SC&A S. Cohen & Associates
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October 20, 2006

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

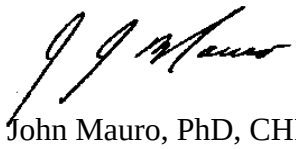
Subject: Contract No. 200-2004-03805, Task Order 1: *Paducah Gaseous Diffusion Plant Site Profile Review*, SCA-TR-TASK1-0016

Dear Mr. Staudt:

SC&A is pleased to submit to NIOSH and the Advisory Board our draft *Paducah Gaseous Diffusion Plant Site Profile Review*. Please note that Attachment 3: Summary of Expert Interviews, is currently being reviewed to confirm that no classified information was inadvertently included, and will be forwarded under separate cover when the review is completed.

If you have any questions or comments on this report, please contact John Mauro at 732-530-0104. We look forward to discussing this draft report with NIOSH and the Advisory Board.

Sincerely,



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ADVISORY BOARD ON

RADIATION AND WORKER HEALTH

National Institute of Occupational Safety and Health

Paducah Gaseous Diffusion Plant Site Profile Review

Contract No. 200-2004-03805
Task Order No. 1
SCA-TR-TASK1-0016

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Disclaimer

This document is made available in accordance with the unanimous desire of the Advisory Board on Radiation and Worker Health (ABRWH) to maintain all possible openness in its deliberations. However, the ABRWH and its contractor, SC&A, caution the reader that at the time of its release, this report is pre-decisional and has not been reviewed by the Board for factual accuracy or applicability within the requirements of 42 CFR 82. This implies that once reviewed by the ABRWH, the Board's position may differ from the report's conclusions. Thus, the reader should be cautioned that this report is for information only and that premature interpretations regarding its conclusions are unwarranted.

S. COHEN & ASSOCIATES: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program***	Document No. SCA-TR-TASK1-0016 Effective Date: Draft – October 20, 2006 Revision No. 0 (Draft)
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Task Manager: _____ Date: _____ Christopher Nelson	Supersedes: N/A
Project Manager: _____ Date: _____ John Mauro, PhD, CHP	

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ACRONYMS AND ABBREVIATIONS

Advisory Board	Advisory Board on Radiation and Worker Health
AEC	Atomic Energy Commission
Al	Aluminium
AMAD	Aerodynamic Median Activity Diameter
ATSDR	Agency for Toxic Substance and Disease Registry
AWE	Atomic Weapons Employer
BJC	Bechtel Jacobs Corporation
Bq	Becquerel
CDC	Centers for Disease Control and Prevention
CEDE	Committed Effective Dose Equivalent
CFR	*Code of Federal Regulations*
Ci	Curie
CIP	Cascade Improvement Project
COI	Conflict of Interest
CUP	Cascade Upgrade Program
DAC	Derived Air Concentration
DCF	Dose Conversion Factor
D&D	Decontamination and Decommissioning
DOL	Department of Labor
DR	Dose Reconstruction
DOE	Department of Energy
dpm	Disintegrations per Minute
DU	Depleted Uranium
FDA	U.S. Food and Drug Administration
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EPA	Environmental Protection Agency
ERDA	Energy Research and Development Agency
ERT	Enriched Reactor Tails
ESE	Entrance Skin Exposure
EU	Enriched Uranium
FEMP	Fernald Environmental Management Project

GM	Geometric Means
GSD	Geometric Standard Deviation
HEPA	High Efficiency Particulate Air
HHS	Health and Human Services
HVL	Half Value Layer
IAAP	Iowa Army Ammunition Plant
ICP-MS	Inductively Coupled Plasma-Mass Spectrometry
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
keV	Kilo electron Volt
KPA	Kinetic Phosphorescence Analysis
kVp	Kilovolt Potential
LANL	Los Alamos National Laboratory
LAT	Lateral
LLNL	Lawrence Livermore National Laboratory
LOD	Limit of Detection
μCi	Microcurie
mA	Milliampere
MAC	Maximum Allowable Concentration
MCW	Mallinckrodt Chemical Works
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDL	Minimum Detectable Level
MeV	Mega-electron Volt
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NOD	Net Optical Density
NRC	Nuclear Regulatory Commission
NTA	Eastman Kodak Nuclear Track Film Type A
NTS	Nevada Test Site
NU	Natural Uranium

OCAS	Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
OTIB	ORAU Technical Information Bulletin
PA	Posterior-Anterior
PACE	Paper, Allied Industrial, Chemical and Energy Workers
pCi	pico curies
PFG	Photofluorography
PGDP	Paducah Gaseous Diffusion Plant
PNNL	Pacific Northwest National Laboratory
POC	Probability of Causation
REM	Roentgen equivalent man
RFP	Rocky Flats Plant
RU	Recycled Uranium
SAIC	Scientific Applications International Corporation
SC&A	S. Cohen and Associates
SEC	Special Exposure Cohort
SID	Source-to-image distance
SNL	Sandia National Laboratory
SOP	Standard Operating Procedure
SRS	Savannah River Site
SSD	Source-to-Skin Distance
TBD	Technical Basis Document
TIB	NIOSH Technical Information Bulletin
TLD	Thermoluminescent Dosimeter
TLND	Thermoluminescent neutron dosimeter
TRU	Transuranics
UF_4	Uranium Tetrafluoride
UF_6	Uranium Hexafluoride
UO_3	Uranium Trioxide
U_3O_8	Urano-uranic Oxide
UO_2F_2	Uranyl Fluoride

USEC	United States Enrichment Corporation
WFO	Work For Others
Y-12	Y-12 National Security Complex

1.0 EXECUTIVE SUMMARY

This report provides the results of an independent audit conducted by S. Cohen and Associates (SC&A) of the technical basis documents (TBDs) that make up the site profile for the Paducah Gaseous Diffusion Plant (PGDP) developed by the National Institute of Occupational Safety and Health (NIOSH). This audit was conducted during the period April 2006–September 2006 in support of the Advisory Board on Radiation and Worker Health (Advisory Board) in the latter's statutory responsibility under the Energy Employees Occupational Illness Compensation Program Act (EEOICPA) to conduct such reviews and to advise the Secretary of Health and Human Services (HHS) on the "completeness and adequacy" of the EEOICPA program.

A subgroup of PGDP workers has been designated as a Special Exposure Cohort (SEC) as defined in 42 CFR Part 83. This designation allows those workers with a covered cancer to be considered for a compensation decision under the provisions of the EEOICPA without requiring a probability of causation (POC) calculation. Other workers with uncovered cancers would have to have such a calculation made as a prerequisite for a compensation decision to be made.

SC&A notes that this review was triggered in part by concerns raised in a letter by Richard Miller (see Attachment 1) to the Secretary of HHS and the Director of NIOSH. Mr. Miller was concerned that the principal author of the Occupational Internal Dose TBD had serious conflict of interest (COI) issues and that the technical credibility of that TBD was questionable as well. As a result, particular emphasis has been placed on the review of that document.

Construction of the PGDP began in 1951 and production started in 1952. The site is located in northeastern McCracken County, approximately 10 miles west of the City of Paducah, Kentucky, and covers 3,423 acres, of which 748 acres are within a security fence. Paducah was initially owned and operated by the U.S. Department of Energy (DOE) and its predecessor agencies. Since July 1, 1993, the DOE has leased the production facilities of the plant to the United States Enrichment Corporation (USEC). The USEC portion of the site is licensed and regulated by the Nuclear Regulatory Commission (USEC 1994). The DOE mission at the site is now one of Environmental Restoration and Waste Management Operations. In some cases, the DOE and USEC operations occur in the same building or area, resulting in potential exposure of USEC employees to DOE radioactive material and vice versa.

Paducah has been actively involved in the enrichment of uranium, its primary function, since its inception. This has not been the sole activity of this site. In Building C-420, there was a facility to convert uranium trioxide (UO_3) and urano-uranic oxide (U_3O_8) to uranium tetrafluoride (UF_4) for further conversion in Building C-410. The C-410 Building was used for conversion of uranium tetrafluoride (UF_4) to uranium hexafluoride (UF_6), the feed material for the process buildings. Paducah was also involved in the reduction of UF_6 to UF_4, and subsequent conversion to metallic uranium. The process used was similar to that at the Fernald site. There are two maintenance shops for fabrication, assembly, maintenance, and repair of process equipment. The Cascade Improvement Program (CIP) in 1954–1961 and the Cascade Upgrade Program (CUP) in 1973–1982 required isolating and opening portions of the cascade system to replace existing process equipment while the plant continued its uranium enrichment operations. These operations and the repair and upgrading of the equipment provided conditions for increased

radiation exposure of workers. An onsite smelter was used to salvage diffusion barriers and other sensitive items for PGDP and other government sites. Waste has either been stored above ground or taken to on-site burial areas (Donnelly 1984). PGDP participated in the work for other program, where they conducted projects for other sites. The DOE mission is currently Environmental Restoration and Waste Management Operations.

SC&A's review focused on the six TBDs that make up the PGDP site profile. These comprise Introduction, Site Description, Occupational Internal Dose, Occupational External Dose, Occupational Medical Dose, and Occupational Environmental Dose as these topics pertain to historic occupational radiation exposure of PGDP workers. These TBDs were issued in 2004–2006. As "living" documents, TBDs are constantly being revised as new information, experience, or issues arise. This review includes consideration of the August 2006 revision of the Site Description and the page change of the Introduction TBDs. The revised versions of the Occupational External Dose and Occupational Medical Dose, published in August and September of 2006, respectively, were not used for this review. SC&A's process included a review of the TBDs, an onsite visit to conduct interviews with site experts and to identify documents for data retrieval, and reviews of retrieved PGDP and other historic records. The TBDs were evaluated for their completeness, technical accuracy, adequacy of data, compliance with stated objectives, and consistency with other site profiles, as stipulated in the *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004). A complete list of the PGDP TBDs and other supporting documents that were reviewed by SC&A is provided in Attachment 2.

Interviews at the site were originally designed to solicit unclassified information; however, a number of site experts were more comfortable with classified interviews. Records reviews included both classified and unclassified documentation. As a result, it was also necessary to have interview notes and documents screened by PGDP classification reviewers to assure that no sensitive material was inadvertently included. Site expert interviews will be released after the interview summary has undergone appropriate classification review. At this time, additional information may be included in other sections of the report.

Issues presented in this report 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 primary findings, because they represent key deficiencies in the TBDs that need to be corrected and which have the potential to substantially impact at least some dose reconstructions. Others have been designated "observations" to both denote their importance for the technical adequacy and completeness of

the site profile, and to indicate that they have been judged by SC&A to have relatively less influence on dose reconstruction or the ultimate significance of worker doses so estimated.

1.1 SUMMARY OF STRENGTHS

In general, the Site Description TBD (Maisler 2006) provides a basic characterization of much of the history and operational activities at the PGDP. The major functions of many of the production and support buildings are outlined, as well as major process changes that occurred through the years.

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

(1) Radiological control personnel have implemented improved procedures and technologies over time to reduce radiation dose to workers, and have improved personnel monitoring programs.

(2) PGDP implemented an environmental monitoring program, including stack monitoring, perimeter monitoring, offsite monitoring, and groundwater monitoring.

(3) Starting in 1961, all employees have been provided with an external dosimeter, creating a potentially useful coworker database.

The PGDP TBDs provided a good description of the in-vivo and in-vitro monitoring programs over the period of operation, including detailed descriptions of dosimeters used for both beta/gamma and neutron monitoring.

1.2 SUMMARY OF FINDINGS

Finding 1: Uranium Enrichment Levels Achieved at PGDP could be Higher than 2%

The Site Description TBD (Turpin 2006, pg. 7) states that PGDP enriched feed material (UF_6) up to about 2.5% ^{235}U. However, the default enrichment level specified in the Occupational Internal Dose TBD (Berger 2004) is only 2%. Although the assayed specific activity of ^{235}U in the cascade product is consistent with this enrichment level, the maximum assayed specific activity of ^{234}U is over 50% higher than the default activity of this isotope The higher activities provide the potential for higher radiation doses from ^{234}U and perhaps from ^{235}U than those calculated using the methodology prescribed in the TBD.

Finding 2: The Number of Workers Assigned a Zero Dose should be Disclosed When Reporting the Mean of the Distribution of Doses

The average recorded doses in Tables 2-2 and 2-3 of the Site Description TBD (Turpin 2006, pg. 11) are biased low and mean little without knowing the numbers of workers assigned dose values of zero when their measured dose was less than the minimum detectable level (MDL).

Finding 3: The TBD Needs to Consider Operations Other than Gaseous Diffusion

There is no mention in the Site Description TBD (Turpin 2006) of the smelting operations that took place in Building C-746B or the smelting of diffusion barriers (including K-25 and Portsmouth barriers) during the cascade improvement and upgrade programs. These activities need to be described, since they could have resulted in significant doses to some workers. Paducah participated in the "Work for Others" (WFO) program providing assistance to other DOE sites. There is no mention of this program in the TBD and how this may affect the dose reconstructions in terms of radionuclides.

Finding 4: The TBD Fails to Adequately Define and Assess Occupational Medical Exposure

The guidelines, as presented by Kathren (2003), go a long way in assuring that all occupational medical exposures are reasonably included in determining the overall dose estimations for claimants. Unfortunately, these guidelines have not been applied in the TBD in a claimant-favorable manner.

Finding 5: Contamination Control and Skin and Extremity Dose

The TBDs do not provide enough information regarding the radiological controls in place (or lack thereof) for the operations at PGDP that pose the potential for exposures. Especially in the early days, lack of adequate radiological controls could lead to worker doses that may be missed in the process of dose reconstruction, particularly for unmonitored workers. Such workers may not be identifiable without this understanding of the effectiveness of the radiological controls in place and for work practices where significant dose was possible. Contamination control was a significant problem and should be examined for its relevance to skin dose. This should be considered in the context of dose reconstruction. Some method for assignment of extremity dose should be developed where this area is affected by cancer.

Finding 6: Onsite Environmental Exposures are based on Site Boundary Data

The TBD purports to describe "potential exposures from ambient sources while working outside the process buildings," but the ambient monitoring data in the document are for measurements at the site boundaries and beyond. There are no corroborating data provided to demonstrate that these measurements are consistent with the levels that workers might experience while working at the site outside the process buildings. Two additional sources of ambient exposure include the burning of contaminated material in onsite pits (i.e., routine and incidentally), and the alleged intentional releases that occurred in the 1950s.

Finding 7: Inadequate Characterization of the Source Term for Internal Exposures

A critical issue is the specification of the source term for the internal exposures. The Internal Occupational Dose TBD (Berger 2004) does not fully utilize information presented in two key documents. One is the exposure assessment of workers at the PGDP (PACE/Utah 2000), which emphasized exposures to neptunium and plutonium. The other is the draft report on recycled

uranium mass balance (BJC 2001), which is cited as a source of information by Turpin (2006), but is not mentioned by Berger.

Finding 8: Isotopic Fractions for Various Enrichments are not Properly Characterized in the TBD

An examination of the data in Table 5-2 of the TBD reveals a number of discrepancies. The most glaring one is the specific activity of ^{235}U in 93% enriched feed. A direct calculation yields an activity of 7.44E-2 Bq/μg, a factor of 10 higher than the listed value of 7.38E-3 Bq/μg.

Finding 9: The Default Isotopic Distribution in Table 5-4 Ignores Many Isotopes Associated with Recycled Uranium

Table 5-4 of the TBD lists default specific activities to be used when only total uranium results are available. This guidance ignores the information presented by Bechtel-Jacobs Corporation (BJC 2001, Table 2.4-1), which lists maximum concentrations of ^{99}Tc, ^{237}Np, and plutonium for 11 different operations at various facilities in the PGDP in various time periods. NIOSH should evaluate these data for use in the dose reconstructions of workers involved in these operations.

Finding 10: Particle Size of Inhaled Aerosols Assumed in the TBD is not Entirely Claimant Favorable

In Table 5-5 of the TBD, the particle size for inhaled aerosols is specified as 5 μm Aerodynamic Medial Activity Diameter (AMAD). Such a particle-size assignment is not supported by the data, nor is it claimant favorable, because particle sizes significantly less than 5 μm AMAD are cited in the supporting literature.

Finding 11: The List and Quantities of Transuranics Addressed in the TBD are not Complete or Claimant Favorable

Table 5-5 of the TBD limits transuranics (TRU) to ^{237}Np and ^{239}Pu. However, the TRU in the Hanford reactor tails and other sources include ^{238}Pu, ^{240}Pu, and ^{241}Am. In addition, information about the occurrence of TRU nuclides at different facilities is presented by PACE/Utah (2000, Appendix D), which lists various radiological data that need to be taken into consideration in the TBD.

Finding 12: Lung Clearance Types need to be clearly Defined and Claimant Favorable

The Lung Clearance Types, referred to as "Absorption Type" in Table 5-5, should be consistent with the chemical forms of each radionuclide. In case of uncertainty, the most claimant-favorable assumption should be adopted. The TBD is not always clear which chemical form and/or clearance type the dose reconstructor should assign to each element.

Finding 13: Radionuclide Intakes based on Bioassay Data Need to Take into Consideration Frequency of Sample Collection

Table 5-6 of the TBD lists the frequency of in-vitro measurements at various facilities at the PGDP during various time periods. The frequencies range from once every 4 weeks to once a year. The last row of the table lists a default frequency of once every 4 weeks. Since the interval between measurements could have been as long as 1 year, this default assumption is not claimant favorable. The longer the interval over which the intakes occurred, be they chronic or acute, the lower the urine concentration for a given total intake. Conversely, the longer the elapsed time (for acute intakes) or exposure duration (for chronic) for a given urine concentration, the greater the derived intake.

Finding 14: Minimum Detectable Concentrations are not Clearly Defined

In many cases, we could not verify the listed MDCs. For example, the MDC for total uranium by in-house fluorimetry is listed as 10 μg/L. None of documents cited as data sources for this table list that value; both PACE/Utah (2000) and SAIC (1999) cite typical MDCs of 5 μg/L. The MDC for natural uranium assayed at ORNL from 1999 to the present is in error; the source document (SAIC 1999) lists the MDC at 0.06 μg/sample, while Table 5-7 lists it as 0.06 mg (60 μg), a 1000-fold discrepancy. The latter value is clearly inconsistent with the value of 5 μg/L for kinetic phosphorescence analysis (KPA), used in 1977–1982. Table 5-7 lists a default detection level of 0.27 pCi/L for urinalysis of individual isotopes of actinide elements (Th, U, Pu, and Am). The TBD cites ICRP Publication 54 (Annals of the ICRP Vol. 19 No. 1–3) , *Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation* (Oxford: Pergamon Press) as the source of these data, which are based on alpha spectroscopy.[1] However, Table 5-7 lists higher levels—1 pCi/L for Pu isotopes and 0.3 pCi/L for uranium isotopes. If the dose reconstructor does not know the analytical method involved, this default assumption is neither claimant favorable nor scientifically correct.

Finding 15: Day of Sample Collection needs to be Taken into Consideration When Deriving Intakes based on Urine Analyses

Section 5.3.3 of the TBD states the following:

> *The practice of offsite collection of samples that takes place 24 to 48 hr after leaving the plant not only minimizes the possibility of sample cross-contamination, but it ensures that samples are collected after the transfer of the rapid clearance component. Some PGDP employees were asked to collect samples after 1 or 2 days off from work; if so, that collection instruction was sometimes noted on the analytical record.*

The TBD fails to note that this practice would lead to a lowering of the calculated intakes, nor does it instruct dose reconstructors to be alert to any cases for which urine samples were

[1] We also note that ICRP Publication 54 has been replaced by ICRP Publication 78: *Individual Monitoring for Internal Exposure of Workers - Annals of the ICRP Volume 27/3-4, Replacement of ICRP Publication 54*, 1998 (Oxford:Pergamon Press).

collected after the worker was off work for any period of time. Appropriate adjustments to the calculated intake should be made to compensate for the lowered concentration in the urine following an absence from work.

Finding 16: Additional Significant Incidents with Internal Dose Potential Need to be Discussed

Significant information that could be useful to dose reconstructors is not included in the TBD. This information includes statements by workers that urine specimens were collected within 30 minutes of an incident or accident, with a potential for elevated exposure. Such a short time period does not allow for equilibrium between the inhaled activity and the concentration in the urine, most likely resulting in a false negative. Follow-up samples were collected from workers who did show elevated levels of radioactive materials in the urine. More important are the statements of former workers that the bioassays performed following such incidents were not always recorded in the database. Thus, the doses from such incidents may not be in the worker exposure records.

Finding 17: The Coworker Model for Applying Bioassay Data to Unmonitored Workers is neither Scientifically Valid nor Claimant Favorable

In the coworker models used by NIOSH, workers are not classified by their jobs or by the buildings where they performed their work. In the TIB describing the coworker model for internal dosimetry (Ikenberry 2005), there is no attempt to sort the urinalysis results by job assignment or location to determine if there were any correlations between the uranium concentration in the urine and the building or department where the job was performed.

In order to apply the coworker model to unmonitored workers, NIOSH needs to demonstrate that there is a low probability that any unmonitored worker could have higher exposures than the monitored workers taken as a group. Ikenberry (2005) fails to do so.

Finding 18: The Method of Converting Mass Concentrations of Uranium in Urine to 24-hour Excretions of Activity of Uranium Isotopes is neither Scientifically Valid nor Claimant Favorable

The TIB describing the coworker model for internal dosimetry (Ikenberry 2005) uses the non-claimant-favorable assumption about the specific activities of uranium isotopes presented in the internal dose TBD (Berger 2004). The default specific activity should be increased from 0.0389 Bq/µg to 0.0541 Bq/µg. The daily excretion of urine should be updated to reflect the latest ICRP (2002) recommendations: 1.6 L/d for male workers and 1.2 L/d for females.

Finding 19: Shallow Dose from Beta Emitters may have been Underestimated

According to the Occupational External Dose TBD (Turner 2005), the film badges used to derive skin doses from beta emitters employed a minimum absorber thickness of 80 mg/cm^2 between the film and the source, but the film badges appear to have been calibrated with a uranium slab

without the absorber. Under these conditions, a large portion of the skin dose from weak betas emitted by uranium would not be detected, and the skin dose significantly underestimated.

Finding 20: Questionable Assumptions for Assigning Skin and Deep Dose for Unmonitored Workers Prior to 1960 by means of Coworker Data

SC&A is concerned that the coworker model described in the Occupational External Dose TBD (Turner 2005) is based on the assumption that prior to 1960, the population of ". . . monitored individuals represents those with the highest exposure potential." It is SC&A's contention that these monitoring data reflect a badging practice that not only included all worker categories (regardless of their potential for exposure), but furthermore diluted the average dose within a given worker category by rotating badge assignments. However, this issue appears to have been resolved with the issuance of ORAUT-OTIB-0031 (Merwin 2005).

Finding 21: Assessment of Neutron Exposures at PGDP Appears to have been Underestimated

The principal sources of neutron exposures at PGDP involve the α, n reaction with fluorine compounds (UF_4, UF_6) and exposures to radionuclides that undergo spontaneous fission (i.e., TRU). However, based on the Occupational External Dose TBD (Turner 2005), it appears that reliable monitoring of neutron exposures did not begin until 1998, and NIOSH had to account for missed neutron dose based on worker activities and models that employ neutron-to-photon ratios. SC&A is concerned that the coworker model may not be dependable for exposures prior to 1961, due to a paucity of photon dosimetry at that time.

Another document is the draft report on recycled uranium mass balance (BJC 2001), which is cited as a source of information by Turpin (2006), but is not mentioned in this TBD. That report breaks down the potential exposures to three contaminants in recycled uranium—^{99}Tc, ^{237}Np, and plutonium—into 11 types of activities and operations at the PGDP, and lists the maximum concentrations of each of these constituents (BJC 2001, Table 2.4-1).

2.0 SCOPE AND INTRODUCTION

This review of the Paducah Gaseous Diffusion Plant (PGDP) site profile was conducted from April 1 through August 31, 2006, in support of the Advisory Board on Radiation and Worker Health (Advisory Board) in the latter's statutory responsibility under the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) to conduct such reviews and advise the Secretary of Health and Human Services (HHS) on the "completeness and adequacy" of the EEOICPA program. This review included interviews of site experts by S. Cohen & Associates (SC&A) regarding specific Technical Basis Documents (TBDs) that make up the PGDP site profile. The review was limited to unclassified materials generated from worker interviews, and document reviews were submitted for classification screening to ensure that no sensitive information was inadvertently included in this report.

The review was triggered, in part, by concerns raised in a letter by Richard Miller (see Attachment 1) to the Secretary of HHS and the Director of NIOSH. Miller was concerned that the principal author of the internal dose TBD had serious COI issues, and that the technical credibility of that TBD was questionable as well. As a result, particular emphasis has been placed on the internal dose portions of the site profile.

A subgroup of PGDP workers has been designated as an SEC, as defined in 42 CFR Part 83:

> *The Special Exposure Cohort (''the Cohort'') is a category of employees defined under 42 U.S.C. 7384l(14). In this definition, Congress specified classes of employees to comprise the Cohort initially, including DOE employees, DOE contractor or subcontractor employees, who were (1) employed an aggregate of at least 250 work days before February 1, 1992 at a gaseous diffusion plant in Paducah, Kentucky, Portsmouth, Ohio, or Oak Ridge, Tennessee, and (2) who were monitored using dosimetry badges or worked in a job that had exposures comparable to a job that is or was monitored using dosimetry badges….*
>
> *As provided in 42 U.S.C. 7384l(9)(A), employees included in the Cohort who incur a specified cancer qualify for compensation (see DOL regulations 20 CFR part 30 for details). Cancer claims submitted by these employees or their survivors do not require DOL to evaluate the probability that the cancer was caused by radiation doses incurred during the performance of duty for nuclear weapons programs of DOE, as is required for other cancer claims covered by EEOICPA.*
>
> *Specified cancers are a limited group of cancers that EEOICPA specifies are compensable under provisions governing compensation for members of the Cohort. Although the list of specified cancers is determined by statute, the list can also be found in this rule under § 83.5.*
>
> *83.5(m) Specified cancer, as is defined in Section 3621(17) of EEOICPA (42 U.S.C. 7384l(17)) and the DOL regulation implementing EEOICPA (20 CFR 30.5(dd)), means:*

(1) Leukemia (other than chronic lymphocytic leukemia) provided that onset of the disease was at least two years after initial occupational exposure;
(2) Lung cancer (other than in situ lung cancer that is discovered during or after a post-mortem exam);
(3) Bone cancer;
(4) Renal cancers;
(5) The following diseases, provided onset was at least 5 years after first exposure:
(i) Multiple myeloma;
(ii) Lymphomas (other than Hodgkin's disease);
(iii) Primary cancer of the:
(A) Thyroid;
(B) Male or female breast;
(C) Esophagus;
(D) Stomach;
(E) Pharynx;
(F) Small intestine;
(G) Pancreas;
(H) Bile ducts;
(I) Gall bladder;
(J) Salivary gland;
(K) Urinary bladder;
(L) Brain;
(M) Colon;
(N) Ovary;
(O) Liver (except if cirrhosis or hepatitis B is indicated).
(6) The specified diseases designated in this section mean the physiological condition or conditions that are recognized by the National Cancer Institute under those names or nomenclature, or under any previously accepted or commonly used names or nomenclature.

While the SEC designation allows a subgroup of workers to file claims under the Act without requiring a dose reconstruction, other workers with uncovered cancers may require a dose reconstruction to be eligible for compensation. The integrity of the dose reconstruction process remains a critical consideration.

2.1 REVIEW SCOPE

Under the EEOICPA and Federal regulations defined in 42 CFR Part 82, *Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program,* of the *Code of Federal Regulations*, the Advisory Board is mandated to conduct an independent review of the methods and procedures used by NIOSH and its contractors for dose reconstruction. As a contractor to the Advisory Board, SC&A has been charged under Task Order 1 to support the Advisory Board in this effort by independently evaluating a select number of site profiles that correspond to specific facilities at which energy employees worked and were exposed to ionizing radiation.

This report provides a review of the following six documents related to historical occupational exposures at the PGDP site:

- Maisler, Daniel 2006, *Technical Basis Document for Paducah Gaseous Diffusion Plant – Introduction,), ORAUT-TKBS-0019-1, (Rev. 00 PC-1)*, ORAU Team Dose Reconstruction Project for NIOSH, August 8, 2006.

- Turpin, Baynard, 2006, *Technical Basis Document for the Paducah Gaseous Diffusion Plant – Site Description, ORAUT-TKBS-0019-2, (Rev. 01)*, ORAU Team Dose Reconstruction Project for NIOSH, August 21, 2006.

- Turner, James E., 2004, *Technical Basis Document for the Paducah Gaseous Diffusion Plant – Occupational Medical Dose, ORAUT-TKBS-0019-3, (Rev. 00)*, ORAU Team Dose Reconstruction Project for NIOSH, February 11, 2004.

- East, James, 2004, *Technical Basis Document for the Paducah Gaseous Diffusion Plant – Occupational Environmental Dose, ORAUT-TKBS-0019-4, (Rev. 00)*, ORAU Team Dose Reconstruction Project for NIOSH, August 24, 2004.

- Berger, Carol D., 2004, *Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose, ORAUT-TKBS-0019-5, (Rev. 00)*, ORAU Team Dose Reconstruction Project for NIOSH, September 30, 2004.

- Turner, James E, 2005, *Paducah Gaseous Diffusion Plant – Occupational External Dose, ORAUT-TKBS-0019-6, (Rev. 01)*, ORAU Team Dose Reconstruction Project for NIOSH, March 29, 2005.

These documents are supplemented by two technical information bulletins (TIBs), which provide additional guidance to the dose reconstructor:

- Ikenberry, Tracy A., 2005, *Internal Dosimetry Coworker Data for Paducah Gaseous Diffusion Plant, ORAUT-OTIB-0037, (Rev 00)*, ORAU Team Dose Reconstruction Project for NIOSH, September 20, 2005

- Merwin, Steven E., 2005, *External Coworker Dosimetry Data for the Paducah Gaseous Diffusion Plant, ORAUT-OTIB-0031, (Rev 00 PC-1)*, ORAU Team Dose Reconstruction Project for NIOSH, August 15, 2005.

Since these TIBs supplement specific TBDs in the site profile, they were also considered in this review.

Implementation guidance is also provided by so-called "workbooks," which have been developed by NIOSH for selected sites to provide more definitive direction to the dose reconstructors on how to interpret and apply TBDs, as well as other available information. The SC&A team did not include review of workbooks related to the PGDP site profile (i.e., the

workbook accompanying ORAUT-OTIB-0031). This workbook is being reviewed under another task order.

SC&A, in support of the Advisory Board, has critically evaluated the PGDP Site TBDs for the following:

- 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 characterizes the facility and its operations, and the use of these data in dose reconstruction. The review was conducted in accordance with *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004), which was approved by the Advisory Board.

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 TBDs 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.

2.2 REVIEW APPROACH

SC&A's review of the TBDs and supporting documentation concentrated on determining the completeness of data collected by NIOSH, the adequacy of existing PGDP personnel and

environmental monitoring data, and the evaluation of key dose reconstruction assumptions. Site expert interviews were conducted at the PGDP, and all notes taken were submitted for screening by appropriate classifiers.

Site expert interviews were conducted to help SC&A obtain a comprehensive understanding of the radiation protection program, site operations, and environmental contamination. Interviews were conducted by SC&A by teleconference or in person during the course of this review. The site experts included current and former staff from radiation control, operations, environmental monitoring, maintenance, and other organizations, as well as other site experts knowledgeable of particular elements of the PGDP environmental safety and health program. 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. A master summary of all interviews conducted is reserved pending review for classification purposes, and will be provided at a later date.

2.3 REPORT ORGANIZATION

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 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 can also be 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.

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 a list of the issues in summary form, and to which objective each particular issue applies.

In many ways, the TBDs 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 METHODS

SC&A 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 *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004). This review is specific to the PGDP Site Profile and supporting TIBs; 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 supplied by USEC and publicly available records relating to the PGDP 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 that occurred at PGDP. 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. For example, if SC&A found that the technical approach used by NIOSH was not scientifically sound or claimant favorable, this would constitute a technical accuracy issue.

3.1.3 Objective 3: Adequacy of Data

SC&A reviewed the site profile with respect to Objective 3, which requires SC&A to determine whether the data and guidance presented in the site profile are sufficiently detailed and complete to conduct dose reconstruction, and whether a defensible approach has been developed in the absence of data. In addition, this objective requires SC&A to assess the credibility of the data used for dose reconstruction. The adequacy of the data identifies gaps in the facility data that may influence the outcome of the dose reconstruction process. For example, if a site did not monitor all workers exposed to neutrons who should have been monitored, this would be considered a gap and, therefore, 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. This objective was accomplished by reviewing key TBD assumptions for determining medical, environmental, internal and external dose from PGDP, and previously reviewed TBDs. 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 reconstructions consist of 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%, a more refined analysis is required. 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 the following:

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

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

Technical Basis Document for Paducah Gaseous Diffusion Plant – Introduction, ORAUT-TKBS-0019-1, Rev. 00 PC-1, (Maisler 2006), explains the purpose and the scope of the TBDs comprising 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. SC&A recognizes 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. 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.

Technical Basis Document for Paducah Gaseous Diffusion Plant – Site Description, ORAUT-TKBS-0019-2, Rev. 01, (Turpin 2006), is an extremely important document, because it provides a description of the facilities, processes, and historical information that serve as the underpinning for subsequent PGDP TBDs. SC&A's review of this section specifically addresses whether all of the potentially important site activities and processes are described, and whether characterization of source terms is complete and sufficient to support dose reconstruction.

Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Medical Dose, ORAUT-TKBS-0019-3, Rev. 00, (Turner 2004), provides an overview of the sources, types of exposure, and the frequency of exams that workers potentially received. The TBD clearly acknowledges the paucity of actual data to substantiate doses or support individual worker dose contributions—especially in the years prior to1975. The TBD draws heavily upon assessments at other sites, e.g., the Mound Plant and the Los Alamos National Laboratory (LANL). SC&A reviewed this section for technical adequacy and consistency with other NIOSH TBDs and procedures.

Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Environmental Dose, ORAUT-TKBS-0019-4, Rev. 00, (East 2004), provides background information and guidance to dose reconstructors for reconstructing the doses to unmonitored workers outside the facilities at the site who may have been exposed to routine and episodic exposures from these facilities. SC&A reviewed this section from the perspective of the source terms, measurements, and assumptions used to derive the external and internal doses to these workers.

Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose, ORAUT-TKBS-0019-5, Rev. 00, (Berger 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 by the workers, the recommended assumptions for use in reconstructing internal doses based on in-vivo and in-vitro 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.

Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational External Dose, ORAUT-TKBS-0019-6, Rev. 01, (Turner 2005), 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.

SC&A has not developed a list of questions, and has not held conference calls with NIOSH to help clarify our understanding of the TBDs. Hence, this review has not had the benefit of this step in the site profile review process.

Comparisons have been made between the methodologies used in the PGDP site profile and other TBDs reviewed to date. In the case of occupational medical exposure, the limited data available warranted additional comparison to TBDs for other sites. These comparisons focus on the methodologies and assumptions associated with dose reconstruction and resultant values used to obtain a POC (see Attachment 4).

There are three levels of review typically employed in the preparation of a site profile review. First, SC&A team members review the report internally (this review has been performed). Second, SC&A project management reviews all aspects of this report (which has been performed), and, if necessary, outside experts are called upon to review selected portions of this report. It was deemed not necessary to engage an independent outside expert review, due to the extensive internal review this report received. The third level, referred to as the expanded review cycle, will consist of a review of this draft report by an Advisory Board-designated Working Group. The issues matrix, which will be provided under separate cover, will be used as the starting point to discuss and resolve issues.

As the Working Group addresses the various issues raised in this report, it will report to the Advisory Board on the status of issues resolution. At such time that the Advisory Board deems appropriate, they will direct SC&A to perform additional analysis and or prepare revisions to this draft report to reflect the findings of the Working Group and the Advisory Board.

Finally, SC&A notes that its 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 are important with respect to dose reconstruction.

4.0 SITE PROFILE STRENGTHS

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

4.1 COMPLETENESS OF DATA

In general, the Site Description TBD (Maisler 2006) provides a basic characterization of much of the history and operational activities at the PGDP. The major functions of many of the production and support buildings are outlined, as well as major process changes that occurred through the years. There are still some gaps related to processing of special materials and recycled uranium. The text provides a short list of special incidents and activities with a high potential for increased radiation exposure for the period 1956–1981. This information can assist dose reconstructors in determining whether additional exposure potential exists, and whether the total dose reconstruction process is claimant favorable. That said, the information presented is a brief summary of the available data. A more comprehensive treatment of these subjects can be found in the PACE/Utah (2000) and Bechtel (BJC 2001) reports. In any case, NIOSH will often have to reconstruct doses for workers in the period before about 1974—and especially before 1961—from very limited film badge and bioassay data. In-vitro bioassay data are generally limited to measurements of the total mass of uranium in urine. Radionuclide-specific estimates of activity intakes must be based on supplementary tables of activity relative to the total mass of uranium from those measurements.

4.2 ADEQUACY OF DATA

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

(1) Radiological control personnel have implemented improved procedures and technologies over time to reduce radiation dose to workers, and have improved personnel monitoring programs.

(2) PGDP implemented an environmental monitoring program, including stack monitoring, perimeter monitoring, offsite monitoring, and groundwater monitoring.

(3) Starting in 1961, all employees have been provided with an external dosimeter.

Although PGDP has significant quantities of personnel monitoring data, as well as environmental data, there are gaps in the information. Only a fraction of the population was monitored for radiation exposure in the early years of operation.

4.3 TECHNICAL ACCURACY/CLAIMANT FAVORABILITY

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

(1) NIOSH has provided a good description of the in-vivo and in-vitro monitoring programs over the period of operation. They have provided detailed descriptions of dosimeters used for both beta/gamma and neutron monitoring. The TBDs and OTIBs made efforts to track the changes in dosimetry methods, calibration standards, and administrative limits for the period of 1952 to 2003 to assist in making the assigned dose claimant favorable. Chronological sequences of events at the PGDP were outlined in the TBDs, along with associated tables of dosimetry methodology, sensitivity, and limitations as dosimetry changed over the years. Major areas of external radiation hazards were addressed as the functions of the PGDP changed over time.

(2) The dose reconstructor generally uses the inhalation Absorption Type (F, M, or S) that gives the best outcome to the claimant.

5.0 VERTICAL ISSUES

This section presents a list and brief description of the issues identified with each TBD. No distinction is made regarding the relative importance of the issues. However, the more important issues are designated as findings in the Executive Summary.

