritium
325	Radiochemistry laboratory including hot cells	Fission Products (FP), Pu, Uranium, ^{147}Pm
326	Diffraction and electron microscopes; tritium testing; non-destructive testing	Tritium, RGDs
327	Post irradiation testing of fuel in hot cells	Activation Products (AP), FP, Pu, Uranium, and others; external
340	Radioactive liquid and solid waste from all facilities; spent fuel development, fuel fabrication, R & D	Alpha and beta contamination of a variety of radionuclides

Table 1. Radionuclides of Concern from a Field Perspective in the 300 Area

Facility	Operation	Radiological Hazard
3745	Accelerator for calibration of instruments	External exposure
300 Area Burial Grounds	Waste Disposal	Plutonium, uranium (especially yellowcake)

Tritium has been identified as a potential internal hazard in various facilities such as the 309 and 326 Buildings. The 326 Building was involved with experimental work relating to tritium. Buildings 308 and 319 have or had positive neutron dose rates. Radiation Generating Devices used onsite include sources (^{60}Co, ^{192}Ir), x-ray diffraction units, backscan rays, and electron microscopes. A number of these are housed in the 300 Area.

There are numerous burial trenches in this area that can contain waste from all over the site. The radionuclides of concern in the 300 Area include uranium and plutonium. It is not uncommon to find contamination outside buildings in the 300 Area. There is an increase in radiation levels at depths in the soil as the amount of uranium encountered increases.

400 Area

During the operations of the Fast Flux Test Facility (FFTF), the radionuclides of concern were ^{22}Na, ^{24}Na, ^{137}Cs, ^{54}Mn, ^{60}Co, ^{237}Np, and ^{241}Am (in the waste stream). After shutdown of the reactor, ^{60}Co and ^{137}Cs became the radionuclides of concern. There are limited amounts of alpha contamination (plutonium) in the facility as a result of failed fuel and washing of this leaking fuel in the Sodium Removal System (i.e., primarily in the waste stream). There is also tritium associated with the primary and secondary Na systems. FFTF had occasional issues with neutron exposure during fuel handling operations. There was no detectable neutron exposure from the operation. Prior to the identification of uranium and plutonium in the wash area, only beta/gamma surveys were required. FFTF is currently shut down and there is not much potential for exposure.

600 Area

The 600 Area is located approximately 1 mile north of the 200 Area. 212-N-P-R Buildings were used as a location for cooling waste. There are also waste sites in this area. The Gable Mountain Pond and the S-Ponds contain low levels of radiation.

Off-Normal Occurrences and Incidents

Over the course of operation at Hanford, there have been a number of both major and minor incidents. Some of the incidents resulted in significant personnel exposure and property damage while others did not impact personnel at all. Incidents reported by site experts included:

- Continuous Air Monitor and Area Radiation Alarm Monitor alarms
- Fires in the Z-vaults
- Disposal of portions of fuel rods in tanks (e.g., Tanks C-106, 101U, and S-102)
- Disposal of boron balls in Tank S-106
- An explosion in the PFP resulting in the largest uptake of ^{241}Am in the complex

- A criticality accident in the 209E Building
- Overexposure (i.e., extremity) from handling of ^{147}Pm in 325 Building.

There have also been numerous spills, contamination spreads, and leaks from contained systems. Site experts present onsite during the explosion at PFP in 1976 (referred to as the McClusky incident), indicated that the site instituted a "Take Cover" alarm. All individuals onsite were held at Hanford into the evening of that day. Each individual had to be surveyed prior to being released from the site.

Another incident involved the P-11 facility located close to the old Hanford High School. There was some sort of criticality testing occurring. There was a "goof up" during this project leading to high radioactivity. The employees involved in the project were reassigned to other jobs. During site assessment activities many years later, plutonium was identified as being present.

When incidents or occurrences happened, the operations group issued an Event Fact Sheet (EFS). The EFS summarized the investigation completed by the operations organization. These reports should be reviewed with caution as the events were often minimized in the reports. Other incident reports issued by Radiological Control are discussed in Attachment 4 of this report.

High-Risk Jobs

There are a number of jobs onsite that pose a higher radiation exposure hazard than others. Workers provided the following examples of high-risk jobs at various facilities onsite.

- Work with highly radioactive material in open-faced hoods.
- Chemical flush of the reactors.
- Handling irradiated Sm balls and other materials removed from reactors
- Charging and discharging the reactors
- Entry into high exposure rate areas of reactors (e.g., 16 foot level, Ball room, Pipe Gallery)
- Fuel Handling and/or movement
- Hot cell entries and removal of material from hot cells
- Filter change outs
- Removal of windows off glovebox
- Work with higher levels of plutonium contamination (Building 308, PFP, separations facilities)
- Hand mixing of plutonium and uranium resulting in extremity exposure
- Ion Exchange Column Removal
- Waste handling activities

- Tank Farms pit entries
- Remediation of unknown areas

Many workers in the reactor areas were put on work restriction for up to several months when they reached or exceeded the administrative dose limits for the site.

The Irradiation Testing group performed reactor surveillance. During reactor surveillance, operators had to look inside process tubes (i.e., top, bottom and side) and identify anomalies in the process tubes and graphite moderator. Following reactor shutdown, boroscopes were used to perform these types of inspection. When the boroscope and vertical manometer were removed from the reactor, the crew used a Kotex to remove the contamination from the instrument as they withdrew it from the reactor. Hand contamination from this operation was common.

Decontamination and Decommissioning

For many years, the solution to disposing of highly radioactive materials at Hanford was to bury it. As a result, Hanford has been involved in extensive Decontamination and Decommissioning (D&D) activities. D&D presents different radiological challenges than those typically encountered during the years of operations. During the operations of a facility, the radionuclides of concern and exposure rates are known. Individuals involved in determining the hazards are also often readily available during D&D. During the process of decommissioning a facility or outside area, operations and radiological control often have to deal with unknowns, and there is a heavy reliance on historical documents. For example, one may locate unlabeled bottles or drums in burial grounds. While radionuclides with short half-lives often predominate during periods of operations, long-lived fission and activation products, transuranics and uranium become the predominant radionuclides during environmental restoration. There are situations where liquid radioactive material, usually in sealed containers, is found. These containers have occasionally leaked due to degradation causing spread of radioactive material.

As Hanford's mission changed from production of plutonium to environmental restoration, the site made use of deactivated facilities. For example, the pipe galleries of the separations facilities became storage areas for radioactive equipment and materials. Many of these materials were stored with lead shielding around them. Storage in deactivated facilities has been used in lieu of disposal of material.

Some decommissioning of facilities began in the 1960s with the reactor shutdowns. Minimal work was done on these facilities at the time such as removal of ventilation systems. Bechtel Hanford, Inc. became involved in the current D&D activities in approximately 1998 with the decommissioning of the 105-C Building. As a result of removal of ventilation systems, D&D crews no longer have the benefit of these engineering controls.

Each remediation site at Hanford has a unique characterization. In terms of remediation work, the radionuclides of concern from a field perspective in the 100 Area are ^{152}Eu and ^{60}Co. ^{137}Cs and ^{90}Sr are of a lesser concern except where encountering fuel or where fuel processing equipment and waste were buried. Inside the reactor building, americium, plutonium, ^{14}C and ^{63}Ni are of concern. The typical ratio of plutonium to americium in the reactor areas is 3:1 for the single pass reactors and 1:1 at N-reactor. The 108F Building in 100F Area housed animal

experiments where animals were injected with or required to inhale radioactive material. They were also externally exposed. The radionuclide constituents of this building are not completely known. In the 200 Area, cesium, strontium and plutonium are of concern. With respect to the 300 Area, uranium including uranyl nitrate, plutonium from fuel separations, and multiple chemicals are of concern. Chemicals rather than radiation are the primary concern in the 300 Area. The constituents are dependent on the facilities and the source of materials originally buried or processed. For example,

- Uranium and plutonium (18:1 ratio of Pu to Am) was found at the 618-2 site. The normal ratio is 3:1 Pu to Am. Lanthium fluoride, which was used to precipitate plutonium out of nitric solution, was also found. There are estimated, based upon field analysis, gram quantities of separated plutonium in this area.
- Construction debris with plutonium and uranium was found at the 618-3 site. Plutonium and uranium were present at 1-10 pCi/g.
- Barium chemical contamination, lead, debris, yellowcake, and uranium shavings in oil from a mill tailing process (depleted uranium oxide) were found at the 618-4 burial site. There were numerous good quality drums with 1,100 – 1,300 lbs of black oxide.
- Garbage, unknown ignitable waste, and low-level waste and soil were found at the 618-5 site.
- Drums of hexone contaminated with uranium and garbage were found at the 618-9 site. Hexone was used in fuel fabrication. This area is located on the west side of the main road.
- 618-8 contains construction debris and uranium but no plutonium.
- Beryllium was identified inside and outside the 313 and 314 Buildings.

