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

February 7, 2008

Mr. David Staudt
Center for Disease Control and Prevention
Acquisition and Assistance Field Branch
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Re: Contract No. 200-2004-03805, Task Order 1: Draft Report SC&A-TR-TASK1-0020, *Review of the NIOSH Site Profile for the Portsmouth Gaseous Diffusion Plant*

Dear Mr. Staudt:

SC&A is pleased to submit its draft report, *Review of the NIOSH Site Profile for the Portsmouth Gaseous Diffusion Plant*, SC&A-TR-TASK1-0020. The main body of this report has been reviewed for classified information and for Privacy Act (PA) information, edited accordingly, and cleared for unrestricted distribution. Attachment 1 of this report, *Summary of Site Expert Interviews*, has not yet been cleared and is not included in this submission. This attachment will be forwarded to you at a later date.

If you have any comments or questions, please contact me at 732-530-0104.

Sincerely,

John Mauro, PhD, CHP
Project Manager

cc: P. Ziemer, PhD, Board Chairperson
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Draft

ADVISORY BOARD ON

RADIATION AND WORKER HEALTH

National Institute for Occupational Safety and Health

Review of the NIOSH Site Profile for the Portsmouth Gaseous Diffusion Plant

Contract No. 200-2004-03805
Task Order No. 1
SCA-TR-TASK1-0020

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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-TASK1-0020
	Effective Date: Draft — February 6, 2008
	Revision No. 0 – (Draft)
REVIEW OF THE NIOSH SITE PROFILE FOR THE PORTSMOUTH GASEOUS DIFFUSION PLANT	Page 2 of 89
Task Manager: _________________________ Date: ___________ U. Hans Behling, PhD, MPH	Supersedes: N/A
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TABLE OF CONTENTS

Acronyms and Abbreviations 5

1.0 Executive Summary 7
- 1.1 Technical Approach and Review Criteria 7
- 1.2 Summary of Findings 8
- 1.3 Report Organization 10

2.0 Scope and Introduction 12
- 2.1 Review Scope 12
- 2.2 Assessment Criteria and Methods 14
  - 2.2.1 Objective 1: Completeness of Data Sources 14
  - 2.2.2 Objective 2: Technical Accuracy 14
  - 2.2.3 Objective 3: Adequacy of Data 14
  - 2.2.4 Objective 4: Consistency among Site Profiles 15
  - 2.2.5 Objective 5: Regulatory Compliance 15

3.0 Relevant Background Information 17
- 3.1 Principal Operations 17
- 3.2 Sources and Quantities of Uranium Hexafluoride Feed Material 17
- 3.3 Recycled Uranium in Feed Materials 18
  - 3.3.1 Other Sources of Uranium Hexafluoride Feed Materials 18
  - 3.3.2 Radionuclides of Concern 19

4.0 Findings Identified in Behalf of the Portsmouth Gaseous Diffusion Plant Site Profile 20
- 4.1 Review of TBD-1 (ORAUT-TKBS-0015-1) Portsmouth Gaseous Diffusion Plant – Introduction 20
- 4.2 Review of TBD-2 (ORAUT-TKBS-0015-2) Portsmouth Gaseous Diffusion Plant – Site Description 20
- 4.3 Review of TBD-3 (ORAUT-TKBS-0015-3) Portsmouth Gaseous Diffusion Plant – Occupational Medical Dose 21
- 4.4 Review of TBD-4 (ORAUT-TKBS-0015-4) Portsmouth Gaseous Diffusion Plant – Occupational Environmental Dose 22
  - 4.4.1 Internal Environmental Exposures 22
  - 4.4.2 External Environmental Dose 26
- 4.5 Review of TBD-5 (ORAUT-TKBS-0015-5), Portsmouth Gaseous Diffusion Plant – Occupational Internal Dose 30
- 4.6 Review of TBD-6 (ORAUT-TKBS-0015-6) Portsmouth Gaseous Diffusion Plant – Occupational External Dose 60

5.0 Worker Interviews, Data Completeness, and Data Integrity 76
- 5.1 Relevant Background Information 76
- 5.2 Site Interviews 77
- 5.3 Document Review 79
- 5.4 The Butler 1996 Report 79
- 5.5 The Cardarelli (1997) Report 81
- 5.6 DOE (2000) Report 81

5.7 Conclusions Pertaining to Data Integrity and the Adequacy of Data from Dose Reconstruction 83

6.0 References 85

Attachment 1: Summary of Site Expert Interviews 89

ACRONYMS AND ABBREVIATIONS

Advisory Board or ABWRH	Advisory Board on Radiation and Worker Health
AEC	U.S. Atomic Energy Commission
AGG	Argon Gamma Graphs
ANTI-C's	Anti-contamination Clothing
BJC	Bechtel Jacob Corporation
Chi/Q	Ratio of average air concentration to release rate at the source
ClF_3	Chlorotrifluoride
CIP	Cascade Improvement Program
CUP	Cascade Upgrade Program
DDE	Deep Dose Equivalent
DOE	Department of Energy
DOELAP	Department of Energy Laboratory Accreditation Program
DOL	Department of Labor
dpm	Disintegrations per minute
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ES&H	Environmental Safety and Health
EWP	Electrical Work Permit
GAT	Goodyear Atomic Corporation
GCEP	Gaseous Centrifuge Enrichment Plant
HASA	High Assay Sampling Area
HEU	Highly Enriched Uranium
HHE	Health Hazard Evaluation
ICN	International Chemical and Nuclear Corporation
K-25	Oak Ridge Gaseous Diffusion Plant
LEU	Low Enriched Uranium
LOD	Limit of Detection
MDC	Minimum Detectable Concentration
MIVRML	Mobile In-Vivo Radiation Monitoring Laboratory
MRD	Minimum Reported Dose
MTU	Metric Ton(ne)s
NIOSH	National Institute for Occupational Safety and Health
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
OCAW	Oil, Chemical, and Atomic Workers Union
OR	DOE Oak Ridge Operations Office
ORAUT	Oak Ridge Associated Universities Team
OW	Open Window

PAL	Plant Allowable Limit
PGDP	Paducah Gaseous Diffusion Plant
POC	Probability of Causation
PORTS	Portsmouth Gaseous Diffusion Plant
PW	Product Withdrawal
RU	Recycled Uranium
SC&A	S. Cohen & Associates
SDE	Shallow Dose Equivalent
SEC	Special Exposure Cohort
SNM	Special Nuclear Material
SPFPA	Security Police and Fire Protection of America
TBD	Technical Basis Document
TBP	Tri-butyl phosphate
TCE	Trichloroethylene
TIB	Technical Information Bulletin
TLD	Thermoluminescent Dosimeter
TRU	Transuranics
UF_4	Uranium Tetrafluoride
UF_6	Uranium Hexafluoride
U_3O_8	Uranium Oxide
UO_2F_2	Uranyl Fluoride
UMH	Uranium Material Handler
USEC	United States Enrichment Corporation
USW	United Steel Workers

1.0 EXECUTIVE SUMMARY

This draft report presents the S. Cohen and Associates (SC&A, Inc.) evaluation of the National Institute for Occupational Safety and Health (NIOSH) Site Profile for the Portsmouth Gaseous Diffusion Plant (PORTS) (ORAUT-TKBS-0015), which was issued as six separate technical basis documents (TBDs) numbered ORAUT-TKBS-0015-1 through ORAUT-TKBS-0015-6. This draft report was prepared at the request of the Advisory Board on Radiation and Worker Health (Advisory Board) and covers all six TBDs identified below.

- ORAUT-TKBS-0015-1, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Introduction*, Vol. 1, Rev. 00 (Portsmouth TBD, 2007a)
- ORAUT-TKBS-0015-2, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Site Description*, Vol. 2, Rev. 00 (Portsmouth TBD, 2006a)
- ORAUT-TKBS-0015-3, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Medical Dose*, Vol. 3, Rev. 00 (Portsmouth TBD, 2006b)
- ORAUT-TKBS-0015-4, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Environmental Dose*, Vol. 4, Rev. 01 (Portsmouth TBD, 2004a)
- ORAUT-TKBS-0015-5, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Internal Dose*, Vol. 5, Rev. 00 (Portsmouth TBD, 2004b)
- ORAUT-TKBS-0015-6, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational External Dosimetry*, Vol. 6, Rev. 00 (Portsmouth TBD, 2006c)

Throughout this report, individual TBDs are referenced simply by number. For example, ORAUT-TKBS-0015-1 will be identified as TBD-1. As part of our evaluation, SC&A also reviewed numerous other documents that were considered relevant, including the following:

- Select documents that were referenced in the PORTS Site Profile
- Documents contained in the NIOSH Site Research Query Database
- ORAUT-OTIB-0036 – *Internal Dosimetry Coworker Data for Portsmouth Gaseous Diffusion Plant*
- ORAUT-OTIB-0040 – *External Dosimetry Coworker Dosimetry Data for the Portsmouth Gaseous Diffusion Plant*

1.1 TECHNICAL APPROACH AND REVIEW CRITERIA

The approach used by SC&A to perform this review follows the procedural protocols described in *Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004). Approved by the Advisory Board on March 18, 2004, SC&A's protocol reflects the following review criteria:

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

Deficiencies pertaining to these review criteria are noted as 15 "findings," because these represent deficiencies that may require correction due to their potential adverse impact(s) on dose reconstruction. The purpose of this review is to provide the Advisory Board with an independent assessment of issues that surround the PORTS Site Profile. Findings identified in our review are expected to provide the Advisory Board with a **preliminary** overview of potential issues that may impact the feasibility of dose assessment.