5.1 SITE DESCRIPTION TBD

5.1.1 Issue 1: The Enrichment Facilities at PGDP are no longer under DOE Regulation

Although Section 2.1.1 of the Site Description TBD (Turpin 2006) states that "the purpose of the gaseous diffusion plant has been and continues to be the enrichment of uranium…", there is no indication in the site profile TBDs that since July 1, 1993, the United States Enrichment Corporation (USEC) has performed these operations at the site under the terms of a lease from DOE, and that those parts of the plant used by USEC are currently operated in compliance with NIOSH and NRC, rather than DOE regulations. The remaining portions of the plant are still under DOE control and regulation. DOE is responsible for DOE Material Storage Areas. Much of this material is stored in otherwise unused portions of buildings leased by USEC. The data reported in the site profile TBDs since 1993 reflect only the USEC portion of the site operations. DOE is responsible for environmental restoration and decommissioning at PGDP, and has been since July 1, 1993.

For some facilities, there may be both USEC and DOE activities, such as the storage of radioactive material in cascade buildings. This not only results in exposure to the DOE source term, but also the USEC source term. A methodology for addressing exposure to workers from both agencies should be clarified in the TBD. This is especially pertinent since the DOE and NRC implement different radiation protection regulations.

USEC anticipates ending its lease of the plant from the DOE at some time in the future. Prior to USEC taking possession of leased areas at PGDP, a major site characterization study was performed. This established a baseline for site contamination, such that USEC would not be held responsible for site contamination the DOE had created. Although SC&A is working with DOE to obtain this report, USEC has not released it to date. This document would provide valuable information on the existing radiological conditions in the environment, as well as the workplace.

5.1.2 Issue 2: The Enrichment Level at PGDP is no longer Limited to About 2.5%

Section 2.1.1 of Turpin (2006) states that PGDP enriched natural uranium up to about 2.5% ^{235}U. Footnote 1 on page 5 of the Occupational Internal Dose TBD (Berger 2004) indicates that the predominant level of the enriched product was 1.5%, although enrichments of up to 5% were eventually performed. The higher enrichment levels could provide the potential for increased radiation doses from ^{234}U and ^{235}U relative to those for ^{238}U.

5.1.3 Issue 3: The Site Map in the TBD is Inadequate

The site description in Turpin (2006) provides very little information in terms of layout of process buildings, office buildings, storage and waste areas, holding ponds, etc. Unfortunately, the site map (Figure 2-2) is also uninformative. The PACE/Utah 2000 report site map (see Figure 5.1-1) shows the relative locations of the principal buildings and also identifies those that are currently leased to USEC. A good site map could provide the dose reconstructor with information such as the location of a worker relative to a nearby exposure source that might otherwise be overlooked.

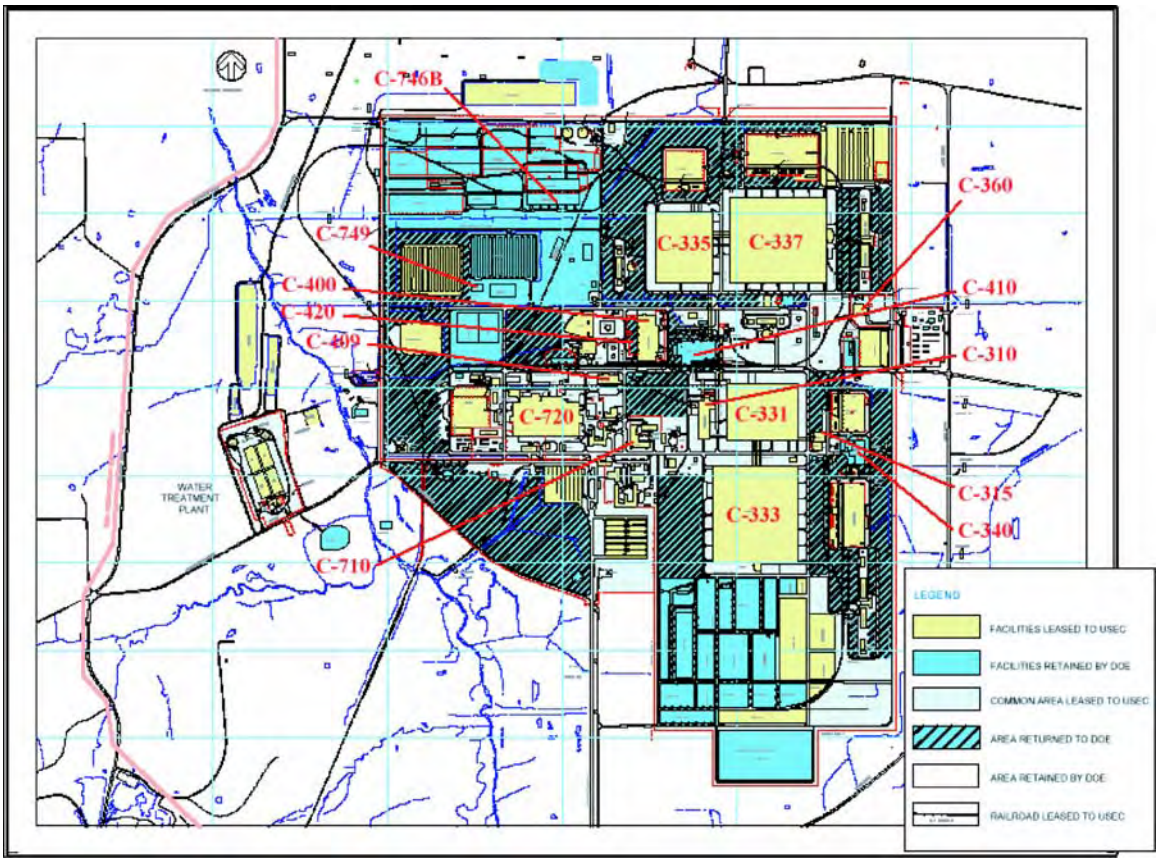


Figure 5.1-1. Schematic Drawing Showing Approximate Locations of PGDP Buildings
(Source: PACE/Utah 2000, Figure C-1)

5.1.4 Issue 4: The Material Processed to make Feed Material was not Just UO_3

The Site Description TBD identifies the material used to produce feed material as UO_3 (pp. 6 and 17). U_3O_8, which also was used in producing feed material (pg. 18), was also processed. Since U_3O_8 is classified as Type S for inhalation, while UO_3 is Type M, the distinction can be important in dose reconstruction.

5.1.5 Issue 5: Average External Doses are biased Low by the Use of Assigned Zero Values

The average recorded doses in Tables 2-1, 2-2 and 2-3 of Turpin 2006 are biased low and mean little without knowing the numbers of workers assigned dose values of zero when their measured dose was less than the MDL.

5.1.6 Issue 6: Incidents and Special Activities need to be Better Referenced

There are no references to reports on the specific incidents or special activities that could be used in reconstructing doses for individuals involved in them. The list of special incidents or activities on page 12 of the TBD is quite brief. Table 3.1 of the PACE/UTAH (2000) report (reproduced here as Table 5.1-1) gives a better chronology of events to be considered in dose reconstruction. The Phase II report (DOE 2000) identifies many incidents at the plant from the beginning of plant operations up to the early 1990s (PACE/Utah 2000). Are there any documents or databases where incident reports are maintained? If so, reference to those reports would be helpful.

5.1.7 Issue 7: The TBD Needs to Consider Operations Other than Gaseous Diffusion

There is no mention of the smelting operations that took place in Building C-746B. In addition to smelting, the diffusion barriers removed from PGDP and other enrichment facilities during cascade improvement and upgrade programs, and special smelting projects were also performed for other facilities. These activities could potentially affect worker doses.

PGDP participated in the "Work for Others" (WFO) program providing assistance to other DOE sites. There is no mention of this program in the TBD and how this may affect the dose reconstructions in terms of radionuclides. Materials and equipment were sent to PGDP from other DOE sites, such as the Savannah River Site (SRS), K-25, the Y-12 Plant, Hanford, Los Alamos National Laboratory (LANL), Lawrence Livermore National Laboratory (LLNL), Sandia National Laboratory (SNL), the Navy, and the National Aeronautics and Space Administration. For example, one such project involved the disassembly and subsequent recovery of precious metals from weapons components. Much of the WFO information is sensitive and would likely require a classified review.

5.1.8 Issue 8: The Intended Use of the Bechtel-Jacobs Radionuclide Concentrations is not Clear

Pages 11–20 of the TBD provide maximum ^{237}Np, ^{239}Pu, and ^{99}Tc concentrations expected for activities at each of the buildings related to the processing of recycled uranium. These relative mass concentrations (BJC 2001 does not associate specific radionuclides with Np or Pu) are associated with the types of exposures associated with worker activities in that building. The reference mass concentration in each case is that of the item shown in parentheses, e.g., MgF_2, ash, UF_6, etc. Bechtel-Jacobs (BJC 2001) uses these values, along with other data, to classify the radiation exposure potential for the worker activities. The TBD has misinterpreted the classification category as applying only to external radiation exposure potential. SC&A notes

that the use of these data for calculation of internal doses would not necessarily be consistent with the methods used in the Occupational Internal Dose TBD (Berger 2004).

Table 5.1-1. Overview of the PGDP Chronology with Emphasis on Events that may have had Some Radiological Consequences

(Source: PACE/Utah 2000, Table 3.1)

Date	Company	Event
Oct 1950	Union Carbide	PGDP Site selected
July 1952		Uranium received
Sept 1952		Cascade Buildings C-331 and C-333 begin operation
1953		Aware that feed from recycled reactor fuel contains trace quantities of plutonium
Sept 1953		2 ash handlers' urinalyses test positive for Pu
July 1954		Cascade buildings C-335 and C-337 begin operation
Aug 1954		First cascade improvement started, plant remains in operation
Aug 1956		C-420 feed plant online
Nov 1956		C-310 fire
Jan 1957		Begin producing U metal and UF4 from depleted uranium
1957		Radiochemical analysis confirms presence of neptunium and plutonium (separate radiochemical analysis indicates entire cascade is contaminated with Np)
Nov 1958		Neptunium recovery started from receiver ash and cylinder heals
Apr 1960		Technetium recovery begins
June 1961		First cascade improvement completed
Sept 1961		MgF2 traps installed to capture Np in effluents
Mar 1962		C-340 explosion and fire due to burnout of Mg bomb where uranium was released to the furnace
Mar 1962		Neptunium recovery ends
Dec 1962		C-337 Explosion and fire
Jan 1963		Technetium traps installed
June 1963		Technetium Recovery Ends
Apr 1968		Radiation overexposure to two maintenance workers*
Mar 1973		Second cascade improvement started
Oct 1973		No longer produces U metal; still produces UF_4 as a byproduct of HF Recovery
Jan 1975		NRC and ERDA assume AEC
Mid 1970's		Tc removed as an environmental protection measure
May 1977		Feed plant ceases operation, receive feed as UF6
Oct 1977		DOE assumes ERDA
Jan 1978		C-315 explosion and fire
Sept 1981		Second cascade improvement completed
Apr 1984	Martin Marietta	
June 1990		Tiger Team Assessment
1992		USEC established
July 1993	Lockheed Martin	
July 1993	USEC (United States Enrichment Corporation)	Leases enrichment production facilities as operations and maintenance contractors
Nov 1996		NRC grants certificate of compliance to USEC
Apr 1998	Bechtel-Jacobs	DOE clean-up contractor
May 1999	USEC	Assumes direct operation of enrichment facilities

5.2 OCCUPATIONAL MEDICAL DOSE

Background & Introduction

The PGDP TBD for Occupational Medical Dose, ORAUT-TKBS-0019-3 (Turner 2004, Section 3), makes reference to occupational dose guidelines in Kathren 2003 as its basis document for estimating medical dose. Kathren 2003 has since been revised as *Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures*, Revision 3 (Kathren and Shockley 2005), which provides a more reasonable basis for assumptions regarding estimation of worker medical exposures at Mound. The SC&A review evidences that the TBD recognizes the total lack of exposure data and protocols existed prior to 1975. In Table 3-3, the TBD details technique factors for x-rays prior to 1975, which are based solely on assumptions with no documentation provided. NIOSH did conduct interviews with PGDP Medical staff to ascertain any personal knowledge on x-ray procedures.

Specific dose estimations in the TBD for occupational x-rays are also derived for the period prior to 1975 from ICRP Report 34 (ICRP 1982) and NCRP Report 102 (NCRP 1989). This is an important issue in that estimated medical doses based upon this research are not actual measurements of dose from the PGDP x-ray unit. For the period of 1995–present, it is stated that there were measurements made by the State of Kentucky and the Food and Drug Administration (FDA); however, no references are provided.

The TBD states there is no physical evidence to show if photofluorography (PFG) to do chest screenings occurred at PGDP. In the absence of documentation, the TBD directs that dose assessors not include the use of PFGs in their dose estimations. This direction contradicts OTIB-0006, which recommends assuming annual fluoroscopic examinations in the early years lacking evidence to the contrary.

Additionally, the TBD does state that all medical x-rays administered in conjunction with routine or special exams are considered as part of the occupational exposure; however, only pre-employment and routine chest exams are mentioned to be used by dose assessors. Review of the subject TBD has resulted in a number of issues and supportive observations, which are important to NIOSH to consider and clarify, due to the risk inherent to medical exposures, to assure that considerations are claimant favorable. These issues and observations are discussed below.

5.2.1 Issue 1: The TBD fails to Adequately Define and Assess Occupational Medical Exposure

The guidelines as presented in ORAUT-OTIB-0006, Revision 2 (Kathren 2003), go a long way in assuring that all occupational medical exposures are reasonably included in determining the overall dose estimations for claimants. Unfortunately, the interpretation to date by the contractor (ORAU) has not been applied conservatively to be claimant favorable. The Occupational Medical Dose TBD (Turner 2004) assumes an interpretation that has been also considered and applied at other sites, such as the Mound Plant and Los Alamos National Laboratory (LANL). To this extent, the assumption that medical procedures are limited to only one pre-employment

chest x-ray and chest x-rays that are part of routine physical exams, may substantially underestimate worker medical exposure, when evaluating occupational medical exposure.

In more recent documentation, Revision 3 of ORAUT-OTIB-0006 (Kathren and Shockley 2005) concluded that other examinations should be included, such as special screening exams (e.g., respiratory protection, beryllium workers, asbestos workers, etc.) and termination exams. The occupational medical TBD does not recognize this change from the previous Revision 2 of the TIB, and also assumes that special chest radiography for respirator certification, beryllium and asbestos workers, and food handlers are accomplished as part of the routine physicals. This is not documented in the medical TBD. Another factor not discussed in the TBD is the potential impact of x-ray procedures utilized by medical authorities to do special screenings that are performed outside the frequency listed in Table 3-1 of the TBD.

The TBD (Turner 2004) concludes that chest examinations are often quite limited after 1974. Only smokers are documented in Table 3-1 of the TBD as having received an exam every 3 years. The only differential noted is that asbestos workers after 1986 had a chest exam performed every 2 years. To the contrary, there is ample evidence that annual chest x-rays were often provided on a voluntary basis to nearly all workers. The majority of workers had chest x-rays each year as a routine at DOE sites until the mid-1980s, when Federal guidelines warning against routine screening were first being enforced.

After discussion with NIOSH personnel, it was their decision to limit occupational medical exposure to those chest exams described above, and to conclude that all other exposures are part of worker background. SC&A believes such an interpretation is not claimant favorable to those most at risk. Our concern is that specified "high-risk" workers, those most likely exposed to radiation and beryllium, would be at risk of having an incomplete dose assessment if not all radiation associated with medical screening for job-related activities were included. Since all radiation provides some risk and, arguably, is cumulative, workers warrant consideration of all forms of work-related x-ray exposure to be claimant favorable. SC&A believes NIOSH should review its interpretation of included medical exposure, and should reasonably adopt a broader interpretation of occupational medical dose, as provided in the most recent version of ORAUT-OTIB-0006 (Kathren and Shockley 2005). This is particularly important for skin cancer, which is not one of the SEC-defined cancers.

5.2.2 Issue 2: Techniques and Protocols Increase Uncertainty of DCFs listed in the TBD

Sections 3.2 and 3.3 of the TBD (Turner 2004) fail to describe information upon which to establish beam quality for x-ray units in use from 1952–1975. In 1975, the site installed a single-phase Picker unit. There is only limited documentation to show that the Picker unit, in use from 1975 through 1995, had added filtration, as first measured by the FDA and the State of Kentucky after 1995. In the absence of definitive tube output measurements, the TBD directs the use of default values and dose conversion factors (DCFs) derived from ICRP Report No. 34 (ICRP 1982). These values are then applied to determine organ doses using Tables A.2 through A.8 of ICRP Report No. 34 (ICRP 1982). An issue of concern is that the DCFs are derived using a default half value layer (HVL) of 2.5 mm aluminum (Al) for Type 1 units, in use from 1946–

1980. Another issue is that the unit, in use from 1975–1995, had an unknown added filtration of Al, which may be different from the default value from NCRP Report 102 (NCRP 1989).

The Occupational Medical Dose TBD (Turner 2004) provides little documentation to support the assumed techniques and protocols applied to calculate the dose, which is mainly derived from NCRP Report 102. The TBD states that a posterior-anterior (PA) chest x-ray was typically the only view. An undocumented assumption in the TBD is that exams required only a PA view. SC&A has inquired whether a definitive protocol existed to validate that chest exams included PA views and lateral (LAT) views only on a limited basis after 1975. NIOSH has acknowledged that the lack of verifiable protocols is a generic problem at many sites, has planned to search all available records, and will include pertinent records and references in any future revision of this section of the TBD.

The Occupational Medical Dose TBD is also deficient in that little documentation exists to validate x-ray protocols and equipment maintenance, and upkeep of records prior to 1995, after which the State of Kentucky commenced surveys of the unit. The TBD uses information derived from the TIB (ORAU 2003) to estimate dose impacts.

5.2.3 Issue 3: Frequency and Type of X-ray Exposure Is Uncertain

The Occupational Medical Dose TBD relies on a very limited review of current medical personnel to establish frequency assumptions. The assumption of one chest radiograph (PA) every 3 to 5 years is not reasonably conservative, in that workers could request an x-ray, or be subject to special screening exams. The frequency of screenings, and the number and type of workers receiving x-rays varies from site to site.

The Occupational Medical Dose TBD in Section 3.2 provides no documentation or references to support the assumption that only a limited group of workers received x-ray exams more frequently than every 5 years after 1952. To the contrary, up until about 1985, most DOE sites performed chest x-rays almost on a voluntary basis. DOE medical program reviews documented during the early 1990s showed many sites still used chest radiography as a general screening exam. Most workers accepted chest x-rays, even though the job did not require it. Also, the assumption that workers in special exposure categories, such as beryllium workers, were given chest x-rays only as part of their routine physical is not well-documented and not consistent with special screening guidelines. The TBD applies no conservative assumption to cover such exams.

Section 3.2 of the Occupational Medical Dose TBD states that PFG units, although generally available at most DOE sites up to the late 1950s, were not used at the PGDP. The undocumented absence of PFG units at PGDP clearly has significant dose implications to workers who may have been given much higher doses from PFG units. The PFG unit provides a dose to the worker greater by a factor of 5–6 more than that delivered by conventional radiography. The TBD does not provide documentation for the types of equipment in use at PGDP prior to 1995. SC&A believes it is not claimant favorable to instruct dose assessors to use kerma (dose) values of 200 mrem and 100 mrem for chest radiography prior to 1975. To be fully claimant favorable, it would be appropriate to instruct dose assessors to use an annual dose of 3.0 rem per year for

chest radiographs, in accordance with guidelines set forth (Kathren and Shockley 2005) until 1959, when you can assume all PFGs were no longer used at DOE sites.

5.2.4 Issue 4: Other Potential Medical Exposures have not been Identified

The Occupational Medical Dose TBD (Turner 2004) does not address the potential use of forms of radiation exposure other than x-ray units to support medical diagnosis. This may involve use of isotopes, sealed sources, etc. The TBD is also deficient in that it does little to catalog the number, types of x-ray equipment, frequency of use, etc., discussed above in Issues 2 and 3.

The below-average performance at the PGDP to perform routine and preventative maintenance during the 1952–1975 timeframe suggests that routine maintenance of x-ray units is not likely, unless performed by an unknown outside contractor. Unfortunately, no records exist to evidence maintenance, calibrations, etc. The lack of defined protocols and basis for approval of radiography procedures suggests that the use of radiography was not closely controlled. The Occupational Medical Dose TBD does not discuss the use of portable radiography to perform screenings and the potential for exposure of medical personnel or other workers without dosimetry devices being utilized. Interviews with medical staff suggest that portable x-ray units were not used. This is potentially an issue for the PFG unit, which was often van-mounted at other sites. Additionally, the TBD fails to document that available x-ray units were not operated at greater than 80–90 kVp prior to 1975. To the contrary, Table 3-3 of the TBD indicates that the kVp after 1975 was set at 100 kVp when performing LAT chest exams.

The conclusion is that the TBD does little to reasonably document the variety of medical occupational exposures, and the lack of documentation on the type of equipment and the maintenance records do little to assure that a conservative and claimant-favorable estimation of dose is possible. This circumstance would suggest the need to reconsider a worst-case approach to establishing dose. NIOSH should revisit and update Sections 3.2 and 3.3 of the TBD as needed.

5.2.5 Issue 5: There are Additional Factors that Contribute to Uncertainties

The Occupational Medical Dose TBD does not consider dose impacts due to less-than-optimal use of technology, such as using screens, grids, or bucky systems. The TBD does not consider these elements as potential contributions to uncertainty.

The TBD does consider the potential contribution to dose that may have resulted in less than optimal use of collimation, at least prior to 1975, as stated in Section 3.5 of the TBD, and offers substitute DCFs for use by dose reconstructors for selected exams, as presented in Tables 3-6 and 3-7. Unresolved is the concern that the DCFs are derived from ICRP (1982), and therefore, are not comparable in terms of beam quality, which varies from unit to unit. These factors can contribute greatly to the dose to the chest and other organs for units in operation prior to 1995, where little or no documentation exists. NIOSH has indicated that it will continue to search for other available records to better define equipment use and beam quality, and include it in an updated version of the TBD, as appropriate.

Uncertainty is defined in the TBD as being due to measurement error, and variation in kilovoltage, tube current, timers, and the source-to-surface distance (SSD). This approach is quite similar to the uncertainty analyses documented in other DOE site profiles. The conclusion in this TBD, and others, is that an uncertainty factor of +30% should be used by dose reconstructors for exposure prior to 1975. SC&A believes the correction factor of 2.0, being applied after 1975, is more appropriate.

SC&A agrees that the TBD conservatively estimates these essential aspects of an uncertainty review. Unresolved is the contribution to uncertainty in dose, due to other errors introduced by lack of quality controls in processing equipment and lack of adherence to established SOPs. A reasonable estimate of these contributions to uncertainty would be an evaluation of retake rates per examination type. NIOSH should revisit the potential for significant retake rates and evaluate its potential effect on dose as part of future revisions of this TBD, especially as it relates to examinations prior to 1975.

The Occupational Medical Dose TBD does not show that PGDP applied dose minimization principles to reduce medical exposures. The document also does not assess or consider the likely exposure to workers who are referred to off-site medical facilities for follow-up. The TBD states that review of selected medical records and files did not reasonably show or match expected x-ray exam frequency, and type of exam. Little evidence exists to document the number of x-ray exams provided to the average worker, or for special exposure needs.

5.3 OCCUPATIONAL ENVIRONMENTAL DOSE

Background & Introduction

The TBD for the PGDP Occupational Site Description, ORAUT-TKBS-0019-4 (East 2004), describes the potential exposures from ambient sources to workers while outside the process buildings. The document presents data for estimating annual intakes from radionuclides in air and external dose from ambient radiation, as well as from cylinder storage yards.

The document does not present radionuclide-specific measurements. Uranium concentrations are inferred from measurements of alpha concentrations in air and Tc-99 concentrations from beta concentration in air measurements. Until September 1981, external gamma radiation levels were monitored with Geiger Mueller counters. Since then, thermoluminescent dosimeters (TLDs) have been used for that purpose. The TBD provides references to numerous environmental reports for the site.

5.3.1 Issue 1: Onsite Environmental Exposures are based on Site Boundary Data

The TBD purports to describe "potential exposures from ambient sources while working outside the process buildings," but the ambient monitoring data in the document are for measurements at the site boundaries and beyond. There are no corroborating data provided to demonstrate that these measurements are consistent with the levels that workers might experience while working at the site outside the process buildings.

5.3.2 Issue 2: There are No Comparisons between Predicted and Measured Environmental Exposures

The TBD states that since 1962, "At PGDP all personnel wore film badges…." This document is supposed to provide information for estimating environmental doses when monitoring was not performed or coworker data could not be used, but there are no comparisons between doses based on personnel badge data with estimates based on ambient environmental exposures. Such comparisons would help to validate the methods for reconstructing environmental doses.

5.3.3 Issue 3: The Pre-operational Measurements could not have detected Ambient Levels of Uranium in Air

East (2004) states that none of the 10 pre-operational samples analyzed in 1952 showed any measurable concentration of uranium in air (0.00 mg/m^3), and only one of the two alpha activity samples showed any measurable alpha activity in air (0.018 cpm/ft^3). There are no comparisons with other ambient measurements. ATSDR (1999) published a table of uranium activity in air values measured in 1984 and 1993 at over 30 sites. The total uranium activities ($^{234}U + ^{235}U + ^{238}U$) in these measurements range from about 4.6 aCi/m^3 (Syracuse, New York, 1984) to 341.4 aCi/m^3 (Lynchburg, Virginia, 1984). An alpha concentration of 0.018 cpm/ft^3 (assuming a 100% counting efficiency) would correspond to about 290,000 aCi/m^3 if all the alpha radiation came from ^{nat}U—about 850 times the Lynchburg measurement. Clearly, the one positive alpha activity cannot be interpreted as due to the pre-operational concentrations of uranium in air. On the other hand, if the MDL for uranium mass concentration measurement was about 0.01 mg/m^3 (10 μg/m^3), that would correspond to a ^{nat}U activity concentration of about 6.9 pCi/m^3, a level about 20,000 times higher than the Lynchburg measurement. In conclusion, it would appear that neither of the methodologies used for determining pre-operational levels of uranium in air at the site was adequate for that purpose.

5.3.4 Issue 4: The Ambient Air Sampling Collection Network was not Intended for Monitoring Onsite Concentrations

The four on-site air monitoring stations located at the security fence (according to Table 4-1 of the TBD) were primarily intended for demonstrating compliance with offsite air quality criteria. There is no map showing the locations of the air sampling stations in the network.

5.3.5 Issue 5: Environmental Dose Gaps Exist in the TBD

It is not clear whether or not air transport modeling was performed for the site. There are a number of considerations identified that limit the use of air transport modeling to estimate airborne concentrations from site releases, but there are no comparisons made between the site boundary data and any such modeling that had been done.

There is little characterization of the releases in Tables 4-2 and 4-3. There is no discussion of what release points were considered or the physical form of the uranium and ^{99}Tc release data in Tables 4-2 and 4-3, other than to indicate that they were obtained from DOE, Bechtel-Jacobs, and United States Enrichment Corporation reports.

Section 4.3, External Dose, does not consider the radionuclides in the depleted uranium cylinders providing the gamma radiation for the increasing external exposure rates near the depleted uranium cylinder storage yards. SC&A questions the use of TLDs as environmental dosimeters starting in 1961. That year seems rather early for the use of these TLDs. Early survey results of about 0.02 mrem/hr (probably measured as 20 μR/hr) would correspond to 40 mrem in a 2,000-hr period or about 175 mrem/yr for continuous exposure. For comparison, the average of the 1993–2001 reported background measurements in Table 4-4 correspond to a 100 mrem/yr rate for continuous exposure. Presumably, the earlier survey was in an era when fallout from weapons testing would have contributed significantly to ambient levels.

Section 4.4, Uncertainty, deals less with estimating the uncertainties in measured or calculated values than with outlining default values and assumptions to be used when data are unavailable.

5.3.6 Issue 6: Comments on Section 4.3.2, Radiological Conditions in Cylinder Storage Yards

(1) The statement that dose rate measurements at 100 m from the UF_6 storage yards are representative of dose rates near the storage yards is not supported with measurements. It would seem likely that dose rates could increase substantially as the cylinders are approached.

(2) The statement, "During recent years this area has been posted as a radiological area, which has reduced the number of unmonitored workers spending any significant time in the area to zero," raises some questions:
   a. What were the criteria for the designation?
   b. Given the 1962 policy that all workers were to wear dosimetry at all times, who were the unmonitored workers this policy was designed to protect? Were they workers whose badges were not normally analyzed?

5.3.7 Issue 7: There is Little Characterization of the Effect of Plant Releases on the Onsite Levels of Exposure

There is no discussion in the TBD of radionuclide-specific concentrations in air, soil, or water. There is no discussion of the environmental fate of uranium and other radionuclide releases. In effect, there is an implicit a priori assumption that the dosimetry systems in use at PGDP will adequately reflect any environmental exposures. Two additional potential sources of ambient exposure include the burning of contaminated material in onsite pits and the alleged intentional releases that occurred in the 1950s. Further investigation into these areas is necessary.

5.4 OCCUPATIONAL INTERNAL DOSE

We have reviewed the *Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose* (Berger 2004), as well as *Internal Dosimetry Coworker Data for Paducah Gaseous Diffusion Plant* (Ikenberry 2005), and have identified a number of key issues,

which are presented below. The next part of this section presents a detailed discussion of these issues, keyed to the sections and tables in the TBD.[2]

5.4.1 Key Issues

Issue 1: Completeness of Data Sources—Inadequate Characterization of the Source Term

A critical issue is the specification of the source term for the internal exposures. This TBD does not fully utilize information presented in two key documents. One is the exposure assessment of workers at the PGDP (PACE/Utah 2000), which emphasized exposures to neptunium and plutonium. Although this report is cited as a source of information on a few narrow topics, the extensive data summarizing the results of radiological surveys were not incorporated in developing the default specific activities, which are needed for reconstructing the doses to workers whose bioassay results were limited to total concentrations of uranium.

The other document is the draft report on recycled uranium mass balance (BJC 2001), which is cited as a source of information by Turpin (2006), but is not mentioned in this TBD. That report breaks down the potential exposures to three contaminants in recycled uranium—^{99}Tc, ^{237}Np, and plutonium—into 11 types of activities and operations at the PGDP, and lists the maximum concentrations of each of these constituents (BJC 2001, Table 2.4-1). Other data, such as those in DOE 2000, are erroneously characterized.

Issue 2: Inconsistencies and Inaccuracies in the Source Term

The TBD is inconsistent in assigning radionuclides to the various facilities at the PGDP. Notable among these are the omission of transuranics (TRU) in the feed plant and the "cascades (general)" facilities.

The default specific activities of nuclides other than ^{235}U and ^{238}U have been underestimated by as much as 5 orders of magnitude. Prior to 1989, most urinalyses in the electronic database only list the chemical assays of uranium. Information on the specific activities of the uranium isotopes is therefore required to estimate intakes of the activities of each isotope, which are needed for dose reconstruction. The specific activities of other radionuclides (i.e., activity of each nuclide per unit mass of uranium) are required to estimate the intakes of these nuclides, based on the mass concentration or daily excretion of uranium in urine.

A more detailed discussion of issues involving the source term is presented in Section 5.4.2 of this review entitled "Section 5.2: 'Source Term'."

Issue 3: Particle Size and Lung Clearance Types

The TBD assumes all respirable aerosols have an AMAD of 5 μm, despite evidence of smaller particle sizes. Such an assignment is, in most cases, not claimant favorable. Incorrect Lung

[2] In the remainder of this section, the term "TBD" is used to refer to the "Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose," ORAUT-TKBS-0019-5.

Clearance Types are assigned to the various nuclides in many facilities. The assigned types are, in many cases, not claimant favorable.

Issue 4: Inadequate and/or Inaccurate Guidance on Estimating Intakes from Bioassay Results

The TBD presents inadequate, and sometimes inaccurate, guidance to dose reconstructors on estimating the intakes of uranium, including the assumed specific activities of other radionuclides not analyzed by the bioassays. The issues include making claimant-favorable assumptions about the time of collection of urine samples with respect to the work week, and the utilization of "recall" samples in cases the original assay indicated a high intake of radioactive materials. A more detailed discussion of issues involving the estimation of intakes is found in Sections 5.4.3 and 5.4.4 of this review entitled "Section 5.3: 'In-Vitro' Measurement Methods" and "Section 5.4: 'In-Vivo' Measurement Methods," respectively.

Issue 5: Erroneous Equations for Estimating 24-hour Excretion

The equations for estimating 24-hour excretion on the basis of spot urine samples are incorrectly written. There is an error in the presentation that can lead to errors in the interpretation and application of these methods. Furthermore, the daily excretion of urine is based on obsolete data.

Issue 6: Invalid Guidance for Estimating Intakes Based on Time of Sample Collection

The guidance for utilizing the time of sample collection in dose reconstruction, or assuming a time if none is listed in the records, is neither scientifically valid nor claimant favorable. Large variability in the calculated intakes can result if the time of collection following exposure is not properly accounted for.

Issue 7: Omission of Guidance on Use of Breath and Fecal Analyses

Limited fecal analyses, and possibly breath analyses, were performed at PGDP. The TBD provides no information on such bioassays.

Issue 8: Inadequate and Internally Inconsistent Discussion of In-Vivo Measurements

Although in-vivo bioassays were not widely performed at the PGDP, the TBD should nevertheless present adequate information to allow dose reconstructors to properly utilize any data that may be available. The discussion of in-vivo measurements and MDAs is inadequate and internally inconsistent.

Issue 9: Inadequate Discussion of Significant Incidents and Accidents

The information on significant incidents and accidents is extremely sketchy and of little use to dose reconstructors.

Issue 10: Unsuitable Use of Coworker Models

According to a more general guidance document on the use of coworker bioassay data (Brackett 2005), the basis for the application of a general coworker model is as follows:

> *... participation in a bioassay program involves workers who have the largest potential for exposure. While there are exceptions to this generality such as accidents involving unmonitored workers, it is unlikely that an unmonitored worker would have received a larger dose than the most highly exposed monitored worker at a site.*

However, Brackett provides no statistical or quantitative technical basis for this hypothesis.

In the coworker models used by NIOSH, workers are not classified by their jobs or by the buildings where they performed their work. The model described by Ikenberry does not identify jobs that may have led to a higher probability of intake by workers, either in routine or special circumstances. Air sampling data from the PGDP show differences in concentrations of uranium dust in the air in the various buildings, which would lead us to expect that workers in buildings that had higher concentrations of uranium in the air would have higher concentrations in the urine. However, there was no attempt in this report to sort the urinalysis results by job assignment or location to determine if there were any correlations between the uranium concentration in the urine and the building or department where the job was performed.

In order to apply the coworker model to unmonitored workers, NIOSH needs to demonstrate that there is a low probability that any unmonitored worker could have higher exposures than the monitored workers taken as a group. Ikenberry fails to do so.

In fact, there is evidence that the monitoring program was designed to encompass all workers. According to PGDP (2003a):

> *The frequency of such industrial urine samples for uranium varies from a maximum frequency of each four weeks for all personnel working routinely in C-310, C-315, C-340, C-400 and C-4l0 to a minimum frequency of each 12 months for those working in some locations where there is little likelihood of any exposure to this element or its compounds.*

The above statement negates the assumption that the unmonitored workers had lower probabilities of radiation exposures. If all workers were supposed to be monitored, unmonitored workers might well be those whose records were lost or who, for some unknown reason, did not participate in the program.

Issue 11: Use of Unverified Bioassay Data

Dodd (2002) states the following:

> *The data that is included in the "PGDP_Historical_Internal_Data_(Unverified).mdb" database consists of in-vivo data and urine card data. This [sic] data has not been through a verification and validation process; therefore it should not be considered data of record.*

This caveat raises a question about the use of the cited database, which is the basis of bioassay data for the 1952–1976 period. This issue needs to be addressed by NIOSH before such data can be used for the coworker model—in fact, before they can be used for any dose reconstruction.

Issue 12: Incorrect Calculation of Intakes from Mass Concentrations of Uranium in Urine

According to Ikenberry:

> *The total uranium results are in units of µg/L; therefore the results were multiplied by 1.4 in order to normalize them to the Reference Man excretion rate of 1400 mL per day. Bioassay results were converted from mass to activity before fitting assuming 0.0389 Bq/µg, characteristic of low-enrichment (2 percent) uranium.*

The current ICRP (2002) reference values are 1.6 L/d for an adult man and 1.2 L/d for an adult woman. Therefore, the TBD should be revised to reflect these values. Furthermore, the default specific activity of uranium, based on the measured concentrations of uranium isotopes in the PGDP cascade product, should be 0.0541 Bq/µg, instead of 0.0389 Bq/µg, as cited by Ikenberry.

Issue 13: Lack of Documentation for Choice of Distribution Type

Brackett (2005) describes the statistical analyses for developing a site-specific coworker bioassay model. These include calculating the goodness-of-fit to determine whether the bioassay results conform to a lognormal distribution. Finally, according to Brackett:

> *All decisions regarding the statistical analysis are recorded in instructions to the statistics analyst and in a site-specific document such as the site profile or a TIB.*

Ikenberry states that the urinalysis results for individual calendar-year quarters have lognormal distributions. However, he cites no statistical results that support this conclusion; furthermore, the TIB fails to indicate whether any such statistical analyses were performed. Absent such data, the compilation of 50^{th} and 84^{th} percentiles have little meaning.

Issue 14: Incorrect Selection of Distinct Time Periods

According to Ikenberry:

> *The analysis started a new chronic intake period if the data indicated a significant sustained change in the bioassay results. By this method, the period from 1952 through 1988 was divided into two chronic intake periods.*

The author gives no detailed explanation of how the division into two time periods was accomplished. Our own examination of the data, followed by a statistical analysis, finds that there are three distinct time periods, as shown in Table 5.4-1 of this review.

Table 5.4-1. Urinalysis Results (Bq/d) for Three Distinct Time Periods

Time Period		**6/1/1952 – 3/31/1959**	**4/1/1959 – 3/31/1980**	**4/1/1980 – 12/31/1988**
Number of quarterly results		**24**	**84**	**35**
50^{th} percentile	Mean	0.398	0.282	0.084
	S.D.	0.084	0.078	0.023
	S.E.	0.017	0.009	0.004
84^{th} percentile	Mean	1.212	0.744	0.256
	S.D.	0.521	0.197	0.064
	S.E.	0.106	0.022	0.011

As shown on the table, the three time periods exhibit mean values of both the 50^{th} and 84^{th} percentiles that are significantly different from each other, as shown by the standard error of each mean, which is far smaller than the difference between the means. Thus, the three time periods encompass three distinct sets of values.