Burial sites also exist in the 100 and 200 Areas. The 100 Area burial sites contained materials and equipment from the reactors. For example, 118-B-1, a remediation site by B-reactor, contains reactor garbage, irradiated wire, and expended fuel. Liquid tritium capsules have also been bound in BC Burial Grounds. Mercury was found in some burial grounds. Hanford staff suspect that some of the material found in the burial grounds were originally from other DOE facilities, but underwent processing in Hanford labs.

Uranium is found frequently in the 300 Area of Hanford. Yellowcake is commonly encountered in the soil at remediation sites. The uranium tends to bind with the soil preventing it from being an airborne hazard in many cases. It is easily detected due to the high-energy beta emitter of its ^{234}Pa daughter. There are numerous forms of uranium on the site; however, most of the uranium is found as UO_2. Other forms observed in remediation include U_3O_8 and uranyl nitrate. The gamma dose rate for a drum of concentrated uranium is about 3 mrem/hour. The open-window reading taken with an ion chamber has been observed as high as 250 mR/hour. Although it is not often encountered during D&D, recycled uranium is easily distinguished from other uranium by the detection of ^{99}Tc. Large concentrations of ^{230}Th and ^{232}Th are also present in the 300 Area sediment.

One site expert made the following comment with respect to D&D:

If you don't find a surprise in the burial grounds, that is the only surprise you get.

The use of engineering controls is difficult in a D&D setting as many facilities do not have existing ventilation and in some cases no support utilities at all. In addition, much of remediation requires the use of large equipment that cannot easily be contained. In general, for outside environments, the area is wet down and fixatives applied where necessary. This type of work relies more heavily on respirators to prevent internal uptake than engineering controls. At one time, decontamination of the facility or area was tried; however, this actually resulted in more exposure than the current method.

Environmental Monitoring

Hanford had an extensive environmental air-monitoring program. Initially the air samplers were placed based on a gridded system. The site replaced the grid system with a point source system. These were areas where the highest levels of release were expected. A lot of the initial air sampling locations in the grid system didn't make sense. Currently, each facility is required to have a Facility Effluent Monitoring plan. This requires the facility to evaluate operations and set up a monitoring program based on actual hazards.

There are a number of environmental issues at Hanford that involve release of radioactive material to the environment and subsequent contamination spread. Environmental release pathways include soil contamination, ground and surface water contamination, airborne releases, and spread of contamination by animals.
Hanford has had both planned and unplanned releases to the environment. In the early years some releases were deliberate. Unplanned releases mentioned by site experts included:

- Pipe ruptures and subsequent release to the environment.
- Carbon tetrachloride release in the 200W area from the vapor extraction process.
- Leakage of U1 and U2 cribs and subsequent contamination of U16 crib.
- Black smoke (suspected uranium) released from the stack of one of the reactors.

The Waste Information Data System contains a list of unplanned releases to the environment. Analysis of episodic and routine releases from facilities was handled by separate groups. The information provided in the annual environmental monitoring reports does not include information and data from episodic releases.

Cribs and trenches were used at facilities to filter or dispose of process waste and reactor coolant water. Prior to the construction and operation of the tanks, waste was dumped into what is referred to as swamps. Some facilities had overflow ponds. Cribs and trenches have led to significant exposure to occupational workers (e.g., BC cribs, Z-9 trench, Z-10 trench, Z-11 trench, Z-12 trench, N-cribs). The Z-9 trench raised criticality concerns as there was excessive plutonium in the trench.

There are a number of cribs and trenches surrounding the reactor areas. The reactor facilities pulled river water from the Columbia River. The water was run through a treatment plant where

it was cleaned but not demineralized. This water was fed into the reactor coolant system. After the water was circulated through the reactor it was fed into a holding basin and ultimately discharged back to the river. Cribs and trenches were equipped with piping to carry the water. The cribs were composed of gravel beds. The pipes were perforated to allow the water to flow into the crib and soil. Soil was an effective filter for some contaminants. Radionuclides like ^{90}Sr, ^{137}Cs, and ^{99}Tc were mobile within the soil matrix. There has been some evidence of colloidal transport of plutonium through the soil. Also, uranium has shown up in the ground water at places onsite. ^{90}Sr is frequently identified in the groundwater. ^{129}I was identified in groundwater under the U-plant. In some situations, even the closed piping systems leaked causing release of material to the environment. In addition to reactor coolant water, sludge was shipped to the cribs and burial sites on Hanford. The contents of "drag-off" boxes were also dumped into the cribs.

One of the methods contamination is spread is by one of the 90 species of animals that make the site their home. For example, when performing work on the 151/152 Diversion boxes, mice spread contamination that resulted in disposal of a vehicle. In 1998, work was being completed on the 152 Diversion Box at B-plant. A sugar-based fixative was applied to control contamination. Fruit flies found the fixative a good place to lay their eggs. As the eggs hatched, the contamination was spread outside the immediate area. Also, mud doppers collect contaminated soil from trenches or other areas to build their nests.

Hanford is located in a desert receiving little rainfall during the year. When there are leaks in systems such as transfer lines, animals and plants are attracted to the area as a source of water. Tumbleweeds and other vegetation absorb the contaminated liquid and become a radioactive hazard. Tumbleweeds then break off and spread contamination throughout the desert. These leaks can also result in skin contamination to workers.

Initially, there was not much done to control the spread of contamination by animals. In the late 1990s, the site implemented a formal program to deal with this type of contamination spread. For example, the site recently implemented engineering barriers around potential waterholes to prevent animals from accessing these areas. Since 1998, the contamination spread from wildlife and plants has decreased. With respect to environmental contamination, constituents are composed of a cocktail of mixed fission products (e.g., Cs, Sr, and Tc). When animals are trapped or found dead, they are surveyed to determine if they are radioactive.

There continue to be a number of environmental issues at the Hanford site including:

- Insufficient characterization of environmental releases across the Hanford Site
- General area contamination from historical PUREX releases
- Use of surrogate data from a similar tank to characterize other tanks
- Insufficient characterization of the VADOSE layer in the 300 Area
- Issues with transient contamination
- Incomplete information in the Solid Waste Burial Ground Environmental Impact Statement.

Chemical

There are a number of chemicals of concern at Hanford including chlorine, hydrazine, hexone and strong acids and bases. Although these chemicals were present, no safety precautions were taken in many cases. There was often direct exposure to the skin. This is also true for some chemicals later determined to be carcinogenic. Chemicals such as barium, lead and beryllium have been identified during remediation of the 300 Area.

Concerns

Many site experts were concerned that Jack Fix and Don Bihl were principal authors of the external and internal technical basis documents, respectively. Jack Fix and Don Bihl have been actively involved in the external and internal dosimetry programs at Hanford for many years. Although they may know the programs very well, they may not be objective in relation to all aspects of the program. This causes a potential conflict of interest.

References

Moore, R.L. (1992), *As I Knew Him: Recollections of a Shy Timid Soul Who Led a Quiet Life Spanning Some Interesting Times*. Published by Robert Lee Moore, Richland, WA, pp. 55-142.