SC&A's draft report with its preliminary findings will subsequently undergo a multi-step resolution process. Resolution includes a transparent review and discussion of draft findings with members of the Advisory Board's Working Group and select personnel representing NIOSH/Oak Ridge Associated Universities Team (ORAUT). This resolution process is intended to ensure that each finding is evaluated on its technical merit in a fair and impartial manner. A final report will then be issued to the full Advisory Board for review and deliberation.

1.2 SUMMARY OF FINDINGS

As stated above, SC&A identified a total of 15 findings as a result of our review of the PORTS Site Profile. An overview of findings identified in each TBD is presented below.

TBD-1 (Introduction). The Introduction describes the purpose and scope of the PORTS Site Profile. SC&A has no findings regarding information provided in TBD-1.

TBD-2 (Site Description). The Site Description TBD provides critical information regarding the historical and current status of facilities, processes, source terms, etc., at PORTS. SC&A's review identified the following finding:

> Finding 4.2.1. Based on site experts interviewed by SC&A, TBD-2 failed to identify/characterize several buildings/locations at PORTS that had the potential for worker exposures.

TBD-3 (Occupational Medical Dose). The Occupational Medical Dose TBD provides guidance for reconstructing doses from diagnostic x-ray procedures required as a condition of employment. SC&A's evaluation of TBD-3 identified the following findings:

> Finding 4.3.1. At PORTS, available records that identify the use of photofluorography are incomplete and do not define a timeframe for its use. Although the TBD identifies two timeframes to dose reconstructors (1954–1960 versus 1954–1957), the **shorter timeframe** was selected and was based on a single record with a recorded date of October 1957.

SC&A views the justification for the shorter time period (i.e., 1954–1957) as technically unsound, whimsical at best, and claimant unfavorable.

TBD-4 (Occupational Environmental Dose). TBD-4 provides data for the reconstruction of doses to unmonitored workers exposed onsite internally and externally from environmental releases. SC&A's review identified the following three findings with regard to **external** occupational environmental dose:

Finding 4.4-1. Use of the generic ambient environmental dose of 35.9 mrem/y is too restrictive for **non-compensable** claims and claimant unfavorable.

Finding 4.4-2. The default ambient environmental dose of 267 mrem/y to workers exposed at the Cylinder Storage Yards is without technical basis and may be too low.

Finding 4.4-3. Ambient environmental doses are confined to the deep dose that may significantly underestimate the potential shallow dose to the skin.

TBD-5 (Occupational Internal Dose). At PORTS, internal exposure is dominated by uranium that existed over a wide range of enrichment. Other radionuclides of concern included transuranics and contaminants associated with recycled uranium.

SC&A's review of TBD-5 identified the following six findings:

Finding 4.5-1. TBD-5 provides activity values for transuranic (TRU) elements and Tc-99 in reactor tails processed at PORTS. Values cited for Tc-99 were understated by **several orders of magnitude.**

Finding 4.5-2. Inconsistent bioassay protocols were employed that significantly affect the interpretation of urine bioassay data used for dose reconstruction.

Finding 4.5-3. Current guidance for estimating internal exposure to recycled uranium (RU) contaminants is unachievable and/or inappropriate.

Finding 4.5-4. Empirical data suggest that the generic default value of 3.5% enrichment for uranium is inappropriate/claimant unfavorable for large segments of worker groupings.

Finding 4.5-5. TBD-5 contains contradictory/erroneous data and guidance that instructs the use of an incorrect minimum detectable concentration (MDC) value.

Finding 4.5-6. Mobile In Vivo Radiation Monitoring Laboratory (MIVRML) chest counts for the detection of uranium, TRUs, and fission products are subject to significant limitations and uncertainties.

TBD-6 (Occupational External Dose). At PORTS, radiation fields contributing to external radiation included photons, neutrons, and betas. SC&A's review of TBD-6 identified the following four findings:

Finding 4.6-1. The assumed LOD value for shallow dose (as defined by the two-element film dosimeter used between 1954 and 1980) lacks technical support and is not claimant favorable.

Finding 4.6-2. Unmonitored shallow doses derived from coworker data suffer deficiencies that are likely the result of dosimeter design limitations and/or process policies.

Finding 4.6-3. External exposures to localized skin and to extremities were inadequately monitored, and guidance to dose reconstructors is too subjective and arbitrary.

Finding 4.6-4. Before 1992, PORTS failed to monitor workers for neutron exposures. Current guidance to account for unmonitored neutron exposures is incomplete.

1.3 REPORT ORGANIZATION

Based on the issues raised in each of the six TBDs, SC&A prepared a summary of findings, which is provided above in the executive summary. These findings are not meant to be exhaustive, but rather issues of dosimetric significance that SC&A investigated in more detail in order to develop suggestions for improvement of any revisions to the PORTS TBD and for use in dose reconstruction, as appropriate.

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) Relevant Background Information
(4) Findings Identified on Behalf of the PORTS Site Profile
(5) Worker Interviews, Data Completeness, and Data Integrity

Following this Executive Summary, Section 2.0 identifies the review objectives that were used to evaluate the PORTS TBD.

Section 3.0 of this report provides a brief summary of relevant background data contained in the PORTS Site Profile. The site profile specifies relevant background information and methods to be used by NIOSH for the reconstruction of internal and external doses. Included herein are brief summaries of materials and quantities processed, facility descriptions, and radionuclides of concern to dose reconstruction.

As a result of our review of the site profile and other documents, SC&A identified a total of 15 findings, which are cited in Section 4 of this report. In behalf of each finding, a discussion is provided that serves to explain the technical basis for our concern. For some findings, support is also provided by one or more documents, which are enclosed as exhibit(s), or are referenced (see Reference List in Section 6.0).

These exhibits frequently contain empirical data and/or personal observations/opinions expressed by key individuals who were involved in PORTS operations and worker/workplace monitoring. As such, SC&A regards these historical documents as highly relevant, credible, and impartial. For this reason, the reader is encouraged to review the enclosed exhibits and independently determine the degree to which they support each of the corresponding findings. For practical reasons, findings are grouped by category in the following subsections of Section 4.0:

- Subsection 4.1: Discusses TBD-1, *Introduction*, and provides recommendations for improvement of this section; however, no findings were identified.
- Subsection 4.2: Findings associated with TBD-2, *Site Description.*
- Subsection 4.3: Findings associated with the assessment of occupational medical doses (TBD-3).
- Subsection 4.4: Findings associated with the assessment of occupational environmental doses (TBD-4).
- Subsection 4.5: Findings associated with the assessment of occupational internal doses (TBD-5).
- Subsection 4.6: Findings associated with the assessment of occupational external doses (TBD-6).

Section 5.0 discusses worker interviews and presents our opinion(s) regarding data completeness and data integrity.

2.0 SCOPE AND INTRODUCTION

Under 42 U.S.C. 7384[14], Congress initially granted the Portsmouth Gaseous Diffusion Plant (PORTS) Special Exposure Cohort (SEC) status. Members of the cohort include U.S. Department of Energy (DOE) employees, contractors, or subcontractor employees who were potentially exposed to radiation and were employed an aggregate of at least 250 workdays before February 1, 1992. As provided in 42 U.S.C. 73841(9)[A], members of the cohort who incur one (or more) compensable cancers, as specified in Section 3621[17] of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) [42 U.S.C. 73841[17], qualify for compensation without the need for the U.S. Department of Labor (DOL) to evaluate the probability that the cancer was caused by occupational radiation exposures. Excluded from SEC compensation are employees with less than 250 days of aggregate employment and employees with cancer(s) not specified as compensable.

In spite of the PORTS SEC statute, NIOSH has determined that it is feasible to reconstruct doses for PORTS employees with sufficient accuracy. Correspondingly, NIOSH developed a site profile along with two technical information bulletins (TIBs) for PORTS that are to be used for dose reconstruction in behalf of claims not covered by SEC criteria.

2.1 REVIEW SCOPE

Under the EEOICPA and federal regulations defined in Title 42, Part 82, *Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program,* of the *Code of Federal Regulations* (42 CFR Part 82), the Advisory Board on Radiation and Worker Health (Advisory Board) is mandated to conduct an independent review of the methods and procedures used by the National Institute for Occupational Safety and Health (NIOSH) and its contractors for dose reconstruction. As a contractor to the Advisory Board, S. Cohen and Associates (SC&A, Inc.) has been charged under Task 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 technical basis documents (TBDs) related to historical occupational exposures at the Portsmouth Gaseous Diffusion Plant:

- ORAUT-TKBS-0015-1, 2007, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Introduction*, Vol. 1, Rev. 00
- ORAUT-TKBS-0015-2, 2006, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Site Description*, Vol. 2, Rev. 00
- ORAUT-TKBS-0015-3, 2006, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Medical Dose*, Vol. 3, Rev. 00
- ORAUT-TKBS-0015-4, 2004, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Environmental Dose*, Vol. 4, Rev. 01
- ORAUT-TKBS-0015-5, 2004, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational Internal Dose*, Vol. 5, Rev. 00

- ORAUT-TKBS-0015-6, 2006, *Technical Basis Document for the Portsmouth Gaseous Diffusion Plant – Occupational External Dosimetry*, Vol. 6, Rev. 00

These documents are referred to in this review as PORTS TBD Volumes 1 through 6. There were two TIBs specific to PORTS: (1) *Internal Dosimetry Coworker Data for Portsmouth Gaseous Diffusion Plant* (ORAUT-OTIB-0036, Rev. 00) and (2) *External Coworker Dosimetry Data for Portsmouth Gaseous Diffusion Plant* (ORAUT-OTIB-0040, Rev. 00). SC&A also reviewed other pertinent documents, including those cited on the NIOSH Site Research database. SC&A, in support of the Advisory Board, has critically reviewed the PORTS TBDs, as well as supplementary and supporting documents, against the following three evaluation criteria:

- Determine the completeness of the information gathered by NIOSH, 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 guidelines for the use of the data in dose reconstructions

SC&A's review of the six volumes that comprise the TBD, along with its supporting supplemental documentation, focuses on the quality and completeness of the data that characterized the facility and its operations, and the adequacy 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 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 volumes of the PORTS Site Profile are supposed to serve as site-specific guidance documents to be used in support of dose reconstructions. While dose reconstructors use other data, information, and guidance documents in making dose estimates, the purpose of site profiles is to provide dose reconstructors 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 the various types of dose reconstruction estimates that NIOSH performs—minimum for compensation only; maximum, with worst-case assumptions to be used for denial only, and "best-case" or "reasonable" dose estimates to be used for both compensation and denial. The criteria for evaluation include whether the TBDs provide a basis for scientifically supportable and claimant-favorable dose reconstructions that systematically resolve uncertainties in favor of the claimant as required by 42 CFR 82, the regulation governing the dose reconstruction process.