Issue 15: Coworker Data Inapplicable to Individual Unmonitored Workers

The applicability of the coworker data presented by Ikenberry to individual unmonitored workers is questionable on several grounds. The models presented by the author represent a best estimate of the constant, chronic intake for each of two time periods by a hypothetical individual whose quarterly urinalyses results correspond to the median values of all workers for a given quarter. A similar estimate is made for a hypothetical worker whose urinalysis results correspond to the 84^{th} percentile values for each quarter. There are three sets of values—one for each uranium Lung Clearance Type; F, M, and S. A distribution of uranium intakes for each time period and each Lung Clearance Type was constructed. The intake of the hypothetical median individual was assigned to the median value of this distribution, while the ratio of the intake of the hypothetical 84^{th} percentile individual to that of the median individual was assigned to the geometric standard deviation (GSD). If the latter value is less than 3, the GSD is assigned a value of 3.

While this distribution might well represent the intakes of the *population* of monitored workers, there are several reasons why it cannot be used as a surrogate for the intake of an unmonitored worker:

- The data for this distribution includes assays of workers with very low potentials for exposure. According to PGDP (2003a, see excerpt quoted previously), all workers were to submit urine samples at least once every 12 months. Consequently, the distribution would not be claimant favorable for workers with job assignments that would have subjected them to significant radiation exposures.

- The model assumes a constant rate of intake; however, the plots of the 50th percentile excretion rate of uranium show considerable variability among the quarters within a given time period. The 84th percentile rates are even more variable. Thus, the assumption of a steady intake is not supported by the data.

- The intake rate is based on a steady exposure during the entire time period. It is unlikely that a given unmonitored worker would have been continually exposed during, say, the period from June 1, 1952, to March 31, 1980—a period of 28 years.

- As pointed out earlier in this review, urinalyses during the period from June 1, 1952, to March 31, 1959, had significantly higher values than those between April 1, 1959, and March 31, 1980, yet they were combined into a single period. A worker exposed during the earlier period would thus be assigned a lower intake rate than had the rates for the three periods been calculated separately.

In conclusion, the lognormal distributions described by Ikenberry are artificial constructs that are not based on any real individuals. The application of such distributions to unmonitored workers is neither scientifically valid nor claimant favorable.

The remainder of Section 5.4 contains a detailed discussion of the TBD, keyed to the sections and tables in that document.

5.4.2 Section 5.2: "Source Term"

The first paragraph of Section 5.2: "Source Term," singles out ^{230}Th and ^{234m}Pa as uranium progeny "of dosimetric interest." This designation of ^{230}Th is apparently an error; DOE (2000) cites ^{234}Th and ^{234m}Pa, the short-lived progeny of ^{238}U. According to DOE, these nuclides are significant, because when the uranium is further processed, significant quantities of these daughter products can remain behind in the form of oxides or ash or on the surface of process vessels. Most of the beta dose from uranium is, in fact, due to these two radionuclides. Consequently, high concentrations of the nuclides would produce much higher fluxes of beta rays than would emanate from unprocessed uranium.

5.4.2.1 Table 5-1: "Elemental Fractions of Uranium, Neptunium, and Plutonium at Specific Facilities"

Table 5-1 of the TBD lists activity fractions of uranium, neptunium, and plutonium at several facilities of the PGDP; the report recommends that these values be used by dose reconstructors. First, as the author acknowledges in footnote "a" to this table, the title, "Elemental fractions of uranium, neptunium, and plutonium at specific facilities," is misleading. Despite how the data appears in the source document, it would be better to use the term "activity fractions" rather than "elemental fractions," if this is indeed what the values represent.

More important, the fractions of neptunium and plutonium at the feed plant (Buildings C-410 and C-420) and at "Cascade (general)"[3] are listed as zero. This ignores the data in Table 7.9 of PACE/Utah 2000, which lists activity fractions of 0.7% ^{237}Np and 9% ^{239}Pu in the C-410 control room and fluorination tower air concentrations, and lower fractions in other feed plant locations. It is also inconsistent with Table 5-5 of the TBD, which lists ^{237}Np and ^{239}Pu in Buildings C-410 and C-420. These activity concentrations can have large dosimetric consequences, given that the 50-y dose to a non-metabolic organ (e.g., testes) from unit intakes of Type M ^{237}Np or ^{239}Pu are more than two orders of magnitude greater than the comparable dose from ^{238}U.

Table 5-1 does indicate neptunium and plutonium activities in Buildings C-333 and C-337, which housed portions of the cascades. However, there is no mention of Buildings. C-331 and C-335, which also contain parts of the cascades. This is again inconsistent with Table 5-5, which lists ^{237}Np and ^{239}Pu in Buildings C-331 and C-335.

5.4.2.2 Table 5-2: "Isotopic Fractions for Various Enrichment Percentages"

The TBD advises dose reconstructors to use Table 5-2, which lists "isotopic fractions" for various uranium enrichment percentages, to determine the amount of each uranium isotope when bioassay results are listed in terms of the mass of uranium. First, the table is mislabeled; the values are specific activities, not fractions. Next, an examination of these data reveals a number of discrepancies. The most glaring one is the specific activity of ^{235}U in 93% enriched feed. A direct calculation yields an activity of 7.44E-2 Bq/μg, a factor of 10 higher than the listed value of 7.38E-3 Bq/μg.

Smith (1984, Appendix 13) tabulated the measured concentrations of ^{234}U, ^{235}U, and ^{236}U in the PGDP cascade product and tails during the period October 2, 1955, to December 31, 1982. Using these data, we calculated the mean and maximum concentrations of these isotopes in 80 cylinders of cascade product withdrawn during this period, along with the corresponding specific activities. These results are listed in Table 5.4-2, together with the default specific activities from Table 5-2 of the TBD.

[3] Some further explanation is required of the term "Cascade (general)."

Table 5.4-2. Concentrations of Uranium Isotopes in the PGDP Cascade Product

Isotope	Mean Concentration		Maximum Concentration		TBD Default
	Wt %	Bq/µg	Wt %	Bq/µg	Bq/µg
U-234	0.0100	2.26E-02	0.0172	3.89E-02	2.52E-02
U-235	1.27	9.99E-04	1.95	1.54E-03	1.58E-03
U-236	0.0072	1.71E-04	0.0625	1.48E-03	3.33E-05
U-238[a]	98.72	1.23E-02	97.97	1.22E-02	1.21E-02
Total	100	3.61E-02	100	5.41E-02	3.89E-02

Source: Smith (1984, Appendix 13)
Note: specific activities are in terms of total uranium
[a] Weight percent of ^{238}U calculated by subtracting concentrations of other isotopes from 100%

When compared to the measured concentrations at the PGDP, the default specific activities of ^{234}U and ^{236}U, listed in Table 5-2, are not claimant favorable. The default specific activity of ^{234}U, while higher than the mean activity, is only 65% of the maximum measured activity, while the default ^{236}U activity is much lower than the mean of the measured values. Absent knowledge of the specific activities or detailed information on the form of uranium dust to which a worker was exposed, NIOSH should assume the maximum measured concentrations listed by Smith, as shown in Table 5.4-2, above.

5.4.2.3 Table 5-3: "Transuranic Element Concentrations in Plant Aerosols (1987)"

Table 5-3 of the TBD, which lists concentrations of ^{230}Th, ^{234}U, ^{238}U, ^{237}Np, and ^{239}Pu in dust samples, is erroneously captioned "Transuranic element concentrations in plant aerosols," although only two of the five radionuclides listed are TRU. The specific activities are listed in units of dpm/g U and "nCi/µg U." The latter heading is incorrect; the units are actually nCi/g U. Although this is obviously an editing error, it results in the values being overstated by a factor of one million. The values in this table are apparently taken from Baker (1987). In fact, the attribution "(1987)" in the title is another editing error—it should read "(Baker 1987)."

Baker (1987, pg. 6) lists radionuclide concentrations in the UO_3 powder area and the Building C-420 Green Salt Plant in several different units. There is an inconsistency in the ^{230}Th concentration values. The concentration is listed as 1 ppb, which is equivalent to about 44 dpm/g U, which in turn is equal to a fraction of 2.9×10^{-5} of the total α activity in the dust. Baker lists a value of 2.8×10^{-5}, which is in good agreement with the value cited above. However, he lists a specific activity of 42,000 dpm/g U, which is inconsistent with the other two values. This is confirmed by subtracting the activities of ^{234}U, ^{238}U, ^{237}Np, and ^{239}Pu, which are reproduced in Table 5-3 of the TBD, from the total activity of 1,500,818 dpm/g U listed by Baker. The difference, which is the ^{230}Th activity, is 48 dpm. Thus, the value of 42,000 dpm is erroneous, an error that is propagated in Table 5-3. Although this may result in higher doses, it is not correct.

The concentrations in the other areas listed by Baker appear to be internally consistent and are correctly reproduced in Table 5-3. However, the table omits information on activities in airborne dust at four additional locations or work areas listed by PACE/Utah (2000, Table 7.9); ash

receivers, pulverizer, Building C-400 converter salvage line, and converter maintenance. Two of these facilities—ash receivers and pulverizer—have the highest fractions of ^{230}Th and ^{239}Pu in the dust.

Finally, the TBD provides no guidance as to how the data on air concentrations can be used in dose reconstruction. Although we agree with the primary reliance on urinalysis, supplemented by in-vivo measurements, where such records are available, measured air concentrations could be used to supplement the bioassay results when the bioassay records are missing or fragmentary. They could be especially useful in assigning relative specific activities of isotopes other than uranium.

5.4.2.4 Table 5-4: "Default Isotopic Distribution"

Table 5-4 of the TBD lists default specific activities to be used when only total uranium results are available. The report does not explain when this table should be used instead of Table 5-1, which gives percentages of uranium, neptunium, and plutonium activities at specific facilities, or Table 5-2, which presents default specific activities of uranium isotopes, as discussed above. This is a source of possible confusion for dose reconstructors. Not only do Tables 5-2 and 5-4 employ different units—Bq/μg and nCi/g, respectively—the unit conversions for the uranium isotope are imprecise. The specific activities of the uranium isotopes in Table 5-4, when converted to Bq/g, are consistently about 3% higher than the default activities listed in Table 5-2. Although this is not a significant difference in terms of dose reconstruction, it is another example of the lack of accuracy and scientific rigor in the report. The table appears to be an amalgam of data from various sources—the report does not indicate how the values were calculated.

As stated earlier in this review, this guidance ignores the information presented by BJC (2001, Table 2.4-1), which lists maximum concentrations of ^{99}Tc, ^{237}Np, and plutonium for 11 different operations at various facilities in the PGDP in various time periods. NIOSH should evaluate these data for use in the dose reconstructions of workers involved in these operations.

We re-evaluated the default activity of each isotope listed in Table 5-4, as well as ones that were believed to be present at the PGDP but were not listed in this table. The discussion of each isotope is presented in order of atomic number.[4] The recommended changes in the values should be used in the generic dose reconstructions for workers, except for those known to have been exposed as a result of the operations described by BJC (2001), for whom the BJC data may lead to a more claimant-favorable dose reconstruction.

5.4.2.4.1 Technetium-99

The text cites a range of mass fractions of ^{99}Tc of 0.041–7 ppm. This range actually represents two discrete estimates—the lower figure, cited by Smith (1984, Appendix 12) is the average concentration in French reactor tails, while the 7 ppm value is taken from DOE 2000. If the value of 7 ppm ^{99}Tc in reactor tails was adopted as the concentration in airborne dust, the correct activity concentration would be 120 nCi/g U, not 0.12 nCi/g, as listed in Table 5-4. This is

[4] The following review encompasses all isotopes relevant to dose reconstruction at the PGDP. It identifies isotopes whose default values are adequately characterized, as well as issues that need to be addressed.

another computational or editorial error. The activity listed in the table, therefore, understates the activity in the reactor tails by a factor of 1,000. However, concentrations of ^{99}Tc in enriched reactor tails (ERT) averaged 16 ppm (uranium basis) (Smith 1984, Appendix 9), which is equivalent to about 274 nCi/g U.

The concentrations in reactor tails do not necessarily reflect airborne concentrations. According to a discussion of TRU and technetium in DOE 2000; "However, these elements are much more hazardous than natural uranium and were concentrated by the cascade at certain specific locations, presenting increased hazards to certain workers." According to PACE/Utah (2000, Appendix D), ^{99}Tc concentrations in breathing zone air samples at ash handling locations ranged from 288 to 2,878 dpm/m^3 β activity, vs. uranium α activities of 15–150 dpm/m^3. These ranges yield geometric means (GMs) of 910 and 47 dpm/m^3, respectively, for a ratio of ^{99}Tc to total uranium activity ≈19. According to PACE/Utah (2000, Table 7.9), the average fraction of uranium α activity in the dust at each of these two locations is 0.7 of the total α activity. Since the ash handled in these locations primarily consists of feed material, it is reasonable to assume the relative fractions of the uranium isotopes to be the same as those of natural uranium, with a total α activity of 686 nCi/g U. Based on these data, the specific activity of ^{99}Tc in the dust is calculated to be about 19 μCi/g U (0.686 × 19.19 ÷ 0.7 = 18.8). This value constitutes a claimant-favorable default specific activity of ^{99}Tc.

5.4.2.4.2 Thorium-230

The default specific activity of ^{230}Th in Table 5-4 is equal to the highest value for this nuclide for the five locations listed in Table 5-3. However, PACE/Utah (2000, Table 7.9) lists two locations—ash receivers and pulverizer—with far higher fractions of ^{230}Th in airborne dust. (The data tabulated by PACE/Utah for the C-400 converter salvage line is internally inconsistent and, therefore, is not being used in this comparison.) The average fraction of ^{230}Th α activity in the dust at each of these two locations is 0.143, while the uranium accounts for 0.7 of the α activity. Since the ash handled in these locations primarily consists of feed material, it is reasonable to assume the relative fractions of the uranium isotopes to be the same as in natural uranium, with a total α activity of 686 nCi/g U. The specific activity of ^{230}Th is thus calculated to be 140 nCi/g U (686 × 0.143 ÷ 0.7 = 140), over 7 times higher than the value of 18.9 nCi/g U listed in Table 5-4. Since the default activity ratios could be used for workers at these locations, the higher value should be adapted for a conservative, claimant-favorable dose assessment.

5.4.2.4.3 Uranium-234

The default specific activity of ^{234}U is listed in Table 5-4 as 702.0 nCi/g U. We believe the more conservative, claimant-favorable value of 0.0389 Bq/μg U listed in Table 5.4-3 should be adopted. This is equal to approximately 1,050 nCi/g U.

5.4.2.4.4 Uranium-235

The default specific activity of ^{235}U is listed in Table 5-4 of the TBD as 43.9 nCi/g. This is in reasonable agreement with the value of 1.54 × 10^{-3} Bq/g (42 nCi/g) listed in Table 5.4-2.

5.4.2.4.5 Uranium-236

The default specific activity of ^{236}U is listed in Table 5-4 of the TBD as 0.93 nCi/g U. The highest enrichment of ^{236}U in the PGDP cascade products were observed in May 1973, with an average concentration of 0.0625 wt% being measured in three cylinders withdrawn that month, which yields a value of 41.6 nCi/g U as a reasonable upper bound of ^{236}U specific activity (see Table 5.4-2). If the dose reconstructor has no detailed information about the time and location of exposure, this conservative, claimant-favorable value should be adopted.

5.4.2.4.6 Uranium-238

The default specific activity of ^{238}U, 337.5 nCi/g, listed in Table 5-4 of the TBD, is approximately that of this isotope in natural uranium.

5.4.2.4.7 Neptunium-237

As was the case with ^{230}Th discussed above, the default specific activity of ^{237}Np listed in Table 5-4 of the TBD is equal to the highest value for this nuclide for the five locations listed in Table 5-3. PACE/Utah (2000, Table 7.9) lists three locations—ash receivers, pulverizer, and converter maintenance—with higher fractions of ^{237}Np in airborne dust. The average fraction of ^{237}Np α activity in the dust in the first two locations is 0.007, while the uranium accounts for 0.7 of the α activity. For converter maintenance, the activity fractions of ^{237}Np and of uranium are both listed as 1, and the average air concentrations of these two nuclides are both listed as 2.3E-11 μCi/cc. The same activity fractions and relative concentrations apply to the maximum values of these nuclides at that location.

These activity fractions are obviously erroneous—the fractions of the two nuclides cannot both equal 1—which led us to question the absolute air concentrations as well. We confirmed the average ^{237}Np concentration from data presented by Davis (1978), who reports activity concentrations in airborne dust during converter disassembly in Building C-409. The average ^{237}Np concentration, derived from 18 samples (including both breathing zone and general area samples) is 50.3 dpm/m^3, which is, in fact, equal to 2.3×10^{-11} μCi/mL. However, the average uranium concentration is equal to 7.8×10^{-11} μCi/mL. Uranium thus accounts for 76.4% of the α activities, while ^{237}Np constitutes 22%, the remainder being identified as ^{230}Th and ^{239}Pu. These uranium and neptunium values are almost exactly the same as those listed in Table 5-1 for Building C-409; similar values are listed for Building C-337. The specific activity of ^{237}Np is therefore about 29% (0.22 ÷ .0724 = 0.288) of the combined specific activities of all the uranium isotopes. The total of the recommended specific activities of the uranium isotopes, as discussed above, is 1,463 nCi/g, yielding a specific activity of ^{237}Np of 544 nCi/g, 100 times higher than the default activity of 5.4 nCi/g listed in Table 5-4. Since the default activity ratios could be used for workers at these locations, the higher value should be adapted for a conservative, claimant-favorable dose assessment.

5.4.2.4.8 Plutonium-238

There is no mention of ^{238}Pu in the TBD. The feed material for the PGDP cascades included Hanford reactor tails (Smith 1984). One source of information on plutonium isotope fractions of different materials produced at Hanford is PNNL 2004 (Chapter 8), which lists examples of mixtures of plutonium isotopes. Because ^{238}Pu has a significantly shorter half-life than ^{239}Pu (87.7 vs. 24,110 y), it has a correspondingly higher specific activity. The data listed by PNNL indicates that the ^{238}Pu activity can exceed the combined activities of ^{239}Pu and ^{240}Pu in some plutonium fuel mixtures. It is not clear if ^{239}Pu α activity listed in reference documents includes ^{238}Pu. Although we are not proposing a default specific activity for this isotope, we believe that this isotope should be accounted for in the PGDP dose assessments. We note that the internal dose coefficients for this isotope are similar to those for ^{239}Pu, so that combining the activities of ^{238}Pu, ^{239}Pu, and ^{240}Pu would be an efficient and scientifically correct method of performing dose reconstruction.

5.4.2.4.9 Plutonium-239

As was the case for ^{230}Th and ^{237}Np, discussed above, the default specific activity of ^{239}Pu listed in Table 5-4 of the TBD is equal to the highest value for this nuclide for the five locations listed in Table 5-3. However, as was the case for the other two nuclides, PACE/Utah (2000, Table 7.9) lists two locations—ash receivers and pulverizer—with far higher fractions of ^{239}Pu in airborne dust. The average fraction of ^{239}Pu α activity in the dust at each of these two locations is 0.15, while the uranium accounts for 0.7 of the α activity. Since the ash handled in these locations primarily consists of feed material, it is reasonable to assume the relative fractions of the uranium isotopes to be the same as natural uranium, with a total α activity of 686 nCi/g U. The specific activity of ^{239}Pu is calculated to be 147 nCi/g U ($686 \times 0.15 \div 0.7 = 147$), more than twice the value of 67.5 nCi/g U listed in Table 5-4. Since the default activity ratios could be used for workers at these locations, the higher value should be adapted for a conservative, claimant-favorable dose assessment.

Higher specific activities of ^{239}Pu have been reported at PGDP. PACE/Utah (2000) reports the following:

> *There was one documented case where a batch of material was sent to Fernald in June of 1980 that was found to have particularly high levels of plutonium. According to one document [8], the plutonium levels ranged from 67 to 7,757 ppb U.*

A concentration of 7,757 ppb ^{239}Pu would correspond to a specific activity of 470 nCi/g. However, since this was the upper end of the range in one batch of material, it is unlikely that such a high value would represent the average specific activity in the airborne dust inhaled by any worker.

5.4.2.4.10 Plutonium-240

No value for ^{240}Pu is listed in Table 5-4 of the TBD. Smith 1984 states that the reported ^{239}Pu activities included ^{240}Pu, calculated as ^{239}Pu. Since the dose coefficients for these two isotopes are virtually identical, this agglomeration does not affect dose reconstruction. However, this fact should be noted in the TBD.

5.4.2.4.11 Plutonium-241/Americium-241

The default specific activity for "^{241}Pu/^{241}Am" in Table 5-4 of the TBD is set to the same value as that of ^{239}Pu. Since feed material for the PGDP cascades included Hanford reactor tails (Smith 1984), the specific activities of these nuclides can be estimated from the relative specific activities of Hanford reactor fuel mixtures, such as those listed by PNNL (2004, Chapter 8). According to PNNL, the ^{241}Pu activity can be up to 144 times as great as the combined activities of ^{239}Pu and ^{240}Pu, while the ^{241}Am activity can be up to four times as great. The default activities of these nuclides should be increased in light of these data.

5.4.2.4.12 Other Fission Products

Baker (1987) notes the presence of ^{106}Ru, ^{95}Zr, and ^{144}Ce in the UO_3 feed, and ^{106}Ru and ^{95}Zr in the ash, while Smith (1984, pg. 18) also cites the presence of ^{137}Cs in the dust. Although these appear to be minor radioactive constituents, they need to be addressed in the TBD.

5.4.2.4.13 Summary of Recommended Changes to Table 5-4

The recommended changes to Table 5-4 of the TBD are listed in Table 5.4-3, below. We have not included fission products, other than ^{99}Tc.

Table 5.4-3. Recommended Changes to Default Specific Activities

Radionuclide	TBD	SC&A	
	nCi/g U	Bq/µg U	nCi/g U
Tc-99	0.12	6.95E-01	18,800
Th-230	18.9	5.18E-03	140
U-234	702	3.89E-02	1052
U-235	43.9	1.54E-03	42
U-236	0.93	1.48E-03	40
U-238	337.5	1.22E-02	329
Total U	1084.33	5.41E-02	1,463
Np-237	5.4	2.01E-02	544
Pu-239/240[a]	67.5	5.44E-03	147
Pu-241[b]	67.5	7.83E-01	21,200
Am-241[b]	67.5	2.18E-02	588

Note: Fission product activities should be investigated and included if they can make a significant contribution to worker doses.

[a] Combined activity of both isotopes—see text.

[b] Based on the recommended default activity of $^{239/240}$Pu and on the discussion in the text.

5.4.2.5 Table 5-5: Facility-Specific Source Radionuclides with Solubility Type and Particle Size

Table 5-5 of the TBD specifies the chemical compound, the lung clearance type (called "absorption type" in the table), and the particle size distribution for various radionuclides at a number of facilities at the PGDP, as well as default parameters listed at the end of the table.

We have reviewed the literature cited in the footnotes to Table 5-5. One reference is listed as "PACE et al. (2000, Table 2.4-1)." This is an error—the table number does not correspond to any table in that document.

The format of this table is needlessly long and cumbersome. Since many buildings have identical characteristics in terms of the radionuclides and their lung clearance types, the table would be much more usable if these buildings were listed together, rather than each building having a separate block of the table. Furthermore, the buildings are not listed in numerical order, nor in any other discernable order, requiring the reader to hunt through the entire table to locate a particular building.

We will first discuss issues that apply to a number of buildings, following which we will discuss individual buildings, grouping together those having identical characteristics.[5]

5.4.2.5.1 Particle Size

The particle size in each case is specified as 5 μm AMAD. Such a particle-size assignment is not supported by the data, nor is it claimant favorable. The author cites a study that characterized the particle sizes of neptunium and plutonium isotopes as 3 μm in seven buildings at the PGDP, then dismisses these results in favor of a default of 5 μm recommended by ICRP 1994, which applies to a generic workplace exposure. Baker (1987, pg. 5) reports: "The UO_2F_2 fume had an AMAD of about 1 μm…" In discussing the exposure of cold trap and refrigeration systems operators, Baker states: "There was also chronic exposure to dust generated in other areas… (AMAD estimated at 4 μm)" (Baker 1987). Finally, Bruner (1960) cites that the dust particles at the PGDP are "about 0.5μ."

This decision to ignore site-specific data is contrary to both the intent and the letter of 42 CFR 82. Furthermore, such a particle-size assignment is not claimant favorable. We have compared the 50-year doses to each organ from 3-μm and 5-μm AMAD particles of ^{230}Th, the three natural uranium isotopes, ^{237}Np, and ^{239}Pu from all lung clearance types listed. Except for the extra-thoracic airways (in some instances), all organs would receive equal or higher doses from the 3-μm particles, the differences ranging from zero to 47%. Since particle sizes ranging from 1–10 μm AMAD are cited by Baker, the dose reconstructors should be instructed to use whatever particle size is the most claimant favorable in each case.

[5] As in our discussion of Table 5-4 of the TBD, our critique of Table 5-5 is comprehensive and cites data that are adequately characterized, as well as those that are at issue.

5.4.2.5.2 Chemical Compounds—General Observations

In a number of buildings, the compound is listed as UF_6; however, upon contact with the water vapor in the air, UF_6 is hydrolyzed according to the following reaction:

$$\mathbf{UF_6 + 2H_2O = UO_2F_2 + 4HF}$$

Since UO_2F_2 has the same lung clearance type as UF_6, this chemical reaction does not affect the internal dose corresponding to a given airborne concentration of uranium, but it should be noted in the TBD in the interest of scientific accuracy.

The only specific chemical compounds listed for each building are compounds of uranium. The structure of the table gives the erroneous impression that these compounds apply to all the radionuclides, including isotopes of technetium, thorium, protactinium, neptunium, plutonium, and americium. For elements for which more than one lung clearance type is listed by ICRP (1994), the chemical compound needs to be listed in order to enable the correct assignment of a clearance type. If the compound is unknown, that should be stated in the table. Terms such as "ash handling" and "neptunium recovery" are improperly listed under the heading "Compound."

5.4.2.5.3 Uranium Progeny

In each instance that ^{234m}Pa is included in the list of radionuclides, it is preceded by ^{230}Th. This is clearly a propagation of the error at the beginning of Section 5.2, which we have noted earlier in this review. Protactinium-234m, which has a half-life of 1.17 minutes, would always be found together with its parent, ^{234}Th. The intake of ^{234m}Pa makes a negligible contribution to internal dose; in fact, this nuclide is not even listed in ICRP Publication 68 (ICRP 1994).

Thorium-230 is a long-lived ($t_{½}$ = 75,380 y) daughter product of ^{234}U that is found in conjunction with uranium that has aged for a number of years following its chemical separation. PACE/Utah (2000, Table 7.9) lists ^{230}Th activities in Buildings C-410 and C-420, and in the cold trap area. Baker (1987) lists ^{230}Th in the UO_3 powder area, the C-420 green salt plant, the fluorination tower area, the cold trap and refrigeration operation, and Building C-410. Although Table 5-3 of the TBD includes ^{230}Th activities in these locations but not in additional ones listed in PACE/Utah 2000, Table 7.9 (see discussion of Table 5-3 earlier in this review), this nuclide is not listed in Buildings 410 and 420 in Table 5-5.

5.4.2.5.4 Transuranics

Table 5-5 limits TRU to ^{237}Np and ^{239}Pu. However, as was discussed in the review of Table 5-4, the TRU in the Hanford reactor tails and other sources include ^{238}Pu, ^{240}Pu, and ^{241}Am. These nuclides should be included with ^{237}Np and ^{239}Pu in Table 5-5, to be consistent with Table 5-4 (which lists all but ^{238}Pu), SAIC (1999, pg. 14), and Hill and Strom (1993, Table 16.2, Note "a").

Information about the occurrence of TRU nuclides at different facilities is presented by PACE/Utah (2000, Appendix D), which lists various radiological data. Air sampling, radiation survey, and urinalysis data indicates the presence of one or more TRU nuclides in the following

facilities: C-310, -315, -331, -333, -335, -337, -340, -360, -400, -409, -410, -420, -710, -720, -746, and -746B. In addition, Hill and Strom (1993, Table 16.2) list TRU nuclides at Buildings C-746A and "C-7460" (perhaps a typo for C-746Q). Finally, SAIC (1999, Table 1.7) lists TRU activities at C-333A, -337A, -404, -411, and -746Q. Since the dose assessments should always give the claimant the benefit of the doubt, the list of radionuclides in Table 5-5 should be all inclusive; therefore, the complete list of TRU nuclides should be included for each of these facilities. This requires expanding the radionuclide lists for C-310, -315, -337, -333A, -337A, -340, and -404.

5.4.2.5.5 Technetium-99

According to SAIC (1999, Table 1.7), ^{99}Tc is found in the same locations as the TRU. (This is to be expected, since both products originate in recycled uranium). Thus, this nuclide should be added to the nuclide lists for C-315, -337, -333A, -337A, -340, -400, -404, and -409. Furthermore, DOE (2000, Appendix B) specifically cites high concentrations of ^{99}Tc in process solutions in Bldg. C-400.

5.4.2.5.6 Lung Clearance Types and Chemical Compounds

The Lung Clearance Types, referred to as "Absorption Type" in Table 5-5, should be consistent with the chemical forms of each radionuclide. In case of uncertainty, the most claimant-favorable assumption should be adopted. It is not always clear which chemical form and/or clearance type the dose reconstructor should assign to each element.

Technetium-99
Technetium-99 at all facilities at the PGDP is assigned to Type F, the default for unspecified compounds (ICRP 1994, Annexe F). However, since technetium halides, as well as oxides and hydroxides, are Type M, that type should also be listed. According to DOE (2000, pg. 31), technetium formed a volatile compound of fluorine. (This was most likely TcF_6, which boils at 55.3ºC, just below the sublimation point of UF_6.) For 3 μm particles, the 50-year doses to the lungs and extra-thoracic airways are orders of magnitude higher from Type M; however, for some organs, the doses from Type M are up to 11% lower. Therefore, both types should be listed, allowing the dose reconstructor to select the more claimant-favorable type for a given organ.

Thorium-230
Thorium-230 is assigned solely to Type S. This assignment is not scientifically correct. According to ICRP (1994), Type S is assigned to thorium oxides and hydroxides, while all other compounds are assigned to Type M. Thorium fluorides (e.g., ThF_4) at the feed plant are mentioned by Baker (n/d, pg. 389); thus, Type M thorium should also be considered. Furthermore, the assignment to Type S may not be claimant favorable; for a given intake, Type M delivers doses to organs other than the lung and the extra-thoracic airways that are 9–10 times higher than those from Type S. However, in cases where ^{230}Th was detected in the urine, the calculated dose based on the concentration in the urine would be higher if the thorium were assigned to Type S. Therefore, both types should be listed, allowing the dose reconstructor to select the more claimant-favorable type for a given organ.

Uranium

Buildings C-310, C-315, C-331, C-333, C-335. Uranium isotopes at Buildings C-310, C-315, C-331, C-333, C-335, which house the cascade, are assigned to Type F. The uranium was in the form of UF_6 which, upon release to the atmosphere, reacted with moist air to form UO_2F_2, as discussed previously. Both of these compounds are Type F, so the assignment is appropriate. It is not clear why the uranium in Buildings C-333A and C-337A is assigned to Type M as well as F. The only compound listed in Table 5-5 is UO_2F_2, which is Type F. If other compounds are believed to be present, this should be explained in the TBD.

Building C-340. Uranium isotopes in Building C-340 are restricted to Types M or S. However, the list of compounds includes UF_6, which is Type F. Type F should therefore be included as an option for use by dose reconstructors for workers exposed at this facility.

Building C-360. Uranium isotopes in Building C-360 are assigned to Types F, M, or S. Since the building appears to have been used for transfer and sampling of UF_6 cylinders, it is not clear why Type M or S compounds would be found there. In the interest of equity in dose reconstruction, the same logic in assigning lung clearance types should be used for all facilities.

Building C-400. Uranium isotopes in Building C-400 are assigned to Types F, M, or S; however, the only compounds listed are UF_6 and UO_2F_2. According to DOE (2000, Appendix B), releases of UF_4 (Type M) and U_3O_8 (Type S) also occurred in this building. These compounds should be listed in Table 5-5 to document the assignment of lung clearance types.

Facility C-404. Uranium isotopes in Facility C-404 are assigned to Types F or M. However, since the wastes disposed of in this area included incinerator ash, the presence of UO_2 is likely—Type S should also be included. For the sake of completeness, the list of compounds should include UO_2F_2 if there is reason to believe that this compound (or the dry form, UF_6) was released there. If not, there may be no justification for including Type F as an option.

Building C-405. Uranium isotopes in Building C-405 are assigned to Types F, M, or S.

Building C-409. Building C-409 is described as the decontamination building, whereas PACE/Utah (2000) and Turpin (2006) call it the stabilization building. The terminology should be corrected. Uranium isotopes in Building C-409 are assigned to Types F or M; however, the only compounds listed are UF_6 and UO_2F_2. These are also the only uranium compounds listed by DOE (2000, Appendix B). Unless there is evidence for UF_4 or other Type M compounds at this facility, Type M should not be included in Table 5-5.

Buildings C-410 and C-420. Buildings C-410 and C-420 housed the feed plant. Thus, uranium compounds include UO_2, UO_3, U_3O_8, UF_4, and UF_6 (which forms UO_2F_2 when released). Table 5-5 includes "UF" (a nonexistent compound) in the list for both buildings and omits U_3O_8. It also omits UF_6 (or UO_2F_2) at C-420. The assignment of the uranium isotopes to Types F, M, or S is correct, given the possible compounds.

Building C-710. Building C-710 housed the analytical laboratories, which performed neptunium and uranium recovery operations. Absent specific information on the uranium

compounds handled in this building, the claimant-favorable assumption should be that all three lung clearance types of uranium—F, M, and S—could have been present. Representative compounds of each clearance type (e.g., $U_2O_2F_2$, UF_4, and UO_2) should be listed in the "Compound" column.

Building C-720. Building C-720 housed various maintenance operations, including converter and compressor disassembly and flange grinding. Workers were primarily exposed to UF_6 and UO_2F_2, but also to UF_4 and oxides such as U_3O_8 or UO_2. These compounds represent all three lung clearance types—F, M, and S. Table 5-5 lists these three types for ^{235}U and ^{238}U, but erroneously omits Type S for ^{234}U. Uranium compounds representing all three classes should be listed in the "Compound" column.

C-746 Facilities. DOE (2000) refers to a number of C-746 facilities: C-746C, C-746E, C-746F, C-746K, C-746R, C-746S, and C-746T, in addition to C-746A, C-746B, and C-746Q, which are listed in Table 5-5. It would be simplest to list "C-746" as a general category that includes all these facilities. Given the diversity of operations, all three clearance types and their representative compounds should be included.

Neptunium

All neptunium compounds are assigned to Type M (ICRP 1994, Annexe F). However, for a number of facilities, Table 5-5 erroneously includes Type F, and sometimes also Type S, for ^{237}Np. Such assignment, for which no dose coefficients have been developed, is a possible source of confusion for dose reconstructors.

Plutonium

All plutonium isotopes at all facilities are assigned to Type S in Table 5-5. This assignment is not scientifically correct. According to ICRP 1994, Type S is assigned to insoluble plutonium oxides, while Type M applies to all other compounds. According to DOE (2000), "… most of the plutonium and technetium was volatilized to the cascade…" This indicates that plutonium was most likely present as a fluoride rather than an oxide; PuF_6 melts at 52ºC, while PuO_2 has a melting point of 2400ºC. Furthermore, such an assignment may not be claimant favorable—for a given intake, Type M delivers doses to organs other than the lung and the extra-thoracic airways that are 9–10 times higher than those from Type S. However, in cases where plutonium was detected in the urine, the calculated dose based on the concentration in the urine would be higher if the plutonium were assigned to Type S. Therefore, both Type M and Type S should be listed, except for facilities where only one or the other class of compounds was present. For instance, it is likely that only Type M would be present in the cascades, since the exposure would be to releases of PuF_6 in the UF_6 gas.

5.4.3 Section 5.3: "In-Vitro Measurement Methods"

5.4.3.1 Calculation of Intakes

5.4.3.1.1 Frequency of Sample Collection

Table 5-6 of the TBD lists the frequency of in-vitro measurements at various facilities at the PGDP during various time periods. The frequencies range from once every 4 weeks to once a year. The last row of the table lists a default frequency of once every 4 weeks. Since the interval between measurements could have been as long as 1 year, this default assumption is not claimant favorable. The longer the interval over which the intakes occurred, be they chronic or acute, the lower the urine concentration for a given total intake. Conversely, for a given urine concentration, the longer the elapsed time (for acute intakes) or exposure duration (for chronic), the greater the derived intake.

To illustrate this effect, we have calculated the acute intake that would result in the excretion of 1 Bq in 24 hours, assuming the inhalation of Type F, M, or S uranium with an AMAD of 5 μm. The results are shown in Table 5.4-4 of this review.

Table 5.4-4. Acute Intake of Uranium (Bq) That Would Result in Excretion of 1 Bq/d

Time After Intake (mos)	Lung Clearance Type		
	F	M	S
1	1.47E+03	3.77E+03	1.30E+05
2	4.41E+03	5.99E+03	1.93E+05
4	1.39E+04	1.02E+04	2.63E+05

As shown in Table 5.4-4, in the case of Type F uranium, assuming that an intake occurred 1 month before the urine sample was collected, when in fact it occurred 4 months ago, would result in a 10-fold error in estimating the intake.

5.4.3.1.2 Converting Spot Samples to 24-hour Samples

Equations 5-1 to 5-3 of the TBD present methods of converting spot samples to 24-hour samples. We have several comments on the formal presentation of these equations, as well as on their content. The first observation is that the physical units should be included in a description of symbols that precedes or follows the equation, not in the equation itself. The equations as written are incorrect and confusing. We recommend that Equation 5-1 be written as follows:

$$C_{cor} = C_m \left[\frac{SG_r - 1}{SG_m - 1} \right] \qquad (5\text{-}1)$$

C_{cor} = corrected activity concentration in urine sample (pCi/L)

C_m = measured activity concentration (pCi/L)

SG_r = reference specific gravity of urine (dimensionless)
= 1.024

SG_m = measured specific gravity of sample

Equation 5-2 in the TBD incorrectly indicates that the corrected concentration is multiplied by 1.4 and divided by 24. We recommend that the equation be written as follows:

$$A_u = C_m V \qquad (5\text{-}3)$$

A_u = activity excreted in 24 h (pCi)

V = excretion rate for Reference Man (ICRP 2002)
= 1.6 L/d (adult man)
= 1.2 L/d (adult woman)

Similarly, Equation 5-3 should be written: C_m = measured activity concentration (pCi/L)

In the line immediately above Equation 5-3, the text "by assuming a reference value *V* of 1.4 and volume *L* of 1.0 L" should be deleted. The text as it stands is incorrect—"L" is the symbol for liters, not some specified volume—and is no longer needed if the above corrections are adopted.