ATTACHMENT 6: CONSISTENCY BETWEEN HANFORD AND SRS SITE PROFILES

Table A.6.1 Occupational Medical Exposure Default Assumption Comparison for Hanford and the Savannah River Site

Description of Assumption	Hanford	SRS
Frequency of chest x-rays (Default)	Posterior-Anterior View: Before 1946 – 1/1982: Pre-employment, annual, and termination 1/1982-1/1983: Pre-employment, annual, and termination for over 50 years; Biennially for 40-49 years; Every third year for 39 years or younger. 1/1983-3/1990: Biennially for over 50 years; Every third year for 40-49; and Every five years for 39 years and younger. 3/1990 – present: Every five years Lateral chest x-rays also given periodically prior to 4/1997.	One annual x-ray procedure for each year or partial year.
Organ Dose Conversion Factors	Obtained from ICRP 34 (1982a)	Obtained from ICRP 34 (1982a)
Substitute dose conversion factors for thyroid, eye/brain, ovaries and analogues, testes, and uterus	Use DCFs for lung for all other organs in thoracic cavity; for organs in abdomen, use DCFs for the ovary (Scalsky 2003, page 10) Use substitute dose conversion values outlined in Table 5.1-1, ORAUT-OTIB-0006.	Use substitute dose conversion values outlined in Table 5.1-1, ORAUT-OTIB-0006 (also Table 2.6, SRS TBD) prior to 1970 (Scalsky 2004bb, page 50).
IREP Radiation Rate	Acute	Acute
IREP Radiation Type	Photons, 30-250 keV	Photons, 30-250 keV
IREP Dose Distribution Type	Constant	Constant
Total uncertainty	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	2.5	2.5
Chest Thickness	Standard chest thickness is 22-24 cm. For chest thicknesses of 25-27 cm, increase the dose by a factor of 1.5. For chest thickness of >27 cm, increase the dose by a factor of 2. (Scalsky 2003, pp. 8 and 17).	PA View: 26 cm Lateral View: 34 cm No corrections for varying chest thicknesses are mentioned in the TBD.
Analogue organ for Thymus	Lung	Lung
Analogue organ for Esophagus	Lung	Lung
Analogue organ for Stomach	Lung	Lung
Analogue organ for Bone Surface	Lung	Lung
Analogue organ for Liver, gall bladder, spleen	Ovary	Lung

Table A.6.1 Occupational Medical Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Analogue organ for Remainder Organs	Ovary	Lung
Analogue organ for Urinary/bladder and colon/rectum	Ovary	Ovary
Analogue organ for Eye/brain	Thyroid	Thyroid
Skin dose	Skin dose was determined by multiplying the ESE by the backscatter factors of 1.35 and 1.4 for HVLs of 2.5 and 3.5 mm Al, respectively (Scalsky 2003, page 158).	Skin dose was determined by multiplying the ESE by the backscatter factors of 1.35 and 1.4 for HVLs of 2.5 and 3.5 mm Al, respectively (Scalsky 2004bb, page 158).
Posterior-Anterior View X-ray Techniques[1,2,3]		
<1946	kVp: Unknown mAs: Unknown SSD: 72" (183 cm) ESE: 120 mR	Not Applicable
2/1946 – 12/ 1950	kVp: 80 mAs: 25 SSD: 72" (183 cm) ESE:79 mR	Not Applicable unless pre-employment is indicted.
1/1951 - 1/1/1957	kVp: 80 mAs: 10 SSD: 72" (183 cm) ESE: 79 mR	kVp: 80 mAs: 30 SSD: 152 cm No Collimation ESE: 108 mR
1/1/57 – 4/1959	kVp: 80 mAs: 10 SSD: 72" (183 cm) ESE: 79 mR	kVp: 80 mAs: 30 SSD: 152 cm No Collimation ESE: 108 mR
4/1959 – 12/1970	kVp: 80 mAs: 10 SSD: 72" (183 cm) ESE: 40 mR	kVp: 80 mAs: 30 SSD: 152 cm No Collimation (prior to 1970) ESE: 108 mR
1/1971 – 1/1983	kVp: 80 mAs: 10 SSD: 72" (183 cm) ESE: 40 mR	kVp: 110-120 mAs: 10 SSD: 152 cm ESE: 44 mR
1/1983 – 7/1985	kVp: 100 mAs: 10 SSD: 72" (183 cm) ESE: 35 mR	kVp: 110-120 mAs: 10 SSD: 152 cm ESE: 44 mR
8/1985 – 3/1990	kVp: 100 mAs: 10 SSD: 72" (183 cm) ESE: 35 mR	kVp: 120 mAs: 7.5 SSD: 152 cm ESE: 33 mR

Table A.6.1 Occupational Medical Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Posterior-Anterior View X-ray Techniques[1,2,3]		
3/1990 – 4/1997	kVp: 110 mAs: 6.7 SSD: 72 " (183 cm) ESE: 21 mR	kVp: 120 mAs: 7.5 SSD: 152 cm ESE: 33 mR
4/1997 – 2/1998	kVp: 110 mAs: 10 SSD: 183 cm ESE: 17 mR	kVp: 120 mAs: 7.5 SSD: 152 cm ESE: 33 mR
2/1998 – 5/1999	kVp: 110 mAs: 5 SSD: 183 cm ESE: 11 mR	kVp: 120 mAs: 7.5 SSD: 152 cm ESE: 33 mR
5/1999 – present	kVp: 110 mAs: 5 SSD: 183 cm ESE: 11 mR	kVp: 120 mAs: 7.5 SSD: 152 cm ESE: 33 mR
Photofluorography		
Technique Factors	kVp: 80 to 100 kVp mAs: not specified SID: 102 cm ESE: 1.53 R Applies 1945 to 1962	kVp: 100 mAs: 60 SID: 102 cm ESE: 1.5 R Applies from 1951-1957

1 Refer to Scalsky, 2004, pp. 41-47 for SRS x-ray technique discussion.
2 Refer to Scalsky 2003, page 18, for Hanford x-ray technique summary.
3 Available data indicate the x-ray beams used at Hanford were collimated. (Scalsky 2003, page 8).
4 N/A = not applicable; PA = posterior-anterior; LAT = lateral; kVp = kilovolt potential; mAs = milliampere-second; SSD = source-to-skin distance; ESE = entrance skin exposure.

Table A.6.2 External Exposure Default Assumption Comparison for Hanford and the Savannah River Site

Description of Assumption	Hanford	SRS
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, (Fix 2004, page 75).	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 2004bb, page 111).
Missed Photon Dose Methodology	Divide the MDL by 2, and multiply by the number of zeros and not monitored periods (Fix 2004, page 75). Table 6E.6 (Fix 2004), provides potential maximum photon dose by year.	(1) For a claimant-favorable maximum potential missed dose, use the limit of detection (LOD) multiplied by the number of zero doses (Scalsky 2004bb, pp. 111 and 238) (2) Divide the limit of detection (LOD)/2, and multiply by the number of zeros and not monitored periods; (Scalsky 2004bb, page 242), or (3) Missed doses are added to measured doses and treated as a constant.
IREP Dose Distribution Type for missed photon dose	Lognormal distribution with a geometric standard deviation of 1.52.[1] The assessment at Hanford was 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, page 27).	(1) When using the Limit of Detection (LOD)/2 methodology, a lognormal distribution with a geometric standard deviation of 1.52 in Parameter 2 of the IREP input is used (Scalsky 2004bb, page 116). (2) When simply adding the missed and measured dose, a constant is used.
Missed Neutron Dose Application	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 (Fix 2004, page 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 (Fix 2004, page 73).	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).

Table A.6.2 External Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Missed Neutron Dose Methodology	A neutron-to-photon ratio is applied to missed and recorded photon dose for unmonitored workers 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).	A neutron-to-photon ratio is applied to missed and recorded photon dose for unmonitored 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 2004bb, pp. 240–241). After 1970, the assignment of missed dose is based on the limit of detection provided in Table E-10 (Scalsky 2004bb, pp. 241–242). It appears that an ICRP 60 correction factor is applied to missed dose; however, this is unclear in the TBD (Scalsky 2004bb, page 110).
IREP Dose Distribution Type for missed neutron dose	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	Acute for beta and photon Chronic for neutron (Fix 2004, pp. 8, 59, and 69, respectively)	Acute for beta and photon Chronic for neutron (Scalsky 2004bb, pp. 87 and 235, respectively).
IREP Radiation Type (default)	Photon, 30-250 keV Electron, > 15 keV Neutron, 0.1-2 MeV (Fix 2004, page 29.)	Photon, 30-250 keV Electron, > 15 keV, Neutron, 0.1-2 MeV (Scalsky 2004bb, pp. 49, 236, and 237, respectively).
Organ dose conversion factor	The dose conversion factors for each organ, radiation type, and energy range from OCAS-IG-001 are used. If the exposure geometry cannot be determined, default values are found in Table 6E-9 (Fix 2004, page 77). No separate value is provided for the maximizing approach.	For the maximizing approach, a value of 1 is used (Scalsky 2004bb, page 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 2002a) 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 2004bb, page 242).