The basic principle of dose reconstruction is to characterize the radiation environments to which workers were exposed, and determine the levels of exposure the workers received in those environments 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.

SC&A's review of the PORTS TBDs was further supplemented by interviews with site personnel in order to gain a better insight into operational practices and the implementation of radiation protection protocols. Attachment 1 provides a transcript of the interviews, in which statements were paraphrased and names of those interviewed have been omitted for privacy reasons. While discussions with site personnel avoided issues that to date may be regarded as classified, the interviews (as summarized in Attachment 1) were submitted to the DOE for review as a precautionary measure.

2.2 ASSESSMENT CRITERIA AND METHODS

Under Task 1, 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 PORTS Site Profile and supporting TIBs. Our review identifies a number of issues, and discusses the degree to which the site profile fulfills the review objectives delineated in SC&A 2004 in behalf of the following objectives.

2.2.1 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.

2.2.2 Objective 2: Technical Accuracy

Objective 2 requires SC&A to perform a critical assessment of the methods used in the site profile to develop technically defensible guidance or instructions, including evaluating field characterization data, source term data, technical reports, standards and guidance documents, and literature related to processes that occurred at PORTS. The goal of this objective is to analyze the data according to sound scientific principles, and then to evaluate this information in the context of dose reconstruction.

2.2.3 Objective 3: Adequacy of Data

Objective 3 requires SC&A to determine whether the data and guidance presented in the site profile are sufficiently detailed and complete to conduct dose reconstruction, and whether a defensible approach has been developed in the absence of data. In addition, this objective requires SC&A to assess the credibility of the data used for dose reconstruction. The adequacy of the data identifies gaps in the facility data that may influence the outcome of the dose reconstruction process. For example, if a site did not monitor all workers exposed to neutrons who should have been monitored, this would be considered a gap and thus an inadequacy in the data. An important consideration in this aspect of our review of the site profile is the scientific validity and claimant favorability of the data, methods, and assumptions employed in the site profile to fill in data gaps.

2.2.4 Objective 4: Consistency among Site Profiles

Objective 4 requires SC&A to identify common elements within site profiles completed or reviewed to date, as appropriate. In order to accomplish this objective, the PORTS TBDs were compared to other TBDs reviewed to date. This assessment was conducted to identify areas of inconsistencies and determine the potential significance of any inconsistencies with regard to the dose reconstruction process.

2.2.5 Objective 5: Regulatory Compliance

Objective 5 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%, assuring compensability to the claimant. In some cases, such partial/incomplete dose reconstructions with a POC greater than 50% may 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 [based on limits of detection (LOD)] 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 defined by federal guidance 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, and that the approach is scientifically supportable.

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, 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 to satisfy this type of a dose reconstruction. 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 a "reasonable" or "best-case" estimate, NIOSH has committed to resolve uncertainties in favor of the claimant. According to 42 CFR 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.]

SC&A's draft report and preliminary findings will subsequently undergo a multi-step resolution process. Resolution includes a transparent review and discussion of draft findings with members of the Advisory Board Working Group, petitioners, claimants, and interested members of the public. This resolution process is intended to ensure that each finding is evaluated on its technical basis in a fair and impartial basis. A final report will then be issued to the full Board for deliberation and a final recommendation.

3.0 RELEVANT BACKGROUND INFORMATION

This section presents summary information that will provide the reader with an overview of key facility processes, production quantities, and radiological source terms that may have contributed to internal and external exposures.

3.1 PRINCIPAL OPERATIONS

The PORTS began operation in 1954. For 47 years (i.e., until 2001), the primary purpose of PORTS was the enrichment of uranium. Enrichment by gaseous diffusion involved uranium hexafluoride (UF_6) gas as feed material, which is passed through a series of semiporous barriers or stages. Each stage allows the lighter U-235 to pass through more easily than the heavier U-238. This results in each successive stage having slightly more U-235 on one side and, therefore, increasing (i.e., enriches) the amount of fissile U-235. The enriched UF_6 gas flows up towards the top of the cascade, while depleted UF_6 gas remains at the bottom portion of the stage.

In total, the PORTS cascade consisted of 4,080 stages that were housed in three separate process buildings. The initial enrichment of UF_6 was performed in Building X-333, followed by intermediate enrichment in Building X-330. Process Building X-326 contained the highest stage enrichment cascade and was able to enrich uranium to more than 97%.

At each stage, variably enriched as well as depleted UF_6 was shipped offsite for further processing and use in military applications, and lower enriched UF_6 was converted to commercial reactor fuel. Depleted UF_6 (or tails) were either re-fed to the cascade or stored onsite.

Because UF_6 is a solid at ambient temperature, it is received and/or shipped as a solid in various cylinder types and sizes, depending on the level of enrichment and associated concerns for nuclear criticality.

3.2 SOURCES AND QUANTITIES OF URANIUM HEXAFLUORIDE FEED MATERIAL

During the 47 years of uranium enrichment at PORTS, about 330,000 Metric Ton(ne)s uranium (MTU) in the form of UF_6 passed through the enrichment cascades. Included were an estimated 121,485 MTU of enriched UF_6 withdrawn from the Paducah Gaseous Diffusion Plant (PGDP) cascades and supplied to PORTS as feed for further enrichment. A second major source of feed material came from the K-25 Plant at Oak Ridge.

Only about 11,890 MTU of UF_6 feed was produced at PORTS between 1958 and 1962 at the Feed Manufacturing Plant, Building X-344. This facility converted UF_4 (green salt provided by Mallinckrodt Chemical Works and National Lead of Ohio) to UF_6 in fluorination towers, where powdered green salt passed through a rising column of fluorine gas.

3.3 RECYCLED URANIUM IN FEED MATERIALS

Among the UF_6 feed materials processed at PORTS were an estimated 1,094.66 MTU of recycled uranium (RU). All but 1.86 MTU of RU-containing feed had been provided by PGDP and the Oak Ridge K-25 Plant between 1955 and 1974. This reactor return feed material consisted of slightly depleted uranium with a U-235 content between 0.64% and 0.68%.

3.3.1 Other Sources of Uranium Hexafluoride Feed Materials

Buildings X-705 and X-705E. Other feed material was produced by the Decontamination, Cleaning, and Recovery (Building X-705) and the Oxide Conversion Facility (Building X-705E). At the uranium recovery facility (Building X-705), uranium-bearing waste streams and uranium scrap were processed to extract uranium in the form of U_3O_8. Between 1958 and 2001, this facility recovered 38.2 MTU of U_3O_8.

The U_3O_8 produced in Building X-705 was converted directly to UF_6 in Building X-705E. This facility operated between 1957 and 1978, and produced a total of 233 MTU of UF_6.

Building X-344 (Feed Manufacturing Plant). Uranium tetrafluoride (UF_4) produced at Mallinckrodt and Fernald was converted in fluorination towers to UF_6. Between 1958 and 1962, this facility produced 11,890 MTU of UF_6 feed material that was processed at PORTS.

Other Support Facilities:

- Building X-342 (Fluorine Generation Facility). This facility produced fluorine gas for the uranium recovery operations in Building X-705E, and housed equipment needed to sample and heat UF_6 to a gaseous state before feeding to the cascades.
- Building X-343 (Fixed Feed Facility). Initially, UF_6 feed and processed material could enter or be withdrawn from the cascades at any location. In later years, feed and withdrawal were reduced to limited points along the cascades. These fixed feed facilities were housed in Building X-343.
- Building X-345 (Special Nuclear Materials Storage). Select areas of Building X-345 contained vaults that were used to store highly enriched uranium (HEU). This facility also contained a **high assay sampling area** (HASA) for assaying the highly enriched UF_6.
- Building X-710 (Analytical and Process and Materials Technology Labs). This facility calibrated radiological instruments, performed industrial radiograph, and evaluated feed and other process materials.
- Building X-744 (Aluminum Smelter and Recovery). Retired process equipment, including cascade compressors with residual contamination, was subject to a smelter process that extracted aluminum for reuse/resale.
- Building X-770 (Test Loop Facility). This facility was involved in the development/ testing of various cascade components (e.g., piping, pumps, sampling lines, etc.).