Furthermore, Equation 5-2 in the TBD is based on an excretion rate of 1.4 L/d of urine. However, this value is based on an earlier reference value for an adult man. The current ICRP (2002) reference values are 1.6 L/d for an adult man and 1.2 L/d for an adult woman. Thus, the value of 1.4 L/d is not scientifically valid for either male or female workers. Furthermore, it is not claimant favorable for males.

5.4.3.2 Minimum Detectable Concentrations

Table 5-7 of the TBD lists minimum detectable concentrations (MDCs) for total uranium and for various radionuclides, using different analytical techniques during different time periods at the PGDP. The column headed "Measurement type" lists analytical techniques in some cases, and simply "ORNL" or "contractor" in others. The types of measurements performed by ORNL and the unspecified contractor need to be presented, or an explanation given as to why this information is not available. Without any information on the type of analysis, it is not possible to independently assess the MDCs for these measurements. The column headings "MDC (mg/L)" and "Recall level (µg/L)" are confusing, since not all values are in these units. A footnote indicating that these are the units "unless otherwise noted" would help to clarify this issue.

In many cases, we could not verify the listed MDCs. For example, the MDC for total uranium by in-house fluorimetry is listed as 10 µg/L. None of documents cited as data sources for this table list that value; both PACE/Utah (2000) and SAIC (1999) cite typical MDCs of 5 µg/L. The MDC for natural uranium assayed at ORNL from 1999 to the present is in error; the source document (SAIC 1999) lists the MDC at 0.06 µg/sample, while Table 5-7 lists it as 0.06 mg (60 µg), a 1000-fold discrepancy. The latter value is clearly inconsistent with the value of 5 µg/L for kinetic phosphorescence analysis (KPA), used in 1977–1982. Table 5-7 lists a default detection level of 0.27 pCi/L for urinalysis of individual isotopes of actinide elements (Th, U, Pu, and Am). The TBD cites ICRP Publication 54 (Annals of the ICRP Vol. 19 No. 1–3) ,

Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation (Oxford: Pergamon Press) as the source of these data, which are based on alpha spectroscopy.[6] However, Table 5-7 lists higher levels—1 pCi/L for Pu isotopes and 0.3 pCi/L for uranium isotopes. If the dose reconstructor does not know the analytical method involved, this default assumption is neither claimant favorable nor scientifically correct.

Given the importance of the MDC in assigning doses in cases of low readings or non-detection, greater attention should be given to this topic to make certain that the values listed result in claimant-favorable dose assessments.

There is no discussion of the uncertainties in the determination of uranium in urine samples for fluorimetry and KPA. The uncertainty of the fluorimetry methodology can be very high; it is strongly dependent on the quenching of the sample. For other radionuclides, there is no description of the technique used for their measurements and the accompanying uncertainties.

5.4.3.3 Day of Sample Collection

Section 5.3.3 of the TBD states the following:

> *The practice of offsite collection of samples that takes place 24 to 48 hr after leaving the plant not only minimizes the possibility of sample cross-contamination, but it ensures that samples are collected after the transfer of the rapid clearance component. Some PGDP employees were asked to collect samples after 1 or 2 days off from work; if so, that collection instruction was sometimes noted on the analytical record.*

The TBD fails to note that this practice would lead to a lowering of the calculated intakes, nor does it instruct dose reconstructors to be alert to any cases for which urine samples were collected after the worker was off work for any period of time. Appropriate adjustments to the calculated intake should be made to compensate for the lowered concentration in the urine following an absence from work.

Table 5.4-5 of the present review illustrates the effect of the day of sample collection on the calculated intake. In this example, we have assumed a chronic intake by inhalation of Type F uranium with an AMAD of 5 μm by a worker who is exposed 8 hours per day, 5 days per week (Monday through Friday). We have calculated the intakes that would result in a rate of excretion of 1 Bq/d on each successive day of the week during the 4th week of the assessment period.

[6] We also note that ICRP Publication 54 has been replaced by ICRP Publication 78: *Individual Monitoring for Internal Exposure of Workers - Annals of the ICRP Volume 27/3-4, Replacement of ICRP Publication 54*, 1998 (Oxford: Pergamon Press).

Table 5.4-5. Intakes Derived from Samples Collected on Different Days of the Week

Exposure Duration (d)	Time of Sample	IRF: 24-hour Urine[a]	Intake (Bq/d)
21	Monday morning	0.0402	24.9
22	Tuesday	0.218	4.59
23	Wednesday	0.223	4.48
24	Thursday	0.226	4.42
25	Friday	0.228	4.39
26	Saturday	0.230	4.35
27	Sunday	0.0504	19.8

Note: Each sample assumed to correspond to an excretion of 1 Bq/d

[a] Intake retention fraction

Table 5.4-5 shows significant differences in the calculated intakes for different days of sample collection. If the sample was collected on Wednesday, the calculated chronic intake during the previous 3½ weeks would be about 4.5 Bq/d. However, if the sample was known to be collected on Monday morning, prior to starting work, the calculated intake during the previous 3 weeks would be about 25 Bq/d, over 5-times higher. The TBD should provide claimant-favorable guidance for dose reconstruction of cases where the date of sample collection is not recorded.

This guidance becomes especially relevant in the case of recall samples, which were normally collected on Monday mornings (PACE/Utah 2000, pg. 38). Instead, the TBD states the following:

> *… contamination of samples from the worker's hands or clothing cannot be ruled out as a contributor to any given result. If a second analysis was performed and if that result was negative, it is reasonable to assume the first result was a false positive due to sample contamination or laboratory error.*

Such guidance violates 42 CFR 82, who states that in cases of uncertainty, the dose reconstruction should use assumptions that are claimant favorable. The TBD should instruct dose reconstructors to consider all results, even if a later analysis had a lower or non-detectable concentration. To do otherwise would create a bias against the claimant.

5.4.3.4 Dietary Intake of Uranium

The TBD gives confusing guidance regarding the possible dietary intakes of uranium by monitored workers:

> *Because studies of the average daily uranium excretion on Paducah residents do not appear to have been performed, it is not possible to make corrections for the contribution of nonoccupational intakes of uranium to a given urine sample result. However, to put a given result into perspective, a nominal daily (24 hr) urinary excretion rate for uranium of 0.43 µg (environmental decision level at 95% confidence)* ***can be used****… No correction for environmental levels of*

> *uranium is required for samples analyzed by fluorimetry or KPA because the MDC is larger than the correction.* [Emphasis added.]

It is not clear if the TBD intends to give dose reconstructors the option of subtracting 0.43 µg from the 24-hr urine sample analyzed by techniques other than fluorimetry or KPA. Such guidance would be neither scientifically correct nor claimant favorable. First, the value cited by SAIC (1999) (the TBD refers to this document as BJC) applies to residents of the Oak Ridge, Tennessee, area, not of the PGDP. Second, it is an upper limit (95th percentile). Subtracting such a high-end value from the urinalysis result would result in a bias against the claimant. The TBD should state unambiguously that the results should not be corrected for any assumed contributions of uranium in the environment.

5.4.3.5 Table 5-9: "In-Vitro Record Codes"

Table 5-9 of the TBD lists the record codes for urine bioassays. First, we note that one of the sources for these data is listed as "PGDP 2003b." That document, *Uranium in Urine Analysis* (n/d), contains no information on record codes. The author most likely meant to cite "Urinalysis Program" (n/d) which does, in fact, contain such data.

For Form WCP-455, the codes for "Reason for Visit" contain one ambiguity; code 33 is identified both as "industrial health recheck" and "termination samples." This ambiguity should be resolved. Furthermore, the midnight shift is identified as "O," whereas on form WCP-885, the shifts are indicated by the letters A–D. This is a possible error that should be resolved. The cited documents do not provide any information on the coding of Form WCP-455; therefore, we could not verify the accuracy of these data.

Urinalysis Program (n/d) presents instructions for coding Form WCP-885. We have several comments about the presentation of the codes for this form in Table 5-9 of the TBD. First, the code for "shifts" refers to four shifts, identified as A, B, C, or D. Table 5-9 erroneously identifies the third shift as "O." Next, the frequency codes are erroneously based on an example of how to fill out this form, presented in *Urinalysis Program* (n/d). The correct interpretation of the frequency codes is: 1 = 1 per month, 2 = 1 per every 12 months. The identity of the analyte is indicated elsewhere on the form and is not relevant to this code.

Form UCN-5242 is stated to have the same column identifiers and codes as "NCP-455." It is most likely that the correct designation is "WCP-455."

Finally, we note that units are listed for breath and fecal analyses. The only other mention of fecal analyses in the TBD is a footnote that indicates that routine fecal analyses were not performed at the PGDP. According to PACE/Utah (2000), fecal bioassays were performed on 16 workers. Should any of these workers or their coworkers become claimants, such data could be useful in performing dose reconstructions. Information on the fecal analysis methodology, the MDCs, and guidance on the use of these data in dose reconstruction should be provided in the TBD.

No information is presented on breath analyses. If such studies were performed, information similar to that needed for fecal analyses should be provided.

5.4.4 Section 5.4: "In-Vivo Measurement Methods"

Table 5-10 presents the MDAs for the measurement of various radionuclides by whole-body or chest counting. The sources cited, however, do not present the values listed, with the sole exception of ^{237}Np for the 1968–1991 period. Some of the values are inconsistent. For example, 2% enriched uranium has about 3 times the concentration of ^{235}U as natural uranium, yet the MDA for "total uranium" is 40 times higher (4 mg vs. 100 μg) than 2% enriched uranium. The ^{235}U in natural uranium would permit the detection of smaller quantities of uranium. The same MDA (4 mg) is listed for ^{238}U. Even the depleted uranium tails at the PGDP contained about 0.27% ^{235}U on a mass basis, which would be sufficient to permit the detection of less than the stated MDA. Uranium-238 can be detected by the radiation emitted by its daughter product, ^{234}Th, which has a 24-day half-life and an MDA of 3 nCi (SAIC 1999, Table 3.2). Assuming the two nuclides remain in secular equilibrium in the body, this corresponds to 10 mg of ^{238}U, and approximately the same quantity of natural uranium. These MDA values should be verified and documented to enable verification by an independent reviewer.

The MDA for in-vivo counting at Y-12 prior to 1967 appears to be overstated. Bruner (1960) reports a lower limit of 7 nCi, which is 2 orders of magnitude lower than the value of 0.5 μCi for the 1960–1967 period.

Table 5-11 lists in-vivo record codes. Of the three references cited, "PGDP 2003a" (Urinalysis Program n/d) is clearly inapplicable. There is no available documentation for the record codes of the "In-Vivo Radiation Monitoring Report;" consequently, these codes could not be verified.

5.4.4.1 Addressing Interferences and Uncertainties

According to Section 5.4.3 of the TBD:

> *For in vivo measurements, contamination could have occurred as external to the body or, in the case of chest counting, as external to the lung. If a follow-up in vivo count (the same day or within a few days) showed a dramatic decrease in activity or no detectable activity, then external contamination should be assumed.*

We agree that if a second in-vivo count performed shortly after the first (i.e., within 1–2 hours) showed a markedly reduced count rate, it is reasonable to assume that the worker underwent external decontamination following the first measurement, and that the second count was more indicative of the internal body burden of a given radionuclide. We do not agree, however, that a count taken several days later should be substituted for the earlier measurement. The first count could have detected activities in the lungs that were later transported to other organs. If the measurements were chest counts, then the counting efficiency of radioactive contaminants in other organs would be lower, both because the detector geometry was optimized for the lungs, and because abdominal organs, for instance, are shielded by a greater mass thickness of tissue, which absorbs more of the emitted photons. Therefore, it is neither correct nor claimant

favorable to discard the earlier counts on the unverified grounds that they may have been due to external contamination.

The TBD gives no instruction to the dose reconstructor on estimating uncertainties in the in-vivo measurements, except to state, "For results near or at the reporting levels, dose reconstructions should apply the prescribed standard deviation of 0.3 times the MDA or reporting level…" While we do not disagree with that recommendation, clearer guidance is needed for higher levels. The uncertainties for in-vivo bioassay include the uncertainty in the counting statistics, including the background that is subtracted from the count rate, the uncertainty of translating a count rate from a given worker into a body burden, given the differences in morphology among different individuals, and the variation in the naturally occurring radionuclides in the human body, primarily ^{40}K, that contribute to the background.

During site interviews, it became apparent that only a subset of workers from a particular group was monitored, indicating cohort monitoring for in vivo counts. This leads to additional uncertainty in the use of in vivo counting and deserves further evaluation.

The TBD instructs dose reconstructors to disregard any measurements of ^{137}Cs, on the grounds that there was no occupational exposure to this radionuclide at the PGDP. We disagree with this conclusion, since this nuclide may have been present in reactor tails processed at the PGDP, and was, in fact, detected in cascade dust (Smith 1984, pg. 18). Therefore, although in-vivo measurements of ^{137}Cs may be due to atmospheric deposition from worldwide testing of nuclear weapons, such measurements can be disregarded only if there were adequate measurements on individuals in the PGDP areas who had not had any occupational exposures to radioactive materials, and who could therefore serve as controls. Absent such data, the ^{137}Cs measurements should be considered evidence of occupational exposure and included in the dose reconstruction.

5.4.4.2 "Significant Incidents with Internal Dose Potential"

Table 5-12 of the TBD is entitled "Input Parameters for Significant Incidents and Events." In fact, this table presents a partial chronology of activities at the PGDP. Some of the entries, such as "guard patrolling," have a "Low Potential for Increased Radiation Exposure" (PACE/Utah 2000, Table 6.1) and do not belong in this table. Furthermore, there is no direction on how these and other events listed in Table 5-12 are to be used in dose reconstruction.

Significant information that could be useful to dose reconstructors is not included in this section of the TBD. This information includes statements by workers that urine specimens were collected within 30 minutes of an incident or accident with a potential for elevated exposure. Such a short time period does not allow for equilibrium between the inhaled activity and the concentration in the urine, most likely resulting in a false negative. Follow-up samples were collected from workers who did show elevated levels of radioactive materials in the urine. More important are the statements of former workers that the bioassays performed following such incidents were not always recorded in the database. Thus, the doses from such incidents may not be in the worker exposure records.

5.5 OCCUPATIONAL EXTERNAL DOSE

Statement of Purpose

SC&A's review of ORAUT-TKBS-0019-6, Rev. 01 (March 29, 2005) is intended to assess the technical merit as well as the completeness of guidance provided for the reconstruction of external doses associated with photon, electron, and neutron radiation environments to which workers may have been exposed at PGDP.

For practical reasons, our review and citation of issues will parallel the sequence of information and data as presented in the TBD.

Relevant Background Information

The PGDP began operation in 1952. The principal process of PGDP involves the enrichment of feed material in the form of UF_6 gas to about 2.5% U-235. Originally, most UF_6 came from depleted tails produced during normal diffusion operations at PGDP, ORNL, and Portsmouth. From 1953 to 1964, some feed material (about 17%) was recycled uranium obtained from spent reactor fuel shipped from Hanford and SRS. In brief, feed material consisted of depleted, natural, and recycled uranium.

The Site Description TBD (Turpin 2006) states the following:

> *. . . an estimated 2500 to 4000 workers worked in areas with 'moderate' to 'high' potential for increased* ***internal*** *and* ***external*** *radiation exposures. This is based on a relative ranking of the potential of radiation exposures at PGDP. These areas included the feed plant (Buildings C-410 and -420) with* ***operators*** *and* ***mechanics*** *receiving the highest doses, respectively, decontamination building (C-400) with* ***decontamination workers*** *receiving the highest doses, and the cascade buildings (C-331, C-333, C-335, and C-337) with* ***operators*** *receiving the highest doses. In addition, workers in Buildings C-340 and C-720 had the potential for increased external radiation exposures. Average doses per* ***department*** *are listed in Table 2-1 . . .* [and] *Average and maximum doses are listed in Table 2-2.*
>
> *. . . Approximately* ***10%*** *of the 2500 to 4000 workers had a potential for higher than average radiation exposures.* [Emphasis added.]

Table 2-1 cited above and Turner's Table 6-5, which incorporates the data from Turpin's Table 2-2, are reproduced herein as Table 5.5-1 and Table 5.5-2, and will be referenced in discussions below. In support of data presented in these tables, Turner (2005, Section 6.2) states the following:

> *From startup until July 1960, PGDP issued dosimeters to a* ***limited*** *number of individuals (PACE/Utah 2000). This population of* ***monitored*** *individuals represents those with the* ***highest*** *exposure potential. After July 1960, PGDP*

*routine practices required the assignment of dosimeters to **all** workers who entered a controlled radiation area…* [BJC 2001] [Emphasis added.]

Table 5.5-1. Average External Cumulative Penetrating Dose (deep dose equivalent) and Number of Workers Assigned to Each Department During 1953–1988

Dept. No.	Job Description	Average Cumulative Dose (mrem)	Number of Workers
5751	Feed plant operators	3,814	185
5760	Decontamination	2,788	116
5034	Feed plant mechanics	2,587	99
5015	Unknown	2,025	17
5676	Unknown	861	14
5730	Cascade operators	627	578
5785	Chemical operators	595	113
5075	Instrumentation	538	245
5020	Unknown	481	17
5008	Transport pool	371	33
5002	Process maintenance	364	578
5108	Environmental control	338	48
5268	Unknown	316	236
5077	Electricians	298	318
5005	Material termination mgr.	295	90
5772	PEMU decontamination	253	22
5759	Unknown	220	4
5049	Unknown	182	12
5725	Unknown	175	20
5044	Mech. Inspection	170	113
5021	Plant services	147	486
5770	Converter test	145	23
5035	Feed plant mechanics	143	160
5019	Unknown	142	13
5740	Nitrogen plant	142	22
5646	Metals building	132	95
5674	Unknown	129	8
5048	Fabrication shops	127	667
5023	Unknown	115	24
5675	Unknown	114	7
5743	Steam plant	111	61
5027	Unknown	110	282
	Total		4,706

Table 5.5-2. Average Recorded External Radiation Doses (deep dose equivalent) per Worker per Year from 1953 to 1988 including Maximum Exposures Recorded for any Single Worker for that Year

Year	Number of Workers	Average Dose, GM (rem)	Maximum Dose (rem)	GSD (rem)
1953	223	0.1398	0.820	2.14
1954	284	0.2835	1.580	2.09
1955	417	0.2419	2.500	2.72
1956	471	0.3586	4.700	3.02
1957	669	0.2517	3.190	2.97
1958	661	0.1853	3.630	3.59
1959	570	0.2015	2.360	2.88
1960	526	0.2011	2.510	2.95
1961	1,690	0.1770	2.530	3.13
1962	1,479	0.1495	2.980	3.01
1963	1,311	0.1441	3.040	3.70
1964	1,289	0.0734	1.860	4.00
1965	1,128	0.0341	1.610	5.23
1966	1,138	0.0371	1.470	5.19
1967	1,143	0.0498	1.120	3.80
1968	1,241	0.0618	1.400	3.82
1969	1,270	0.0733	1.970	4.11
1970	1.273	0.0417	0.840	3.63
1971	1,254	0.0624	1.380	3.78
1972	1,288	0.0589	1.760	4.30
1973	1,404	0.0530	1.830	4.57
1974	1,624	0.0265	1.030	4.81
1975	2,013	0.0501	1.049	3.69
1976	2,426	0.0351	1.224	4.59
1977	2,643	0.0232	0.742	4.42
1978	2,613	0.0399	0.359	2.57
1979	2,487	0.0082	0.364	5.09
1980	2,308	0.0182	0.344	3.53
1981	1,840	0.0076	0.420	5.60
1982	1,617	0.0065	0.350	5.53
1983	1,452	0.0067	0.340	5.39
1984	1,434	0.0092	0.420	5.15
1985	1,365	0.0061	0.350	5.69
1986	1,244	0.0096	0.490	5.41
1987	1,275	0.0080	0.470	5.74
1988	1,359	0.0065	0.720	7.54

Presented below are summary descriptions of issues that conflict with basic assumptions stated in the TBD, and which may impact the credibility/claimant favorability of dose reconstructions.

5.5.1 Issue 1: Questionable Assumptions for the Assessment of Recorded Skin Dose

Section 6.3.2.1 of the TBD states the following:

> *…The nonpentrating response of the **two-element** dosimeter was calculated as the difference between the "unshielded" and "shielded" portions of the film based on a uranium calibration. The two-element dosimeter workplace nonpenetrating (i.e., beta or shallow) dose response based on the uranium calibration should **adequately** represent $H_P(0.07)$ or at least be claimant-favorable because of the significant **over-response** of the unshielded portion of the film to any lower energy photons that could have been present…* [Emphasis added.]

The above quotation references a "two-element" dosimeter, yet Table 6-1 of the TBD only identifies four-element film dosimeters in use between 1953 and 1980. This "technical" discrepancy has significant implications when viewed in context with the statement that the resultant beta dose is likely to be claimant-favorable ". . . because of the significant **over-**response of the unshielded portion of the film to any lower energy photons that have been present." [Emphasis added.]

Inspection of Figure 6-1 in the TBD shows that, while this statement may apply to the **two-element** film, it appears **not** to apply to the **multi-element** film.

The method for deriving the nonpenetrating response as ". . . the difference between the unshielded and shielded portions of the film" has been questioned by the National Research Council (1989), which stated the following:

> Pages 39-41:
>
> *When a film badge with only a lead filter and an open area is exposed to unknown mixtures of beta and photon energies, it is not possible to determine contributions from each component to NOD* [net optical density] *in the film open area.*
>
> *. . . The final attempt to evaluate and report beta exposure with film badges during atmospheric testing was . . . during Operation PLUMBBOB in 1957. . . This badge has four filter areas: lead-tin laminate, open window, copper, and aluminum. This combination was thought to be capable of providing beta exposure, but the analytical procedures used were faulty . . . As stated previously, the function of NOD versus exposure is **not** linear, and NOD from a film must by converted to exposure with a common calibration curve because an increment of NOD can represent a different amount of exposure at different locations on a calibration curve. . . Thus, beta particle monitoring with personnel badges was **not** successful during atmospheric nuclear testing series.* [Emphasis added.]

Section 6.3.3.1 (page 11) of the TBD provides the following statements:

> *For a number of years, ORNL used uranium beta as well as Ra-226 gamma calibration curves to interpret film densities… The ratio of beta-to-gamma responses was tested in several ways. Films wrapped in a **7 mg/cm^2** absorber were placed in **contact** with a slab of **natural** uranium…* [Emphasis added.]

However, it was also acknowledged that because the actual dosimeter worn by individuals:

> *. . . had a **minimum** absorber thickness of **80 mg/cm^2** between the film and the source, the effective beta energy is needed to interpret the film density in terms of H_P (0.07). The radiation was routinely treated as **1.7 MeV** beta particles from uranium, which are about 40% absorbed in 80 mg/cm^2. The determination of beta dose was thus specific to uranium* [U-238].

Section 6.4.2 of the TBD ironically concludes the following:

> *The early film dosimeters were calibrated to uranium for nonpentrating radiation. **No** numerical adjustment of recorded shallow doses is recommended.* [Emphasis added.]

The high-energy beta of 2.29 MeV is emitted from Pa-234, which is the short-lived daughter of U-238. However, this simple "calibration" factor ignores the following lower-energy betas from short-lived daughters of U-238 and U-235, and from Tc-99 that would be completely absorbed by the 80 mg/cm^2 absorber:

	Beta Max (MeV)	Yield
Th-234	0.103	21%
	0.193	79%
Th-231	0.206	13%
	0.287	12%
	0.288	37%
	0.305	35%
Tc-99	0.294	100%

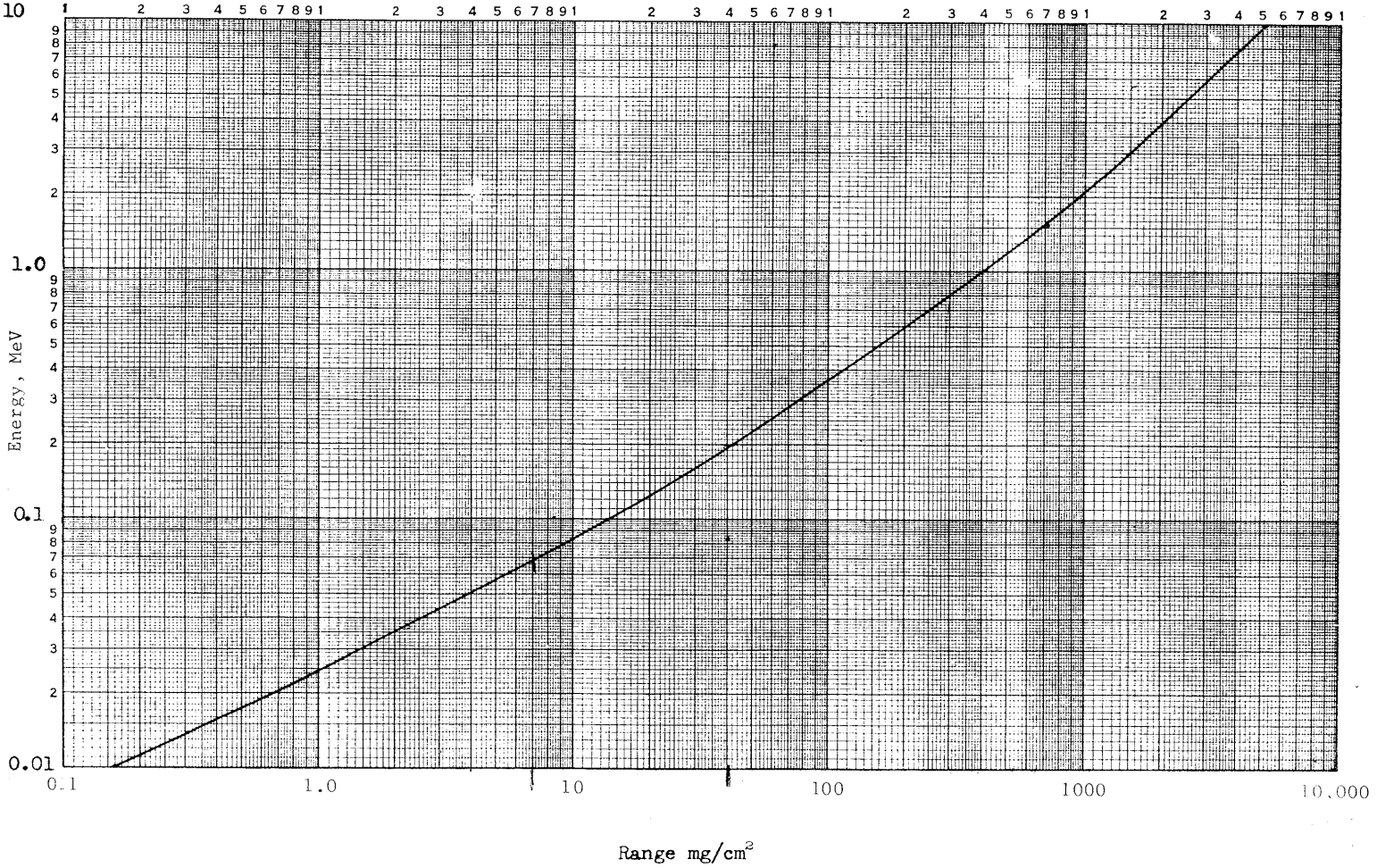
While **all** of these betas would contribute to the 7 mg/cm^2 skin dose, **none** of these betas would have penetrated the 80 mg/cm^2 absorber and registered as a "dose" on the open window (see Exhibit #1, which defines the range of betas in an absorbing medium). Inspection of Exhibit #1 shows that a beta particle with energy >300 keV is required to penetrate an absorber of 80 mg/cm^2. The magnitude of this deficiency is substantial and variable, since it is affected by the degree of U-235 enrichment and the contamination level of Tc-99.

SC&A concludes that the combination of these factors is likely to have resulted in skin dose estimates that are inaccurate, low, and not claimant favorable.

EXHIBIT #1

BETA PARTICLE

RANGE ENERGY CURVE



5.5.2 Issue 2: Questionable Assumptions for Assigning Skin and Deep Dose for Unmonitored Workers Prior to 1960 by Means of Coworker Data

As previously stated in the TBD:

> *…From startup until July 1960, PGDP issued dosimeters to a limited number of individuals…. This population of monitored individuals represents those with the* ***highest*** *exposure potential.* [Emphasis added.]

Section 6.5 of the TBD provides guidance for assigning skin and deep dose to unmonitored workers prior to 1960 by defining the following three categories of unmonitored workers:

> Worker Group 1: A zero dose was recorded, but the worker was not monitored (majority of workers from 1953 to July 1960).
>
> Worker Group 2: A zero dose was recorded from the dosimeter system for any response less than the MDL.
>
> Worker Group 3: There was no recorded dose, because workers were not monitored or the dosimetry record is not available.

Section 6.5.1 of the TBD provides the following "guidance" for assigning dose to unmonitored workers based on the "worker categories" described above.

- For **Group 2**, the missed dose should be based on *n*(MDL)/2 using data defined in Table 6-1 of the TBD.
- "If it is definitely established that the employee was **not** a radiation worker, then the unmonitored deep dose for the period can be assigned as the on-site ambient dose."
- "Otherwise, an individual in Groups 1 or 3 should be treated as a radiation worker. The unmonitored dose can then be approached in two ways."

> ***First****, the same assignment of missed dose as for Group 2, from the last column of Table 6-1, can be considered. However, for the period 1953 through July 1960, with the frequent (weekly) dosimeter exchange and relatively large MDL, the resulting implied annual missed dose of 1 rem is probably unrealistically large for many unmonitored persons in both Groups 1 and 3. Figure 6-2 shows the distribution of individual annual deep dose equivalent for monitored workers for the years 1953 to 1974…. Few of these individuals received as much as 1 rem in any given year.*
>
> *A* ***second****, alternative approach for Group 1 or 3 is to base the unmonitored dose estimate on exposure data compiled in (PACE/Utah 2000) for monitored PGDP workers. The first four columns in Table 6-5 (taken from Table 7.4 of the PACE report) show the number of monitored*

> *workers, their average recorded deep dose, and the maximum individual deep dose for each year from 1953 through 1988.* [Note: This table has been reproduced above as Table 5.5-2}.
>
> *For the shallow dose, the TBD provides Table 6-6. For Groups 1 and 3, the TBD recommends the use of the GM and GSD for assigning dose to "unmonitored workers" for the period 1953 to 1960.*

For the large number of workers who were **not** monitored before 1960, guidance provided in the TBD for assigning dose is at best incomplete/confusing and subject to speculative decisionmaking. However, of greater concern are the underlying assumptions and interpretation of monitoring data, which provide the quantitative basis for assigning dose to unmonitored workers.

It is SC&A's opinion that the monitoring data cited in Table 6-5 of the TBD (and reproduced above in Table 5.5-2) have been misrepresented/misinterpreted, and the use of these data has the potential for significantly underestimating worker doses. At the center of this concern is NIOSH's claim that prior to 1960, the population of ". . . monitored individuals represents those with the highest exposure potential."

It is SC&A's contention that these monitoring data reflect a badging practice that not only included all worker categories (regardless of their potential for exposure), but furthermore diluted the **average** dose within a given worker category by rotating badge assignments. The following circumstantial evidence supports our contentions:

- Table 5.5.-1 above purports to provide **average** external cumulative penetrating dose and number of workers assigned to each department during **1953**–1988. No fewer than **32** different departments are identified with **average** cumulative doses ranging from 3,814 mrem to a **low** of 110 mrem.

  In light of the claim that only workers with the highest potential for exposures were monitored, it is inconceivable that such selective monitoring would allow estimates for exposures back to **1953** in behalf of all **32** departments. For example, Table 5.5-2 identifies a total of only 223 workers that were monitored in 1953. The ability to monitor such a broad spectrum of workers could only be achieved by cohort badging, in which a select few individuals for any given worker category are monitored.

- According to Table 5.5-2 between 1953 and 1960, the number of monitored workers increased from 223 to 526. After this time, **all** workers who entered a controlled radiation area were monitored. Table 5.5-2 identifies that for 1961, a total of 1,690 workers were monitored. Thus, as part of this transition, an additional 1,164 workers were newly monitored workers.

  If, in fact, these **newly** monitored workers represented workers with **low** potential for exposures, the **average** worker's dose for 1961 would have been expected to drop precipitously between the years 1960 and 1961. Inspection of Table 5.5-2, however,

shows that the average value of 201 mrem for 1960 dropped only 24 mrem to 177 mrem in 1961. Furthermore, this drop more than likely had nothing to do with the "expanded" monitoring practice that now included low-exposed workers, but reflected a **general** downward trend. Thus, in 1962, the average again dropped by 27 mrem, and in 1964 dropped by 71 mrem.

- A 1964 *Paducah Plant Health Physics Program* report (Report No. KY-204, Rev. 3) authored by R.A. Winkel provides the following summary statistics, which are reproduced herein as Table 5.5-3 below.

Table 5.5-3. Summary of Radiation Exposure from External Sources Paducah Gaseous Diffusion Plant (1953–1963)
(Source: Winkel 1964)

Whole-Body Dose Range (rad or rem)	Number of Monitored Persons in Dose Range										
	Year										
	1953	1954	1955	1956	1957	1958	1959	1960	1961	1962	1963
0–1	220	273	389	418	624	631	554	506	1,609	1,418	1,261
1–2		11	25	51	36	25	15	17	75	57	40
2–3			3	10	8	5	2	4	5	4	8
3–4				4	2	1					
4–5				1							
> 5											
Total Workers	220	284	417	484	670	662	571	527	1,689	1,479	1,309

Data for the transition years of 1960 and 1961 clearly contradict the assumption/claim that prior to 1960, the monitored population represents workers with the highest potential for exposure. If that statement represented fact, then the addition of 1,162 newly monitored workers in 1961 would have been **exclusively** added to the 0–1 dose category. Instead, the data reveal that for 1960 and 1961, the number of workers with annual doses between 1 and 2 rem **increased nearly 5-fold**, from 17 to 75.

In brief, these data further suggest that the post-1960 total rad worker population did **not** appear to differ from the pre-1960 monitored worker population, as claimed by NIOSH. It further suggests that the 57 additional workers with doses between 1 and 2 rem in 1961 were **not** part of the monitored pre-1961 monitored worker group. Under current protocol prescribed by NIOSH, these 57 workers would likely be assigned the GM dose values for any years prior to 1961.

- The discrepancy between the claim of monitoring workers with the highest potential for exposure and the failure of dosimetry data to support this claim may very well be explained by yet another curious approach to monitoring workers. From a very limited body of available data, SC&A suspects that for the pre-1960 period, when badges were read on a **weekly** basis, workers may have been monitored on a "rotating basis," as suggested by Exhibits #2, #3, #4, and #5.

Exhibits #2, #3, #4, and #5 represent excerpts from summary **Inter-Company Correspondence** for the Carbide and Carbon Chemical Company that operated the PGDP in 1953. These exhibits show the following:

Exhibit #2: For the **month of December 1952**, 32 film dosimeters were used to monitor PGDP workers.

Exhibit #3: For the second week of January 1953, 47 badges were assigned weekly.

Exhibit #4: Between February 19 and March 18, 1953, weekly film dosimeter assignments increased from 56 to 77 badges.

Exhibit #5: Over a 5-week period (i.e., from April 16 to May 20, 1953) ". . . the number of [**weekly**] film badges in use was increased from 77 to 96.

These exhibits imply the following:

(1) The total number of workers monitored on a weekly basis was small, but increased over time.

(2) Locations selected from monitoring workers showed some variations.

(3) The number of monitored workers was likely weighted by the total number of workers at each location but nevertheless represented only a fraction of the total number of workers at each location.

(4) It appears that for any given location, worker monitoring was done on a rotational basis.

In summary, data in Table 5.5-2 appear to be misleading. Table 5.5-2 identifies a total of 223 workers as having been monitored in 1953. Implied in that statistic is that the same 223 individuals were monitored every week, and that their GM dose represented an annual dose of 140 mrem. The source for the average dose values in Table 5.5-2 is Table 7-4 of PACE/Utah (2000), where it is made clear that the average dose is that of the **recorded** doses. Those authors note that, "The large numbers of zero values in the database would reduce the average values below what they realistically may have been." Similarly, the maximum dose is the maximum **recorded** value. The decision to use these values to represent the median and 95th percentile of a geometric distribution seems quite arbitrary. (Note that the GSD is a dimensionless quantity and should not have been labeled "rem" in Table 6-5 of Turner (2005).)

Exhibits #2 through #5, however, suggest that in any given week, only a fraction of the total annual number of 223 workers were monitored on a rotating basis. For example, Person A may have been among the 223 monitored persons in 1953. However, it is possible that this worker may have been monitored for only 10 weekly cycles over the course of 52 weeks. Thus, records for this individual may show up to 10 positive weekly exposures, along with 42 or more zeros for cycles during which the worker was **not** monitored.

If, as SC&A suspects, such a monitoring practice took place at PGDP, then all doses identified in Table 2 have been misrepresented, and are low and not claimant favorable.

EXHIBIT #2

INTER-COMPANY CORRESPONDENCE

(ERT AME) COMPANY CARBIDE AND CARBON CHEMICALS COMPANY LOCATION Post Office Box 748 PADUCAH, KY.

Unclassified

TO H. S. Gardner, M. D.
LOCATION Medical Director

DATE January 8, 1953

ANSWERING LETTER DATE

ATTENTION
COPY TO File

SUBJECT Health Physics and Hygiene Report for December, 1952

The industrial health recheck program which was begun in November was expanded to include the Laboratory Division and the Process Maintenance and Utilities Maintenance Departments. Departmental survey and hand counting programs together with instrument assignments were outlined in conferences with Maintenance Division personnel. Personnel monitoring with film badges was initiated with 32 badges being used by operators in the feed vaporization area and by operators and maintenance mechanics in the compressor disassembly pit.

A major release of UF_6 at the temporary feed vaporization area resulted in 29 persons reporting to the dispensary and extensive uranium contamination. Uranalyses of specimens collected following this and other minor releases, during industrial health rechecks and recalls showed 91 of 143 samples to contain detectable quantities of uranium.

UF_6 Release

A major release of UF_6 gas occurred in the Temporary Feed Building at approximately 4:45 a.m. on December 1, 1952, when cylinders of liquid UF_6 were accidentally opened to the atmosphere through an evacuation line. As a result of this release, 29 Carbide personnel and 13 Construction personnel reported to the dispensary for checkup or treatment. Analysis

Unclassified

THIS FORM FOR INTER-COMPANY CORRESPONDENCE ONLY

EXHIBIT #3

K-S. MONTHLY PROGRESS REPORT FOR JANUARY, 1953

Unclassified

RADIATION EXPOSURES

No significant exposures of personnel to radiation were reported. Film badge usage up to the 14th of January showed a maximum of 60 mr/week. Forty seven badges are being used weekly. Reports with results for the remainder of the month have not been received from Oak Ridge National Laboratory.