Table A.6.2 External Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Exposure geometry	Default exposure: Likely non-compensable workers - 100% AP Compensable worker – 50% AP, 50% ROT Compensable supervisor – 50% AP, 50% ISO. (Fix 2004, page 77).	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 2004bb, page 242).
Photon Adjustment Factors (Recorded Dose)	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, page 73).	Multiply by 1.119 for years prior to 1987. Multiply by 1.039 for 1987. No adjustment is needed post-1987, (Scalsky 2004bb, page 238). Note: Taylor et al. (1995) indicate 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.
IREP Dose Distribution Type for recorded photon dose	Constant.[1]	Constant. The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included.[1]
Recorded Neutron Dose Adjustment Factor (Prior to 1971 – SRS; Prior to 1972 Hanford)	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, page 48).	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 2004bb, page 238).
Recorded Neutron Dose Adjustment Factor (7/1978-12/1983)	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 correction factor (Fix 2004, page 74).	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 2004bb, pp. 235–238).

Table A.6.2 External Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Recorded Neutron Dose Adjustment Factor (1/1972-6/1978, 1/1984 – present)	Divide the recorded neutron dose into the facility specific neutron energy bins, and multiply by the ICRP 60 correction factor (Fix 2004, page 74).	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 2004bb, pp. 235–238).
IREP Dose Distribution Type for recorded neutron dose	Constant[1]	Constant: The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included.[1]
Shallow Dose Adjustment Factors	Shallow dose adjustments factors are not addressed in the TBD.	Shallow dose adjustments factors are not addressed in the TBD or SRS TIBs.
Low-energy photons (< 30 keV)	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, page 26). For 200 Area workers prior to 1957, 20% of the open window dose is added to the penetrating dose (page 14).	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)
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 2004bb, page 97).
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 (Fix 2004, page 29).	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 2004bb, page 98).

[1] These parameters were obtained from review of several dose reconstruction IREP input sheets.

Table A.6.3 Internal Exposure Default Assumption Comparison for Hanford and the Savannah River Site

Description of Assumption	Hanford	SRS
Particles Size (default)	5 micron (Bihl 2004, page D-10)	5 micron (Scalsky 2004bb, Section 4.0, Attachment D)
Intake Type (default)	Chronic (Bihl 2004, pp. 7-9)	Chronic (Scalsky 2004bb, Section 4.0, Attachment D).
Default Excretion Volume	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, page 27).	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 2004bb, page 70)
Solubility Class	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, page 24).	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 2004bb, page 85).
Intake Date for Hypothetical Intake (excluding tritium)	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, page 8).	Acute inhalation on January 1 in the first year of employment (Scalsky 2004bb, page 85; Bracket 2003, page 3).
Tritium Missed Dose Application	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).	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 2004bb, page 84; Duncan 2003, pp. 6 and 12)

Table A.6.3 Internal Exposure Default Assumption Comparison for Hanford, SRS (continued)

Description of Assumption	Hanford	SRS
Basis for Tritium Missed Dose	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 as 1 uCi/L. Later in 1982 the MDA changed to 10 dpm/ml and in 1991 to 20 dpm/ml, (Bihl 2004, page 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).	Dose calculated based on the tritium reporting level for a particular time period (Scalsky 2004bb, page 67; Duncan 2003, page 6).
Hypothetical Intake Application	Applied to individuals who wore a dosimeter but did not have any bioassay (Bihl 2004, page 48).	Applied to claims with non-metabolic and digestive tract cancers (Scalsky 2004bb, page 85; Bracket 2003, page 2).
Basis for missed internal dose from radionuclides other than tritium	(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose, (Bihl 2004, page 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, page 49). (3) From 1989 through the present, a daily exposure at 5% of the limiting air concentration for 4 hrs per week was assumed, (Bihl 2004, page 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, page 47).	(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose (Scalsky 2004bb, page 84; Bracket 2003, page 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 dose (Scalsky 2004bb, page 84; Bracket 2003, page 8). (3) Highest five intakes for various nuclides are applied to those individuals with non-metabolic or digestive system cancers (Bracket 2003, page 2).

Table A.6.3 Internal Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Radionuclides included in the Hypothetical Intake	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 in 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 some facilities and periods of time it is specified (Bihl 2004, pp. 51–52).	$^{241}Am/^{241}Pu$ (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, page 9)
Default Activity Ratios Pu Mixture	Not specified in the TBD.	Ten-year old 12% plutonium mix (Scalsky 2004bb, page 66).
Activity Fractions for other Mixtures	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, page 10, Attachment D).	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.
Radionuclides of Concern for Monitored Workers	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, page 13).	Radionuclides of concern were based on the in vivo and in vitro bioassay data of the individual (Scalsky 2004bb, 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.

Table A.6.3 Internal Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Tritium Dose for Monitored Workers	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 (page 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 (Bihl 2004, pp. 12 and 22)	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, page 6).
Internal Dose for radionuclides other than tritium	Based on 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, page 47).	Based on either actual bioassay values or detection levels for bioassay techniques. For non-metabolic cancers, the maximizing approach is used (Scalsky 2003, page 85).
Basis for pre-bioassay program doses	Air concentration tolerance or limits (Bihl 2004, page 7).	Not included in the TBD.
Ingestion	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, page 8).	Not included in the TBD.
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, page 9).
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, page 9).
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, page 9).

Table A.6.3 Internal Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
IREP Radiation Types for Hypothetical Intake	Alpha[1] Beta: >15 keV[1] Photon: > 250 keV[1] Tritium: < 15 keV[1]	Alpha Beta: >15 keV Tritium: < 15 keV (Bracket 2003, pp. 8 and 12)
IREP Dose Distribution Type	Constant[1]	Constant (Brackett 2003, page 12)
Internal Dose Uncertainty	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. Actually report errors can be used if available (Bihl 2004, page 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, page 7).	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.
Other Comments	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, page 32)	None.

[1] These parameters were obtained from review of several Hanford dose reconstruction IREP input sheets.

Table A.6.4 Environmental Exposure Default Assumption Comparison for Hanford and the Savannah River Site

Description of Assumption	Hanford	SRS
Application	Environmental doses are assigned to personnel with no bioassay and no evidence of having worn a dosimeter at the Hanford Site (Bihl 2004, page 48).	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 2004bb, page 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 worker's badge (Scalsky 2004bb, page 62).
Sources of Environmental Releases 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, page 18).	The TBD heavily references the Cummins (1991) and CDC (2001) documents, and does 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.
Source Term Basis	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).	*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.
Methodology	Puff advection (RATCHET) model (Savignac 2003, page 14)	Gaussian model (Scalsky 2004bb, Section 3.1.1)
Type of Releases	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.	The TBD heavily references the Cummins (1991) and CDC (2001) documents, and does not include many of the base assumptions from those reports in the TBD.

Table A.6.4 Environmental Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

Description of Assumption	Hanford	SRS
Ventilation Rate (m3/year)	2,400 (default); Based on 1.2 m^3/hour ± 0.4 m^3/hour (Savignac 2003, page 16).	2,400 (default); Adjustments can be made for light and heavy work (Scalsky 2004bb, page 162).
Exposure Time (hours/week)	40 (Savignac 2003, page 24.)	40 with a 1.25 conversion factor to increase the exposure time to 50 hours per week (Scalsky 2004bb, page 61).
Mobile Workforce	Information not included in the TBD.	Assign the maximum dose listed for any area onsite.
Facility Specific Workforce	Information not included in the TBD.	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 2004bb, page 61).
Radionuclides Considered for External Dose	^{41}Ar,^{131}I, ^{106}Ru (Savignac 2003, pp. 19 and 23).	^{41}Ar, (Scalsky 2004bb, page 60).
Radionuclides Considered for Submersion Dose	^{41}Ar (Savignac 2003, page 17).	^{41}Ar, (Scalsky 2004bb, page 59).
Submersion DCF	Federal Guidance Report No. 13, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides*, 1999.	Assumed values from the Federal Guidance Report 12 (EPA 1993). (Scalsky 2004bb, page 60).
Radionuclides Considered for Internal Dose.	^{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, page 8).	^{3}H,^{131}I, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{234}U, ^{235}U, and ^{238}U (Scalsky 2004b, page 51.)
Soil	Not included in the TBD.	Density = 1,600 kg/m3 Surface Factor = 0.08 Resuspension Factor =1E-9/m (Scalsky 2004b, page 59).
Liquid Effluents	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 2004b, page 61).
IREP Rate	Chronic[1]	Chronic (Scalsky 2004b, page 61).
IREP Radiation Type	Photon, 30-250 keV[1]	Photon, 30-250 keV ^{41}Ar , 100% photon, > 250 keV (Scalsky 2004b, pp. 60 and 61).