3.3.2 Radionuclides of Concern

Workers at PORTS were engaged in many process operations and maintenance activities that had the potential for both external and internal exposures to a host of radionuclides, as defined in Table 1. A credible assessment of worker exposures to these radionuclides is hampered as a result of the following:

- Many radionuclides existed in various chemical states that impacted their physical and biological behavior in the environment and in the human body.
- Chemical processes altered the relative abundance of individual radionuclides by selectively concentrating some radionuclides in finished products, while concentrating others in waste streams and tailings.
- Although isotopes of uranium were the dominant concern, feed materials at PORTS contained significant quantities of uranium decay products.
- Among feed materials processed at PORTS was ~1,095 MTU of RU containing transuranics and the fission product Tc-99. An estimated total of 60,000 to 90,000 grams of Tc-99 was fed into the cascade.
- For the first 40 years of PORTS operation, routine in-vitro bioassays (fluorophotometry and gross alpha counting of urine samples) were limited to uranium. Only as recently as 1994, urinalysis assessed the excretion of Np-237, Pu-238/-239/-240, and Am-241. Starting in 1965, the inhalation of uranium compounds was assessed by means of chest counting.

Table 1. List of Radionuclides of Concern

Radionuclides of Concern
U-234
U-235
U-236
U-238
Tc-99
Np-237
Pu-238c
Pu-239c
Pu-240c
Am-241c
Th-228
Th-230
Th-231
Th-232
Th-234d
Pa-234md

4.0 FINDINGS IDENTIFIED IN BEHALF OF THE PORTSMOUTH GASEOUS DIFFUSION PLANT SITE PROFILE

This section identifies findings that resulted from our review of the six TBDs that represent the PORTS Site Profile. Findings are grouped by their corresponding TBD and follow the order in which the TBDs are sequenced in the Site Profile. For some findings, supportive information is provided by one or more exhibits. For ease of tracking, findings and their associated exhibits are numbered in a manner that provides a linkage. For example, all findings associated with internal dose reflect TBD-**5** (of ORAUT-TKBS-0015-5) and are discussed in Section 4.**5** below. The first finding pertaining to internal dose is, therefore, identified as Finding 4.5-1. There are two exhibits that support Finding 4.5-1, which are further labeled as Exhibit 4.5-1A and 4.5-1B.

4.1 REVIEW OF TBD-1 (ORAUT-TKBS-0015-1) PORTSMOUTH GASEOUS DIFFUSION PLANT – INTRODUCTION

The *Introduction* explains the purpose and the scope of the site profile. SC&A was attentive to this section, because it provides a useful overview and explains the role of each TBD in support of the dose reconstruction process. Hence, the introduction helps in framing the scope of the site profile. As will be discussed later in this report, NIOSH may elect to include additional qualifying information in the introduction describing the dose reconstruction issues that are not explicitly addressed in the TBDs that follow. NIOSH may also want to include a roadmap to the dose reconstructor expounding on the entire dose reconstruction process and showing how the TBDs fit into the process.

In spite of its brevity, SC&A identified no findings in behalf of TBD-1.

4.2 REVIEW OF TBD-2 (ORAUT-TKBS-0015-2) PORTSMOUTH GASEOUS DIFFUSION PLANT – SITE DESCRIPTION

The *Site Description* is an important document, because it provides a description of the facilities and processes, as well as historical information that serve as the underpinning for subsequent TBDs. Specifically, this document describes the history and current status of key facilities and processes, and the associated source terms that are relevant to dose reconstruction. SC&A's review of this section focused on whether all the potentially important site activities and processes are described, and whether characterization of source terms is complete and sufficient to support dose reconstruction.

It should be noted that much of the relevant background information contained in Section 3.0 of this report was taken from TBD-2. A positive feature of TBD-2 is a series of tables (i.e., Table 2-1 through Table 2-16) that provide a comprehensive overview of key facilities, material quantities, and radionuclides of concern. However, based on site experts interviewed by SC&A, several facilities/locations at PORTS with the potential for exposures were not included.

Finding 4.2-1: TBD-2 (PORTS Site Description) may not be Complete with regard to Buildings/Locations with the Potential for Worker Exposures

Site experts, with employment as far back as 1954, informed SC&A that the current TBD-2 (ORAUT-TKBS-0015-2) is incomplete with regard to buildings/locations where workers may have received radiation exposures. The following were identified by site experts:

Building	Description	Comments
X-746	Cylinder Storage	Incident occurred in southeast corner
X-760	Test loop from the 710 Laboratory	
X-744H	Storage	Storage of potentially contaminated components
X-744J	Storage	Storage of potentially contaminated components
X-745G	Empty Cylinder Storage Yard	
X-745C	Empty Cylinder Storage Yard	
X-745F	Heel Cylinder Yard	

The extent to which this deficiency may impact dose reconstruction for select workers who may have been assigned to these locations depends on whether these workers were either monitored and/or qualify for inclusion in the coworker models, as defined in ORAUT-OTIB-0036 and ORAUT-OTIB-0040.

4.3 REVIEW OF TBD-3 (ORAUT-TKBS-0015-3) PORTSMOUTH GASEOUS DIFFUSION PLANT – OCCUPATIONAL MEDICAL DOSE

Between 1954 and 1989, annual chest x-rays were mandatory for all employees. Starting in 1990, the frequency of chest x-rays was reduced to every 3, 5, or 10 years, depending on age. Moreover, these diagnostic exams were optional with the exception of asbestos/beryllium workers, for whom an annual chest x-ray remained mandatory. Because records pertaining to x-ray equipment and operating parameters are scarce, generic values cited in ICRP Publication 34 (ICRP 1983) were used to derive organ-specific doses for conventional 14 in × 17 in posterior-anterior and lateral chest x-rays for the years 1961 and beyond. For the earlier years, the absence of data raises the potential that photofluorography may have been used in lieu of conventional chest x-rays. To account for the potential use of photofluorography, higher organ doses were derived that were based on a generic entrance kerma of 3 rads for the years 1954 to 1957.

In the absence of site-specific data, the use of generic/surrogate data defined in the ICRP and/or NCRP publication is not unreasonable for the assignment of organ doses, as given in Tables A-1, A-2, A-3 and A-4 of ORAUT-TKBS-0015-3. However, there exists an inconsistency for the timeframe for the potential use of photofluorography, as noted in Finding 4.3-1 below.

Finding 4.3-1: Two Different Timeframes for Assigning Organ Doses Derived for Photofluorography Exams are Cited in the TBD

Section 3.3.1 of TBD-3 states the following:

> *Even-though* ***no evidence*** *of the use of photofluorography was found at PORTS it is reasonable to presume that at least some of the occupational medical diagnostic chest x-rays with the DOE and its predecessor organizations were accomplished by photofluorography. The use of photofluorography should be assumed to ensure claimant-favorable dose reconstructions from the time-period of* ***1954 through 1960****.* [Emphasis added.]

Section 3.4.1, entitled *Photofluorography (1954–1957)*, identifies a more restrictive time period and cites the following justification:

> *. . . The PFG period of applicability is based upon the discovery of* ***one PFG*** *performed as appeared in one claimant's file in* ***October of 1957****.*

The restrictive time period of 1954 to 1957 for PFG is selected for organ doses defined in Tables A-1 and A-2 of the TBD.

SC&A not only regards the above-cited statements inconsistent, but views the justification of restricting the time period of 1957 whimsical at best and claimant **un**favorable.

4.4 REVIEW OF TBD-4 (ORAUT-TKBS-0015-4) PORTSMOUTH GASEOUS DIFFUSION PLANT – OCCUPATIONAL ENVIRONMENTAL DOSE

The application of data cited in TBD-4 is intended for use in dose reconstruction for unmonitored workers. Derived estimates are defined for internal and external exposures.

PORTS has estimated that approximately 10,545 kg of uranium corresponding to 8 Ci and 27 Ci of Tc-99 have been released to the atmosphere from 1955 to 1993. Nearly half (or about 4,800 kg) of the estimated release of uranium was attributable to one accidental release from a 14-ton cylinder in 1978, and another 3,250 kg of the total uranium were released in the first 8 years of plant operation when the Feed Production Plant was operational. After the shutdown of the Feed Production Plant in 1962, routine environmental releases of uranium dropped significantly (Rumble 1978).

Although Tc-99 was introduced into the cascade system along with feed produced from recycled uranium, it was not until 1975 that its presence was first noted. Most of the estimated 27 Ci of Tc-99 was likely released during maintenance of contaminated cascade equipment (i.e., Cascade Improvement Program/Cascade Upgrade Program or CIP/CUP).

4.4.1 Internal Environmental Exposures

The TBD acknowledges the fact that environmental monitoring only began in 1964, and for most years thereafter, focused on monitoring locations that were either offsite or near the PORTS' perimeter. For these reasons, upper-bound estimates of potential intakes of airborne contaminants were based on a model that employed annual release quantities and empirical yearly Chi/Q values. For years prior to 1964, the maximum of all empirical Chi/Q values was used to estimate airborne levels and intakes based on a yearly inhalation of 2,400 m^3. Table

4.2.5-1 of the TBD provides maximum annual intakes and their uncertainties for uranium isotopes, uranium daughters, and Tc-99. Environmental releases of Np and Pu were considered insignificant and were not included.

As part of SC&A's evaluation, we compared the modeled intake data defined in Table 4.2.5-1 against reported environmental air concentrations for select years. For example, for the years 1964 and 1968, Table 4.2.5-1 of TBD-4 cites total uranium intakes of 17.76 Bq/yr and 8.88 Bq/yr, respectively. Empirical air sampling data for these 2 years are provided in GAT-449 (GAT 1964) and GAT-R-568 (GAT 1969).

Exhibits 4.4-1A and 4.4-1B contain the reported air sampling data for the years 1964 and 1968 at various locations. Thus, air sampling data for the year 1964 as given in Exhibit 4.4-1A shows an average air concentration of 0.01×10^{-11} μCi/cc (or 3.7 Bq/m^3) for a total of 28 sample locations. For the inhalation value of 2,400 m^3/y, this would correspond to 8.8 Bq/y, as cited in Table 4.2.5-1 of TBD-4.