RADIATION & CONTAMINATION

Radiation intensities accessible to personnel were some higher. The maximum intensities measured of beta + gamma (mostly beta) was 1000 mrep/hr. at the open ends of piping in the C-331 temporary UF_6 vaporization area during dismantling operations. At one foot of these lines a maximum of 55 mrep/hr was detected. Personnel on the above mentioned job were advised of the conditions and followed instructions of staying clear of close contact at open ends of piping. Also film badges were issued for use on this job. (Results not obtained). A maximum of 400 mr/hr gamma radiation was measured at surface and at bottom of empty ten-ton type UF_6 cylinders with 55 mr/hr at one foot.

Except for small confined areas contaminated from UF_6 releases, the extent and intensity of surface contamination in various operating areas remained fairly low.

CONTAMINATION & RADIATION STATUS

Month of January, 1953

Location:	Contamination Index*	Radiation Index**
C-315 Surge & Tails Building	6.6	55.
C-400 Chemical Processing Building	17.0	15.
C-310 Purge & Product Building	3.4	10.
C-400 Compressor Pit	27.7	5.
C-331 Temporary UF_6 Vaporizer		55.
C-410 Feed Vaporization Area	0.2	5.

Unclassified

EXHIBIT #4

Unclassified

Material Releases

No major releases occurred during the month. A total of three releases were reported with 11 persons reporting to the dispensary for medical attention as a result. Eleven persons, including one Goodyear employee, reported to the dispensary as a result of four unreported releases.

The majority of the above 22 people would probably have had no exposures if they had used respiratory protective equipment required for the operations which caused the releases.

Noise

Preliminary noise surveys in the C-310 Purge and Product Building show that excessive noise exists in a large portion of the building. It has been recommended that ear plugs be worn by all employees working regularly in the major portion of the C-310 Building.

Personnel Monitoring

1953

Film badge results from February 19, through March 18, showed no exposures of personnel to penetrating radiation exceeding plant acceptable amounts. A maximum exposure of 105 mr/wk was shown by badge film with an average of 8 mr/wk. The number of film badges in use has increased from 56 to 77. All pocket chambers used by personnel operating the industrial x-ray unit in C-720 showed an exposure less than ten mr/day.

Location	No. badges in use/wk	Maximum exposure (mr)	No. >50 mr	Average exposure (mr)
C-410 Vaporizer Area	33	60	1	8
C-400 Chem. Processing	35	105	2	6
Temp. Laboratory	1	30	0	4
C-315	8	75	1	Film service in C-315 was started last week of this period.--
	77			

1953

Spot checks for contamination by Health Physics

	Total No. checked	No. above PAL	Per cent above PAL
Hands	23	7	30.4
Personal Shoes	8	0	0
Personal Clothing	6	0	0
Plant Issued Shoes	14	0	0
Plant Issued Clothing	16	2	12.5
Gloves	14	6	43

PQO

Unclassified

EXHIBIT #5

Unclassified

Department		No. Pos. / Total Samples	Per Cent Positive	Average* mg U/liter
5002	Proc. Maint.	10/44	22.7	0.003
5005	Util. Maint.	0/13	0	0
5027	Conv. Maint.	2/18	11.1	0.002
5075	Inst. Maint.	2/44	4.5	<0.001
5080	Maint. Training	4/40	10.0	0.002
5731	Cascade	13/24	54.2	0.026*
5733	Cascade	7/21	33.3	0.026
5751	Feed Plant	24/55	43.6	0.006*
5760	Chem. Proc.	9/41	22.0	0.003

* In taking the average, specimens collected immediately following material releases were ignored.

Personnel Monitoring

Film badge results from April 16 through May 20 showed no exposures of personnel to penetrating radiation exceeding plant acceptable amounts. A maximum exposure of 65 mr/wk was shown by badge film with an average of 3.18 mr/wk. These results are considerably lower than those for previous months. The number of film badges in use was increased from 77 to 96. Below is a summary of film badge results for each location using them.

Location	No. Badges in use/wk	Maximum Exposure	No. > 50
C-410 Vaporizer Area	34	65	2
C-400 Isotopic Laboratory	1	0	0
C-315 Surge & Tails	8	15	0
C-720 Feed Plant Maintenance	4	40	0
C-400 Chemical Processing	47	55	1
C-720 Utilities Maintenance	2	0	0
	96		

From the results of personnel spot checks for contamination, no significant change has taken place. Below is a summary of the results of these checks.

	Total No. Checked	No. Above PAL	Per Cent Above PAL
Hands	19	4	21
Personal Shoes	6	0	0
Personal Clothing	0	0	0
Plant Issues Shoes	10	0	0
Plant Issued Clothing	19	3	15.8
Gloves	14	6	42.8

Reports from locations which routinely take and record hand counts have shown no hand counts above plant limits. Following is a list of these

Unclassified

5.5.3 Issue 3: Issues related to ORAUT-OTIB-0031

ORAUT-OTIB-0031, *External Coworker Dosimetry Data for the Paducah Gaseous Diffusion Plant* (Merwin 2005), was issued for use on August 15, 2005, or 5 months after the issuance of ORAUT-TKBS-0019-6. The purpose of OTIB-0031 is to provide guidance for assigning external doses from penetrating and non-penetrating radiation in behalf of PGDP workers with no or limited monitoring data based on site coworker data.

A comparison of ORAUT-OTIB-0031 with ORAUT-TKBS-0019-6 shows significant changes regarding the use of coworker data. These changes include (1) the use of "prorated" coworker doses that account for partial years of employment and/or incomplete monitoring records, and (2) the addition of missed doses for null dosimeter readings based on the formula n(LOD/2).

As a result of these changes, coworker data that may be assigned to unmonitored PGDP workers under OTIB-0031 increased significantly. Table 5.5-4 compares the 50th percentile (or GM if the doses are from a lognormal distribution) deep dose coworker data cited in TKBS-0019-6 with revised coworker data as given in OTIB-0031 for the years 1953 through 1960. Inspection of the data shows that the revised values are several-fold higher than those cited in TKBS-0019-6. It appears that issues raised by SC&A pertaining to the coworker deep dose model in TKBS-0019-6 have been resolved in OTIB-0031. Since both the 50th and 95th percentile doses are tabulated, the GSD for the corresponding lognormal distribution can easily be calculated.

While similar adjustments were made for the non-penetrating dose in OTIB-0031, one outstanding issue that was not addressed in the TIB is the issue of the 80 mg/cm^2 attenuation of lower-energy betas.

Table 5.5-4. Comparison of Median Annual PGDP External Coworker Deep Doses: TKBS-0019-6 versus OTIB-0031

Year	TKBS-009106[1] Pen. Dose (rem)	OTIB-0031[2] Pen. Dose (rem)
1953	0.1398	1.128
1954	0.2835	1.183
1955	0.2419	1.067
1956	0.3586	1.073
1957	0.2517	1.072
1958	0.1853	1.040
1959	0.2015	1.083
1960	0.2011	0.672

[1] ORAUT-TKBS-0019-6, Table 6-5.
[2] ORAUT-OTIB-0031, Table 2.

5.5.4 Issue 4: Issues Related to the Assessment of Neutron Exposures at PGDP

The principal source of neutron exposure at PGDP involves the α, n reaction with fluorine compounds (UF_4, UF_6). This reaction would have the highest potential for exposure to workers in areas with storage cylinders that contained either depleted UF_6 (tails) or enriched UF_6.

Other sources of neutrons involve uranium and transuranic elements that may undergo spontaneous fission and neutron-induced fission of fissile elements.

Section 6.3.2.2 of ORAUT-TKBS-0019-6, however, states that ". . . the use of the commercial Harshaw thermoluminescent neutron dosimeters (TLND) to assess neutron dose routinely (along with deep and shallow dose) began [only] in 1998." Thus, before 1998, workers were in effect unmonitored for neutron exposure. To account for missed neutron doses prior to 1998, Section 6.5.3 of the TBD provides the following guidance:

> *A neutron component should be added to the annual dose of individuals who worked in the* ***cylinder*** *yard before 1998. However, careful consideration should be given to work history. In general, only workers who were near cylinders for extended periods have the potential for neutron exposure. Estimates should be based on the neutron-to-photon ratio of 1 to 5 for dose equivalent, as determined from the survey conducted at PGDP (Meiners 1999). The neutron dose equivalent should then be multiplied by the ICRP (1990) factor of 2.* [Emphasis added.]

SC&A reviewed the referenced 1999 study by Meiners, *Paducah UF_6 Cylinder Painting Project Neutron Dose Study*. In this study, TLD measurements were taken that assessed neutron-to-gamma dose ratios for radiation fields defined by depleted UF_6 stored in cooled cylinders. Those measurements were compared to tissue-equivalent proportional counters. The study yielded neutron-to-photon ratios that ranged from 1:24 to 1:72, with an approximate average ratio of 1:5. The study also acknowledged the fact that the neutron-to-photon ratio is affected by the moderating/reflecting material in proximity to the cylinders, and the amount of UF_6 contained in the cylinder(s), as well as the degree of uranium enrichment (an increase in U enrichment increases the alpha activity and, therefore, the α, n reaction).

Lastly, SC&A reviewed guidance contained in ORAUT-OTIB-0024, Rev. 00, *Estimation of Neutron Dose Rates from Alpha-Neutron Reactions in Uranium and Thorium Compounds*, for consistency. Relevant data in OTIB-0024 are contained in Table 5-2, which identifies the neutron dose rate of 6.62E-10 rem/hr-gram of natural uranium without alpha-emitting progeny at a distance of 1 foot. For a large storage cylinder containing 100 lbs, a dose rate of 0.03 rem/hr at 1 foot is calculated. This value is consistent with empirical measurements cited by Meiners (1999) and with guidance contained in ORAUT-TKBS-00190-6.

Because workers were not monitored for neutrons prior to 1998, potential exposures to select workers have to be based on a neutron-to-photon dose ratio. The recommended neutron-to-photon ratio of 1 to 5 for estimating dose equivalent that is further multiplied by 2 to account for

the ICRP 60 neutron quality factor is regarded by SC&A as scientifically defensible for radiological environments defined by depleted UF_6 in storage cylinders.

However, a potential difficulty may arise for deriving unmonitored neutron doses for workers prior to 1961 who were **unmonitored** for photons as well.

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

The SC&A procedures 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 pertaining to how the profile satisfies its intended purpose and scope. This section addresses the latter objective in a summary manner by evaluation of (1) how, and to what extent, the site profile satisfies 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 found that the NIOSH site profile for PGDP presents an adequate accounting of the primary internal and external issues related to predominant radionuclides, such as uranium, plutonium, and some fission products. The PGDP site profile falls short in fully characterizing a number of key underlying issues that are fundamental to guiding dose reconstruction. In some cases, these issues may impact other site profiles. Many of the issues involve lack of sufficient conservatism in key assumptions or estimation approaches, or incomplete site data or analyses of these data. Section 6.0 summarizes the key issues. A detailed evaluation of these issues is provided elsewhere in the report.

Objective 1: Completeness of Data Source

The breadth of data sources used as a basis for the PGDP site profile is evident in the several hundred reports available for PGDP in the Site Profile Research Database. Over 100 reports were cited in the site profile references, while others served to provide confirmatory information, or were only recently retrieved. The NIOSH/ORAU team consulted health physics personnel with long histories at PGDP who have extensive knowledge of key dosimetry historical processes and personnel monitoring data. There was a meeting with the PGDP with the Construction Council and PACE on February 10 and February 11, 2005, respectively, in order to identify worker concerns and discuss the TBDs. This interaction has helped to provide valuable insight into site operations and processes. In addition, the issuance of supporting TIBs reflect the ongoing effort by NIOSH to continually improve guidance provided to dose reconstructors.

The current version of the TBD is silent on the number of workers with zero dose, which makes it difficult to characterize the external dose distribution for the purpose of building a coworker model. Although the TBD has addressed major operations that occurred at PGDP, there are gaps with respect to the smelting operations, Work for Others program, onsite burning of contaminated material, and work supporting the weapons program. PGDP was integral in the

destruction of diffusion barriers removed as a part of the Cascade Upgrade Program for not only itself, but also for K-25 and the Portsmouth Gaseous Diffusion Plant. This introduces some uncertainty into the type of radionuclides handled and the relative concentrations of these radionuclides.

Information in the TBD regarding the radionuclide composition of recycled uranium, as provided in the PACE report, is incomplete, which places constraints on the ability to reconstruct both the external and internal exposures. In addition, the list of transuranic radionuclides handled at the site appears to be incomplete. Table 5-5 of the TBD limits TRU to ^{237}Np and ^{239}Pu. However, as was discussed in the review of Table 5-4, the TRU in the Hanford reactor tails and other sources include ^{238}Pu, ^{240}Pu, and ^{241}Am. These nuclides should be included with ^{237}Np and ^{239}Pu in Table 5-5, to be consistent with Table 5-4 (which lists all but ^{238}Pu), SAIC (1999, p. 14), and Hill and Strom (1993, Table 16.2, Note "a"). Finally, a full disclosure of incidents at the site is not provided, nor are directions provided to dose reconstructors where incident data can be found.

NIOSH did not make use of key information sources. The data in PACE/Utah 2000, which lists activity fractions for TRU and fission products associated with specific processes, was not adequately considered. A draft report on recycled uranium mass balance (BJC 2001), which is cited as a source of information by Turpin (2006), was also not adequately considered. In terms of environmental data, the USEC Characterization Study was not considered, although it provided an extensive characterization of the site prior to leasing the facilities to United States Enrichment Corporation.

Objective 2: Technical Accuracy

Several issues have been identified with regard to the accuracy of the dose reconstruction protocols recommended in the site profile. The principal technical accuracy issues include underestimating the enrichment levels of the uranium to which workers were exposed, and not addressing the complete list of TRU and fission products that were present in recycled uranium. In addition, for those TRU and fission products that are addressed, it appears that the quantities may have been underestimated.

The TBD presents inadequate, and sometimes inaccurate, guidance to dose reconstructors on estimating the intakes of uranium, including the assumed specific activities of other radionuclides not analyzed by the bioassays. There is no guidance as to how the data on air concentrations can be used in dose reconstruction. Although we agree with the primary reliance on urinalysis, supplemented by in-vivo measurements where such records are available, measured air concentrations could be used to supplement the bioassay results when the bioassay records are missing or fragmentary. They could be especially useful in assigning relative specific activities of isotopes other than uranium. There is no discussion of the uncertainties in the determination of uranium in urine samples for fluorimetry and KPA. The uncertainty of the fluorimetry methodology can be very high—it is strongly dependent on the quenching of the sample. For other radionuclides, there is no description of the technique used for their measurements and the accompanying uncertainties.

The practice of offsite collection of samples that takes place 24- to 48-hrs after leaving the plant would lead to a lowering of the calculated intakes. The dose reconstructor is not instructed to be alert to any cases for which urine samples were collected after the worker was off work for any period of time. Appropriate adjustments to the calculated intake should be made to compensate for the lowered concentration in the urine following an absence from work.

It is not clear if the TBD intends to give dose reconstructors the option of subtracting 0.43 μg from the 24-hr urine sample analyzed by techniques other than fluorimetry or KPA. Such guidance would be neither scientifically correct nor claimant favorable. First, the value cited by SAIC (1999) (the TBD refers to this document as BJC) applies to residents of the Oak Ridge, Tennessee, area, not of PGDP. Second, it is an upper limit (95^{th} percentile). Subtracting such a high-end value from the urinalysis results would result in a bias against the claimant. The TBD should state unambiguously that the results should not be corrected for any assumed contributions of uranium in the environment.

Table 5-10 presents the MDAs for the measurement of various radionuclides by whole-body or chest counting. The sources cited, however, do not present the values listed, with the sole exception of ^{237}Np for the 1968–1991 period. Some of the values are inconsistent. The TBD gives no instruction to the dose reconstructor on estimating uncertainties in the in-vivo measurements, except to state, "For results near or at the reporting levels, dose reconstructions should apply the prescribed standard deviation of 0.3 times the MDA or reporting level…" While we do not disagree with that recommendation, clearer guidance is needed for higher levels. The uncertainties for in-vivo bioassay include the uncertainty in the counting statistics, including the background that is subtracted from the count rate; the uncertainty of translating a count rate from a given worker into a body burden, given the differences in morphology among different individuals; and the variation in the naturally occurring radionuclides in the human body (primarily ^{40}K) that contribute to the background. These MDA values should be verified and documented to enable verification by an independent reviewer.

Table 5-4 lists default specific activities to be used when only total uranium results are available. The report does not explain when this table should be used instead of Table 5-1, which gives percentages of uranium, neptunium, and plutonium activities at specific facilities, or Table 5-2, which presents default specific activities of uranium isotopes, as discussed above. This is a source of possible confusion for dose reconstructors. Not only do Tables 5-2 and 5-4 employ different units—Bq/μg and nCi/g, respectively—the unit conversions for the uranium isotope are imprecise. The specific activities of the uranium isotopes in Table 5-4, when converted to Bq/μg, are consistently about 3% higher than the default activities listed in Table 5-2. Although this is not a significant difference in terms of dose reconstruction, it is another example of the lack of accuracy and scientific rigor in the report. The table appears to be an amalgam of data from various sources—the report does not indicate how the values were calculated.

Information about the occurrence of TRU nuclides at different facilities is presented by PACE/Utah (2000, Appendix D), which lists various radiological data. Air sampling, radiation survey, and urinalysis data indicates the presence of one or more TRU nuclides in buildings or facilities C-310, C-315, C-331, C-333, C-335, C-337, C-340, C-360, C-400, C-409, C-410, C-420, C-710, C-720, C-746, and C-746B. In addition, Hill and Strom (1993, Table 16.2) list

TRU nuclides at Buildings C-746A and "C-7460" (perhaps a typo for C-746Q). Finally, SAIC (1999, Table 1.7) lists TRU activities at C-333A, C-337A, C-404, C-411, and C-746Q. Since the dose assessments should always give the claimant the benefit of the doubt, the list of radionuclides in Table 5-5 should be all inclusive; therefore, the complete list of TRU nuclides should be included for each of these facilities. This requires expanding the radionuclide lists for C-310, C-315, C-337, C-333A, C-337A, C-340, and C-404.

The particle sizes of at least some of the aerosols appear to be substantially smaller than the 5 micron AMAD assumed as the default particle size throughout the TBD. The most claimant-favorable lung clearance types were not always recommended for the purpose of reconstructing internal doses. Recommended default assumptions regarding bioassay frequencies are not always claimant favorable, and no guidance is provided in the TBD to address the possible effect of the day of urine sample collection on the ability to reconstruct radionuclide intakes based on bioassay data.

Tables presented with external monitoring summary data appear misleading. Table 5.5-2 identifies a total of 223 workers as having been monitored in 1953. Implied in that statistic is that the same 223 individuals were monitored every week, and that their GM dose represented an annual dose of 140 mrem. The source for the average dose values in Table 5.5-2 is Table 7-4 of PACE/Utah (2000), where it is made clear that the average dose is that of the **recorded** doses. Those authors note that, "The large numbers of zero values in the database would reduce the average values below what they realistically may have been." Similarly, the maximum dose is the maximum **recorded** value. The decision to use these values to represent the median and 95th percentile of a geometric distribution seems quite arbitrary. Information suggests that in any given week, only a fraction of the total annual number of workers were monitored on a rotating basis.

The principal source of neutron exposure at PGDP involves the α, n reaction with fluorine compounds (UF_4, UF_6). This reaction would have the highest potential for exposure to workers in areas with storage cylinders that contained either depleted UF_6 (tails) or enriched UF_6. Other sources of neutrons involve uranium and transuranic elements that may undergo spontaneous fission and neutron-induced fission of fissile elements. Because workers were not monitored for neutrons prior to 1998, potential exposures to select workers have to be based on a neutron-to-photon dose ratio. The recommended neutron-to-photon ratio of 1 to 5 for estimating the dose equivalent that is further multiplied by 2 to account for the ICRP 60 neutron quality factor is regarded by SC&A as scientifically defensible for radiological environments defined by depleted UF_6 in storage cylinders.

The methods recommended for reconstructing shallow doses may result in an underestimate of doses, because no consideration was given to the methods used to calibrate the dosimeters. The coworker model for reconstructing doses to workers in the early years (i.e., pre-1960s) may be unreliable because of possible significant differences in work practices in the early years as compared to the later years where more abundant external dosimetry data are available.

After discussion with NIOSH personnel, it was their decision to limit occupational medical exposure to those chest exams described above, and to conclude that all other exposures are part

of worker background. SC&A believes such an interpretation is not claimant favorable to those most at risk. Our concern is that specified "high-risk" workers, those most likely exposed to radiation and beryllium, would be at risk of having an incomplete dose assessment if not all radiation associated with medical screening for job-related activities were included. Since, all radiation provides some risk and, arguably, is cumulative, workers warrant consideration of all forms of work-related x-ray exposure to be claimant favorable. SC&A believes NIOSH should review its interpretation of included medical exposure, and should reasonably adopt a broader interpretation of occupational medical dose, as provided in the most recent version of Kathren and Shockley (2005). This is particularly important for skin cancer that is not one of the SEC-defined cancers. The methods used to reconstruct occupational medical doses appear to underestimate the doses and do not characterize the full extent of the uncertainty in the doses.

The TBD does little to reasonably document the variety of medical occupational exposures, and the lack of documentation on the type of equipment and the maintenance records do little to assure that a conservative and claimant-favorable estimation of dose is possible. This circumstance would suggest the need to reconsider a worst-case approach to establishing dose. The Occupational Medical Dose TBD in Section 3.2 states that PFG units, although generally available up to the late 1950s at most DOE sites, were not used at the PGDP. The undocumented absence of PFG units at PGDP clearly has significant dose implications to workers who may have been given much higher doses from PFG units. SC&A believes it is not claimant favorable to instruct dose assessors to use kerma (dose) values of 200 mrem and 100 mrem for chest radiography prior to 1975. To be fully claimant favorable, it would be appropriate to instruct dose assessors to use an annual dose of 3.0 rem per year for chest radiographs, in accordance with guidelines set forth (Kathren and Shockley 2005) until 1959, when you can assume all PFGs were no longer used at DOE sites.

The TBD purports to describe "potential exposures from ambient sources while working outside the process buildings," but the ambient monitoring data in the document are for measurements at the site boundaries and beyond. There are no corroborating data provided to demonstrate that these measurements are consistent with the levels that workers might experience while working outside the process buildings. Radionuclide-specific concentrations in air, soil, and water were not discussed in the TBD. Exposures from burning of contaminated materials were not considered in the assessment of ambient internal and external dose. There is little characterization of the releases in Tables 4-2 and 4-3. There is no discussion of what release points were considered, or the physical form of the uranium and ^{99}Tc release data in Tables 4-2 and 4-3, other than to indicate that they were obtained from DOE, Bechtel-Jacobs, and United States Enrichment Corporation reports.

Section 4.3, External Dose, does not consider the radionuclides in the depleted uranium cylinders providing the gamma radiation for the increasing external exposure rates near the depleted uranium cylinder storage yards. Early survey results of about 0.02 mrem/hr (probably measured as 20 μR/hr) would correspond to 40 mrem in a 2000-hr period, or about 175 mrem/yr for continuous exposure. For comparison, the average of the 1993–2001 reported background measurements in Table 4-4 correspond to a 100 mrem/yr rate for continuous exposure. The uncertainty for environmental dose deals less with estimating the uncertainties in measured or

calculated values than with outlining default values and assumptions to be used when data are unavailable.

Objective 3: Adequacy of Data

There appears to be a paucity of external dosimetry data prior to the early 1960s, and the reconstruction of external doses during those early years requires the application of a coworker model. Similarly, it appears that reliable neutron monitoring did not begin until 1998. As a result, reconstruction of neutron exposures from alpha/n reactions and spontaneous fissioning must rely on a coworker model. Data gaps also exist in the environmental dose TBD, because the data are limited to measurements made at the site boundary and do not include external exposure measurements performed onsite at worker locations.

It was the policy of PGDP when film badge and urine sampling showed a "lack of value" that these services be discontinued (Emlet 1957). No explanation for the term "lack of value" is provided. This determination to discontinue these services introduced a potential opportunity for unmonitored exposures. During site expert interviews, there was indication that not all individuals participated in in-vivo monitoring, and that a cohort-monitoring program was utilized. A quantitative neutron-monitoring program was not implemented until 1998. Prior to this date, there was no documented neutron dose. These conditions should be considered during the dose reconstructions.

The Occupational Medical Dose TBD is also deficient in that little documentation exists to validate x-ray protocols and equipment maintenance, and upkeep of records prior to 1995, after which the State of Kentucky commenced surveys of the unit. The TBD uses information derived from the TIB (Kathren 2003) to estimate dose impacts.

Objective 4: Consistency among Site Profiles

An extensive comparison was performed by SC&A to compare and contrast the methodologies used in the PGDP site profile and other site profiles reviewed to date. These comparisons focus on the methodologies and assumptions associated with dose assessments and the derivation of values used to obtain a probability of causations (POC) for individual claimants. A detailed analysis is provided in Attachment 4 to this report.

The default values assigned for determining medical exposure are relatively consistent among site profiles. The site profiles do not always apply the same revision of this ORAUT-OTIB-0006, as is the case with the ORNL site profile. This should be corrected in subsequent revisions of the TBDs. A more consistent approach to determining when dose from PFG is to be assigned is needed in cases where there is an absence of site-specific information. Other deviations from the standard assumptions are based on site-specific information.

The default assumptions used for the calculation of environmental occupational dose are the same or similar to other site profiles. The PGDP TBD assumed an intake calculation based on a chronic intake over a year, with a breathing rate of 2,400 m^3 of air in a year. External ambient exposure was based on gamma exposure rates on mrem by location. An occupancy of

2,000 hours per year was used. This is consistent with other site profiles, such as LANL, ORNL, and others. Review of site profiles to date indicates that the NIOSH/ORAU team has not come to a consensus on what components should be considered in the environmental dose. In the case of the PGDP Occupational Environmental Dose TBD, no consideration was given to dose from contaminated soil resuspension considered in the RFP site profile.

There is an inconsistency within the TBD related to the type of dosimeters used between 1953 and 1980. In one location, the TBD indicates a "two-element" dosimeter was used, while Table 6-1 of the TBD only identifies four-element film dosimeters in use between 1953 and 1980. This "technical" discrepancy has significant implications when viewed in context with the statement that the resultant beta dose is likely to be claimant favorable, "…because of the significant **over**-response of the unshielded portion of the film to any lower energy photons that have been present."

At PGDP, whole-body counting was performed using a mobile counter provided by the Y-12 Plant that was sometimes referred to as the Martin Marietta Energy Systems (MMES) Counter. Table 5-10 on page 18 of the PGDP Occupational Internal Dose TBD (Berger 2004) lists general information about the detection capabilities of this counting system for various periods, and provides MDAs for various operational periods at PGDP. In the Fernald Occupational Internal Dose TBD (Rich 2004, pg. 34), Table 5-22 provides a similar table of MDA for the same mobile counter, which shows that the MDA for non-uranium radionuclides was 0.1 pCi/sample throughout most of the years. Uranium in-vivo MDAs for the lung at Fernald are provided in Table 5-26 on page 36. Whereas the MDA at PGDP was 4 mg for U-238 from 1968 to 1980 (Berger 2004, Table 5-10, pg. 18), the MDA for U-238 from 1989 to 2001 at Fernald was 7.4 mg (Rich 2004, Table 5-26, pg. 36). The lung counting detection levels at Y-12 using two 9" x 2.5" NaI detectors varied from 13.5 mg in 1959 down to 4.5 mg when the HP germanium detectors were installed (Rich and Chew 2006, Table 5-12, pg. 32).

Exposure geometry is not dealt with in the PGDP Occupational External Dose TBD. At ORNL, the Occupational External Dose TBD (Burns and Mohrbacher 2004, pg. 23) points out that 100% Anterior-Posterior (AP) exposure geometry has been assumed. The Occupational External Dose TBDs for Y-12 (Murray 2003), SRS (Scalsky 2005), and Hanford (Scalsky 2003) base their default exposure geometry on the compensability or non-compensability of the claim. The MCW (Westbrook 2005) and RFP (Furman and Lopez 2004) Occupational External Dose TBD bases default exposure geometries on job titles. The INEEL (Rohrig 2004) Occupational External Dose TBD defaults to 100% AP exposure.

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 TBDs for adherence to general quality assurance policies and procedures utilized for the performance of dose reconstructions. NIOSH has complied with the hierarchy of data required under 42 CFR Part 82 and its implementation guides.

The TBDs' 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.

The particle size in each case is specified as 5 μm AMAD. Such a particle-size assignment is not supported by the data, nor is it claimant favorable. This decision to ignore site-specific data is contrary to both the intent and the letter of 42 CFR 82.

6.2 USABILITY OF SITE PROFILE FOR INTENDED PURPOSES

SC&A has identified seven criteria that reflect the intent of the EEOICPA 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 PGDP site profile TBDs 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 Department of Labor POC estimate at the upper 99% confidence level

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

Inconsistencies and Editorial Errors in the Site Profiles

Several inconsistencies and editorial errors were noted in the TBD and are provided below.

- Americium-241 is misidentified as a decay product of ^{239}Pu. The TBD (pg. 7) states "…^{241}Am, a decay product of ^{239}Pu, builds up as ^{239}Pu decays." Americium-241 is not a decay product of ^{239}Pu; it is a decay product of ^{241}Pu (EPA 1999, Table G.1).

- The air sampling flow rate characterization is confusing. Section 4.2.2 states that according to the September 1962 Air Sampling Procedure, the sampling flow rate was 0.3 cfm (11 cm/s). The 11 cm/s value is the face velocity and not a flow rate equivalent to 0.3 cfm, as the sentence implies.

- The predominant wind direction in Section 4.2.2 is misstated. The wind rose in Figure 4-1 shows the frequency of winds blowing **from** the indicated directions. The text on pg. 7 of Section 4.2.2 incorrectly identifies the wind as blowing predominantly **toward** the southeasterly direction, rather than as predominantly **from** the south-southwest direction.

- There appear to be some discrepancies between Tables 4-3 and 4-4 and their descriptions in the text. In Table 4-2, alpha concentrations are assigned for 1952–1957, 1990, and 1993 (rather than 1952–1956, 1990, and 1994, as described in Section 4.2.4). In Table 4-3, beta concentrations are assigned for 1954–1960, 1995, and 1997–2001 (rather than 1952–1958, 1989, 1993, and 1995–2001, as described in Section 4.2.4). The assigned values are not consistent with the calculations described.

- The first paragraph of Section 5.2: "Source Term," singles out ^{230}Th and ^{234m}Pa as uranium progeny "of dosimetric interest." This designation of ^{230}Th is apparently an error; DOE (2000) cites ^{234}Th and ^{234m}Pa, the short-lived progeny of ^{238}U.

These errors may cause confusion in the dose reconstruction process and should be modified in subsequent revisions of the TBD.

6.3 UNRESOLVED POLICY OR GENERIC TECHNICAL ISSUES

A number of issues were identified that are common in the PGDP and other site profiles reviewed to date 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, metal tritides, high-fired plutonium oxide), 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 indicated that it may develop separate TIBs

in order to address these more generic issues. The following represents those issues identified in the PGDP and previous site profile reviews 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 generic TIBs to individual dose reconstructions for particular sites is absent.

(2) Mobility of the 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 further considered and substantiated.

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

(5) The significance of various exposure pathways and the assumptions made that influence dose contributions need to be considered (most notably) for solubility, oro-nasal breathing, and ingestion.

(6) Analysis needs to be performed regarding how "frequent or routine incidents" should be addressed, given the possibility that such "spike" exposures often may be missed by routine monitoring as a function of how often and in what manner it was conducted.

(7) Availability of monitoring records for "transient or outside workers," e.g., subcontractors, construction workers, and visitors, who may have potential exposure while working on or visiting a facility should be ascertained.

(8) Dose to decontamination and decommissioning (D&D) workers should be assessed. Many facilities have large-scale D&D operations, which extend back many years. Decontamination and decommissioning operations often require working in unknown situations, which may provide unique exposure situations.

(9) Dose from non-traditional chemical forms of radionuclides, such as high-fired oxides and tritides, requires evaluation.

(10) Dose reconstruction for occupational medical exposures remains incomplete. NIOSH needs to reconsider the definition to include all forms of medical radiation exposure to ensure its considerations are claimant favorable.

Quality Assurance on records provided by the site to the NIOSH/ORAU team is necessary to ascertain whether complete information is being provided.

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ATTACHMENT 1: RICHARD MILLER LETTER TO HHS AND NIOSH

Letter from Richard Miller to the Secretary of the Department of Health and Human Services and the Director of the National Institute for Occupational Safety and Health

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December 9, 2005

Hon. Michael Leavitt, Secretary
Dept. of Health and Human Services
200 Independence Ave., SW
Washington, DC 20201

Dr. John Howard, Director
National Institute for Occupational Safety and Health (NIOSH)
2000 Independence Ave., SW
Washington, DC 20201

Dear Secretary Leavitt and Dr. Howard:

We are writing to request that you personally review a NIOSH Contract Oversight Assessment Report which (1) blesses a NIOSH subcontractor's conflict of interest who was estimating radiation doses for use in the Energy Employees Occupational Illness Compensation Program (EEOICP), and (2) overlooks how this conflict of interest led to the use of incorrect data and contributed to a loss of credibility in the science used within the NIOSH program.

The "Assessment of Potential Conflict of Interest Involving ORAUT-TKBS-0019-5 Technical Basis Document for Paducah Gaseous Diffusion Plant–Occupational Internal Dose" has prompted NIOSH staff to weaken—rather then strengthen– its governing Conflict of Interest (COI) Policy. This will increase the number of people with conflicts of interest writing key scientific documents on behalf of the federal government, and adversely impact the credibility of the government's radiation risk estimates used to approve or deny compensation claims.

The attached white paper discusses how a NIOSH-funded subcontractor, Carol Berger, was allowed, due to loopholes in the program's conflict of interest restrictions, to review her own previous radiation dose assessments for a DOE contractor (Martin Marietta Energy System) at the Department of Energy's (DOE's) Paducah, Kentucky site. It appears that bias towards her own previous work caused her to use faulty science in preparing a NIOSH site profile at the Paducah site. This caused radiation dose to be underestimated.

NIOSH does not dispute that Ms. Berger reviewed, cut and pasted tables from her 1992 report for Martin Marietta into a September 2004 NIOSH site profile used for reconstructing radiation doses and making compensation decisions. NIOSH found she never considered published data which contradicted her previous work for Martin Marietta. NIOSH did not find a violation of its conflict of interest policy. While NIOSH acknowledges that the Paducah site profile should be reviewed and revised to account for the data overlooked by Ms. Berger, they took the wrong message from this case.

Instead of admitting a mistake and fixing it, NIOSH staff used this oversight report to weaken conflict of interest restrictions nationwide. Given comparable conflict of interests impacting NIOSH site profiles in Hanford (WA), Rocky Flats (CO), Pantex (TX) and Idaho

National Labs (ID), there is a need for NIOSH to develop a broad and effective remedy.

Attached is a white paper which:

1. Outlines where the NIOSH contractor oversight team report failed to identify three violations of the conflict of interest policy in its report "Assessment of Potential Conflict of Interest Involving ORAUT-TKBS-0019-5 Technical Basis Document for Paducah Gaseous Diffusion Plant–Occupational Internal Dose";
2. Explains that this NIOSH oversight report overlooked the nexus between the appearance of a conflict of interest and potential bias which undermines NIOSH's scientific credibility; and
3. Analyses how the revised NIOSH/ORAU Conflict of Interest Policy (effective October 11, 2005) weakens important safeguards against conflict of interest, expands the number of people writing critical documents who have a conflict of interest, and how the new policy endangers the scientific credibility of the NIOSH program determinations under EEOICPA.

We recommend that you personally review the new COI policy; request the Advisory Board on Radiation and Worker Health to review the technical and policy issues in the Oversight Team Report and report back to you; and in the interim, suspend the revised COI policy pending your assessment. Detailed recommendations on next steps are outlined in the attached white paper.

We would be pleased to brief you in detail. Please contact me at 413-536-3858 or 413-531-5787 (cell) if you have any questions.

Sincerely,

Richard Miller
Senior Policy Analyst

Encl: White Paper Assessing NIOSH Program Conflict of Interest

Cc: Advisory Board on Radiation & Worker Health
Bill Cossler, Vice President, USW Local 5-550 (Paducah)

WHITE PAPER ASSESSING NIOSH CONTRACTOR CONFLICTS OF INTEREST AS THEY IMPACT THE ENERGY EMPLOYEES OCCUPATIONAL ILLNESS COMPENSATION PROGRAM ACT

GOVERNMENT ACCOUNTABILITY PROJECT (GAP)

DECEMBER 8, 2005

SUMMARY

This white paper:

1. Outlines how a National Institute for Occupational Safety and Health (NIOSH) oversight report failed to find three violations of the Oak Ridge Associated Universities (ORAU) Conflict of Interest policy in its report *"Assessment of Potential Conflict of Interest Involving ORAUT-TKBS-0019-5 Technical Basis Document for Paducah Gaseous Diffusion Plant–Occupational Internal Dose" (OCAS-COT-0015);*
2. Shows that this NIOSH Oversight Report skirts the essential issue: whether a conflict of interest possessed by a "Subject Expert" has undermined the quality of the science and led to the underestimation of radiation risk from exposure to transuranic (TRU) compounds such as plutonium and neptunium;
3. Analyses how recent revisions to the NIOSH/ORAU Conflict of Interest policy weaken important safeguards against conflict of interest and expand the number of people with a conflict of interest writing critical documents, and explains how these changes threatens the scientific credibility of the NIOSH program determinations; and
4. Recommends interim steps to review and remedy the problems.

I. CONTRACT OVERSIGHT TEAM REPORT FAILED TO IDENTIFY THREE VIOLATIONS OF THE ORAU CONFLICT OF INTEREST POLICY:

ORAU, which is under a $200 million contract to NIOSH to perform radiation dose assessments under the Energy Employees Occupational Illness Compensation Program Act (EEOICPA), hired a subcontractor, (Carol Berger), to write "the bulk of" the site profile for the Paducah site, yet this individual was previously employed under contract to Martin Marietta Energy Systems, the DOE's Management and Operating (M&O) contractor at Paducah, Kentucky, to assess worker exposure to transuranic (TRU) compounds from processing over 100,000 tons of "recycled" uranium[1]. While consulting for Martin Marietta, the company was a defendant in claims involving worker exposure to radiation. NIOSH concluded there was no violation of the COI Policy, yet several violations were overlooked or were not investigated.