Table A.6.4 Environmental Exposure Default Assumption Comparison for Hanford and the Savannah River Site (continued)

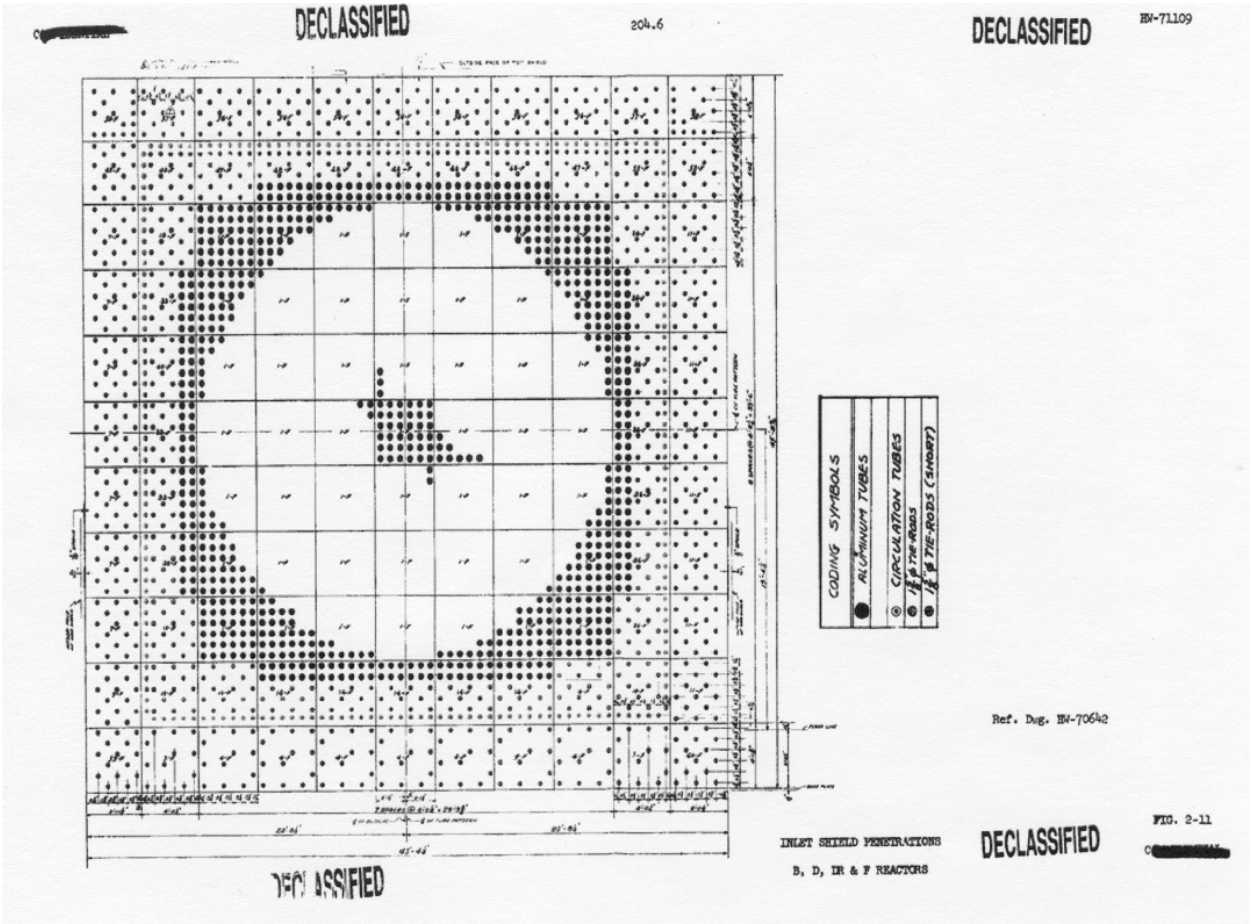
Description of Assumption	Hanford	SRS
IREP Dose Distribution Type	Constant. Doses and intake quantities provided with a geometric mean and standard deviation. There is no direction on how these values should be entered into IREP.	Constant. Doses and intake quantities provided with a 50th-percentile and a geometric standard deviation. A 95th percentile for the source term is estimated as 25% greater than the 50th percentile (Scalsky 2004b, page 60).
Special Considerations for Uranium and Plutonium	Not applicable.	The isotope yielding the maximum organ dose was assumed at 100% rather than applying a mixture (Scalsky 2004b, page 59).
Other	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).	1955 values are assigned to 1952, 1953, and 1954 (Scalsky 2004b, page 54).

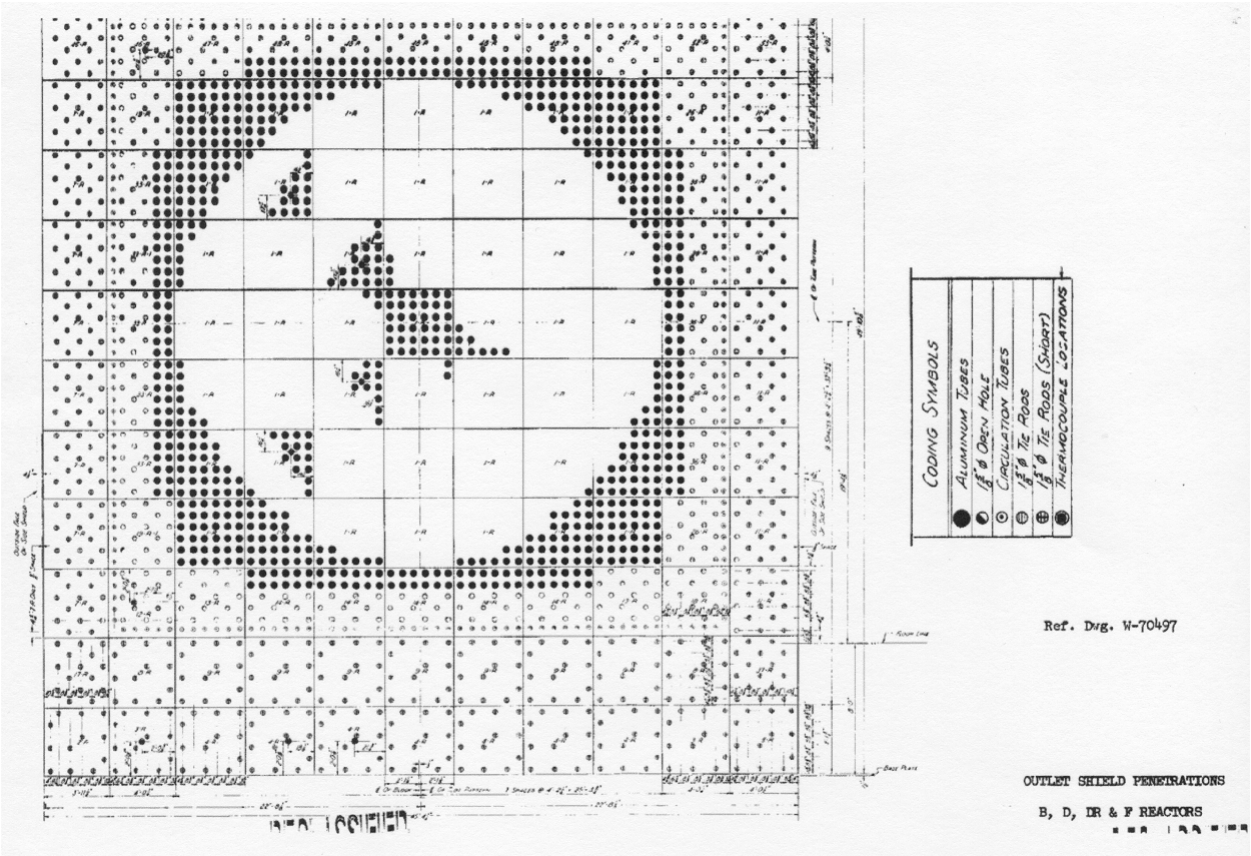
[1] These parameters were obtained from review of several Hanford dose reconstruction IREP input sheets.