For 1968, Exhibit 4.4-1B shows that the **Phone Building** was the location with the highest airborne alpha activities, yielding a yearly average of 1.334×10^{-3} dpm$_\alpha$/ft^3 (or 0.047 dpm/m^3). For the inhalation volume of 2,400 m^3/y, the inhalation intake at the **Phone Building** is estimated at 113 dpm/y or 1.88 Bq/y.

EXHIBIT 4.4-1A
(Source: GAT 1964)

AIR-BORNE SAMPLE RESULTS

January - June, 1964

Map Sample Location Number	Number of Samples Taken	Alpha Activity, μc/cc* High Concentration	Low Concentration	Average Concentration	Percentage of MPC† for Average
2	6	0.04×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	7.0
3	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
4	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
6	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
9	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
12	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
13	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
15	5	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
16	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
17	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
18	6	0.02×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
19	6	0.02×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
20	6	0.02×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
21	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
22	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
23	6	0.02×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
24	6	0.07×10^{-11}	0.01×10^{-11}	0.02×10^{-11}	11.0
25	6	0.03×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
26	6	0.02×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
27	6	0.01×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
28	6	0.02×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0
Over-all	125	0.07×10^{-11}	0.01×10^{-11}	0.01×10^{-11}	5.0

*Microcuries per cubic centimeter
†Maximum permissible concentration
Limit of sensitivity: 1.0×10^{-14} μc/cc

EXHIBIT 4.4-1B
(Source: GAT 1969)

Table IX

OFFSITE CONTINUOUS AIR SAMPLING (1968)

	Phone Building		West Environmental Station		Warehouse #15		Weather Station		Water Treatment Plant*		Water Treatment Plant	
	Avg. α d/m/ft³	Avg. β-γ d/m/ft³	Avg. α d/m/ft³	Avg. β-γ d/m/ft³	Avg. α d/m/ft³	Avg. β-γ d/m/ft³	Avg. α d/m/ft³	Avg. β-γ d/m/ft³	Avg. α d/m/ft³	Avg. β-γ d/m/ft³	Avg. α d/m/ft³	Avg. β-γ d/m/ft³
	x 10^{-4}	x 10^{-3}	x 10^{-4}	x 10^{-3}	x 10^{-4}	x 10^{-3}	x 10^{-4}	x 10^{-3}	x 10^{-4}	x 10^{-3}	x 10^{-4}	x 10^{-3}
Jan.	0.73	4.03	6.60	3.45	0.97	1.49	0.57	1.73	Out of service		**	2.20
Feb.	14.16	5.38	34.02	3.34	**	3.68	3.58	1.13	Out of service		1.42	5.10
March	4.57	10.48	1.40	5.66	1.83	0.47	3.42	3.43	11.50	15.00	4.55	11.99
April	2.80	16.65	2.13	17.72	1.42	5.73	4.97	8.78	17.50	11.00	4.21	11.83
May	5.46	38.05	13.31	26.20	7.83	40.28	7.48	31.84	20.00	45.70	5.19	19.45
June	1.52	42.78	21.90	25.05	17.50	33.45	2.53	18.63	8.70	26.00	1.38	30.53
July	6.66	45.25	12.93	34.73	5.13	27.03	2.15	17.16	14.09	31.87	23.29	31.49
Aug.	23.38	55.90	15.40	18.18	19.25	38.00	22.90	33.78	11.00	59.00	2.84	42.40
Sept.	40.00	49.95	**	1.92	31.68	21.02	**	14.76	3.90	15.00	**	3.68
Oct.	60.00	2.50	1.12	0.96	**	7.18	62.95	1.55	16.00	3.20	**	1.13
Nov.	**	2.49	2.05	1.17	38.98	1.20	**	0.63	Out of service		2.08	0.22
Dec.	1.28	0.64	3.80	1.52	0.70	1.82	2.15	4.39	0.31	2.02	2.71	1.52

Yearly Avg = 1.334 x 10^{-3} dpm/ft^3

* This ~~is a~~ continuous recording air monitor. ~~The~~ All others are ~~vacuum pumps~~. non-recording samplers.

**Not detectable.

The empirically derived alpha intake of 1.88 Bq/y is nearly 5 times lower than the maximized modeled value of 8.88 Bq/y proposed by NIOSH. Although SC&A recognizes that the empirical value of 1.88 Bq/y represents an **offsite** location, it is nevertheless concluded that the modeled doses defined in Table 4.2.5-1 are likely to represent reasonable, if not maximized, intake values.

SC&A has no findings pertaining to occupational environmental **internal** dose.

4.4.2 External Environmental Dose

Between 1954 and 1981, onsite ambient dose rates were measured periodically with an open-window GM tube. Due to instrument limitations, these measurements are of questionable value for dose reconstruction. Starting in 1981, onsite environmental gamma radiation levels were monitored by thermoluminescent dosimeters (TLDs) at select outdoor locations proximal to Perimeter Road that circumscribes PORTS buildings/facilities. Table 4.3.1-1 of TBD-4 provides yearly, location-specific onsite doses that are based on a 2,000 hr/y exposure. For all years and all locations, the highest annual environmental dose of 35.9 mrem is cited for 1993 at location 3 on the south side of PORTS near the south holding pond. However, in order to simplify matters, Section 4.3.1.1 provides the following guidance to dose reconstructions:

> ***If background is not to be subtracted, the maximum value of 35.9 mrem from Table 4.3.1-1*** *can be used to assign annual ambient environmental dose to workers in areas near Perimeter Road such as the general employee parking lots; the guard gates on the outer perimeter of the security area; the switchyards; warehouses X-744S, T, and U; process buildings, and wastewater facility X-611. (The values in Table 4.3.1-1 for location 874 are much higher, but are specific to the* *UF_6* *cylinder storage yards and not to the rest of the facility.)* ***If background is to be subtracted,*** *the ambient radiation for these areas should be assigned a value of* ***0 mrem*** *because of the argument presented above.* [Emphasis added.]

For location 874, which includes the **Cylinder Storage Yards**, guidance includes the following:

> *Given the above information, an ambient radiation dose of* ***267 mrem*** *(2,000 hour work-year) should be applied to cylinder yard workers as claimant favorable. The portion of this dose attributable to neutrons can be assumed to be* ***178 mrem per year****. This value for the annual neutron dose is obtained from Table 4.3-3 as the maximum annual deep dose equivalent reported in the period 1998 to 2001 for the cylinder storage yards* ***X-745C*** *and* ***X-745E****. This assumes that the neutron dose is the only contributor to the deep dose equivalent. All of these assumptions are claimant favorable.* [Emphasis added.]

For convenience, Table 4.3-3 and Table 4.3.1-1 of TBD-4 are reproduced herein as Exhibits 4.4-2A and 4.4-2B, respectively.

EXHIBIT 4.4-2A

Designation (Other designations used over the years for measurements taken in this vicinity)	Location
X-7725	Waste Storage Facility.
X-326	Process Building.
X-330	Process Building.
X-333	Process Building.
X-344	Containment Building
X-345	Special Nuclear Material (SNM) Storage Building.
X-705	Decontamination Building.
X-720	Maintenance and Stores Building.
X-744G	Bulk Storage Building.
X-745C, X-745E	Depleted Uranium Cylinder Storage Yards.

Site Interior Ambient Radiological Conditions

Mean annual dose by location[(a), (b)] (mrem/year)												
	X-7725		X-326		X-345		X-744G		X-745C		X-745E	
Year	**Avg. deep dose**	**Avg. shallow dose**	**Avg. deep dose**	**Avg. shallow dose**	**Avg. deep dose**	**Avg. shallow dose**	**Avg. deep dose**	**Avg. shallow dose**	**Avg. deep dose**	**Avg. shallow dose**	**Avg. deep dose**	**Avg. shallow dose**
1998	21	36	2	4	19	26	—	—	35	39	20	21
1999	5	10	1	4	1	3	23	19	37	30	56	47
2000	14	—	0	—	0	—	285	—	122	—	178	—
2001	23	—	0	—	0	—	1,056	—	142	—	175	—

a. 2,000-hour work-year, prorated from an 8,736-hour year.
b. Includes beta, gamma, and neutron. Neutrons included in deep dose. — No data.

EXHIBIT 4.4-2B

Mean annual dose by location[(a), (b)] (mrem/year)