Violation #1: ORAU failed to issue COI disclosure statement for Carol Berger

The ORAU COI Policy (in effect at the time of the Paducah site profile) states:

[1]Personnel Exposure Potential to Transuranic Materials at the PGDP ("IT Report"), Martin Marietta Energy Systems, Inc., September 2, 1992, IT Corporation / Nuclear Sciences, Carol D. Berger, CHP

> "ORAU is committed to full and open disclosure as the best way to prevent conflicts of interest. ORAU agrees completely that "Sunshine is the best disinfectant."
>
> "The ORAU Team will disclose, for each company and for each individual involved in dose reconstruction, preparation of Site Profiles, research supporting determinations of whether or not to add a class of employees to the SEC, or any other work done by primary authors or reviewers for NIOSH on dose reconstruction or SEC petitions on behalf of the EEOICPA program, information about their past and present work at DOE sites."

When the site profile was released in September 2004, Carol Berger's conflict of interest disclosure was not made public. She was not listed on the ORAU web site for the team preparing the Paducah site profile. Her COI disclosure only became public after GAP notified ORAU Project Director Dick Toohey in February 2005. Failure to follow the disclosure requirement was not noted in NIOSH's Oversight Team Report, yet the Report concluded that a "violation of the then-current COI Policy did not occur." Transparency is a central element in managing the COI, and failure to comply is a *de facto* violation of the COI policy. NIOSH appears to have made no efforts to ensure disclosure of Ms. Berger's conflicts, and neglected to point out their own failures to carry out adequate oversight in this regard.

Violation #2: Apparent Failure to Maintain Adequate COI Data Base

NIOSH determined that having someone who was part of a Department of Energy (DOE) contractor's health physics program write "the bulk of" a chapter in a site profile was not a violation of the prohibition in the ORAU policy. However, ORAU's policy in effect at the time requires that it maintain a database accessible to ORAU's EEOICPA team, to NIOSH and to the public (subject to privacy act limitation) which identifies:

> "Whether ORAU and its subcontractors and their employees are reviewing reports, assessments, surveys, documents and records that they organizationally or individually have been responsible for authoring, developing or submitting to DOE or its contractors, ORAU will further indicate if ORAU, a subcontractor or individual employees of ORAU or a subcontractor was an unidentified contributor to any such reports, assessments surveys, documents or records."

NIOSH's Oversight Report doesn't disclose whether the previous work of Ms. Berger was identified in this COI database. Since the Report did not discuss compliance with this disclosure requirement, we presume there was no such examination. If this is in error, NIOSH should clarify. Surely if ORAU was managing its conflicts of interest, and NIOSH was policing the process, a disclosure that Carol Berger was reviewing her own previous work would have put up a red flag to ORAU management, NIOSH and the public, and triggered efforts to mitigate the problem.

Violation #3: Individuals were allowed to perform, review or approve site profiles at sites where they were previously employed or for contractors for whom they were previously employed.

The ORAU COI Policy states:

> "No individual will perform, review or approve radiation dose reconstructions, site profiles or determination of whether or not to add a class of employees to the SEC for co-workers, DOE facilities at which they were formerly employed, or for contractors by whom they have been employed. Site experts may be employed to advise on site specific issues and incidents as necessary."

It is not in dispute that Carol Berger was employed by a DOE contractor at Paducah or that she wrote "the bulk of" the NIOSH site profile for Paducah. NIOSH explains away the evident conflict. They assert that writing the "bulk of" a site profile does not exceed the limited role allowed to "Site Experts" in the COI Policy, which (as cited above) is limited to "advise" on site specific issues "as necessary." Citing OCAS Senior managers Larry Elliott and Jim Neton (who themselves have a conflict of interest for failing to police conflict of interest and failing to identify technical weaknesses in the Paducah site profile, respectively), the Oversight Report concludes that "Subject Experts may be tasked with drafting the document for which they are designated as the Subject Expert." What NIOSH has done is make restrictions on those with an undisputed conflict of interest so elastic as to defeat the protections in the COI Policy.

Given the "management philosophy" to allow individuals with a conflict to write site profiles, the NIOSH Oversight Team found that a violation of "the then-current COI Policy did not occur." The Team concluded that while the "language of the [COI] policy was ambiguous, the underlying policy intent was followed."

Today, Carol Berger is listed on ORAU web site as a "primary author" of the Paducah Site Profile and indicates that she has no conflicts of interest. NIOSH's new policy, as implemented, allows individuals with a conflict of interest to serve as primary authors and review their previous work without limitation. Rather than fixing the conflict and admitting a mistake, NIOSH has chosen to water down the conflict of interest restrictions.

II. Scope of NIOSH Oversight Report Ignores the Core Issue: Is Failure to Manage Conflict of Interest Creating Faulty Science?

The NIOSH Oversight Report selected only two of the many issues raised in GAP's February 20, 2005 memo:

> 1) Did the involvement of an individual who had previously performed health physics work at the Paducah Gaseous Diffusion Plant, and subsequently served as the Subject Expert on ORUT-TKBS-0019-5, violate existing conflict of interest policies;

2) Does the Technical basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose developed under the circumstances presented in question 1 above take full advantage of and use the best available data for completing dose reconstructions.

NIOSH's framing of the issues skirts two central questions:

- Did the contractor's conflict of interest in reviewing her previous work for a DOE contractor taint the quality of science that was used in the NIOSH dose reconstruction program? If so, what does this imply about ORAU subcontractors with comparable conflicts of interest at Hanford, Rocky Flats, Idaho, and Pantex?

- How could ORAU's designated "Subject Expert" overlook data in publicly available literature (specifically, the alpha activity ratios used to estimate intakes of transuranics) which indicates that radiation risks were far greater? Why didn't four layers of quality assurance/quality control within ORAU and NIOSH catch Ms. Berger's errors?

NIOSH does not dispute that Carol Berger reviewed and duplicated her own previous work performed for Martin Marietta Energy Systems. She simply cut and pasted alpha activity ratio tables from her 1992 report for Martin Marietta into the NIOSH site profile.

NIOSH acknowledges that the Paducah site profile should be reviewed and revised as necessary to ensure that the best available data is used. Specifically, NIOSH concurs that the transuranic concentrations documented during operations of the ash receiver and pulverizer, and during the converter salvage line incident, should be evaluated to determine their applicability to the dose reconstruction process, because it may not represent expected or bounding .[2]

One key scientific question rests on whether Ms. Berger properly accounted for air samples at Paducah which had up to 90% alpha activity from neptunium-237 (Np-237) compared with a range of only 6.9%-22.8% in Ms. Berger's tables. Berger's work for NIOSH also underestimated plutonium-239 levels by up to 7-fold. In the face of significant uncertainty, NIOSH procedures require bounding estimates of potential uptakes to avoid underestimating radiation dose to workers.

Carol Berger told NIOSH that "the PACE Report[3] was used extensively in the

[2] NIOSH asserts, without citation, that "converter disassembly work [at Paducah] was infrequent, requiring eight shifts over a two month period" and indicates that only a limited number of workers could have been affected by using the wrong data in the NIOSH site profile. This is inaccurate. Paducah's "CIP-CUP" program required nearly constant converter assembly and disassembly between 1958-1962 and 1974-1982. Worker interviews indicate converter disassembly was also done in the cascade buildings, where exposure to fission products was high. High Np-237 exposures were also found during neptunium refining, cylinder heel cleaning and hydrogenation tower cleaning. Far more than a handful will be impacted by the use of the wrong data, particularly those who have multiple non-SEC primary cancers.

[3] NIOSH incorrectly references a "PACE Report" on Paducah as a PACE document. This report was published by

development of the site profile document at issue," however, she managed to overlook extensive references to monthly health physics reports which, NIOSH now concedes, would have resulted in higher internal doses for certain worker classifications than Berger had used in her previous work for Martin Marietta. She "did not recall a specific assessment of activity fractions or other related reference materials cited in the PACE Report," according to NIOSH, yet the validity her Martin Marietta report was pointedly challenged in a chapter of the PACE report entitled "Prior Radiological and Health Assessments at Gaseous Diffusion Plants[4]." Her use of the PACE report appears to be selective.

In cutting and pasting her 1992 work for Martin Marietta into the NIOSH site profile, Berger underestimated alpha activity fractions for Np-237 by at least 4-fold. If she had not done so, she would have contradicted her previous work for Martin Marietta. This is a case where a scientist is allowed, due to loopholes in the conflict interest protections, to review her previous work, and where, it appears, that bias towards her own work caused her to use faulty science. NIOSH asserts that any bias would have been caught by its multiple tiers of review. But it wasn't, and NIOSH's Oversight Team never asks why.

In 2000, Carol Berger's work for Martin Marietta was critiqued in the PACE report, but these critiques were not addressed in the September 2004 NIOSH Site Profile, nor were they spotted in 3 ORAU reviews by Jay Maisler, the "ORAU Team Lead" (who is employed by Carol Berger), Judson Kenoyer, Dick Toohey, or in NIOSH's review by Jim Neton (Science Director).

The NIOSH's Oversight Report glosses over the troubling conclusion that conflicts of interest have tainted the quality of science developed by the NIOSH Office of Compensation and Analysis Support (OCAS). The NIOSH staff had a chance to root out these conflicts, but chose instead to recommend changes that will institutionalize these conflicts. Regrettably, this will taint

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the DOE's Office of Environment, Safety and Health, not by the PACE Union, and is entitled "Exposure Assessment Project at the Paducah Gaseous Diffusion Plant" (December 2000), which was prepared under contract by PACE and University of Utah, Division of Radiobiology. Since this is a DOE document, which was subject to DOE-HQ and DOE-Oak Ridge review, as well as a peer review by John Till and Eula Bingham, prior to DOE-HQ publication in 2000, NIOSH should properly cite it as DOE 2000. It appears that repeated reference to this as a "PACE document" is a not-so-subtle effort to label it as an advocacy document produced by a labor union, instead of a peer-reviewed radiation exposure assessment developed by a university-labor consortium for a government agency whose past performance they were auditing.

[4] *Exposure Assessment Project at the Paducah Gaseous Diffusion Plant* states with regards to Berger's previous study for Martin Marietta: "Some serious concerns were noted regarding the study methodology including: 1) the use of average values of statistically insignificant numbers (including negative bioassay results) to determine doses, 2) the determination of dose assuming an acute intake 1, 5 and 10 years before the bioassay sample does not address the question of what transuranic intakes might have been 16-39 years earlier, 3) the workers selected for inclusion within the fecal bioassay study were hired in the mid-1970s however, the majority of the reactor return work was conducted from 1953 through 1977, 4) dose estimates based on uranium urinalysis results were based on only two years of uranium urine data, and 5) dose estimates based on air sampling were based on air sampling data collected from 1989-1991. It is unlikely that samples collected during this time period represent air concentrations from 1953 to the late 1970s. The items noted above contributed to the limited value of this report for the current exposure assessment."

the scientific credibility of site profiles and SEC determinations.

III. THE ORAU CONFLICT OF INTEREST POLICY, AS RECENTLY AMENDED, IS SEVERELY WEAKENED AND THREATENS NIOSH'S SCIENTIFIC CREDIBILITY:

NIOSH amended its Conflict of Interest Policy governing the conduct of ORAU and its subcontractors on October 11, 2005. Several changes are noteworthy:

New Policy: "Site experts may be employed to advise on site specific issues and incidents as necessary. Site or Subject experts may be employed in the preparation of key Project functions documents such as Site Profiles or draft SEC petition Evaluations."

Old Policy: "No individual will perform, review or approve radiation dose reconstructions, site profiles or determination of whether or not to add a class of employees to the SEC for co-workers, DOE facilities at which they were formerly employed, or for contractors by whom they have been employed. Site experts may be employed to advise on site specific issues and incidents as necessary."

Analysis: Allowing "subject experts" with conflicts of interest to write site profiles or perform SEC evaluations infects the program with avoidable bias. When individuals are employed by NIOSH and have to review their previous work, or that of their co-workers, they are naturally going to be inclined to stick by what they have said previously–right or wrong. It is a human response, and a point of professional pride, not to want to contradict one's previous work. This potential bias has to be fire-walled out, not built in. These modifications effectively paper over the Berger conflict of interest, and legitimize the plethora of conflicts of interest which have infected OCAS. Instead of gutting its COI Policy, NIOSH should have brightened the lines to preclude those with conflicts from writing key documents used to make compensation decisions. NIOSH set up a system where experts with obvious conflicts are holding the pen, and waiting for others to catch their biases in the review process. This is akin to letting influenza infected nurses to work in hospitals with face masks, and waiting for others to find out and remove them.

New Policy: "TBD (Site Profile) team members may contribute to documents prepared for sites where they or their employer may be conflicted, but cannot serve as document owner."

Old Policy: None

Analysis: This loophole facilitates evasion of the conflict of interest policy. Non-conflicted team leaders can be bystanders while conflicted team members "hold the pen" and

carry out the bulk of the writing and research.

New Policy: "Routine access to completed forms ORAUT-FORM-0051 (corporate conflict of interest) will be limited to ORAU employees with an established need to know the information and who have agreed not to disclose the contents of the completed forms to other individuals who do not have a need to know. Upon request, access to the completed forms may also be granted to certain NIOSH personnel, but only to individuals who have also agreed not to disclose the information to other parties not having an established need to know. As described above, information concerning specific sites where it has been established that an ORAU Team subcontractor has an active COI will be provided in the Website Disclosure Statements of the individual employees of that subcontractor. No distinction will be made, however, between sites where the employee has a personal conflict and those where the employee's company is conflicted."

Old Policy: "The critical consideration is not whether COI exists, it does.....Therefore, given these inherent potential conflicts, the contractor selected for the dose reconstruction project, along with NIOSH, must do everything possible to prevent of manage actual and perceived conflicts of interest and disclose all potential conflicts of interest."

"ORAU is committed to full and open disclosure as the best way to prevent conflicts of interest. ORAU agrees completely that 'Sunshine is the best disinfectant'."

"The ORAU Team will disclose, for each company and for each individual involved in dose reconstruction, preparation of site profiles, research supporting determinations of whether or not to add a class of employees to the SEC, or any other work done by primary authors or reviewers for NIOSH on Dose reconstruction or SEC petitions on behalf of the EEOICPA program, information about their past and present work at DOE sites."

"Printouts about the persons (and their companies) performing dose reconstructions, preparation of site profiles, research supporting determination of whether or not to add a class of employees to the SEC or any other work done by primary authors or reviewers for NIOSH on dose reconstructions or SEC petitions on behalf of the EEOICPA program will be available upon request, subject to legal requirements concerning the protection of privacy interests."

Analysis: Without full disclosure (subject to Privacy Act restrictions), there is no way for the public to verify whether a corporate conflict exists or not. Transparency used to be a guiding NIOSH principle, but this principle has been downgraded without clear justification. NIOSH and ORAU will police themselves without any mechanism for accountability.

New Policy: "In addition to specific responsibilities delineated for Team Leaders in ORAUT-PROC-0031, DOE Site Profile Development, Review and Approval Process, also ensure that data and information provided by team members (including persons designated as site or subject experts) is not biased due to previous work at or for the site in question."

Old Policy: None on "bias".

Analysis: To the extent "bias" is identified, it should be rooted out. However, this new policy does nothing to close the loopholes that were opened up in the revised COI policy. The term "bias" is not defined, and it is not clear what type of bias is unacceptable (stated bias vs. unstated bias? Dislikes Special Exposure Cohorts vs favors Special Exposure Cohorts?). There are no metrics prescribed or objective criteria specified to determine bias. NIOSH removed the objective limitations on Subject Experts and Primary Authors which helped control conflicts and replaced it with subjective judgments. Health physics professionals, like most people, resent inferences that because they worked for a DOE contractor, they must have biases towards their previous work at a site or that of their employer. NIOSH provides no practical means for Team Leaders to objectively identify and manage bias. NIOSH has no mechanism to ensure that Team Leaders don't bring a preconceived bias of their own to the task. NIOSH has no mechanism to screen out bias on its own staff of federal employees, nor does it take action when contractor bias is brought to their attention. There are also practical limits: will a team leader (who is a subordinate to the corporate management), preclude their superior from working on a project because the subordinate has deemed him/her as "biased"?

IV. RECOMMENDED ACTIONS FOR THE DIRECTOR OF NIOSH:

1. NIOSH should obtain and release a redlined version of its new COI policy which shows what has been added and removed compared with its previous COI policy. This will more clearly reveal the degree and extent to which NIOSH/ORAU has:

a) watered down the protections against conflict of interest,
b) increased the degree of secrecy governing disclosures about corporate conflict of interest, and
c) expanded the numbers people with of conflicts of interest involved in undertaking radiation dose assessments.

2. NIOSH should prepare an accounting of the number of people by site who are preparing/revising site profiles, or have previously prepared site profiles, and who have COIs but are exempted from COI restrictions under NIOSH's newly articulated policy. NIOSH should identify which individuals on each site profile team who must conform to the COI restrictions.

3. Provide the Advisory Board on Radiation and Worker Health (ABRWH) with a redlined copy of the COI policy. Ask the ABRWH to convene a working group and provide recommendations to your office within 60 days on the modified Conflict of Interest Policy.

4. Ask the ABRWH to review the technical issues raised in the NIOSH Oversight Team Report, and assess whether TRU exposures should be reassessed across the Paducah plant site.[5] The assessment should also address whether site profiles at Oak Ridge K-25 and Portsmouth GDP will have to be changed as a result of changes made at Paducah.

5. Suspend the revised COI policy pending a review by the NIOSH Director to determine its potential impact on scientific credibility and bias.

6. Request NIOSH Program staff respond to GAP's February 20, 2005 memo regarding the Paducah site profile and provide the requested information. NIOSH has not responded to this memo for 9 months.

[5] NIOSH's Oversight Team Report suggests only limited revisions are needed to the Paducah internal dose site profile, and declares that the Berger ratios for most of the Paducah plant are acceptable, based on comparison with the PACE report. This is an incorrect application of the PACE report. The PACE report only looked at TRU exposures for a few areas (presumed to be the highest), and did not perform a plant wide TRU assessment. NIOSH's recommended revisions are too narrow. NIOSH failed to assess the 0.5 micron particle size for Np-237 cited in the 1960 AEC memo by Dunham and Brunner and whether this would lead to greater dose consequences.

ATTACHMENT 2: TECHNICAL DOCUMENTS CONSIDERED DURING THE REVIEW

- *Technical Basis Document for Paducah Gaseous Diffusion Plant – Introduction*, ORAUT-TKBS-0019-1, Rev. 00 PC-1 (Maisler 2006).

- *Technical Basis Document for the Paducah Gaseous Diffusion Plant – Site Description*, ORAUT-TKBS-0019-2, Rev. 01 (Turpin 2006).

- *Technical Basis Document for the Paducah Gaseous Diffusion Plant – Occupational Medical Dose*, ORAUT-TKBS-0019-3, Rev. 00 (Turner 2004).

- *Technical Basis Document for the Paducah Gaseous Diffusion Plant – Occupational Environmental Dose*, ORAUT-TKBS-0019-4, Rev. 00 (East 2004).

- *Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational Internal Dose*, ORAUT-TKBS-0019-5, Rev. 00 (Berger 2004).

- *Technical Basis Document for Paducah Gaseous Diffusion Plant – Occupational External Dose*, ORAUT-TKBS-0019-6, Rev. 01 (Turner 2005).

These documents are supplemented by two technical information bulletins (TIBs), which provide additional guidance to the dose reconstructor:

- Ikenberry, Tracy A., 2005. *Internal Dosimetry Coworker Data for Paducah Gaseous Diffusion Plant*, ORAUT-OTIB-0037, Rev 00, ORAU Team Dose Reconstruction Project for NIOSH. September 20, 2005.

- Merwin, Steven E., 2005. *External Coworker Dosimetry Data for the Paducah Gaseous Diffusion Plant*, ORAUT-OTIB-0031, Rev 00 PC-1, ORAU Team Dose Reconstruction Project for NIOSH. August 15, 2005.

ATTACHMENT 3: SUMMARY OF EXPERT INTERVIEWS

To be provided upon clearance.

ATTACHMENT 4: CONSISTENCY AMONG SITE PROFILES

The default site profile assumptions and methodologies for Paducah Gaseous Diffusion Plant (PDGF) are summarized below and were compared to those of other site profiles reviewed to date or in current review. Site profiles completed to date by the SC&A team include Bethlehem Steel, Mallinckrodt Chemical Works (MCW), Iowa Army Ammunition Plant (IAAP), Hanford, the Savannah River Site (SRS), the Y-12 National Security Complex (Y-12 Plant), Idaho National Engineering and Environmental Laboratory (INEEL), the Nevada Test Site (NTS), and the Rocky Flats Plant (RFP). Additional site profiles in the process of review are Oak Ridge National Laboratory (ORNL), the Mound Plant (Mound), and Fernald (FEMP) and Los Alamos National Laboratory (LANL). ORNL had multiple missions which overlap with a number of other sites in the DOE complex such as weapons research, the heat source program, reactor research, assembly and disassembly operations, and tritium operations to name a few.

To ascertain the differences in assumptions between what assumptions are used for the PGDP site profile versus other site profiles, the assumptions from each PGDP TBD must first be understood. The core assumptions for each TBD have been outlined below.

Dose Reconstruction Assumptions for Occupational Medical Exposure

The Paducah Occupational Medical Dose TBD (Turner 2004) has relatively good information on the site specific data related to x-ray equipment and techniques used at the Paducah Gaseous Diffusion Plant (PGDP) which is summarized in a table and discussed in the TBD.

> *Table 3-2 lists the diagnostic medical X-ray equipment used at PGDP during specified periods. The initial General Electric (GE) machine was used from the opening of the plant in 1952 through February 1975. It was replaced by the Picker unit, which served from March 1975 through December 1995. The present equipment has been in operation since January 1996. Quality assurance (QA) has been verified regularly by the Food and Drug Administration and the State of Kentucky, as well as by inhouse surveys. Interviews with the staff provided much of the information in this TBD. The X-ray facility has been operated by the present technician since November 1974.* (Turner 2004, pg. 5)

Table 3-2 on page 6 provides information on types of filtration, collimators, Bucky grids and film development processors. Table 3-3, also on page 6 of the TBD, provides further information by time periods on the type of x-ray (PA and LAT), the kVp (V) for each type of film, and the current, which in all cases is 300 mA.

> *From March 1975 to the present, the dose with either of the two more recent machines listed in Table 3-2 has been comparable for a given procedure. Therefore, organ dose equivalents are determined for two periods: 1952 to February 1975 and March 1975 to the present.* (Turner 2004, pg. 6)

Table 3-9 on page 12 of the TBD discussed the potential sources of uncertainty in organ dose equivalent assessment, discussed conservative default values (referring to Kathren et al. 2003),

and refers to Table B.3 in Report NCRP 102 (NCRP 1989) for a list of air kerma per mAs at different distances. ICRP 34 (ICRP 1982) is referred to for organ-absorbed doses per unit entrance kerma.

> *In the last row of Table 3-9, the same conversion factors from entrance kerma to organ doses are used for all individuals, a distinction being made between male and female for some organs. In any case, the conversion factors are representative for an exposed individual (for the assumed kerma) to the extent that the anatomical features of the individual match those of the phantom on which the tables are based. The variation introduced in this step is not known. An indication can be gained through comparison with dose conversion factors in ICRP (1982) for the 5-year-old pediatric phantom under the same irradiation conditions.* (Turner 2004, pg. 12)

The Fernald Occupational Medical Dose TBD (Chew 2004) has more specific information for the dose reconsrtructor in Section 3.0 of the TBD.

> *Only exposure to X-rays that were performed as a condition of employment is included; X-rays that resulted from injuries or medical testing are not included. Medical exposures are assigned or assumed based on the actual or calculated frequency of chest X-rays required for employment.*
>
> *There is no difference between the exposures experienced by unmonitored versus monitored workers since dosimeters normally worn by workers were not worn during diagnostic medical X-ray examinations. All exposures are external based on the assumption that internal radioisotopes were not administered for diagnostic purposes (with the exception that some radioisotopes might have been used at specific sites for stress testing as part of working conditions). Specific organ doses to be attributed for posterior-anterior (PA) chest X-rays should be calculated on the basis of the dose conversion factors found in International Commission on Radiological Protection (ICRP) Publication 34 (ICRP 1982).*
>
> *Organ doses from lateral chest radiography should be estimated at 2.5 times greater than those from the corresponding PA doses, based primarily on the greater mAs exposure per radiograph and the somewhat smaller source-to-skin distance (SSD). If technique factors can be identified for Type I equipment, organ doses for Type I equipment may be calculated by multiplying the organ dose estimates for Type II equipment by 2.5. This approach is reasonable when compared to other U.S. Department of Energy (DOE) sites (e.g., Hanford) where more information on X-ray equipment from the early period is available. For organs not listed in ICRP (ICRP 1982) but specified in the Interactive RadioEpidemiological Program (IREP) code, doses should be determined by analogy with anatomical location. Using this logic, IREP code organs in the thoracic cavity that are not mentioned in ICRP (ibid) can be assigned*

the same dose as the lungs; doses to organs in the head and neck can be assigned the same dose as the thyroid. Head and neck organ (i.e., eye and brain) dose estimates should be somewhat greater than doses actually incurred (and therefore are claimant-favorable) because of geometry considerations and, at least in the case of the brain, because of attenuation by the bony cranium. To ensure claimant-favorability in view of the variations in organ dose described in ICRP (ibid, p. 51), apply the dose values for females, which are slightly higher than those for males.

For both PA and lateral views, a standard source-to-image distance (SID) of 72 inches (183 cm) may be assumed unless specifically noted otherwise. If not specified, assume that all X-ray machines were single-phase and that there was no air gap between the patient and the film.

If there is an indication that mobile X-ray units were used, this could result in higher doses. This would imply that this type of unit (which could have been a photofluorographic machine) might have delivered higher exposures. (Chew 2004, pg. 5)

The Fernald Occupational Medical Dose TBD spends a whole page describing the type of x-ray equipment and provides a summary Table 3-3 on page 12 of the types of x-ray equipment used at FEMP. Table 3-4 (Chew 2004, pg. 12) provides detailed information on a 1961 survey of the x-ray units. Later surveys in 1995 and 1997 provided much additional information and assisted in learning how to decrease dose to the workers.

A 1995 survey by the DHHS noted a significant increase in the entrance skin exposure over the 1993 survey (i.e., from 15.3 mR to 28.0 mR). The FEMP response was that since the 1993 inspection (and following a risk/benefit analysis) FEMP had switched (for approximately 1 year) to the Kodak InSight Thoracic Imaging System. This system was designed to improve overall clinical performance related to chest examinations. It enabled FEMP to obtain significantly more usable information during a single session in comparison to the former "conventional" imaging system. The new system required changes in equipment settings (i.e., the technique) to accommodate and allow for the best diagnostic utilization of the technically superior film. The "increased" measured ESE of 28 mR using the new technology was still below the recommended Federal ESE guideline of 30 mR for routine chest radiography.

A 1997 DHHS survey found that the measured ESE was 32.2 mR. The technique used was a manual mode at 110 kVp and 8 mAs. The survey noted that the film optical density (OD) was outside the defined range. DHHS recommended that FEMP contact the service representative for the X-ray system and the Kodak film representative to discuss methods to lower patient ESE, and change exposure techniques from a manual to a phototimed process. FEMP made the recommended changes to improve the OD, and in a return inspection DHHS

measured the ESE to be 19.2 mR. The technique used was 4 mAs at 126 kVp. For the assessment of organ doses during this period, the higher ESE of 32.2mR was used. (Chew 2004, pg. 15)

The Y-12 Occupational Medical Dose TBD (Murray 2006a) points out that the type of x-ray equipment is not known for the period from 1948 to 1968. A description of what is known about the x-ray equipment at Y-12 is listed in Table 3-2 on page 8, as well as a more detailed discussion of the types of x-ray equipment on pages 8 and 9.

According to information provided by the Y-12 Medical X-Ray Department, pre-employment chest X-rays were always taken with a conventional medical diagnostic X-ray machine (Wiley 2002). They found no evidence of PFG chest X-rays in the employee medical X-ray folders and all chest X-rays in these folders were 14"x17" films. However, in reviewing the medical X-ray folders of workers from the 1940s, approximately 1400 4"x10" chest X-ray films were found in the medical X-ray folders of workers who were employed at Y-12 from 1943 to 1947. Originally these X-rays were thought to be copies of conventional X-rays that were taken elsewhere and sent to Y-12 when the person was hired there. In fact, these were PFG chest X-rays.

In February 1945, a General Electric (GE) stereoscopic photoroentgen unit is listed as an equipment item in the Y-12 Medical Department (Wolf 1945). Reexaminations and other chest films were done on conventional 14"x17' films. On October 12, 1945, the Tennessee Eastman Corporation (TEC) Medical Director sent a telegram to the GE X-Ray Corporation, requesting them to set up a Photoroentgen Unit 4x10 (Leggo 1945). In June 1946, TEC placed an order with the Oak Ridge Hospital for 6000 Eastman-Single Coated 4"x10" X-ray films for the period from August 1, 1946 to July 31, 1947 (Graham 1946). Thus, it is clear that pre-employment chest X-rays were taken with a PFG unit from 1943 to 1947, as evidenced by the 4"x10" films found in the medical records and the purchasing records mentioned above. All chest X-rays done since then are conventional 14"x17" X-rays.

No record has been located to determine what type of diagnostic X-ray machine was in use at Y-12 from 1948 until the GE-type machine mentioned above that was used in the early 1960s. This date may not be correct either. In a meeting with the X-ray technologist who provided the information for that report, the technologist said that more recent documentation indicated that the GE-type machine was installed in 1969, not in the early 1960s (Beck 2003). Thus, it is not possible at this time to state with certainty what X-ray machine was used from 1948 to 1968. From 1969 until January 1982, the X-ray machine in use was similar to a GE Model DXD-350. This machine was replaced with a GE Model DSX-650II in February 1982 (Wiley 2002).

A description of the X-ray equipment used at Y-12 is included in Table 3-2. The specific technique factors for these machines are shown in Table 3-3. Since no

> *technique factors were identified by Y-12 for Types I and II equipment, organ doses, based on assumed technique factors, were developed on the basis of X-ray techniques contemporary with the time period (1943–1968), with due consideration given to claimant favorability.* (Murray 2006a, pg. 8)

The Y-12 Occupational Medical Dose TBD (Murray 2006a) devotes a section to providing the information needed by the dose reconstructor, but some of the needed information is not available requiring the development of default values.

> *No actual X-ray output measurements are available. The X-ray technique factors provided may not be reliable, especially for the Type II equipment. Thus, default values for entrance kerma will be used in the calculation of organ dose conversion factors (DCFs) for use in dose reconstruction. Default values have been developed for the three of the most commonly used occupational medical diagnostic X-ray procedures: PA, lateral; and PFG chest films (ORAU 2003).. The default values are considered to be maxima developed from reviews of patient doses reported in the literature, machine characteristics, and knowledge of X-ray procedures used during the time periods indicated. Sufficient conservatism was included in the determination of the default values to ensure with near certainty (>99% confidence) that the actual exposures from the specified procedures would not exceed the default values, thus ensuring claimant favorability.*
>
> *In determining the default factors in Table 3-4, it was assumed that minimum filtration was used, along with low kilovolt peak (kVp) techniques, slow film speeds with standard development procedures, and no additional collimation or use of cones. The default entrance kerma values for the three procedures are given in Table 3-4* (Murray 2006a, pg. 9)
>
> *A source-to-image distance (SID) of 72 inches (in) (183 centimeter [cm]) was standard for the time for the PA chest, and 42 in (106 cm) for the PFG chests. The X-ray machines used at Y-12 were most likely single-phase, and typically no air gap was used between the patient and the film. Before 1982, it is assumed that the X-ray equipment was operated at 80 kVp, had at least 1.5 millimeters (mm) aluminum (Al) total filtration (see Table 3.1 of National Council on Radiation Protection and Measurements [NCRP] [1989]), and that the half value layer (HVL) was approximately 2.5 mm Al equivalent [eq.] (see Table B.2 of NCRP [1989]). These were typical machine parameters for chest X-rays performed in this time period. After 1982, the X-ray equipment was operated at 110 kVp and had at least 2.5 mm Al total filtration. The HVL was approximately 3.5 mm Al eq. After 1982, the machine parameters were the same but the exposures were photo-timed (Wiley 2002). The default values for entrance kerma were also used for the PA chest X-rays after 1982 because the exposure time would not be known for a photo-timed (automatic) exposure.* (Murray 2006a, pp. 9–10)

The default values for Occupational Medical Dose are the same as those used in other site profiles as noted on the following pages of the Occupational Medical Dose TBD (Turner 2004).

- Exam included posterior-anterior (PA) and limited use of lateral (LAT) views (pg. 5).

- Dose conversion factors were taken from International Commission on Radiological Protection (ICRP) Publication 34 (ICRP 1982) in Sections 3.4.3 and 3.4.4 to evaluate the dose equivalent in various organs (pg. 7). The skin is the only organ listed in Table 3-7 that does not involve ICRP (1982) dose conversion factors (pg. 10). For 1952–February 1975, the dose conversion factors of Kathren et al. (2003) for pre-1970 are used (pg. 20).

- The air kerma and resultant organ doses are proportional to the time integrated current (mAs) (pg. 6).

- Default values for skin-entrance kerma have been developed for use in such instances by Kathren et al. (2003) (pg. 8).

- As in (Kathren et al. 2003), a beam quality HVL of 2.5 mm Al was assumed, and substitute projections were used for some organs to approximate the lack of good collimation (pg. 9).

- Total filtration is equivalent to 2.5 mm Al. The air kerma value $K_o = 0.19$ cGy/(100 mAs) for single-phase generators at a source-to-image distance (SID) $r_o = 183$ cm (= 72 in) (pg. 7).

- Air gaps between the patient and the film (not addressed in the TBD).

- During the period from March 1975 to the present, when both PA and LAT views were made, the posterior skin (Table 3-7) received a dose equivalent of 140 mrem (with backscatter) from the PA view plus radiation entailed from the LAT view (without backscatter) (pg. 10).

- In the absence of other information, initial organ dose equivalents HLS to be used for lumbar spine dose reconstruction are estimated to be those from all five views, roughly approximated as 2.5 times the sum of the AP and LAT dose equivalents. Values of HLS are given in Table B-1 (ICRP 1982) (pg. 19).

- "*For estimating dose equivalents with the Interactive RadioEpidemiological Program (IREP) for organs not included in ICRP (1982), these organs are classified in three anatomical regions, as listed in the first column of Table 3-7. In the second column, a single organ from ICRP (1982) is selected from Table 3-6 as representative of the dose to all organs in that region. Column three lists other body organs according to the region in which they are located*" (pp. 9–10).

- Rough estimate of uncertainty for x-ray procedures is 50% (ICRP 1982) (Table 3-9, pg. 12).

If one looks at the consistency between the Paducah, Fernald and Y-12 Occupational Medical Dose TBD the following summary helps to put this into perspective.

At Paducah, the frequency of x-ray examinations is covered in Table 3-1 (Turner 2004, pg. 5). For non-smokers, and smokers up to 40 years of age, it is every 5 years. For non-smokers over 40 years of age, it's every three years. For asbestos workers, it's every two years. At FEMP (Chew 2004, Table 3-2, page 11), it's annually during the time period, 1952 to 1981 and from 1982–2002, for employees over 45 it is annually, but employees under 45 are offered a x-ray every two years. For Y-12 (Murray 2006a, Table 3-1, pg. 7), the frequency changed during four time periods, 1943–1988, 1988–1993, 1993–1998, March 1988 to present. Although, during the early period, 1943–1988 x-rays were taken annually for pre-placement, annual and termination examination. Form 1988 to 1993, chest x-rays changed based on age from annually to every 10 years. Starting in 1993, Y-12 annual chest-rays were only given to asbestos and beryllium workers and annual exams were not taken, in general, unless it was for a preplacement or termination examination.

The PGDP (Turner, pg. 10) and Fernald (Chew 2004, Table 3-6) site profiles refer to ICRP 34 (ICRP 1982) for organ dose conversion factors. The Fernald Occupational Medical TBD provides an additional Table 3.7 (Chew 2004, pg. 10) that provides dose conversion factors that are not provided in ICRP 34. The Y-12 Occupational Medical Dose TBD (Murray 2006a, pg. 19) provide two tables, Table 3C-2 (1943–1947) and Table 3C-3 (1948–1968) that provide dose conversion factors for the various organs.

The Fernald Occupational Medical Dose TBD (Chew 2004, Table 3-1, pg. 8) provides a consolidated summary of the relationship of beam intensity and various technical factors. A simplified table like this does not appear in the Paducah Occupational Medical Dose TBD, although a table of this type is helpful to the dose reconstructor.

Whereas the Y-12 Occupational Medical Dose TBD (Murray 2006a, pg. 17) defines the IREP radiation type as photons, from 30–250 keV, the Paducah and Fernald Occupational Medical Dose TBD do not spell this out clearly.

The Paducah Occupational Medical Dose TBD (Turner 2004, pg. 12) provides Table 3-9 that discussed potential sources of uncertainty in organ dose equivalent assessments that is most helpful to the dose reconstructor. On page 11 (Turner 2004), it mentions that the uncertainty is a positive ± 30%. The Fernald Occupational Internal Dose TBD (Chew 2004, pg. 8) in Table 3-1 lists uncertainty as ± 30%, assuming all errors are positive. The Y-12 Occupational Medical Dose TBD (Murray 2006a, pg. 12) assumes an uncertainty of ±30 % at the 99% confidence level may be assumed. For further conservatism, it may be appropriate to assume that errors are all positive, and only the + 30% should be used. The Y-12 Occupational Medical Dose TBD also has a whole Section 3.5 on page 11 that discusses uncertainty.

Dose Reconstruction Assumptions for Occupational Environmental Exposure

The Paducah Occupational Environmental Dose TBD (East 2004) provides an overview and describes the potential exposures to Paducah Gaseous Diffusion Plant (PGDP) workers from ambient sources while working outside the process buildings. Section 4.2 of the TBD also discusses internal dose from the breathing of airborne concentrations of radionuclides released while on the PGDP site. Section 4.3 describes potential external dose from sources of radiation outside process buildings. Section 4.4 discusses possible sources of uncertainty associated with both of these sources of environmental dose. Some of the major assumptions made in the TBD are included below.