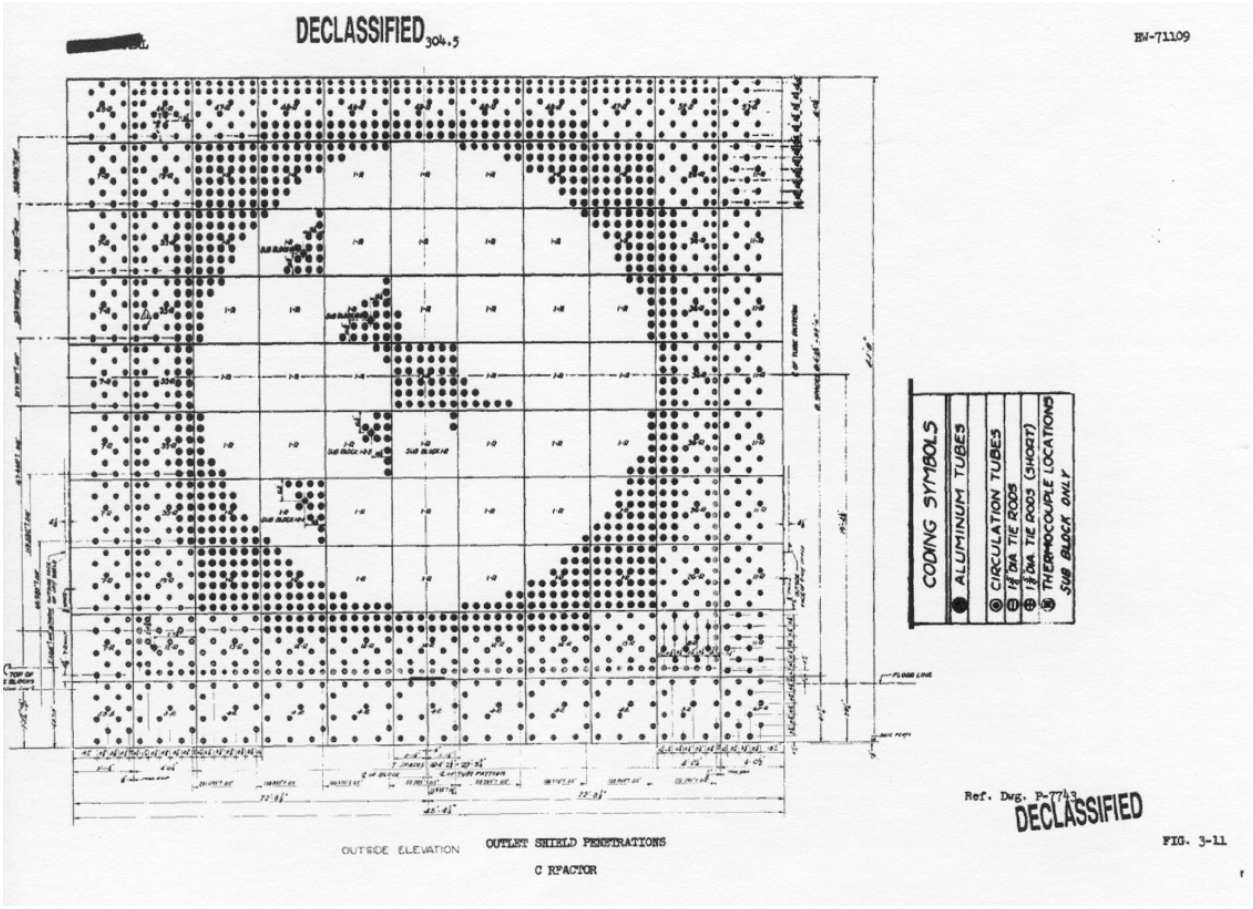
ATTACHMENT 7: ENGINEERING DRAWINGS ON HANFORD REACTOR SHIELDING DESIGN AND SIGNIFICANT HANFORD REACTOR REPAIRS 1956-1967

Engineering Drawings on Reactor Shielding Design

The following contains engineering drawings that provide some details regarding shielding design and the extent of penetrations in the bio-shield to allow for "hands on" worker operations and maintenance. As reactor power and heat levels were increased beyond original design capacities, problems such as graphite distortion, structural stress of the reflectors, and deterioration of biological shielding material led to potentially excessive exposure to neutrons and photons. See Section 5.6 for more discussion on these problems.




CODING SYMBOLS
ALUMINUM TUBES
1 1/8" Ø DRIV HOLE
CIRCULATION TUBES
1 1/8" Ø TIE RODS
1 1/8" Ø TIE RODS (SHORT)
THERMOCOUPLE LOCATIONS
Ref. Drg. W-70497
OUTLET SHIELD PENETRATIONS
B, D, DR & F REACTORS



Significant Hanford Reactor Repairs 1956-1967

Leakage of photons and neutrons from fuel tube penetrations and other inlets were chronic. These problems in turn, required an increase in workers exposed to radiations resulting from operations, maintenance and repair, and experiments to address ongoing problems. A partial chronology of major reactor upgrades and repairs is provided below.

DUN-3232

SIGNIFICANT REACTOR MAINTENANCE WORK PERFORMED ON
"MAINTENANCE JOB AUTHORIZATIONS"
(1956 to July, 1967)

MJA No.	Title
2	Rehabilitate Pile Gas Moisture Monitoring System - 105-B, C, D, DR, F and H
3	Replace Thermohms - 105-KE and KW
7	Replacement of Individual Tube Temperature Thermocouples - 105-D and DR
8	Ball 3-X Electrical Modifications - 105-B, D, DR, F and H
9	Restoration of Vertical Safety Rods - 105-DR and H
12	Modification of Process Water System for I&E Slugs - 105-D
15	Improved Gas Seals for Front and Rear Bellows - 105-C
18	K Graphite Stack Restoration
20	Replace C Reactor Tube Outlet Thermocouples with Improved Type
21	Pressure Monitor Repairs and Modification - 105-C
24	Rear Face Outlet Connectors - 105-F
25	Pressure Monitor Repairs and Modification - 105-KE and KW
29	Pressure Monitor Repairs and Modification - 105-H
32	Modification for Bumper Fuel Elements - 105-D and H
33	Thermocouple Replacement - 105-D
35	Replacement of Thermal Shield Thermocouples and Leads - 105-KE and KW
37	Conversion of Front Face Nozzles and Modification of Charging Machines at B, DR, F, KE and KW Reactors for Use of Bumper Fuel
43	Replacement of Rear Face Thermocouples, Terminal Board, Extension Wire and Stepping Switches - Temperature Monitoring System - B Reactor
47	Pressure Monitor Modification and Repair - 105-B, D, DR and F

Significant Reactor Maintenance Work Performed on "Maintenance Job Authorizations" - contd.