Year	874	PP518	29 (10, PP862)	24 (PP906)	12 (PP933)	PP1406	3 (39, X-230-J2, A35)	A40 (35, (X-104)	A36 (X-611)	A39 (PP722)
1954	23.8	23.8	23.8	23.8	23.8	23.8	23.8	23.8	23.8	23.8
1972	—	—	23.6	23.8	22.4	—	24.0	—	—	—
1973	—	—	27.8	26.4	25.6	—	27.2	—	—	—
1974	—	—	28.2	27.0	26.0	—	26.8	—	—	—
1975	—	—	19.2	18.8	18.4	—	18.8	—	—	—
1976	—	—	20.6	20.4	20.0	—	20.0	—	—	—
1977	—	—	23.8	24.0	24.2	—	24.4	—	—	—
1978	—	—	21.8	21.4	20.6	—	21.2	—	—	—
1979	—	—	21.8	21.2	22.0	—	20.2	—	—	—
1980	—	—	21.4	20.8	20.8	—	21.6	—	—	—
1981	—	—	21.0	22.6	20.2	—	19.8	—	—	—
1982	—	—	14.4	15.6	14.6	—	16.2	17.4	—	—
1983	—	—	15.6	12.8	13.4	—	15.2	15.8	—	—
1984	—	—	12.4	6.4	12.8	—	20.4	11.4	—	—
1985	—	—	13.8	13.8	13.8	—	13.8	13.8	—	—
1986	239.3	—	20.3	16.4	19.1	—	16.9	16.4	20.6	19.1
1987	**267.5**	—	21.6	20.0	21.4	—	18.9	19.4	24.5	22.8
1988	260.6	—	19.6	17.4	19.4	—	20.8	16.6	23.2	21.6
1989	257.2	—	22.7	20.8	20.7	—	20.7	17.4	22.5	22.7
1990	250.7	—	23.7	17.9	24.7	—	20.0	19.6	21.0	—
1991	261.0	16.7	21.5	17.0	21.7	21.7	20.1	NR	20.9	—
1992	246.5	29.1	36.6	17.4	19.9	44.8	33.7	NR	25.6	—
1993	150.4	24.8	25.4	24.0	23.6	12.2	**35.9**	26.0	26.9	—
1994	27.8	22.9	24.6	30.3	24.6	22.1	29.8	19.9	32.3	—
1995	90.0	20.0	16.3	16.8	17.4	17.3	18.1	15.5	19.4	—
1996	112.0	14.0	22.1	26.9	28.4	13.4	15.7	10.5	29.7	—
1997	131.4	13.8	21.5	25.9	27.5	13.9	15.2	10.4	28.5	—
1998	135.0	22.0	23.3	23.3	24.3	23.1	24.7	20.4	24.3	—
1999	131.4	18.5	19.2	16.9	20.6	20.8	20.4	15.8	20.1	—
2000	149.2	24.5	22.0	20.6	29.5	22.0	24.0	19.7	23.8	—
2001	150.4	19.1	23.2	16.8	27.2	19.2	20.8	15.7	19.5	—
2002	142.4	18.4	23.3	15.7	27.1	18.8	20.4	14.8	18.4	—

a. 2,000-hour work-year, prorated from an 8,736-hour year.
b. Data for all years are from PORTS annual environmental reports with the exception of 1993 – 2002 for locations PP518, PP862,
PP906, PP933, PP1406, A35, A40, A36 and 874, which were obtained from USEC by separate communication (ATL 2003f).
— No data.

Finding 4.4.2-1: Use of the Generic Ambient Environmental Dose of 35.9 mrem/y is too restrictive for non-compensable claims

SC&A interprets the above-cited statements/guidance to mean that the generic ambient environmental dose of 35.9 mrem/y is to apply for **non**-compensable claims. We infer this from the statement that "…if background is to be subtracted, the ambient radiation for these areas should be assigned a value of 0 mrem because of the argument presented above."

Implicit in this statement is the assumption that there were no significant environmental releases at PORTS that would give rise to radiation dose rates above natural background. SC&A disagrees with this implicit assumption.

Finding 4.4.2-2: The Default Ambient Environmental Dose of 267 mrem/y to Workers Exposed at the Cylinder Storage Yards (i.e., location 874) is without Technical Basis and may be Too Low

Table 4.3.1-1 of TBD-4 (see Exhibit 4.4-2B) identifies the origin of the 267 mrem/y value as that corresponding to location 874 for the year 1989; and Table 4.3-3 of TBD-4 identifies the value 178 mrem/y as the average deep dose for location X-745E (depleted uranium cylinder storage yard) for the year 2000.

The difficulty in mating these two independent measurements is due to the following facts:

- TLD measurements in 1989 and in 2000 represent two very different time periods.
- TLD measurements in 1987 were recorded using Department of Energy Laboratory Accreditation Program (DOELAP) TLDs, which were neither calibrated nor processed for neutron exposure. To what extent the annual dose of 267 mrem incorporates a neutron dose component is therefore unknown. In the absence of data that might resolve this question, a more favorable approach would combine the 267 mrem dose of 1987 with the 178 mrem neutron dose of 2000.
- Section 4.3.1.2 of TBD-4 states that:

> *. . . This value [i.e., 178 mrem/y] for the annual neutron dose is obtained from Table 4.3-3 as the* ***maximum annual deep dose equivalent*** *reported in the period 1998 to 2001 for the cylinder storage yards X-745C and X-745E.* [Emphasis added]

Inspection of Exhibit 4.4-2A (i.e., Table 4.3-3 of TBD-4), however, identifies the much larger annual dose of 1,956 mrem/y for X-744G. TBD-4 provides no explanation why this dose was not considered.

Finding 4.4.2-3: Ambient Environmental Doses are Confined to the Deep Dose that may Significantly Underestimate Potential Exposures to the Skin

Claims involving **skin cancer** as well as other surficial tissues must be evaluated on the basis of shallow dose estimates. Dose estimates involving external ambient environmental doses are restricted to the **deep dose**, which may significantly underestimate the shallow dose.

4.5 REVIEW OF TBD-5 (ORAUT-TKBS-0015-5), PORTSMOUTH GASEOUS DIFFUSION PLANT – OCCUPATIONAL INTERNAL DOSE

Relevant Background Information Pertaining to Recycled Uranium

In 1952, the Atomic Energy Commission (AEC) approved the enrichment processing of production reactor returns or RU through the gaseous diffusion process. Between 1955 and 1974, a total of 1,094.6 MTU of RU was processed as feed material that had been made at PGDP and the Oak Ridge Gaseous Diffusion Plant. Subsequently, smaller quantities of RU from other sources were also processed at PORTS, including 1.9 MTU of RU-UF_6 extracted and produced by PORTS at the Decontamination and Uranium Recovery Building (Building X-705), Oxide Conversion Facility (Building X-705E), and Feed Production Facility (Building X-344).

At the time, it was known that RU contains low concentrations of transuranics (e.g., Np-237, Pu-238/-239/-240, and Am-241) and select long-lived fission products, most notable Tc-99. When feed materials containing RU contaminants are processed, their contaminants tend to concentrate in either finished products or in liquid raffinates and waste streams, such as tower and filter ashes. For example, when TRU isotopes are fluorinated and fed into the cascades, they tend to follow the heavier isotope of U-238, which is differentially concentrated in the bottom of cascade cells near the feed points. Conversely, the lighter Tc-99 contaminant tended to concentrate in top cascade cells and cascade vent alumina traps (and after 1975 in MgF_2 traps).

Neither the presence of RU contaminants and their tendency to concentrate at select process locations nor their potential for worker exposure was fully recognized for many years at PORTS. In their investigative report (DOE 2000), DOE's Office of Oversight provided the following observations, findings, and opinions:

> *In 1957, radiological surveys at the Paducah Plant identified that neptunium-237 was present in the enrichment cascade. Although the AEC recognized the potential for transuranic contamination of the cascades,* ***it was not until a 1965 appraisal that OR identified a potential problem with transuranics and fission products*** *in X-705E and recommended studies to determine where these materials could concentrate in the process. Records reflect that PORTS then reviewed the potential problems posed by feeding reactor returns to the oxide conversion plant; however, detailed studies were not performed. PORTS correspondence also indicates that* ***health physics staff did not fully understand the presence of transuranics and technetium-99, and appropriate analytical procedures were not developed as late as 1976.*** *During the 1970s, PORTS health physics and Plant managers participated in preplanning for receipt and subsequent*

processing of recycled uranium known to contain trace quantities of neptunium-237, plutonium-239/240, and technetium-99.

During the 1960s, the PORTS health physics group became concerned with increasing alpha radiation levels in process and support facilities at the site. While no records were identified to demonstrate that this issue was satisfactorily resolved, the period coincides with the processing of ***recycled uranium*** *at the* ***Paducah Plant****.* [Emphasis added.]

From page 21:

Some workers had extremely high intakes of uranium detected by bioassay or in-vivo testing that put them on work restriction for months or years. For example, in 1965 ten employees sustained lung exposures greater than one-half the permissible level, and eight were reported to the AEC as overexposures in accordance with AEC regulations. In addition, a worker who had a massive intake of UF_6 in 1973 was still excreting uranium six months later, and ***two workers in 1965 were exposed to uranium levels high enough that, as late as 1973, in-vivo testing showed greater than 50 percent of the maximum allowable body burden for uranium****. Finally, one worker, still living, was put on permanent restriction in 1981, and his in-vivo monitoring before his 1985 retirement still showed high uranium readings in his lungs.* [Emphasis added.]

From page 69:

Starting in 1975, Plant records reveal that elevated ***technetium*** *and* ***transuranic contamination*** *was unexpectedly discovered in liquid process effluents from the X-705. Before then, radiological effluent monitoring was only conducted for uranium and indicator parameters. The PORTS environmental monitoring program did not include these contaminants, which were known by Plant management to have been introduced into PORTS industrial facilities from the processing of reactor returns and from Paducah production feed material. Based on the information collected, it does not appear that personnel responsible for environmental monitoring were aware of the presence of these contaminants at PORTS…*

…The increase in technetium-99 discharges occurred shortly after the initiation of equipment changeout in the X-330/X-326 process buildings. A technetium treatment system was proposed in the late 1970s and installed in the early 1980s to reduce the levels being discharged into the environment.

By 1976, transuranics had also been identified in raffinates generated by the recovery of uranium from contaminated equipment and materials processed in X-705. These raffinates were discharged to the X-701B pond. Subsequent monitoring detected transuranics at significant levels in sludges from this pond and in the effluents from the pond to the east drainage ditch. Transuranics in the *effluent originated primarily in reactor-return materials processed in the X-705 Building. As an outcome of these findings, a committee was formed in December 1976 to study Plant-wide aspects of the transuranic contamination.*

*…**In 1979, isotopic analysis of two cascade deposits revealed relative high concentration of neptunium-237 (i.e., 55 percent and 60 percent of the total alpha activity in the samples was due to Np-237, respectively)**. However, there was no indication of a change in the radiological control program to address this issue, even though data was available to indicate that some level of transuranic contamination was present in the cascade. Transuranic sampling for work planning and control was not actively conducted until the 1990s.*

In June 2000, Bechtel Jacobs Company (BJC), in response to a request by the DOE, issued a report entitled *Recycled Uranium Mass Balance Project Portsmouth, Ohio Site Report* (BJC 2000). The report provided a detailed analysis regarding quantities and flow paths for uranium reprocessed from spent fuel from plutonium and tritium production reactors at other DOE sites. Among the study objectives was the identification of work locations, which had the highest potential for exposures to RU contaminants.