> *Using the maximum airborne radionuclide concentration in a year (either from actual data or an estimated value), annual intakes for the radionuclides of concern were derived by using an assumed annual respiration rate of 2,400 m^3/yr. Most years have only a potential for contributing only about 1 mrem CEDE or less. Intakes from inhalation are potentially significant only for operations before 1963.* (East 2004, pg. 10)
>
> *Unmonitored workers in the early years did not have significant inventories of depleted uranium to contribute to external dose. Later, unmonitored workers would not spend their entire work year at the depleted cylinder storage yards and, therefore, would not reach the maximum dose recorded by fence line monitoring. No other significant sources of external exposure are associated with the PGDP operations. An assumed deep dose equivalent rate of 200 mrem/yr for all years would be reasonable, and deficiencies in earlier measurement techniques thereby become immaterial.* (East 2004, pg. 12)
>
> *All external environmental dose data were adjusted to reflect a 2,000-hr work year. The data were originally reported in site environmental reports as representative of an employee who worked at the site 24 hours a day, 365 days a year. Using an employee permanently on the site, however, is an unrealistic assumption that would clearly overstate onsite environmental exposures.* (East 2004, pg. 15)
>
> *Uncertainty related to internal exposures presents similar concerns. The highest internal exposures occurred during the early years, and lessened over time with increased controls and better equipment. Assumption of the maximum uptake for all years reduces the need to include an uncertainty factor (other than default values) for intakes.* (East 2004, pg. 15)
>
> *In summary, external exposure rose over time as the depleted uranium inventory grew, and internal exposure decreased as releases were reduced, providing offsetting factors to ensure claimant favorability. Therefore, the external dose equivalent that can be applied though the history of PGDP is 200 mrem/yr. Reconstructions should estimate annual intakes using the information in Section 4.2.5.* (East 2004, pg. 15)

This TBD relies on several data sources for estimating ambient airborne radionuclide concentrations and for annual airborne releases by radionuclide. The principal sources were the annual environmental reports for PGDP from 1958 and through 2001. Measurements from the air-monitoring locations have been provided in Tables 4-2 and 4-3 and applied using the method described in the TBD. Attachment 4A provides an expanded version of this data, including additional background monitoring points.

The Fernald Occupational Environmental Dose TBD (Chu 2006) describes the type of environmental exposure at the Feed Materials Production Plant (FEMP) by stating the following:

> *....the primary radiation exposure pathways are inhalation of airborne radionuclides and exposure to direct radiation from emission plumes, radioactive materials in the process plants, and radioactive substances deposited on the ground or surfaces. Dose reconstructors can determine the internal dose from inhalation of radioactive materials from radionuclide air concentrations, and the external dose from exposure to radioactive materials outside the body from historical data of external dosimetry monitoring.* (Chu 2006, pg. 8)

The TBD provides in Table 4-4 a list of annual airborne emissions due to thorium operations at Plants 8 and 9 and the Pilot Plant from 1951 through 1988. The TBD points out that thorium process emissions represent the best approximation possible given the lack of specific production information and assumes that, at best, this estimate is accurate within an order of magnitude. Because of the conservative assumptions used for scrubber and dust collector efficiencies and the intake of material to the collection equipment, the TBD cautions the dose reconstructor that actual thorium emissions should not exceed those listed in Table 4-4.

Some of the assumptions made in the Fernald Occupational Environmental Dose TBD include the following.

> *The average thoron activity concentration in outside air is comparable to the environmental background concentration of radon; therefore, it can be assumed that FEMP thoron environmental concentrations vary with fluctuations in radon concentrations (Tomes 1997)* (Chu 2006, pg. 10)

> *...for 1972 and subsequent years, an activity concentration equal to that of the radon background concentration is assigned to areas close to buildings where thorium was stored.* (Chu 2006, pg. 10)

> *This TBD analysis assumed that the radioactive aerosol concentration varies significantly in various sections of the Plant. The dose received by a worker is highly dependent on the amount of time spent in specific work areas.* (Chu 2006, pg. 10)

> *The intake calculation assumed that an individual would breathe in 2,400 m^3 of air in a year. The concentration and annual intake values for uranium and*

thorium in Table A-1 are in milligrams per cubic meter and milligrams respectively. Table A-4 lists specific activities and isotopic content (or activity fraction of an isotope) for converting milligrams to Becquerel of individual isotopes. Radon (^{222}Rn) and thoron (^{220}Rn) concentration values are in pCi/L. (Chu 2006, pg. 20)

For ^{222}Rn, 1 WL= 100 pCi/L and the derivation of WLM is based on an individual who continuously breathed air at the FEMP for 2,000 hr for the year and an assumed environmental radon daughter product equilibrium factor of 70%. [This ratio for ambient outside air is in accordance with widespread sampling conducted throughout the United States that is referenced in NCRP (1984)]. For ^{222}Rn, 1 WL=7.47 pCi/L and the equilibrium factor have been documented to range between 0.02 and 0.1 at several locations (Tomes 1997). (Chu 2006, pg. 20)

The radionuclide release rates used to estimate EA concentrations were the average of those release rates calculated for AMS 8 and 9, using ground release (χ/Q)s. The source of release was assumed to be at the center of the Production Area (Plant 4). EA concentrations and intakes were calculated using these input data. (Chu 2006, pg. 25)

The Y-12 Occupational Environmental Dose TBD (Murray 2006b) point outs that there are two exposure pathways for environmental dose; (1) inhalation of uranium in ambient air due to operational releases, and (2) direct external radiation from radionuclides in soils and outdoor surfaces, as well as shine from buildings and operational units. The TBD states that there are two potential sources of external exposures received by workers at the Y-12 facility: (1) exposures from the deposition of radionuclides released as a consequence of facility operations, and (2) exposures received from radiation levels emanating from buildings and storage areas.

Due to the complexity of the terrain surrounding Y-12 and the release mechanisms from the production facilities compounded by the limited dispersion distances, traditional dispersion and transport models were considered unsuitable. An empirical approach based on the limited ambient air monitoring at Y-12 was used to estimate uranium air concentrations. This approach used the annual release estimates independently reconstructed by previous studies to generate annual air concentrations for four locations within Y-12. (Murray 2006b, pg. 9)

The ORDR Project estimated releases from 1944 to 1995. Thus, this source cannot be used to estimate air concentrations from 1996 to 2002. The release estimates in Table 4.2.4-1 show a definite downward trend for the years preceding 1995 (Figure 4.3-2). Thus, it is conservative to assume that the air concentrations reported from 1996 to 2002 are equal to the concentrations

> *reported for 1995. Air concentrations and intakes for the four on site locations are presented in Tables D-1 to D-4 in Attachment D. The uranium air concentrations are based on the 50th and 95th percentile value of the lognormally distributed Chi/Q.* (Murray 2006b, pg. 29)
>
> *Estimates of intake assume 2,000 h y-1 exposure. An inhalation rate of 1.2 m^3 h-1 generates an annual intake off 2,400 m^3 y-1.* (Murray 2006b, pg. 30)
>
> *The on site concentrations of ^{234}/^{235}U and ^{238}U were estimated based on an empirical approach. The air concentrations and intakes were estimated using the 50th and 95th percentile values for the empirical dispersion coefficient (Chi/Q) for each station. Air concentrations (becquerel/cubic meter [Bq m-3]) for all years from 1948 to 2002 were estimated based on the quantities of uranium released. Intakes (Bq y-1) were estimated from the air concentrations based on an exposure assumption of 2000 h y-1 and an inhalation rate of 1.2 m^3 h-1.d* (Murray 2006b, pg. 42)

A comparison of the assumptions made with respect to environmental dose from the Paducah Occupational Environmental Dose TBD (East 2004) are compared below with those in the Fernald and Y-12 Occupational Environmental Dose TBDs.

- In the Paducah Occupational Environmental Dose TBD (East 2004) it is pointed out that measurements from the air-monitoring locations listed in Tables 4-2 and 4-3 were collected from documents and applied using a method described in the TBD. Attachment 4A provides an expanded version of this data, including additional background monitoring points. Table 4-4 provides PGDP external gamma exposure rate of mrem/2,000 hr by year and location.

- In the Fernald Occupational Environmental Dose TBD (Chu 2006) Attachment D shows background concentration measurements. The intake calculation assumed that an individual would breathe in 2,400 m^3 of air in a year. The concentration and annual intake values for uranium and thorium in Table A-1 are in milligrams per cubic meter and milligrams respectively.

- The Y-12 Occupational Environmental Dose TBD (Murray 2006b) provides the following information about measurements.

> *No measurements of external exposures were reported for the twelve on site stations at Y-12, which began operations in 1983. With the exception of the limited data from environmental monitoring reports, there are two major characterizations of external exposures that have been performed for Y-12. A series of aerial radiological surveys was performed in 1973–1974, 1980, 1989 and 1992 for the ORR that included Y-12. These surveys consisted of a reservation wide, high level survey and low-level facility specific surveys. The second major characterization was*

performed from 1985 to 1987 and involved an outdoor radiological and chemical scoping survey of the 800+ acres occupied by the Y-12 facility.

External environmental dose equivalent rates outside buildings on Y-12 site from 1948–2002 are provided (Murray 2006b, Table 4.4.4-1, pg. 33).

- The Paducah Occupational Environmental Dose TBD (East 2004) indicated that PGDP had established by 1958 a network of permanent stations on and off the site to collect continuous ambient air samples. Sampling data are available in annual environmental reports published since 1958 for four perimeter locations (inside the fence) and varying numbers of offsite locations. The Paducah TBD assists the dose reconstructor by providing Table 4A-1 (pg. 21) that provides outdoor beta concentrations (Bq/m^3) as 12 monitoring stations for each year from 1952 to 2001. Likewise for outdoor alpha concentration, the TBE provides a Table 4A-2 (pg. 22) that provides outdoor beta concentrations (Bq/m^3) as 12 monitoring stations for each year from 1952 to 1996.

- Similarly in the Fernald TBD (Chu 2006, pg. 9) it is reported that Table 4-1 lists annual airborne uranium emissions due to FEMP operations throughout the operating history (1951 through 1988) from each production plant and processing facility. The Fernald TBD has an Attachment A, which provides an airborne radionuclide concentration and intake summary that lists the radionuclide concentration and intake summary by year for 1951 to 1988 for uranium, thorium, and Rn-222. (Bq conversion factors are provided in Table A-4). The TBD also provides an Attachment C, which lists the FEMP radionuclide concentration and intake results by exposure area and year.

- The Y-12 TBD (Murray 2006b, pp. 44–50) discusses occupational environmental doses in Attachment D. Attachment D provides (in separate sections) calculated $^{234/235}U$ and ^{238}U air concentrations and intakes for Stations 2, 4 8 and 12; site-wide $^{234/235}U$ and ^{238}U air concentrations and intakes based on average air concentrations for Stations 2, 4, 8 and 12; maximum $^{234/235}U$ and ^{238}U air concentrations and intakes; and external dose rates outside buildings on Y-12 site.

Dose from the resuspension of contaminated soil has not been considered in the current version of the Paducah Occupational Environmental Dose TBD (East 2004). It is hoped that NIOSH is considering further investigation into this route of exposure. Soil resuspension dose was considered in relation to environmental exposure at both the Rocky Flats Plant (McDowell-Boyer and Little 2004) and the SRS (Scalsky 2005). It appears that further investigation into environmental source terms is needed. The NIOSH/ORAU team is encouraged to continue the screening of source terms and the inclusion of these results in the TBD.

Dose Reconstruction Assumptions for Occupational Internal Exposure

The Paducah Occupational Internal Dose TBD, ORAUT-TKBS-0019-5 (Berger 2004) describes the default assumptions for occupational internal dose at PGDP. The assumptions were derived from historical records relating to the in vivo, the in vitro, and the air monitoring programs.

Assumptions from the Paducah Occupational Internal Dose TBD (Berger 2004) are noted below; page numbers are from the TBD:

A review of in-house procedures used to assess the concentration of uranium in urine indicates that a variety of quality control steps were an integral part of the process. For example, duplicates were consistently run, and comparison of results to known quantities was a critical step. Therefore, the in vitro results from in-house processing, typically reported in units of micrograms of uranium per liter, can be considered generally reliable. However, interpretation of those results can be difficult, primarily because of uncertainty about enrichment, solubility, and the contribution of environmental uranium, but also because samples were collected at work and during the middle of the workweek, meaning that cross-contamination and the inability to separate soluble from insoluble intake fractions contribute to the uncertainty.

Nonetheless, dose reconstructors can prepare reliable estimates of dose from the dates of employment, the employment locations, and the urine bioassay results. (In vivo results, because they were acquired primarily in response to an incident, are less reliable for assessing routine intakes.) Assumptions such as absorption types and the presence or absence of TRU elements can be derived from the historical records. (Berger 2004, pg. 6)

For dose assessment purposes, a nominal distribution of radionuclides must be assumed because not all analytical methods were capable of detecting many of the radionuclides in the PGDP source term. If only total uranium results are available for a particular measurement result, Table 5-4 provides a default isotopic distribution. (Berger 2004, pg. 7)

As of the date of this report, there is little information to support a reduction in the types or forms of radionuclides listed in this table on an annual basis, and few effective decommissioning efforts that could have resulted in downgrading these radiologically restricted areas. Therefore, it is assumed that all of the radionuclides shown were present in the buildings beginning in 1953. Before 1953, reactor tails were not used as feed to the cascade, and the TRU materials do not apply. (Berger 2004, pg. 12, footnote a, Table 5-5)

Unless site-specific information is available, the particle size is assumed to be 5 μm AMAD, as recommended in ICRP (1994, paragraph 5). (Berger 2004, pg. 12, footnote c, Table 5-5)

In some cases, a detection level for a particular radionuclide or analysis method is not available. In that case, dose reconstructors should use the nominal detection levels in Table 5-8. In addition, if it is not clear from the monitoring records how/where a particular claimant's sample was analyzed, it should be assumed that they were analyzed in-house (i.e., at the PGDP) and the MDC from that measurement type used to assess missed dose. Finally, if a record contains a

> *notation of "less than X micrograms/L" or "< x pCi/sample", that value should reflect the MDC for that sample.* (Berger 2004, pg. 15)
>
> *Urine samples were typically collected in the workplace at PGDP, predominantly on Wednesdays. Therefore, contamination of samples from the worker's hands or clothing cannot be ruled out as a contributor to any given result. If a second analysis was performed and if that result was negative, it is reasonable to assume the first result was a false positive due to sample contamination or laboratory error.* (Berger 2004, pg. 15)

A footnote to the above paragraph provided the following additional information: *"At some point, the sample collection day changed to Monday. However, the date of that procedural change is unclear."*

> *MDA = minimum detectable activity or amount. The MDAs shown for uranium, while given in units of mass, are presumed to have been based on measurement of Th-234 activities along with an assumed isotopic ratio. It is presumed that the results for enriched uranium are based on measurement of the measurement of U-235 activities and an assumption of enrichment. However, these presumptions cannot be confirmed thus cautious use of these MDA values is recommended.* (Berger 2004, pg. 18, footnote b, Table 5-10. In-vivo measurement types and detection levels for various periods.)
>
> *For in vivo measurements, contamination could have occurred as external to the body or, in the case of chest counting, as external to the lung. If a follow-up in vivo count (the same day or within a few days) showed a dramatic decrease in activity or no detectable activity, then external contamination should be assumed.* (Berger 2004, pg. 18)

NIOSH has developed a technical information bulletin to assist in dose reconstruction for unmonitored workers by using coworker data. OTIB-0037, Internal Dosimetry Coworker Data for Paducah Gaseous Diffusion Plant (Ikenberry 2005) includes a number of assumptions on intake modeling as noted below.

> *The IMBA Expert OCAS-Edition computer program requires urine results to be in units of activity per day. The total uranium results are in units of μg/L; therefore the results were multiplied by 1.4 in order to normalize them to the Reference Man excretion rate of 1400 mL per day.* (Ikenberry 2005, pg. 6)
>
> *Bioassay results were converted from mass to activity before fitting assuming 0.0389 Bq/μg, characteristic of low-enrichment (2 percent) uranium. Low-enrichment uranium feed is the default value in ORAU* [Berger 2004] *when the specific location where a claimant worked is not available.* (Ikenberry 2005, pg. 6)

The uncertainty for each result in the intake calculation was assumed to be normally distributed. All results were equally weighted by applying a uniform absolute error of 1, indicating to IMBA that all results are (assumed to be) equally precise. (Ikenberry 2005, pg. 6)

A chronic exposure pattern was assumed for PGDP workers; this pattern also approximates a series of acute intakes with unknown intake dates. Intakes were assumed to be by inhalation using a default breathing rate of 1.2 m3/hr and a 5-µm activity median aerodynamic diameter (AMAD) particle size distribution. (Ikenberry 2005, pg. 6)

The database reported all results as "uranium." The bioassay data were assumed to represent excretion of ^{234}U. (Ikenberry 2005, pg. 6)

All uranium isotopes considered have long half-lives in relation to the assumed intake period so radioactive decay is not a consideration. Also, all uranium isotopes are biokinetically identical so there is no effect on the fitting of the data for intake determination. ^{234}U was the isotope selected because it would result in the highest internal dose; the International Commission on Radiological Protection (ICRP) Publication 68 dose coefficients (also referred to as dose conversion factors) for ^{234}U are 7% to 31% larger than those for ^{235}U, ^{236}U, and ^{238}U (ICRP 1995). (Ikenberry 2005, pg. 6)

Because of the isotopic compositions of the source terms, the assumption of ^{234}U will yield claimant-favorable doses. PGDP received uranium and began enrichment operations during June 1952 and first withdrew enriched uranium during November 1952. The November 1952 period is consistent with uranium urinalysis data; however, the first intake period was conservatively assumed to begin on June 1, 1952. (Ikenberry 2005, pg. 6)

The lognormal distribution is selected in the Interactive RadioEpidemiological Program (IREP), with the calculated dose entered as Parameter 1 and the associated GSD as Parameter 2. The GSD is associated with the intake, so it is applied to all annual doses determined from the intake period. (Ikenberry 2005, pg. 9)

The Fernald Occupational Internal Dose TBD, ORAUT-TKBS-0017-5 (Rich 2004) describes the default assumptions for occupational internal dose at the Fernald Environmental Management Project (FEMP). The assumptions were derived from historical records relating to the in-vivo, the in-vitro, and the air monitoring programs.

Assumptions from the FEMP Occupational Internal Dose TBD (Rich 2004) are noted below; the page numbers are from the TBD:

On February 14, 1966 an accidental release of approximately 1200 kg of uranium occurred during conversion operations that resulted in elevated personnel exposures (Warner 1966). The enrichment is unknown and 2% should be assumed for any claimant identified as a subject of this incident. In addition, more than 70% of the thorium at FEMP was handled and processed from 1964 to 1979 in the Pilot Plant. (Rich 2004, pg. 8)

In the absence of specific enrichment information, and considering the above available data related to processing experience of uranium enrichments at FEMP, the default assumption for time periods after 1964 is 2% enrichment for bioassay data in milligram quantities of uranium. Prior to 1964 natural uranium should be assumed. (pg. 10)

Table 5-3 lists the primary assumptions for FEMP uranium enrichments and the isotopes associated with these enrichments. The mass percentages, relative activities in pCi/µg, and the total pCi/µg values are based upon IMBA NIOSH default values. (pg. 10)

Uranium -1% enriched (EU assumption used in early in vivo calculations) (pg. 10, Table 5-3).

Uranium–2% enriched (Recommended dose evaluation default for this TBD) (pg. 10, Table 5-3).

U-236 is less than 1% activity in DU, Ntl, 1% EU, and 2% EU. (pg. 10, Table 5-3)

In the absence of specific enrichment information, and considering the above available data related to processing experience of uranium enrichments at FEMP, the default assumption for time periods after 1964 is 2% enrichment for bioassay data in milligram quantities of uranium. Prior to 1964 natural uranium should be assumed. (Rich 2004, pg. 10)

Before 1989 no TRU analyses for radiological safety were performed on a routine basis for either airborne or urine activity, and exposure controls remained based on chemical toxicity under the assumption that these controls would be sufficient for all the radiological issues (Bassett et al. 1989). Although the alpha activity from the TRU alpha emitters would have been collected and detected on the air samples, the reported results were all considered to be uranium and compared to the MAC. (Rich 2004, pg. 14)

Before February 1989, no smears or air sampling filters were analyzed specifically for plutonium, neptunium, or thorium isotopes (Basset 1989), although these radionuclides would have been detected by gross alpha counting. In 1989, several sets of air and surface smear samples from Plants 4 and 8 were

analyzed for total uranium, $^{239}Pu/^{240}Pu$, ^{238}Pu, ^{237}Np, ^{228}Th, ^{230}Th, and ^{232}Th. Table 5-8 lists the results in derived air concentrations (DAC) from these analyses. DAC ratios are used not for dose reconstruction purposes, but only to indicate the measured relative levels of TRU contaminants in the plant in the 1989 time period, which in turn lends credence to the default assumptions in Table 5-8 for accounting for the unmeasured TRU in the plants. (Rich 2004, pg. 15)

The chemical forms of the RU contaminants are not known, although it is apparent from the chemical processes to which the materials were subjected during uranium processing, a variety of forms would be expected. Hence the dose reconstructor should assume the most claimant favorable solubility type for the target organ. (Rich 2004, pg. 17)

For the purposes of dose reconstruction, effective equilibrium is a logical, claimant-favorable assumption because thorium was present from the earliest times and was stored at FEMP after the industry need for thorium products was past. However, for purposes of perspective Figure 5-1 presents thorium equilibrium growth curves. (Rich 2004, pg. 19)

The ICRP has assigned oxides and hydroxides of thorium to inhalation type S. All other compounds of thorium are type M. The claimant-favorable assumption would be either type M or S, based on the organ of interest, because all of the compounds in Table 5-14 were handled and could have resulted in intakes. The default isotopes would be ^{228}Th and ^{232}Th in equilibrium, since the degree of equilibrium is impossible to estimate due to the variable times of separation of the isotopes from the feed stock (see Figure 5-1). Table 5-15 lists the estimated thorium emissions from FEMP plant facilities in comparison with uranium emissions. (Rich 2004, pg. 21)

Realizing that the thorium data are not measurements, but are the best values that the TBD technical staff (Dolan 1988) could reconstruct on the basis of available records, recollections of professional engineers, and best estimates on the basis of process knowledge, this information represents the best available. From these data estimates it is clear that thorium represents less than about 5% of the total emissions from the plant processes and that processing occurred during fewer years. In addition, these emissions give some qualitative indication of the estimated availability of the materials to the airborne pathway. Because the contamination controls for thorium processing (ventilation, etc.) were the same or equivalent to those for the uranium processes, certain assumptions in relation to contaminants in the work place apply to both processes. (Rich 2004, pg. 21)

The data from the report (Dolan 1988) indicates that just the Pilot Plant, Plant 8, and Plant 9 processed thorium. A single air sampling data sheet was found that indicated a thorium equipment repair operation in Plant 4 during which there were air activity concentrations above MAC. Therefore, the three plants

mentioned should be considered the primary processing sites, although there is some evidence that a few isolated thorium operations occurred in other locations. Based on evaluation of the available information, dose reconstructors should assume thorium exposure for any employee whose records establish work, and therefore exposure potential, primarily in the Pilot Plant from 1964 to 1979, in Plant 9 in 1954 or 1955, or in Plant 8 in 1966, 1969, 1970, or 1971. (pg. 22)

Some thorium as a contaminant follows the uranium streams through the plant in trace quantities but will constitute <1% of the thorium default assumptions below. In vivo counting was performed on the workers in the more likely exposed groups at least once each year. There is some evidence of urine analyses for thorium in claimant files as early as 1955, but to date no information has been found regarding how to interpret it. Although urinalysis can offer some information regarding thorium intake, it is not the preferred bioassay technique, since the material is is predominantly insoluble. Fecal sampling and in vivo analyses are the preferred default. This is a difficult default to derive with any degree of technical basis. (Rich 2004, pg. 22)

There was primarily gross alpha and some gross beta air monitoring during thorium operations for the purpose of controlling worker exposures to below MAC levels. A few in vitro analyses for thorium were discovered primarily in claimant file records; only a few in vivo analyses were found. The thorium results are questionable because of the lack of information for readily interpreting them (e.g., there is no information regarding the in vitro separation method or counting procedure/equipment, nor is there information regarding the assumptions made to derive the in vivo results). (pg. 22)

3 - Limited operation times and smaller volumes and mass (which also would presuppose a more effective ventilation confinement) reduced the exposure potential, all of which would result in an assumption for limited periods of higher-level contamination.

4 - The MAC of 100 dpm m-3 (4.5 × 10-11 μCi cm-3) is 20 to 100 times larger than the current derived air concentrations for ^{232}Th.

Based upon the above information and assumptions, the recommended claimant-favorable default exposure approach to assign thorium intakes is to assume:

• An intake for an exposure period of 100 hours per year at an assumed exposure of 10 MAC is judged adequate to account for the higher levels of exposure indicated by air sampling, since few samples above 10 MAC were reported and these primarily represented short term maximized sampling (based upon descriptions on the sample sheets). Also typical and more extensive uranium air sample data demonstrate that 10 MAC is a reasonable assumption.

• No respiratory protection factor, although not wearing respirators when air concentrations were above MAC represented procedural violations. This violation was known to have occurred and was not unusual.

• An intake for an exposure period of 500 hours per year at an average air activity of 0.1 MAC during normal operations using these assumptions, the claimant-favorable assumption would be:

500 hr × 0.1 MAC + 100 hr × 10 MAC = 1050 MAC-hr exposure (5-1)
This exposure results in an intake of about 60 nCi per year. Therefore, in the absence of monitoring data a claimant-favorable default intake is 30 nCi/y (82 pCi/d) each of Th-232 and Th-228 (the alpha emitting isotopes detected on the air samples) plus a 60 nCi/y (164 pCi/d) intake of Ra-228. The Ac-228 beta emitter adds about 0.1% to the effective dose and therefore can be ignored in the dose calculations. (Rich 2004, pg. 23)

Assuming that those workers receiving the highest weekly doses were those workers who would also be the most likely to be in the highest particulate air activity, the external dose limits would restrict exposure times to approximately 3 months per year, i.e., (4000 mrem/312 mrem wk-1) (4.3 wks/month)-1 = 3 months. This assumes that doses above 4 rem would not be administratively planned to avoid exceeding the 5 rem/year limit. In any event external dose limits would restrict personnel exposure to the recorded maximum air activity to approximately 3 months per year. (Rich 2004, pg. 27)

From a single radon sample data sheet on which the analyses of two samples were recorded on 10/29/53, the higher of the two samples indicated a result of 230 pCi/L radon gas, which verifies the logical assumption that radon gas was released as the drum lids were removed. In addition to the default particulate intake (determined as previously stated), a conservative/bounding analysis of possible radon plus daughter product exposures can be derived:

• Assume 230 pCi/L (2.3 WL) with 100% daughter product equilibrium for 1304 hrs. (163 day x 8 hr/day)/ 74 months of the dumping operations = 17.6 average hours/month exposure. Then 2.3 WL 17.6/167(the fraction of a full working month) x 12 months = 2.9 WLM exposure per year.

This assumed bounding exposure to radon plus daughter products would be in addition to any assigned exposure that may be derived from the Section 4, Occupational Environmental Dose calculations. (Rich 2004, pg. 27)

It is evident that these estimates are based upon assumptions that are cumulatively conservative, claimant favorable, and establish an upper bound of intake for workers involved in the transfer operation of the 13,000 barrels of the stored MCW raffinates to the K-65 silos. Calculations of internal intakes resulting from exposures to the raffinate dusts generated during dumping

operations should be used only for claimants for whom a work history on this specific project can be established. An examination of external penetrating radiation dose for workers who were known to have worked with and handled these drums of raffinate wastes show significant (several 100 mrem per week) penetrating dose accumulation. Therefore a criteria to determine and/or verify that a worker had indeed been exposed to internal intake from raffinate dusts would be a record of penetrating external dose, i.e., no detectable dose would clearly indicate little direct contact or work with the barrels of waste. (Rich 2004, pg. 27)

For internal dose reconstruction of workers in the Silos 1 & 2 areas in the absence of specific in vitro and/or in vivo data, a maximization approach, based upon air sample results, is proposed:

• assume a worker was assigned to all of the dumping operations during 6 weeks of each of the years in the 6 year period;

• maximum dumping rate, and hence maximum exposure rate, 80 drums/day; and

• 100 MAC (4.5 E-9 uCi/ml alpha analysis) exposure levels with no respiratory protection at 9.6 E+06 cm3 per day breathing rate.

The calculation is: 4.5 E-9 µCi/cc × 9.6E+06 cc/day × 30 days = 1.3 µCi intake per year for the period of July 1953 to September 1958. The contaminants are assumed to be composed of Table 5-16 Isotopes in their indicated activity fractions. For purposes of dose reconstruction, it should be noted that the MAC air concentrations were determined by gross alpha counts, requiring the addition of the appropriate percentage of the beta emitting isotopes. (Rich 2004, pg. 27)

Assume 230 pCi/L (2.3 WL) with 100% daughter product equilibrium for 1304 hrs. (163 day x 8 hr/day)/ 74 months of the dumping operations = 17.6 average hours/month exposure. Then 2.3 WL 17.6/167(the fraction of a full working month) x 12 months = 2.9 WLM exposure per year.

This assumed bounding exposure to radon plus daughter products would be in addition to any assigned exposure that may be derived from the Section 4, Occupational Environmental Dose calculations. (Rich 2004, pg. 27)

According to (Rich 2004, Table 5-19, pg. 30), when doing a fecal analysis using fluorophotometry, if the MDL is unknown, the dose reconstructor should "assume environmental levels of 2 µg per sample.

Industrial Hygiene & Radiation Department Internal Deposition Action Levels procedures from about 1970 indicate actions related to the determination of percent maximum permissible lung burden to either uranium or thorium. Uranium-235 was detected primarily by the emission of its 186 keV photon.

> *Uranium-238 was calculated from measurement of the Th-234 progeny assumed to be in equilibrium with the U-238. Thorium-232 and Th-228 activities were determined based on equilibrium assumptions and detection of their progeny, most likely Ac-228 for Th-232, but Pb-212 may have been used for assessment of both Th isotopes. Thorium-230 is not readily detectable by in vivo measurements. There appeared to be no attempt to detect TRU contaminants with the MIVRML. In fact, the only determination made with the mobile van was a quantification of the uranium lung burden in micrograms of uranium, with the assumption of 1% enrichment and of occasional thorium lung burdens as indicated by some claim records*. (Rich 2004, pg. 35) [This has been used later]
>
> *The results from the MIVRML were calibrated in μCi of ^{235}U and reported in mg of uranium in the lung, which was translated to MPLB based on the assumed enrichment (generally 1%). The percent body burden was then multiplied by 15 rem to obtain the assigned annual dose in rem.* (Rich 2004, pg. 35)
>
> *The recorded MDA reported in Scott et al. (1969, p. 169). Values reported in a claimant record were below 10% MPLB, which implies an MDA of less than 1 nCi total 1% (assumed) U-235.* (Rich 2004, pg. 36, footnote to Table 5-26).

The following general assumptions were noted in the Y-12 Occupational Internal Dose TBD (Rich and Chew 2006). Enrichment assumptions at Y-12 for dose assessment for various analytical techniques were provided in Table 5-1 (Rich and Chew 2006, pg. 13)

> *Based on the changing workplace conditions after partial restart in 1998, the default assumptions were modified to return to chronic exposure but to use type S solubility.* (Rich and Chew 2006, pg. 12)
>
> *Before stand-down, the Y-12 program default modeling assumption was class Q (90% Super-W, 10% Y). During the stand-down, the Y-12 program default assumption of chronic exposure was modified to assume acute exposures occurring at the midpoint of a quarterly sampling frequency.* (Rich and Chew 2006, pg. 11)
>
> *Exposure to type M material from 1948 to July 1998 appears to be the more likely absorption type. After July 1998, exposure to absorption type S material is more likely. However, the absorption type can be based on the monitoring data, claimant-favorable assumptions, or both.* (Rich and Chew 2006, pg. 12)
>
> *For lung counts a combination of the information in the Type Analysis and Material Type (see Section 5.A.1.3 for details) reporting fields can be used to determine whether the count was believed to be due to NU or DU. For records through 1971, a Type Analysis of 1 with a Material Type of 2 or 3 indicates DU, while a Material Type of 7 indicates NU. For records after 1971, a Type Analysis of 4 with a Material Type of 2 or 3 indicates DU, while a Material Type of 7 indicates NU.* (Rich and Chew 2006, pg. 13)

As a rule, routine uranium exposures at Y-12 were considered to be of a chronic nature. For example, in 1958, Patterson stated, "…our interpretation of urinalysis results and our assignment of internal dose assume an exposure under equilibrium conditions of intake and elimination" (Patterson 1958, p. 58). For the stand-down period from September 1994 until August 1998, acute exposures should be assumed to be the more likely mode of exposure. After this time, while complete equilibrium is not expected in modern internal dosimetry models, the presumption of chronic exposure conditions for uranium remains in place as noted in a 2003 report: "The most likely exposure potential for uranium work at the Y-12 Complex is chronic in nature" (BWXT Y-12 2003, p. 107). (Rich and Chew 2006, pg. 10)

Reference to Table 5-8 with relationship to the plutonium contaminant, the recommended isotope Pu-239 should be assumed for Oralloy only, since Oralloy used at Y-12 did not come directly from the primary RU generating sites, i.e., it was enriched at the GDP facilities in which Pu-239 was dominant compared to Pu-238. (Rich and Chew 2006, pg. 19)

It is provisionally assumed that L_D was 46 dpm/d before 1965 and 25 dpm/d after 1965. However, given the limitations of the rate method of estimating daily urine volumes, uncertainty in the excretion volume is expected to add substantially to the uncertainty associated with the detection limit of a single measurement. Because recoveries were based on batch rather than individual measurements, uncertainties in recovery would also add to the uncertainty of the detection limit of a single measurement. (Rich and Chew 2006, pg. 23)

The results in the claim files are assumed to be in units of activity per sample unless otherwise indicated (BWXT Y-12 2003, p. 7). (Rich and Chew 2006, pg. 24)

Fluorometric urinalysis continued to be used for NU and DU until 1989, when alpha spectroscopy began to be used. Fluorometry yielded results in mass units (e.g., μg/L), but results were often converted to disintegrations per minute per day with the assumed specific activity of 1.55 dpm/μg for NU. The fluorometric technique had an industry standard sensitivity of about 5 ppb (5 μg/L) (e.g., UCC 1949, p. 7). (Rich and Chew 2006, pg. 25)

Almost all of the restrictions that did take place would have occurred on the basis of the urinalysis program alone. In those cases in which restriction was based on in vivo analysis alone, lung retention times were often observed to be considerably longer than the assumed 120-d lung half-life on which the urinalysis program was based. In other words, the problem was largely due to the inapplicability of the lung model for some materials and individuals, rather than a failure of the urinalysis program. (Rich and Chew 2006, pg. 26)

Due to uncertainty in the process, claimant-favorable assumptions should be made about solubility and uranium activity ratios. (Rich and Chew 2006, pg. 27)

The beta activity of the tritium is then measured by liquid scintillation counting…. Exposure should be assumed to be due to HTO. (Rich and Chew 2006, pg. 27)

Assume that the ratio of ^{234}U to ^{238}U is 1:1 for dietary uranium. (Rich and Chew 2006, pg. 29)

Due to the uncertainty about actual methods for determining the lung counting detection limits, for dose reconstruction purposes and based on review of the Y-12 data, the ^{235}U and ^{238}U lung count detection thresholds are assumed to be 130 µg and 13.5 mg, respectively, through 1990. Beginning in 1991, the critical level or MDA is supplied with the individual results. (Rich and Chew 2006, pg. 31)

No description of the conversion count rate to activity to mass has been found. The analyte reported was based on the area in which the employee worked. Individuals working in NU or DU areas had results reported as ^{238}U, and workers in enriched areas had results reported as ^{235}U. Claimant-favorable assumptions should be based on conversions of 93% enrichment for ^{235}U and natural isotopic abundances for ^{238}U. (Rich and Chew 2006, pg. 32)

This means that the maximum dose conversion factor per milligram of ^{232}Th would be less than that for ^{232}Th in full equilibrium with its progeny. However, unless specific information is available in the claimants' data, the dose reconstructor must make claimant-favorable assumptions. The thorium sensitivity varied due to the dependence of this technique on the ratios of ^{232}Th and ^{228}Ra to the progeny being measured. (Rich and Chew 2006, pg. 33)

Exposure to type M material from 1948 to June 1998 appears to be the most likely absorption type. After June 1998, exposure to absorption type S material is more likely. However, the absorption type can be based on the monitoring data or claimant-favorable assumptions. (Rich and Chew 2006, pg. 46)

For dose reconstruction purposes and based on review of the Y-12 data, the ^{235}U and ^{238}U lung count detection thresholds are assumed to be 130 µg and 13.5 mg, respectively, through 1990. Beginning in 1991, the critical level or MDA is supplied with the individual results. (Rich and Chew 2006, pg. 47)

The Paducah Occupational Internal Dose TBD (Berger 2004, pg. 6) describes the radionuclides of concern at PGDP in the following statement:

The mission of PGDP was to enrich uranium in the form of UF_6 (for use in domestic and foreign commercial power reactors) from roughly 0.7% ^{235}U (natural enrichment) to 2.5% ^{235}U (DOE 2000, p. 6). In addition, other compounds of uranium were present throughout the plant's history, including

> *UO_2F_2, UF_4, and UO_3. The primary radionuclides of concern for the plant are ^{238}U, ^{235}U, and ^{234}U. The progeny of dosimetric interest for these radionuclides includes ^{230}Th and ^{234}mPa….* (Berger 2004, pg. 6)

At Fernald, the Occupational Internal Dose TBD (Rich 2004) points summarize three main radionuclides of concern.

> *This material remains on the Fernald site in the K-65 Silos and represents a continuing internal exposure potential (for any operations in which direct contact with these residues is required) from the unusually high concentrations of ^{226}Ra and its progeny, ^{210}Pb and ^{210}Po (these three radionuclides comprise approximately 90% of the total activity in the K-65 materials).* (Rich 2004, pg. 7)

The Fernald Occupational Internal Dose TBD goes much farther than the Paducah internal dose TBD in providing a detailed list of radionuclides of concern, (Rich 2004, Table 5.2). The Y-12 Occupational Internal Dose TBD (Rich and Chew 2006), likewise provides a fuller listing of radionuclides of concern.

The Paducah Occupational Internal Dose TBD (Berger 2004) fails to even mention airborne releases.

The Fernald Occupational Internal Dose TBD has numerous places where airborne release potentials are mentioned.

> *Plant 6, the Metals Fabrication Plant, and Plant 9 produced metal parts in rolling mills and machining lathes. Plant 6 began operations in 1953. Uranium metal fires were common, resulting in elevated airborne uranium concentrations*. (Rich 2004, pg. 9)

> *In addition to the routine releases at FEMP, there were frequent "upset" conditions (i.e., spills, effluent filter ruptures, etc.) that produced episodic airborne radioactivity in the work areas and plant effluents, and were of a magnitude that the ventilation systems were unable to contain all of the releases.* (Rich 2004, pg. 11)

> *Before 1989 no TRU analyses for radiological safety were performed on a routine basis for either airborne or urine activity, and exposure controls remained based on chemical toxicity under the assumption that these controls would be sufficient for all the radiological issues (Bassett et al. 1989). Although the alpha activity from the TRU alpha emitters would have been collected and detected on the air samples, the reported results were all considered to be uranium and compared to the MAC.* (Rich 2004, pg. 14)

> *The information on the data sheets indicates that in spite of the fact that the contents of the drums were wet, the operations resulted in significant airborne contamination.* (Rich 2004, pg. 26)

The Y-12 Occupational Internal Dose TBD (Rich and Chew 2006), also mentions airborne releases on three different pages.