MJA No.	Title
48	Install Ten Knuckle Type VSRs in C Reactor
49	Install New Rear Face Connectors and Connector Elbows, Modify Thermocouple Well Connector; Remove Original Thermocouples from Crossheader and Install 1/4" Pipe Plugs - C Reactor
51	Replace Rear Face Temperature Monitoring Wire and Wells, Relocate Terminal Boards, Install New Lead Wire and Zone Temperature Terminal Boards - H Reactor
53	Installation of Rear Crossheader Expansion Joints - 105-B, D and H
54	Thermocouple Temperature Monitoring Stepping Switch Replacement and Associated System Repairs - C, D, DR, F and H Reactors (Work Authorized at B Reactor on MJA-43)
55	Zone Temperature Monitor Terminal Board Relocation and Controller Revision - B, C and D Reactors
59	Resistance Temperature Detector Extension Lead Replacement and Relocation 105-KE and KW Process Tube Temperature Monitoring System
60	Resistance Temperature Detector and Lead Wire Replacement - 105-KE and KW Tube Temperature Monitoring System
61	Steam Boiler Repair, 165-KE Building
62	Repair Liner Cracks 105-C for Downcomer and Near Downcomer and Effluent Expansion Boxes
64	Repair to 107-C Tanks and Diversion Box
65	Downcomer Approach Box Strengthening 105-KE and 105-KW
66	Steam Boiler Repair - 165-KW
67	Steam Boiler Repair - 165-KE
68	Rear Face Piping Supports and Other Modifications - 105-B and D
69	107-KE and KW Retention Basin Repair

MJA No.	Title
70	107-B Retention Basin Repair
71	K Outfall Drainage Trenches
73	Repair of Effluent Pipe Joints and Retention Basin Valves for the 105-KE and KW Plants

ATTACHMENT 8: AN EVALUATION OF THE BETA EMISSIONS FROM RU-106

PENETRATION OF β RAYS FROM ^{103}Ru and ^{106}Ru

^{103}Ru, which has a 39.26 d half-life, decays by β emission to ^{103}Rh. As shown in the decay scheme illustrated in Figure 1, 92% of the decays are to a short-lived excited state of ^{103}Rh. This state, in turn, decays by γ emission to the ^{103}Rh ground state, which is stable.

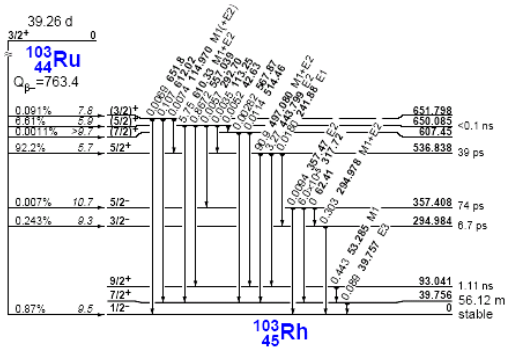


Figure 1. ^{103}Ru to ^{103}Rh Decay Series

The maximum energy of the β particles is 763.3 ! 536.8 = 226.5 keV. About 98% of the β particles have energies below 300 keV. The decay of the excited state of ^{103}Rh is accompanied by γ emission, with a primary energy of 497 keV.

^{106}Ru, which has a 39.26 d half-life, decays by β emission to ^{106}Rh. As shown in the decay scheme illustrated in Figure 2, 100% of the decays are to the ground state of ^{106}Rh, which has a half-life of ~30 s. The maximum β energy is from this transition is 39.4 keV, with 57% of the β particles having energies > 10 keV.

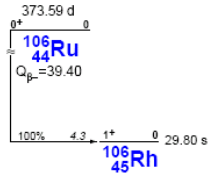


Figure 2. ^{106}Ru to ^{106}Rh Decay Series

As shown in Figure 3, ^{106}Rh decays by β emission to ^{106}Pd. About 79% of the transitions are to the ground state of ^{106}Pd, which is stable. The maximum energy of this transition is 3.54 MeV, with 45% of the β energies falling in the range of 1.3 ! 2.5 MeV.

The range and stopping power of electrons in various media is illustrated in Figure 4. Water is a suitable surrogate for clothing, which is primarily composed of organic fibers with a similar average atomic number. As shown, 100 keV electrons, a typical energy for β particles emitted during the decay of ^{103}Ru, have a range of about 0.015 g/cm^2. Typical cotton fabrics have densities on the order of 20 – 40 mg/cm^2 and would thus shield the skin from most of these radiations, especially since the clothing might consist of more than one layer of fabric. However, as stated earlier, typical β particles emitted during the decay of ^{106}Rh have energies 1.3 ! 2.5

MeV. These radiations have ranges of 0.6 – 1.2 g/cm^2, which is equivalent to 15 to 30 layers of a heavy fabric such as denim. Thus, the β radiation from ^{106}Rh would penetrate most garments. It is therefore inappropriate to neglect the dose to the skin from ^{106}Rh deposited on the outer surface of clothing.