Locations and processes where worker exposure to TRU and Tc-99 was most probable were thought to include the following:

- <u>Decontamination and Uranium Recovery Building (X-705) and the Oxide Conversion Building (X-705E)</u>

  These facilities recovered uranium from decontamination solutions and incinerator ash in preparation for conversion to UF_6 for feed into the cascade. Between **1959** and **1961**, uranium oxides from spent reactor fuel was processed. The BJC Report acknowledged that ". . . A limited amount of information is available that describes the recycled constituents of the oxide processed in this facility . . . [and] How much TRU was present in each year of operation is **not** known, however, sample results do verify that TRU contamination was present." [Emphasis added.]

  DOE's assessment of this facility included the following statements (DOE 2000):

  > *…Probably as the most hazardous operations at PORTS…[where] Processing of transuranic-contaminated material was not adequately anticipated in the original or subsequent designs or operation. Samples obtained after shutdown showing the presence and level of transuranic contamination in the facility indicate that worker airborne exposures could have exceeded the acceptable standards, especially given the apparent lack of discipline in respirator use.*

- Buildings X-333, X-330, and X-326

  While TRUs were differentially concentrated/deposited in lower cascade cells near feed points, Tc-99 would sequentially concentrate in top cascade cells, as well as in alumina traps or MgF_2 traps. Exposure to TRUs and Tc-99 (as well as uranium isotopes and their daughters) would have resulted from **routine** equipment repair and maintenance. However, the most intense maintenance and modification to the PORTS cascades occurred between 1972 and 1983 in behalf of a program called the cascade improvement program and cascade upgrade program (CIP/CUP).

  CIP/CUP replaced or upgraded key cascade components (e.g., converters, compressors, transformers, motors, etc.) in order to improve the diffusion process efficiency and reliability.

- UF_6 Cylinder Cleaning and Storage (Building X-745)

  UF_6 cylinders of several diameter dimensions containing HEU RU were used at PORTS. The repeated filling of UF_6 cylinders without washing/removing the non-volatile heel has the effect of concentrating TRU and FP in the heel. The degree to which the constituents concentrate depends on the amounts added with each filling and the fraction removed during each feeding. It is assumed that a fraction of cylinders experienced several cycles before they were cleaned. It was further assumed that some cylinders were never cleaned and contained unknown concentrations of TRU.

Finding 4.5-1: Incorrect Values for Tc-99 were Derived for Reactor Tails

Table 5.1.2.6-3 in TBD-5 provides activity values for TRU and Tc-99 in reactor tails in dpm/g of U on an annual basis. These values were supposedly derived from data contained in Table 3.2-1 and Table 5.1-1 of the BJC 2000 report. For convenience, Tables 3.2-1 and 5.1-1 of the BJC 2000 report are reproduced herein as Exhibits 4.5-1A and 4.5-1B.

SC&A reviewed these data and concludes that Tc-99 values in Table 5.1.2.6-3 of TBD-5 are understated by several orders of magnitude, as illustrated below in a sample calculation.

Sample Calculation for Year 1955:

(1) Exhibit 4.5-1A (BJC 2000 Table 3.2.1) identifies that for 1955, PORTS received 105,873 kg of RU (or 1.059×10^8 g RU).

(2) Exhibit 4.5-1B (BJC 2000 Table 5.1-1) estimates that in 1955, the 105,873 kg of RU contained the following: 8 g of Np-237 and 3.45 **kg** of Tc-99.

(3) From the $t_{1/2}$ value of 2.14×10^6 years for Np-237, the specific activity of 7.14×10^{-4} Ci/g is derived; and the 8 g of Np-237 in 105,873 kg RU yields the following activity:

$$\text{Np-237 activity in RU} = \frac{(8\,g)(7.14\,x\,10^{-4}\ Ci/g)}{105{,}873{,}000\,g\ \ RU}$$

$$= 5.394 \times 10^{-11}\ Ci/g\ RU$$

$$= 53.94\ pCi/g\ RU$$

$$\text{Np-237} = 118\ dpm/g\ RU$$

SC&A's derived value of 118 dpm/g RU for Np-237 is fully consistent with NIOSH's value of 1.2×10^2 or 120 dpm/g RU, as given in Table 5.1.2.6-3 of TBD-5.

(4) For Tc-99, however, SC&A derives an activity value as described below:

With a half-life of 2.12×10^5 years, the specific activity of Tc-99 is equal to 0.01725 Ci/g, and the total activity of 3.45 kg of Tc-99. Thus, the activity of Tc-99 in RU yield the following:

$$\text{Tc-99 activity/g RU} = \frac{(3450\,g)(0.01725\,Ci/g)}{105{,}873{,}000\,g\ \ RU}$$

$$= \frac{59..5\,Ci}{105{,}873{,}000\,g\ \ RU}$$

$$= \frac{5.62\,x\,10^{-7}\,Ci}{g\,RU}$$

$$= 1{,}236{,}400\ dpm/g\ RU$$

$$\text{Tc-99/g RU} = 1.24 \times 10^6\ dpm/g\ RU$$

SC&A's value of 1.24×10^6 dpm/g RU for Tc-99 is **24,000 times higher** than NIOSH's value of 56.1 dpm/g RU cited in Table 5.1.2.6-3 of TBD-5. A review of Tc-99 values cited in Table 5.1.2.6-3 indicates that all values suffer an error of this magnitude.

EXHIBIT 4.5-1A

Table 3.2-1 PORTS Receipts Summary (RU Only)
(Source: BJC 2000)

Shipping Facility	Uranium Form	Net Weight (kgU)								
		FY 1955	FY 1956	FY 1957	FY 1958	FY 1966	FY 1967	FY 1968	FY 1969	FY 1972
Allied Chemical	UO_3									
Babcock & Wilcox	UF_6									
Division of International Affairs	UF_6							151		
	UNH					7	39			
Fernald	U_3O_8									
France	UF_6									65
Germany	UNH									
K-25	UF_4			865						
	UF_6		296,504							
	UO_2			418						
	UO_3			3,319						
NUMEC	UF_6									330
Paducah	UF_6	105,873	54,649	6,156	64,311				567,620	
United Kingdom	UNH									
USAEC Office Safeguards & Materials Management	UF_6								2,833	
Y-25	U_3O_8									
Grand Total		**105,873**	**351,154**	**10,758**	**64,311**	**7**	**39**	**151**	**570,453**	**395**

EXHIBIT 4.5-1A (Continued)

Shipping Facility	Uranium Form	Net Weight (kgU)							
		FY 1973	FY 1974	FY 1975	FY 1976	FY 1976.5	FY 1977	FY 1978	Grand Total
Allied Chemical	UO_3	1,376		1,403	1,295				4,074
Babcock & Wilcox	UF_6						153		153
Division of International Affairs	UF_6								151
	UNH								46
Fernald	U_3O_8				7,798				7,798
France	UF_6	202	324	128	273	112	152	235	1,586
Germany	UNH				6,860				6,860
K-25	UF_4								865
	UF_6								296,505
	UO_2								418
	UO_3								3,319
NUMEC	UF_6								330
Paducah	UF_6								798,609
United Kingdom	UNH			7					7
USAEC Office Safeguards & Materials Management	UF_6								2,833
Y-25	U_3O_8						104		104
Grand Total		**1,578**	**324**	**1,538**	**16,226**	**112**	**409**	**235**	**1,123,658**

EXHIBIT 4.5-1B

Table 5.1-1. Annual Inventory of RU Constituents (Source: BJC 2000)

Fiscal Year	Np (g)	Pu (g)	Tc-99 (kg)	Fiscal Year	Np (g)	Pu (g)	Tc-99 (kg)
1955	8.00	0.0	3.45	1977	49.62	0.23	61.61
1956	16.00	0.0	6.90	1978	49.73	0.23	58.88
1957	24.00	0.0	10.35	1979	49.73	0.23	56.78
1958	32.00	0.0	13.80	1980	49.73	0.23	55.31
1959	32.00	0.0	16.36	1981	49.73	0.23	52.83
1960	32.00	0.0	18.92	1982	49.66	0.21	50.46
1961	32.00	0.0	21.48	1983	49.46	0.16	49.15
1962	32.00	0.0	24.03	1984	49.90	0.13	48.38
1963	32.00	0.0	26.59	1985	48.90	0.13	47.68
1964	32.00	0.0	29.15	1986	48.90	0.13	47.18
1965	32.00	0.0	31.71	1987	48.90	0.13	46.84
1966	32.01	0.0	34.27	1988	48.90	0.13	46.55
1967	32.02	0.0	36.83	1989	48.90	0.13	46.33
1968	35.47	0.02	39.75	1990	48.90	0.13	46.01
1969	75.61	0.04	43.94	1991	48.90	0.13	45.82
1970	75.61	0.04	46.50	1992	48.90	0.13	45.51
1971	75.61	0.04	49.06	1993	48.90	0.13	44.40
1972	75.72	0.05	50.32	1994	48.90	0.13	44.23
1973	76.01	0.06	56.59	1995	48.24	0.12	43.89
1974	45.42	0.06	63.99	1996	47.58	0.12	43.54
1975	45.69	0.07	65.26	1997	46.92	0.11	43.20
1976	48.03	0.17	64.72	1998	46.26	0.11	35.80
				Mid-1999	44.30	0.11	35.11

Finding 4.5-2: Variable Bioassay Protocols were Employed that Significantly Affect the Interpretation of Urine Bioassay Data used for Dose Reconstruction

At PORTS, worker exposure to uranium isotopes was assessed by measuring the concentration of uranium in urine samples. Important to dose reconstruction are urine bioassay data that reflect **routine, special**, and **termination samples**.