> *The 86-inch Cyclotron started operation on November 11, 1950, and operated until the early 1980s. This cyclotron is no longer in operation. Polonium isotopes and alpha airborne activity are the mentioned internal dose concerns. The 86-inch cyclotron was later used to create neutron-deficient radionuclides as a part of the R&D program above.* (Rich and Chew 2006, pg. 10)

> *If a work area has been determined to have predominantly insoluble airborne uranium, it is current practice for workers with a moderate to high exposure potential (CEDE greater than 100 mrem) to submit both urine and fecal samples. The relative elimination by urinary and fecal pathways is used to determine the solubility mixtures for each individual.* (Rich and Chew 2006, pg. 28)

The Paducah Occupational Internal Dose TBD (Berger 2004) highlights the following in regard to monitoring for intakes of uranium.

> *At the PGDP, monitoring for intakes of uranium, whether in vivo or in vitro, often resulted in reports of elemental uranium concentration in urine or the mass of elemental uranium in organs or the whole body. However, internal dose assessment requires the use of isotopic concentrations as input to the assessment process. Therefore, reconstructors should use Table 5-2 to derive the isotopic fractions associated with each microgram of uranium reported in an analytical result*. (Berger 2004, pg. 7)

Other assumptions related to *in vitro* analysis in the Occupational Internal Dose TBD (Berger 2004) include those in vitro measurement frequencies that are noted in Table 5-6 on page 13 and in vitro measurement types and detection levels for various periods noted in Table 5-7 on page 14.

At Paducah, whole-body counting was performed using a mobile counter provided by the Y-12 plant that was sometimes referred to as the Martin Marietta Energy Systems (MMES) Counter. Table 5-10 on page 18 of the Paducah Occupational Internal Dose TBD (Berger 2004) lists general information about the detection capabilities of this counting system for various periods and provides MDAs for various operational periods at PGDP. In the Fernald Occupational Internal Dose TBD (Rich 2004, pg. 34), Table 5-22 provides a similar table of MDA which shows the MDA for non-uranium radionuclides was 0.1 pCi/sample throughout most of the years. Uranium *in vivo* MDAs for the lung at Fernald are provided in Table 5-26 on page 36. Whereas, the MDA at Paducah was 4 mg for U-238 from 1968 to 1980 (Berger 2004, pg. 18, Table 5-10), the MDA for U-238 from 1989 to 2001 at Fernald was 7.4 mg (Rich 2004, pg. 36, Table 5-26). The lung counting detection levels at Y-12 using two 9" × 2.5" NaI detectors varied from 13.5 mg in 1959 down to 4.5 mg when the HP germanium detectors were installed (Rich and Chew 2006, Table 5-12, pg. 32).

In the Paducah Occupational Internal Dose TBD (Berger 2004, pp. 5 and 8), only two references to potential thorium exposure are noted. These involve processing small quantities of transuranic (TRU) elements, primarily neptunium and plutonium present in various workplaces, as well as the thorium and protactinium progeny of uranium and the fission product ^{99}Tc. At Fernald, (Rich 2004, pg. 8) it mentions that more than 70% of the thorium that was handled and processed occurred during the period from 1964 to 1979 in the Pilot Plant. There are numerous discussions of thorium in the Fernald Occupational Internal Dose TBD. In Table 5A-1 on page 36 of the ORNL Occupational Internal Dose TBD (Bollenbacher et al. 2006), data used to determine historical MDAs for radionuclides in urine and feces are listed. Thorium is one of the listed radionuclides. In the LANL internal dose TBD (Argall 2004), there is more discussion regarding the isotopic mixture for thorium based on professional judgment. In the case of a maximizing approach, the LANL TBD assumes Th-230. In the LANL internal dose TBD, it advised the dose reconstructor that the absorption class (M or S) should be selected based on the expected compound or matrix (pg. 33).

The use of IMBA in the Paducah Occupational Internal Dose TBD (Berger 2004, pg. 7) is only mentioned as the source of the isotopic fractions for various enrichment percentages in Table 5-2. In the Fernald Occupational Internal Dose TBD (Rich 2004, pg. 10) it is noted that the mass percentages, relative activities in pCi/µg, and the total pCi/µg values are based upon IMBA NIOSH default values that are provided in Table 5.3 (Rich 2004, pg. 10). Thus, the Fernald Occupational Internal Dose TBD more fully explains the associated activities determined by IMBA for different categories at Fernald. In the LANL internal dose TBD, the isotopic composition for uranium is defaulted to the values in the Integrated Modules for Bioassay Analysis IMBA NIOSH Phase I database USDOE Version 1.0.42, Table 5-11 (depleted uranium), or Table 5-12 (enriched uranium) (Argall 2004, pp. 25–26).

Uranium background determination is not discussed in the Paducah Occupational Internal Dose TBD. The Fernald Occupational Internal Dose TBD (Rich 2004, pg. 28) does mention background regarding minimal occupational internal exposure above normal expected radon/thoron backgrounds from the Silos source and mentions that a default of 5% of 0.4 WL could be added to plant exposures. It is pointed out that this equals an exposure of 0.24 WLM per year. The Oak Ridge National Laboratory Internal Dosimetry Program Technical Basis Document (McLaughlin 2002) documented the ORNL assumptions on the handling of uranium exposure:

> *….A urinary uranium background study was conducted in the mid 1990's using non-occupationally exposed employees to quantify the range of typical background uranium excretion. Based upon the results of that study, a discrimination level (set at the 99th percentile level) of 0.14 dpm/day was established for both U-234 and U-238 to differentiate between environmental and occupational exposure to uranium. A value of 0.25 dpm/day is applied to total uranium results*. (McLaughlin 2002) (Also cited in the internal dose TBD on page 18)

The Paducah Occupational Internal Dose TBD (Berger 2004) is very thorough in providing solubility classifications and assumed particle size (µm AMAD) for a number of radionuclides

by facility or building in Table 5.5 on pages 8–12. A general statement is also made about particle size as well (Berger 2004, pp 7).

> *At the PGDP, and unless site-specific information is available, the particle size is assumed to be 5 µm aerodynamic median activity diameter.* (Berger 2004, pg. 7)

It is stated in the Fernald Occupational Internal Dose TBD (Rich 2004, pg. 19) that "Table 5-14 lists a summary of the thorium plant processes with chemical forms and solubility types." However, a review of Table 5.14 on page 20 only lists production years and chemical form, leaving the dose reconstructor to try to attempt to assign the proper solubility type based on the chemical form provided. At Y-12, the following statement was made in the Y-12 Occupational Internal Dose TBD.

> *Particle sizes tended to be relatively larger (0.86 to 1.6 µm), and the solubility tended to be lower, for materials processed at higher temperatures…. While the exceptional cases with unusually protracted lung clearance are important, it is more important to note that, for the vast majority of individuals, lung clearance took place in approximate accordance with the ICRP Publication 2 (1960) insoluble model, which fits within the current type M framework.* (Rich and Chew 2006, pg. 10)

> *The predominant material encountered after partial restart in August 1998 is uranium oxide, which fecal sampling has shown to be more consistent with 100% type S material. Based on the changing workplace conditions after partial restart in 1998, the default assumptions were modified to return to chronic exposure but to use type S solubility.* (Rich and Chew 2006, pg. 11)

> *For a workplace as varied as Y-12, it is clear that no single solubility or particle size would apply to all workers. Furthermore, accurate assignment of the uranium lung clearance type to a given bioassay result was considered virtually impossible because of uncertainties about chemical form and limitations of the personnel-tracking system (Barber and Forrest 1995, p. 669)*. (Rich and Chew 2006, pg. 12)

In the Paducah Occupational Internal Dose TBD (Berger 2004, pg. 15), Table 5-8 indicates that alpha spectrometry was used as the analytical method for urine bioassay samples, and a detection level of 0.27 pCi/L was given for use with Th-228, Th-232, U-234, U-235, U-238, Pu-238, Pu-239, Pu-240 and Am-241.

In the Fernald Occupational Internal Dose TBD, the following explanation was provided on the spectrometer used at Fernald.

> *The method currently being used at Fernald for urinalysis is inductively coupled plasma mass spectrometry (ICP-MS) which has an a priori MDA of 0.15 µg L-1.* (Rich 2004, pg. 32)

At Fernald, there also is no mention of evaluation dose for strontium exposure but the Occupational Internal Dose TBD did provide the dose reconstructor with the following information on polonium and cesium.

> *The following radionuclides that could interfere with in vivo analysis of uranium and thorium were quantified so that their effects on the spectra could be taken into account:*
> • *^{40}K, ^{137}Cs*
> *Other radionuclides addressed included:*
> • *^{60}Co and $^{95}Zr/^{95}Nb$ for organizations outside Y-12 (Cofield 1961)*
> • *^{210}Po* (Rich and Chew 2006, pg. 13)

In regard to the potential for Cs-137 exposure at PGDP, the Occupational Internal Dose TBD provides the following statement.

> *On occasion, in vivo measurement results included ^{137}Cs. However, those PGDP workers could have had body burdens of ^{13}7Cs from nonoccupational sources (e.g., fallout and consumption of specific foodstuffs). There is no evidence of occupational intakes of ^{137}Cs at the PGDP, thus no dose of record should be associated with these measurement result, if any.* (Berger 2004, pg. 19)

The Fernald Occupational Internal Dose TBD provides no information of possible cesium exposure to FEMP workers.

The Y-12 Occupational Internal Dose TBD does discuss the contribution of dose from Cs-137.

> *The ^{137}Cs contribution to the regions of interest changed over time as fallout levels varied, requiring frequent updates to the control subject spectrum library*. (Rich and Chew 2006, pg. 31)

> *Therefore, it is concluded that the primary internal radiation exposure to Y-12 workers was from uranium. However, the internal dosimetry program has included limited monitoring for ^{137}Cs, ^{99}Tc, thorium, plutonium, ^{228}Ac, and tritium, among other radionuclides. There are difficulties in interpreting these measurement data that cannot be resolved satisfactorily at this time. These issues should not have a large impact on the worker's internal dose because exposure to uranium is the source of the greatest part of the internal dose.* (Rich and Chew 2006, pg. 46)

The Paducah Occupational Internal Dose TBD discusses thorium exposure only in one place.

> *Over the years, workers at PGDP handled mainly UF_6 and slightly oxidized forms of uranium. The facility processed both virgin feed material and recycled or reprocessed reactor fuel to enrichments of up to 5% (by weight) of ^{235}U in the*

final product.1 This processing involved small quantities of transuranic (TRU) elements, primarily neptunium and plutonium, present in various workplaces, as well as the thorium and protactinium progeny of uranium and the fission product ^{99}Tc. (Berger 2004, pg. 5)

The Fernald Occupational Internal Dose TBD provides guidance on assumption for environmental levels of thorium and its contribution to internal dose.

For the purposes of dose reconstruction, effective equilibrium is a logical, claimant-favorable assumption because thorium was present from the earliest times and was stored at FEMP after the industry need for thorium products was past. However, for purposes of perspective Figure 5-1 presents thorium equilibrium growth curves. (Rich 2004, pg. 19).

Realizing that the thorium data are not measurements, but are the best values that the TBD technical staff (Dolan 1988) could reconstruct on the basis of available records, recollections of professional engineers, and best estimates on the basis of process knowledge, this information represents the best available. From these data estimates it is clear that thorium represents less than about 5% of the total emissions from the plant processes and that processing occurred during fewer years. In addition, these emissions give some qualitative indication of the estimated availability of the materials to the airborne pathway. Because the contamination controls for thorium processing (ventilation, etc.) were the same or equivalent to those for the uranium processes, certain assumptions in relation to contaminants in the work place apply to both processes. (Rich 2004, pg. 21)

The data from the report (Dolan 1988) indicates that just the Pilot Plant, Plant 8, and Plant 9 processed thorium. A single air sampling data sheet was found that indicated a thorium equipment repair operation in Plant 4 during which there were air activity concentrations above MAC. Therefore, the three plants mentioned should be considered the primary processing sites, although there is some evidence that a few isolated thorium operations occurred in other locations. Based on evaluation of the available information, dose reconstructors should assume thorium exposure for any employee whose records establish work, and therefore exposure potential, primarily in the Pilot Plant from 1964 to 1979, in Plant 9 in 1954 or 1955, or in Plant 8 in 1966, 1969, 1970, or 1971. (p. 22)

Some thorium as a contaminant follows the uranium streams through the plant in trace quantities but will constitute <1% of the thorium default assumptions below. In vivo counting was performed on the workers in the more likely exposed groups at least once each year. There is some evidence of urine analyses for thorium in claimant files as early as 1955, but to date no information has been found regarding how to interpret it. Although urinalysis can offer some information regarding thorium intake, it is not the preferred bioassay technique, since the material is predominantly insoluble. Fecal sampling and in vivo analyses are the

preferred default. This is a difficult default to derive with any degree of technical basis…(Rich 2004, pg. 22).

According to (Rich 2004, Table 5-19, pg. 30), when doing a fecal analysis using fluorophotometry, if the MDL is unknown, the dose reconstructor should "assume environmental levels of 2 µg per sample.

Industrial Hygiene & Radiation Department Internal Deposition Action Levels procedures from about 1970 indicate actions related to the determination of percent maximum permissible lung burden to either uranium or thorium. Uranium-235 was detected primarily by the emission of its 186 keV photon. Uranium-238 was calculated from measurement of the Th-234 progeny assumed to be in equilibrium with the U-238. Thorium-232 and Th-228 activities were determined based on equilibrium assumptions and detection of their progeny, most likely Ac-228 for Th-232, but Pb-212 may have been used for assessment of both Th isotopes. Thorium-230 is not readily detectable by in vivo measurements. There appeared to be no attempt to detect TRU contaminants with the MIVRML. In fact, the only determination made with the mobile van was a quantification of the uranium lung burden in micrograms of uranium, with the assumption of 1% enrichment and of occasional thorium lung burdens as indicated by some claim records. (Rich 2004, pg. 35)

In regard to the potential dose for the actinides, the Y-12 Occupational Internal Dose TBD, Table 5-9 on page 24 lists the analytical laboratory L_D values used at Y-12 from 1988 for the radionuclides: Am-241, Th-228, Th-232, Np-237.

The in vivo lung count was the only monitoring technique for monitoring thorium exposure in the body during the plant's first decades. Thorium lung activity was inferred from ^{228}Ac and/or ^{212}Pb lung activity. Thorium lung counting was conducted from 1958 to 1984 with routine lung counts, scheduled at approximately six-month intervals, starting in 1961. (Rich and Chew 2006, pg. 32)

Dose Reconstruction Assumptions for Occupational External Dose

The Paducah Occupational External Dose TBD (Turner and Thomas 2006) lists the following assumptions for use by the dose reconstructor.

Summary of reasonable but claimant favorable assumptions:

1. Neutron to photon ratio developed using the average midline and bottom photon dose rates.
2. Neutron to photon ratio developed using the low enriched uranium neutron dose rate
3. Assumption that all recorded and missed photon dose was the result of exposure to enriched UF_4 material
4. Assumption that all of the neutron energies are between 0.1–2 MeV

5. Assumption that neutron dosimeter used a quality factor of 10 to convert rad to rem. (Turner and Thomas 2006, pg. 24)

…this TBD has assumed that the TLD was used in a wrist dosimeter configuration, and that dose calculations involved use of a modifying factor to provide some estimate of actual extremity exposure. The sampling of records reviewed showed that while extremity doses were often near limits, the imposed whole-body restrictions limited worker extremity exposure to less than the extremity limit. It was concluded that the dose of record is the best to use for reconstruction of the extremity dose. (Turner and Thomas 2006, pg. 11)

The use of the mrep unit is somewhat unique to FEMP because it declined in use after the 1950s. There are few references, including the Radiological Health Handbook (PHS 1970), so this TBD assumes that a rep is approximately 93 ergs/g of tissue. Because FEMP used this term interchangeably with rad (100 ergs/g of any receptor), there is a further inherent conservativeness of approximately 7%. (Turner and Thomas 2006, pg. 13)

It is reasonable to assume that the early versions of the FEMP dosimeters reacted similarly to the ORNL dosimeter system, given that FEMP used the ORNL system until changing to a commercial system in 1985. (Turner and Thomas 2006, pg. 13)

Results of tests of FEMP dosimeters used during the 1960s (Heatherton 1960) included the conclusion that the half-value thickness of absorption of UX-2 (234Pa) beta energy was approximately 110 mg/cm2. It was determined that "the combined dose rate from the surface of uranium metal in equilibrium with its two daughters, UX-1 (234Th) and UX-2 (234Pa), is about 240 mrad/hr." It was also determined that approximately 95% of the surface dose rate, or approximately 228 mrad/hr, originated from the UX-2 in the metal. The processing of the metal resulted in separation of uranium daughter products, which produced much higher dose rates in portions of the product, process equipment, and byproducts. The reason for this increase in dose rates is the loss of self-shielding afforded by the mass of the in-process uranium. Further studies involved the absorption afforded by such materials as film wrapping paper, polyethylene, cardboard, and Lucite. These materials were used because they were assumed to be nearly tissue equivalent. (Turner and Thomas 2006, pg. 14)

While it was not explicitly stated in the documentation (Heatherton 1960), it is assumed that the dosimeter in use at FEMP was the ORNL version and that the film used was a DuPont type (Johnson 1963). There is some general data of results between film and TLD dosimeters dated November and December 1982. After review of these data, the conclusion is that the two types of dosimeters did not agree and that the ratio of film to TLD varied with the location of the exposure. In some instances the ratio was greater than 1, and in others it was less than 1. Most often film results provide higher dose estimates than TLD results, which support the conclusion that early film dosimeter results are

claimant-favorable. However, TLD results have been documented to be more accurate than film dosimeters, and provide a more representative measure of the true exposures (this is particularly true when measuring beta and low-energy photons). (Turner and Thomas 2006, pg. 14)

The most claimant favorable neutron energy group (i.e., the neutron energy group with the largest REF) is the 0.1 – 2.0 MeV energy group. Since the neutrons at Fernald are assumed to be generated (born) in this energy group and although some fraction will scatter to a lower (less claimant favorable) energy groups, all of the neutron dose should be assumed to result from the 0.1 – 2 MeV energy group. In the absence of data, this is a reasonable and claimant favorable assumption. The default neutron dose fractions are given in Table 6-11. (Turner and Thomas 2006, pg. 22)

*It can be assumed with some certainty that there have been missed doses in the recorded doses for FEMP workers. This could have resulted when a dosimeter was lost or a worker was not monitored, or a zero was entered because the dosimeter result was less than the MDL. Various methods were used to estimate lost dosimeter results such as using coworker results, products of time and dose rate, or previously recorded results for similar work. All of these required considerable review and examination. Missed dose from MDLs is especially important when there were short exchange periods, generally through the 1950s and 1960s. That period also had higher MDLs. The recommended procedure for missed dose is to assign with a log-normal distribution, with zero as a minimum value, (LOD/2 * the number of zero measurements) as the central tendency and (LOD * the number of zero measurements) as the upper 95% estimate. This procedure is applicable to both Hs(0.07) and Hp(10) since the same dosimeter was used for both until the introduction of a TLD finger dosimeter. The MDL per period and exchange frequencies along with the product (LOD * exchanges) are listed in Table 6-13.* (Turner and Thomas 2006, pg. 23)

The Fernald Occupational External Dose TBD (Faust 2004) listed the following assumptions for use by the dose reconstructors.

General assumptions were listed together in one location in the Fernald Occupational External Dose TBD for use by the dose reconstructor, as was the case with the Paducah Occupational External Dose TBD.

Summary of reasonable but claimant favorable assumptions:

1. Neutron to photon ratio developed using the average midline and bottom photon dose rates.

2. Neutron to photon ratio developed using the low enriched uranium neutron dose rate

3. Assumption that all recorded and missed photon dose was the result of exposure to enriched UF4 material

4. Assumption that all of the neutron energies are between 0.1–2 MeV

5. Assumption that neutron dosimeter used a quality factor of 10 to convert rad to rem. (Faust 2004, pg. 24)

More specific additional Fernald Occupational External Dose TBD assumptions were also provided

TLDs were introduced in or around 1978 or 1979, but only on an experimental basis. An exception was the extremity dosimeter program, which introduced TLDs in 1977; however, no data were found in the literature that described these TLDs other than "they were the Teledyne Teflon impregnated with calcium sulfate type" (Dugan 1981). Therefore, this TBD has assumed that the TLD was used in a wrist dosimeter configuration, and that dose calculations involved use of a modifying factor to provide some estimate of actual extremity exposure. (Faust 2004, pg. 11)

This TBD assumes that when FEMP began the practice of using bags for contamination control, attendant calibration procedures were revised to accommodate the new practice. Small changes in film density can mean large changes in recorded exposure. Therefore, it is desirable to calibrate dosimeters using the same radiological sources and energies to which worker dosimeters will be exposed; this practice was implemented at FEMP. (Faust 2004, pg. 13)

...this TBD assumes that a rep is approximately 93 ergs/g of tissue. Because FEMP used this term interchangeably with rad (100 ergs/g of any receptor), there is a further inherent conservativeness of approximately 7%. (Faust 2004, pg. 13)

It is reasonable to assume that the early versions of the FEMP dosimeters reacted similarly to the ORNL dosimeter system, given that FEMP used the ORNL system until changing to a commercial system in 1985. (Faust 2004, pg. 13)

Results of tests of FEMP dosimeters used during the 1960s (Heatherton 1960) included the conclusion that the half-value thickness of absorption of UX-2 (^{234}Pa) beta energy was approximately 110 mg/cm^2. It was determined that "the combined dose rate from the surface of uranium metal in equilibrium with its two daughters, UX-1 (^{234}Th) and UX-2 (^{234}Pa), is about 240 mrad/hr." It was also determined that approximately 95% of the surface dose rate, or approximately 228 mrad/hr, originated from the UX-2 in the metal. The processing of the metal resulted in separation of uranium daughter products, which produced much higher dose rates in portions of the product, process equipment, and byproducts. The reason for this increase in dose rates is the loss of self-shielding afforded by the mass of the in-process uranium. Further studies involved the absorption afforded by such materials as film wrapping paper, polyethylene, cardboard, and Lucite. These materials were used because they were assumed to be nearly tissue equivalent. It was determined that the half value thickness for tissue was approximately 110 mg/cm^2 and, therefore, the dose to the eyes or

gonads was approximately 15% of the skin dose. It was also determined that coveralls worn by workers (about 30 mg/cm2) reduced uranium beta exposure to the skin by approximately 20%. Figure 6-6, at the end of this TBD, summarizes these data. (Faust 2004, pg. 14)

While it was not explicitly stated in the documentation (Heatherton 1960), it is assumed that the dosimeter in use at FEMP was the ORNL version and that the film used was a DuPont type (Johnson 1963). (Faust 2004, pg. 14)

Since the neutrons at Fernald are assumed to be generated (born) in this energy group and although some fraction will scatter to a lower (less claimant favorable) energy groups, all of the neutron dose should be assumed to result from the 0.1 – 2 MeV energy group. In the absence of data, this is a reasonable and claimant favorable assumption. The default neutron dose fractions are given in Table 6-11. (Faust 2004, pg. 22)

The probability of causation calculations in IREP uses a Radiation Effectiveness Factor (REF) to estimate the probability of causation. The most claimant favorable neutron energy group (i.e., the neutron energy group with the largest REF) is the 0.1 – 2.0 MeV energy group. Since the neutrons at Fernald are assumed to be generated (born) in this energy group and although some fraction will scatter to a lower (less claimant favorable) energy groups, all of the neutron dose should be assumed to result from the 0.1 – 2 MeV energy group. In the absence of data, this is a reasonable and claimant favorable assumption. The default neutron dose fractions are given in Table 6-11. (Faust 2004, pg. 22)

*It can be assumed with some certainty that there have been missed doses in the recorded doses for FEMP workers. This could have resulted when a dosimeter was lost or a worker was not monitored, or a zero was entered because the dosimeter result was less than the MDL. Various methods were used to estimate lost dosimeter results such as using coworker results, products of time and dose rate, or previously recorded results for similar work. All of these required considerable review and examination. Missed dose from MDLs is especially important when there were short exchange periods, generally through the 1950s and 1960s. That period also had higher MDLs. The recommended procedure for missed dose is to assign with a log-normal distribution, with zero as a minimum value, (LOD/2 * the number of zero measurements) as the central tendency and (LOD * the number of zero measurements) as the upper 95% estimate. This procedure is applicable to both Hs(0.07) and Hp(10) since the same dosimeter was used for both until the introduction of a TLD finger dosimeter. The MDL per period and exchange frequencies along with the product (LOD * exchanges) are listed in Table 6-13.* (Faust 2004, pg. 23)

The Y-12 Occupational External Dose TBD (Kerr 2006), which has been recently revised, has assumptions that are based on a different approach from the approach used in the Paducah and Fernald Occupational External Dose TBDs.

Claimant-favorable assumptions should be made using guidance in Table F-2 for beta particles and for photons (x-rays and gamma rays) to assure that dose is not underestimated. The values presented in this table are intended to provide a reasonable estimate of parameters used to calculate the organ dose without significant numerical error for long-term Y-12 workers in the respective facilities. There is no direct evidence to select the specific values shown other than considerations of the radiation sources and usual work tasks. In those cases where there is some doubt in the values, a range of realistic values should be selected for comparison and the most claimant-favorable option selected. (Kerr 2006, pg. 43)

A detailed discussion of the conversion of measured dose to organ dose equivalent is provided in Appendix A of NIOSH (2002). Appendix B of NIOSH (2002) contains the appropriate dose conversion factors (DCFs) for each organ, radiation type, and energy range based on the type of monitoring performed. In some cases, simplifying assumptions are appropriate. (Kerr 2006, pg. 49)

The two general types of neutron dosimeters that have been used at Y-12 differ significantly in their response to neutrons of different energies as illustrated in Figure 6.3.2.2-1 (IAEA 1990). An NTA emulsion was included in the same holder used for the Y-12 beta/gamma dosimeter until 1980. Between 1980 and 1989, there is a serious gap in the neutron dosimetry information for Y-12. It is known that Y-12 had become increasingly dependent over the years on ORNL to process the NTA films because of the small numbers of neutron-exposed workers at Y-12. Thus, the neutron dosimetry at Y-12 is assumed to be the same as that at ORNL from 1980 to 1989. During this period, workers at both ORNL and Y-12 were provided with a two-element TLD dosimeter for beta-particle and photon dosimetry. Those ORNL and Y-12 workers who were exposed to neutrons were provided with a separate neutron dosimeter. This neutron dosimeter contained both an NTA film for measurement of the fast-neutron dose and a TLND for measurement of the neutron dose from lower energy neutrons (Gupton 1978; Berger and Lane 1985). From 1980 to 1985, the neutron doses to Y-12 workers were determined at ORNL using both the NTA and TLND dosimeters as discussed by Gupton (1978). From 1986 to 1989, they were determined at ORNL using only the TLNDs (Berger and Lane 1985). (Kerr 2006, pp. 18–19)

This dosimeter was a modification of the film badge dosimeter previously used at both ORNL and Y-12. During the switch from film to TLD, the film badge dosimeter was modified to hold four TLD chips in a polyethylene mount, 1 mm thick. For workers exposed to neutrons, the modified badge contained a combination of two TLD chips and two TLND chips for low and intermediate energy neutron dosimetry plus an NTA film for fast neutron dosimetry. The MDL of this neutron dosimetry is assumed to about the same as that of the NTA film alone because most of the neutron dose at Y-12 comes from neutrons above the 500-keV threshold of the NTA film. In the mid-1980s, the NTA film was removed

because of poor film quality, its large MDL, and the labor intensive processing requirements (Berger and Lane 1985). (Kerr 2006, pg. 34)

During casting operations, the decay products of ^{238}U float to the top surface of the molten metal and remain as surface residues. These surface residues result in an increased exposure potential because of the high beta and photon energies associated with the ^{234}Pa nuclide. The ^{234}Pa nuclide emits a number of high-energy photons and has a specific activity that is approximately 2 x 1015 times larger than the specific activity of its ^{238}U parent (Henderson 1991). For ^{234}Pa, the percentages of photons with energies of 30 – 250 keV and 250 keV or more are about 7 and 93%, respectively, and for ^{238}U in equilibrium with its short-lived ^{234}Th, ^{234}mPa, and ^{234}Pa, the percentages of photons with energies of 30–250 keV and 250 keV or more are about 82 and 18%, respectively. Thus, an artificially high percentage of photons with energies greater than 250 keV was assumed in Table 6.3.4.1-1 for the normal and depleted uranium process areas. This produces doses that are claimant favorable because of the increased exposure potential to high energy photons from the short-lived ^{234}Pa decay product of ^{238}U. (Kerr 2006, pg. 22)

The neutron dose distributions by energy for the various neutron exposure areas at Y-12 are summarized in Table 6.4.2.2-1. By multiplying the recorded neutron dose by the area-specific correction factors, the neutron dose equivalent is calculated as follows. Consider security personnel who inventory fissile material in the Enriched Uranium Storage Area of Building 9212. Assume that the worker receives a recorded annual neutron dose of 100 mrem. The corrected neutron dose is 151 mrem for neutrons with energies between 0.1-2 MeV, 28 mrem for neutron with energies between 2-14 MeV, and 179 mrem for neutrons of all energies. These corrections should be applied to both measured neutron dose and missed neutron dose. The dose fractions by energy and the associated ICRP 60 (1990) correction factors for various neutron exposure areas at Y-12 are summarized in Table 6.4.2.2-1. (Kerr 2006, pg. 32)

The dose fractions by neutron energy group and the associated ICRP 60 correction factors for the various neutron exposure areas at Y-12 are summarized in Table F-5. As an example, consider security personnel who inventory fissile material in the Enriched Uranium Storage Area of Building 9212 and assume that such a person receives a neutron dose of 100 mrem. The corrected neutron dose is 151 mrem for neutrons with energies between 0.1–2 MeV and 28 mrem for neutrons with energies between 2–14 MeV. Thus, the total corrected neutron dose is a total of 179 mrem. These corrections should be applied to both recorded dose and missed dose. (Kerr 2006, pg. 45)

OTIB-0031 (Merwin 2005) provides some additional insight into general assumptions for external dose reconstruction.

Footnotes in Table 1 provide some of these assumptions.

> *c. The TBD indicates that either monthly, quarterly, or annual exchange frequencies were used during this period, depending on work locations and the potential for exposure. A review of the data indicates that quarterly exchanges were predominant; thus, quarterly exchanges have been assumed here to calculate the maximum annual missed dose.*
> *d. The TBD indicates that either quarterly or annual exchange frequencies were used during this period, depending on the potential for exposure. A review of the data indicates that quarterly exchanges were predominant; thus, quarterly exchanges have been assumed here to calculate the maximum annual missed dose.* (Merwin 2005, pg. 7)

OTIB-0031 (Merwin 2005) also points out the following.

> *As described in ORAUT-OTIB-0020… some cases not having complete monitoring data can be processed based on assumptions and methodologies that do not involve coworker data. For example, many cases falling in the first category above can be processed by assigning ambient external and internal doses based on information in the relevant site Technical Basis Documents (TBDs).* (Merwin 2005, pg. 4)

> *As described in the PGDP External Dosimetry TBD…operations at the site began in 1952, and in 1953 the site began using dosimeter and processing technical support provided by the Oak Ridge National Laboratory (ORNL). Until July 1960, dosimeters were issued to a limited number of individuals, i.e., those with the highest potential for exposure, and the badges were exchanged weekly. After that time, dosimeters were assigned to all workers who entered a controlled area, and the badges were exchanged and processed on a monthly or quarterly schedule. There does not appear to be any significant administrative practice that would jeopardize the integrity of the recorded dose of record.* (Merwin 2005, pg. 4)

Inconsistencies between Site Profiles

There is variation among site profiles related to the standard chest thickness and whether an adjustment factor is applied. The Paducah medical dose TBD (Turner 2004, pg. 7) states that, "For the PA view, an allowance of 5 cm is made for cassette thickness and 26 cm for chest thickness between the source and image. Therefore, the SSD is rPA = 183 – 31 = 152 cm. For the LAT projection with an assumed chest thickness of 34 cm, rLAT = 183 – 39 = 144 cm."

The Fernald medical dose TBD (Chew-2004, pg. 8) states that for units of 25–27 mm, the dose increased by a factor of 1.5 and with units of greater than 27 mm, the dose increased by a factor of two.

The Y-12 TBD (Murray 2006a, pg. 12) states that: "For a given individual, the source-to-skin distance (SSD) will be determined largely by the thickness of the patient and the accuracy in positioning the patient."

The SRS TBD (Scalsky 2005) and INEEL TBD assumes a chest thickness of 26 cm and 34 cm for PA chest x-rays and lateral chest x-rays, respectively. OTIB-0006 Revision 2 and Revision 3 both indicate that the average worker chest size is 22–24 cm (Kathren and Shockley 2005). The OTIB recommends that adjustments be applied for larger individuals with chest thicknesses of 25–27 cm and > 27 cm chest thickness, resulting in an increase in dose by a factor of 1.5 and 2.0, respectively. In the absence of site specific data, consistent default assumptions are warranted.

In the Paducah Environmental Dose TBD (East 2004), there is no mentioned of doing soil sampling in order to better understand soil resuspension and its potential impact on inhalation dose. Doses from the resuspension of contaminated soil have not been given any attention in the Fernald Environmental Dose TBD (Chu 2006), as well. The Y-12 Environmental Dose TBD (Murray 2006b, pg. 9) does acknowledge that one pathway for environmental dose is direct external radiation from radionuclides in soils and outdoor surfaces, as well as shine from buildings and operational units. Surveys were taken at Y-12 as discussed on page 31.

> *The second major characterization was performed from 1985 to 1987 and involved an outdoor radiological and chemical scoping survey of the 800+ acres occupied by the Y-12 facility. This survey included both radiological and chemical assessments and included measurements of both gamma ray exposure rates and the collection of surface soil samples.* (Murray 2006b, pg. 9)

Dose reconstructors are not given and guidance in the Paducah, Fernald, or Y-12 TBDs to quantify the level of exposure associated with contaminated soils.

There has been no consideration of ingestion dose for Paducah, Fernald, or Y-12 Although engineering controls were implemented over time, internal contamination from incidents, reused personal protective equipment, and deposition on food and beverages in areas where eating was allowed may have led to ingestion.

The Paducah Occupational External Dose TBD (Kerr 2006) indicates that it was as late as 1998 before quantitative monitoring of neutron dose was implemented.

> *Quantitative monitoring for neutron dose began at PGDP in 1998. TLNDs were used in conjunction with appropriate work field calibration factors. Before 1998, the beta/photon badge assembly contained a neutron-sensitive element (NTA, Eastman Kodak Type 2 film). This element was processed only when requested. (NTA film had an energy threshold of about 0.5 MeV.) A review of data does not indicate the assignment of neutron dose before 1998.* (Turner and Thomas 2006, pg. 17)

At Fernald, the "limitations of NTA film were well documented including an MDL of approximately 40 mrem for fast neutrons" (Faust 2004, pg. 18) as noted in the Fernald

Occupational External Dose TBD. The Fernald TBD does not mention NTA film in any other place than this one site.

The Y-12 Occupational External Dose TBD (Kerr 2006, pg. 16) indicates that an additional ORNL neutron badge containing NTA film and TLND dosimeters was issued to Y-12 personnel who were exposed to neutrons. It is noted that:

> *An NTA emulsion was included in the same holder used for the Y-12 beta/gamma dosimeter until 1980. Between 1980 and 1989, there is a serious gap in the neutron dosimetry information for Y-12.*

In general, PGDP, FEMP, Y-12, ORNL, Hanford, SRS and other site profiles discount the use of NTA film as an adequate measure of neutron dose. By the late 1960s, ORNL apparently abandoned the idea of assigning a thermal neutron dose on the basis of the results of NTA film dosimeters. It wasn't until around 1980–1985 that ORNL implemented the use of TLDs for personnel neutron monitoring (Burns and Mohrbacher, pp. 21 and 66). Y-12 became increasingly dependent on Y-12, particularly during the years from 1980 to 1989. NTA file was still used for measurement of fast neutron dose even after the neutron TLD was implemented. (Kerr 2006, pg. 18). The SRS Occupational External Dosimetry TBD (Scalsky 2005) distinguishes neutron energies and neutron-to-photon ratios for reactors, fuel fabrication, plutonium production, and radionuclide production and calibration (Scalsky 2005). The INEEL Occupational External Dose TBD (Rohrig 2004) considers the reactors, the processing plant, waste handling operations, calibration sources, and uranium handling. Neutron energy spectra and neutron-to-photon ratios for Pu-238 and Pu-239 operations are segregated at SRS (Scalsky 2005). Further evaluation of neutron-to-photon ratios should include more specific categories including neutron sources (RaBe, Cf, etc.), accelerators, early subcriticality experiments, initiator development, and neutron spectra from alternate fissile materials. NIOSH/ORAU has proposed to include a Pu-238 specific neutron-to-photon ratio in the next revision.

Exposure geometry is not dealt with in the Paducah Occupational External Dose TBD. There is only one mention of exposure geometry.

> *The adequacy of dosimetry methods to measure radiation dose accurately is determined from radiation type, energy, exposure geometry, and other factors described in this section. (*Turner and Thomas 2006, pg. 11)

In the Fernald Occupational External Dose TBD exposure geometry is only mentioned once in Table 6-15 on page 24 indicating that the exposure geometry was AP.

At Y-12, the Y-12 dosimeter system calibrated using A-P laboratory irradiations (Kerr 2006, pg. 23. It was reported that the NTA dosimeter response increases with increasing exposure angle and TLND response decreases with increasing exposure angle (Kerr 2006, pg. 28).

At ORNL, the Occupational External Dose TBD (Burns and Mohrbacher 2004, pg. 23) points out that 100% AP exposure geometry has been assumed. This is the only mention of exposure geometry. The Occupational External Dose TBDs for Y-12 (Murray 2003), SRS (Scalsky 2005),

and Hanford (Scalsky 2003) base their default exposure geometry on the compensability or non-compensability of the claim. The MCW (Westbrook 2005) and RFP (Furman & Lopez 2004) Occupational External Dose TBDs base default exposure geometries on job titles. The INEEL (Rohrig 2004) Occupational External Dose TBDs default to 100% Anterior-Posterior (AP) exposure.