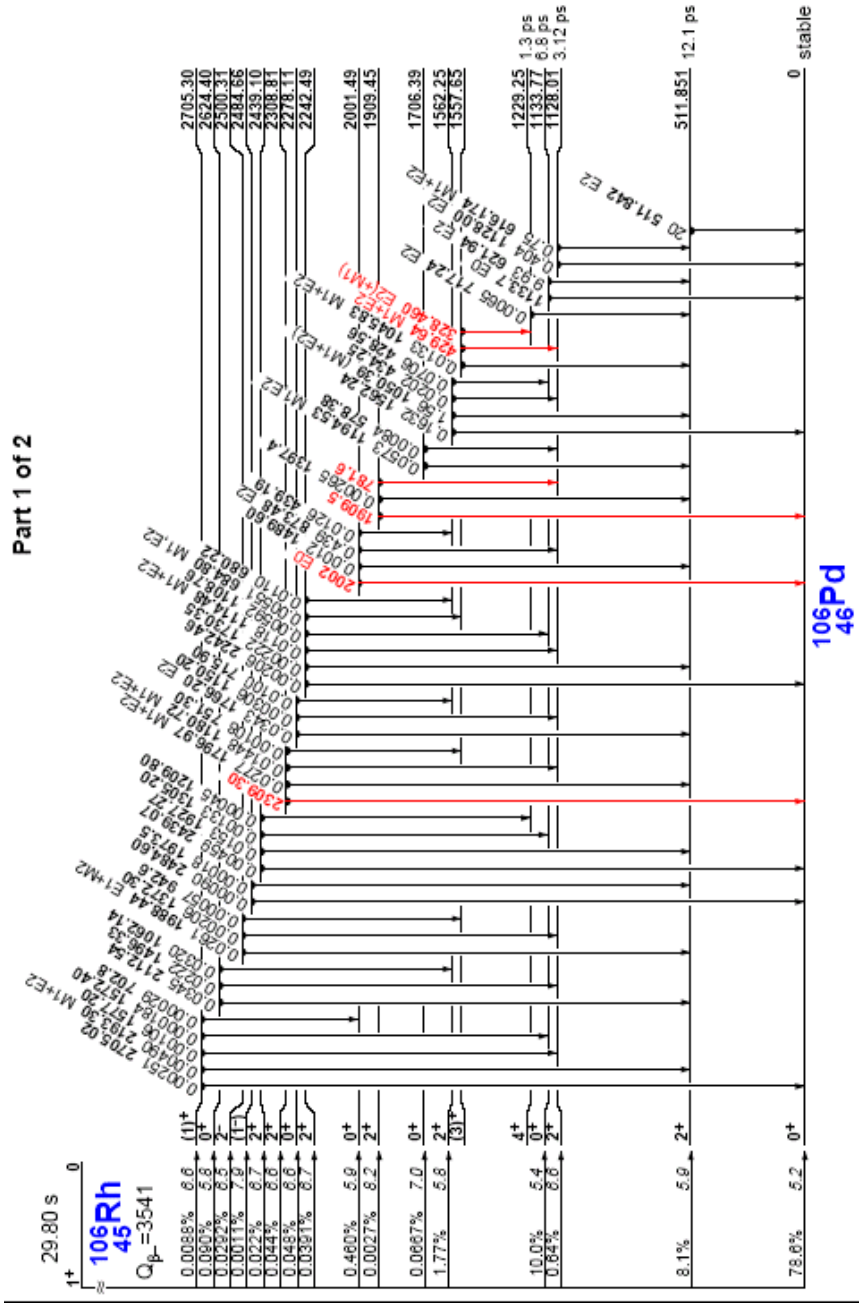


Figure 3. ^{106}Rh to ^{106}Pd Decay Series

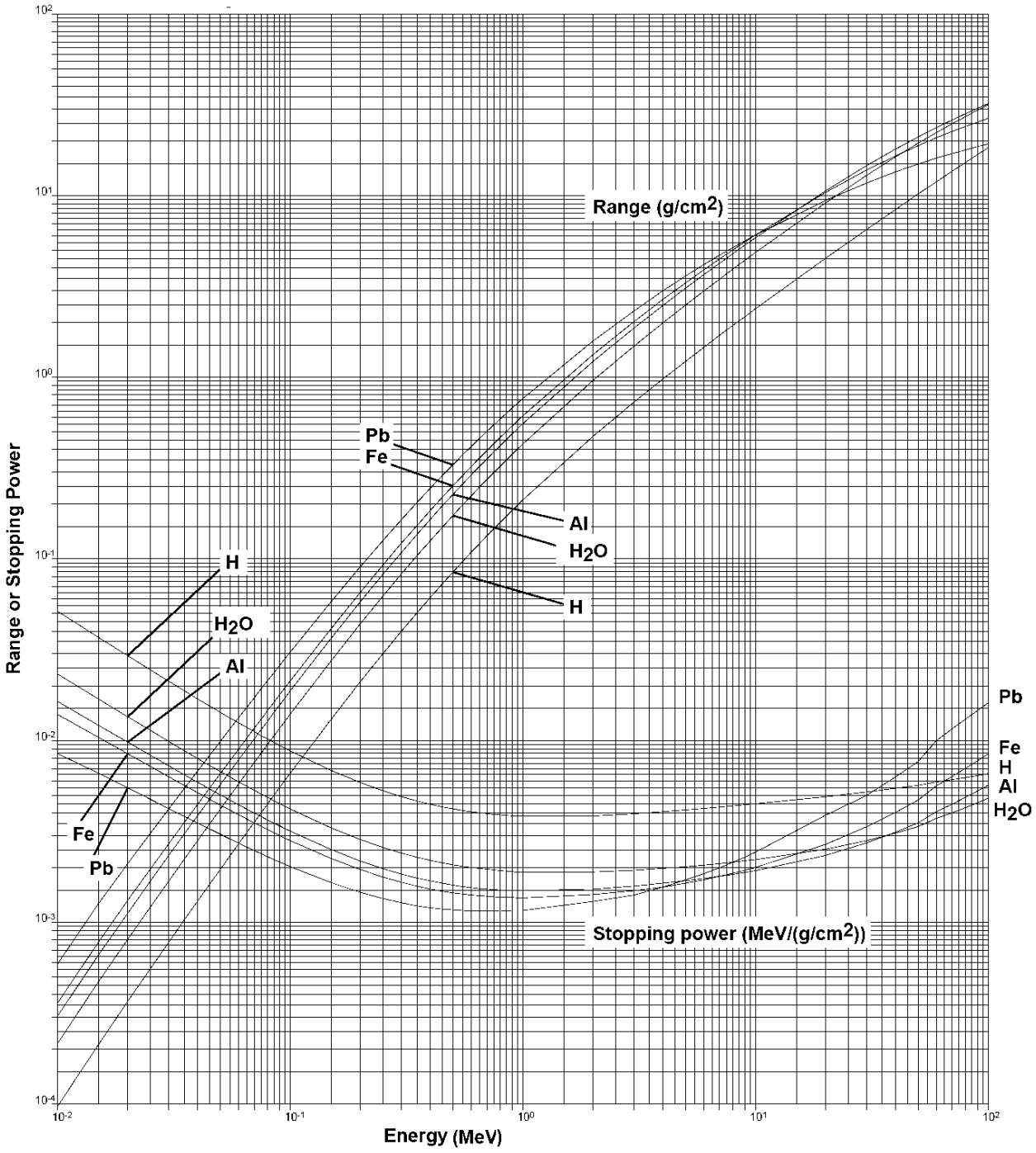


Figure 4. Range of Stopping Power versus Energy (MeV) for Various Elements

ATTACHMENT 9: LETTER ISSUED BY R.P. CORLEW IN 1972 TO THE U.S. ATOMIC ENERGY COMMISSION (AEC) REGARDING NEUTRON RADIATION AT 234-5-Z BUILDING

Federal B. Jr.
Richland, Washington 99352
Telephone 509 942 7411

NOV 2 1972

U. S. Atomic Energy Commission
Richland Operations Office
Richland, Washington 99352

Attention: Mr. C. J. Elgert, Director
Production and Waste Management
Programs Division

Subject: NEUTRON RADIATION AT 234-5-Z BUILDING
Contract AT(45-1)-2130

Gentlemen:

The introduction of the new multipurpose thermoluminescent dosimeters (TLD) in January 1972 alerted us to the fact that some Atlantic Richfield Hanford Company employees working in the 234-5-Z Building on the RMC Button Line were receiving higher neutron dosimetry readings than previously measured. Prompt actions were taken to reduce these exposures, and at the same time an investigation of the measurement system was started. The Richland Office of the U. S. Atomic Energy Commission has been kept informed of our progress in this study.

In-plant changes have continued and reductions in radiation levels have been observed. No employee is expected to receive as much as five rems whole body penetrating radiation exposure this year. Installation of shielding around Hood 9-8 on the RMC Button Line will proceed but will be completed too late to have a significant benefit this year. Studies of the extrapolated employee radiation exposure at the production rates forecasted for fiscal years 1974 and 1975 are being conducted to determine whether additional shielding will be required.

An estimation of prior years' exposure to employees working in the 234-5-Z Building has been made, based on the increased knowledge provided by the new dosimetry system. A summary of the methodology used has been prepared by Battelle-Northwest and ARHCO and is attached as Appendix A. This gives a practical maximum radiation dose which could have been received by each employee. Past experience has shown that the recorded gamma dose for each worker was accurately measured. The ratio of neutron exposure to gamma

U. S. Atomic Energy Commission
Attention: Mr. O. J. Elgert
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exposure was determined for periods related to work conduct and shielding. These ratios and the recorded gamma exposure for prior years were used to estimate a total whole body penetrating Hanford radiation history. Table I summarizes the results of these calculations for the 16 employees whose estimated exposure was the highest. Exposures for all these people are within the radiation control limits established in Atomic Energy Commission Manual Chapter 0524. Two short-term plutonium workers having the highest average exposures have been reassigned to reduce their average annual Hanford radiation exposure.

The Manager, Personnel Protection, will describe the methodology used to estimate prior years' exposure of 234-5-Z Building employees and discuss the results with the 234-5-Z Building management. Following this discussion, the Hanford Atomic Metal Trades Council will be notified that a neutron exposure study has been completed and that it will be reviewed with the Council's Radiation Committee and concerned employees. Each current employee listed in Table I will then be contacted individually by the Personnel Protection manager and will be given a review of his exposure history. All contacts are expected to be completed within a week with the employees who are available. If other building employees are concerned about their radiation exposure, the Personnel Protection manager will be available for personal discussions. A letter noting that an employee worked in the 234-5 Building prior to January 1, 1972 will be placed in his radiation exposure records file (see Appendix B). If the need arises for a revised estimate of the employee's neutron exposure, it can be obtained using the methodology described in Appendix A.

Please contact us if you require additional information on this subject.

Very truly yours,

for R. P. Corlew
R. P. Corlew
Vice President - Operations

RPC:BJM:jgp

Att. (3)

TABLE I

Estimated Total Whole Body Exposures
for Years Prior to 1972
Plutonium Finishing Section Button Line Employees

Employee No.	Total Measured Hanford Dose (a) (rems)	Total Calculated Maximum Hanford Dose (b) (rems)	Age	Permissible Dose to Present Age	
				Allowed (rems)	Percent Used
2	51.1	110	52	170	65
3	42.8	85	52.5	172	49
4	56.5	105	56	190	55
5	46.3	95	62	220	43
6	45.2	105	49	155	68
7	44.1	100	56	190	53
8	41.9	85	56	190	45
9	45.6	100	49.5	157	64
10	25.1	50	48.5	152	33
11	34.0	80	43	125	64
12	40.9	85	59.5	207	41
13	43.7	95	53.5	177	54
14	45.1	90	57.5	197	46
15	13.7	35	33.5	77	45
16	19.6	50	33	75	67
17	18.0	45	32	70	64
18	10.2	25	34.5	82	30
20	19.0	25	32.5	72	35

(a) Measured by film dosimeters.

(b) Estimated using the previous gamma exposure data and the estimated neutron-to-gamma ratio. Estimated for groups of workers by work location.

B. J. McMurray
11/2/72