Included among the many variables that affect the interpretation of urinalysis data for dose reconstruction is the time at which the urine sample is collected relative to the worker's most recent exposure. This critical time interval apparently changed over time, as indicated by the following:

DOE 2000 (page 37):

> *. . . During the 1950s and 1960s, urine samples were* ***typically*** *analyzed for uranium, and in most cases for alpha activity.* ***Typically****, the sample collection procedure involved the collection of* ***Monday morning*** *urine specimens (the morning following two or more days off the job). This was non-conservative, and*

> *the collection date* ***evolved*** *to a "Friday" sample during the 1970s and 1980s….* [Emphasis added.]

The use of "Monday morning" and "the end of the shift on the last day of the employer's workweek" urine samples are documented in the GAT *Standard Operating Procedure, Industrial Hygiene and Health Physics Department 212* (GAT 1963b) and in GAT-S-54 *Urinalysis Parameters* (GAT 1985b) (see Exhibits 4.5-2A and 4.5-2B). However, SC&A was unable to determine from available data/documents whether this transition for urine collection protocols occurred at a fixed moment in time or oscillated over a period of years, as suggested by DOE (2000).

NIOSH is fully aware that the uncertainty regarding the urine collection protocol will have a significant impact on the interpretation of bioassay data by IREP.

A review of TBD-5 for PORTS, however, not only fails to mention the change in protocol for obtaining urine samples over time, but incorrectly **implies** that "Monday morning" samples were consistently used in the past up to the present time, as given in the following:

> *…Routine samples were submitted on* ***Monday*** *of every week and recorded on form A-551. A special sample was given 4 hours after exposure and one voiding for suspected inhalation incidents. Total uranium analysis required 2 ml and a total alpha analysis required a 100-ml volume. Until around 1995, spot samples were the norm at PORTS.* [Emphasis added.]

EXHIBIT 4.5-2A

Samples are coded as Types I, II, or III. Type I indicates a routine sample, Type II a recall sample and Type III a supervisory request. Type II results are entered on a "hold" file in the department. All Type III results are entered in a department folder with the pertinent details of why the individual is being called in for a supervisory check, and the name of the supervisor requesting the sample.

If the accumulated positive values within a continuous positive series exceed either 0.30 milligrams of uranium per liter or 45 disintegrations per minute per 100 milliliters, the Medical Direction may then recommend limiting the work assignment of the individual involved until recall samples indicate an excretion rate below 0.06 milligrams of uranium per liter and/or 9.0 disintegrations per minute per 100 milliliters. Individuals on restriction submit daily samples until the excretion rate falls below the values indicated. At that time the work restriction is removed but two additional consecutive "Monday" samples are collected ("Monday" being defined as the first day back to work after being scheduled off for two or more days). The "Monday" samples must be below the established levels or the work restriction is reapplied. All restrictions are investigated as to the causes of the excessive excretion rates and an "Accident Investigation Report" is prepared. The investigation is conducted to determine the validity of the indicated exposure and to recommend measures to prevent recurrences. Records are maintained for future reference.

The following are established as plant acceptable limits for various urinary contaminants:

Contaminant	Plant Acceptable Limits
Uranium	0.06 mg/ liter
Alpha Activity	9.0 D/M/ 100 ml
Fluorides	4.0 mg/ liter
Mercury	0.1 mg/ liter

Film Badges: The film badge is a combined film packet holder and security badge designed to be attached to the outer layer of clothing. Incorporated within the badge are features which make it adaptable to estimate exposures to neutrons in the event of a critical reaction. The badges are assigned in pairs to each employee. Each badge is identical except for the color of the insert. This color differential is utilized to facilitate distribution, collection, and identification in the processing cycle. The Industrial Hygiene and Health Physics Department coordinates the necessary functions as to loading, identifying, processing, interpreting, and recording the results.

EXHIBIT 4.5-2B
(GAT 1986a)

TABLE I

URINALYSIS SAMPLING PARAMETERS

Contaminant	Class	Submission Criteria	Type
Uranium (Total)	Routine	Postshift at end of workweek	Spot
	Special	4 to 24 hours after exposure	Spot or Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Alpha (Total)	Routine	Postshift at end of workweek	Spot
	Special	4 to 24 hours after exposure	Spot or Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Technetium-99 (Total)	Routine	Postshift at end of workweek	Spot
	Special	4 to 24 hours after exposure	Spot or Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Fluoride (Total)	Routine	Postshift at end of workweek	Spot
	Special	4 to 24 hours after exposure	Spot or Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Mercury (Total)	Routine	Postshift at end of workweek	Spot
	Special	24-hours following exposure	Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Lead (Total)	Routine	Postshift at end of workweek	Spot
	Special	24-hours following exposure	Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Chromium (Total)	Routine	Postshift at end of workweek	Spot
	Special	24-hours following exposure	Composite
	Recall	2 Samples (Pre-and Postshift)	Spot
Nickel (Total)	Routine	Postshift at end of workweek	Spot
	Special	24-hours following exposure	Composite
	Recall	2 Samples (Pre- and Postshift)	Spot

EXHIBIT 4.5-2B (Cont.)

6.1.4 Once an employee is selected for inclusion in the routine urinalysis program, participation is mandatory until they are removed from the program by HP-IH.

6.2 Routine Urinalysis Program

6.2.1 HP-IH shall mail the RUNS to each Custodian (or designee) on the first work day of each week.

6.2.2 The Urinalysis Program Custodian shall distribute the RUNS to the appropriate FLM.

6.2.3 FLMs shall distribute the RUNS to each scheduled employee, and ensure employees receive notification of submission (RUN) within appropriate time frame.

6.2.4 HP-IH shall ensure that the information required on the "Urine Sample Record or Receipt" is complete when collecting samples.

6.2.5 FLM or Urinalysis Program Custodians shall notify HP-IH of personnel who are unable to submit routine samples (i.e. LOA).

6.2.6 Employees scheduled for sampling shall submit routine urine samples within three weeks of the date listed on the RUN (or the last day of day shift), bring their RUN to a sample collection location and submit a urine sample with a volume $\geq$ 100ml. (Routine sample collection locations are shown in Appendix E.)

A. Employees should submit scheduled samples prior to leaving site if their return is NOT within the three week time frame from the scheduled date.

B. An employee that has lost their RUN shall report to Health Services or Fire Services (after hours) and submit a sample as scheduled.

C. Health Services and/or Fire Services shall complete an NRUF indicating that the sample is a Lost Notice on the "Urine Sample Record of Receipt"..

6.2.7 The employee shall submit the routine urinalysis sample as follows:

A. Report to sample collection station

Finding 4.5-3: Current Guidance for Estimating Internal Exposure to RU Contaminants Unachievable and/or Inappropriate

From the foregoing discussion, it is reasonable to conclude that any assessment of internal exposure to TRU and Tc-99 on the basis of in vitro urinalysis and in-vivo chest counting is severely limited. In behalf of these limitations, NIOSH provides the following guidance (Section 5.1.2.6, p. 13 of TBD-5):

> *If specific* ***source term*** *information to which the employee has been exposed is available, the dose reconstructor should utilize that information. However, if no* ***source term*** *information is available, the values and parameters in Table 5.1.2.6-4 provide an adequate input to the process. Note that the activity fraction and contaminant ratio to uranium columns are based upon 1993 through 1999 PORTS* ***air sampling data*** *and is therefore a snapshot of the temporal radionuclide matrix of the* ***facilities****.* [Emphasis added.]

SC&A regards NIOSH's guidance for internal dose assessment involving TRU and Tc-99 (as well as for thorium isotopes and Pa-234m listed in Table 5.1.2.6-4) as either **unachievable** or **inappropriate**, as explained below.

The term **unachievable** is a reference to NIOSH's guidance to make use of "source term" information. Source term information implies the availability of data over time and location. Source term materials (e.g., filter ash, tower ash, cascade deposits, cylinder heels, pond sludges, MgF_2 traps, raffinates, etc.) that may have concentrated select radionuclides and given rise to airborne contamination. Such data do not exist and will, therefore, not be available to the dose reconstructor. The term **inappropriate** is a reference to the use of **general air** sample data collected in the **1990s** that only define activity fractions for select isotopes relative to uranium (see Table 5.1.2.6-4 of TBD-5). For use in dose reconstruction, these **relative activity** values defined in the 1990s must be applied to uranium bioassay data that may have been taken decades earlier and reflect processes/facilities/radiological conditions that have little to no relevance. The following examples illustrate SC&A's concerns regarding the use of Table 5.1.2.6-4.

<u>Example #1: DOE (2000), page 35, provides the following statement</u>:

> *… In 1979, isotopic analysis of two cascade deposits revealed relative high concentrations of neptunium-237 (i.e.,* ***55 percent*** *and* ***60 percent*** *of the total alpha activity in the samples was due to Np-237, respectively). However, there was no indication of a change in the radiological control program to address this issue, even though data was available to indicate that some level of transuranic contamination was present in the cascade. Transuranic sampling for work planning and control was not actively conducted until the 1990s.* [Emphasis added.]

Such high concentrations of TRU contaminants are likely to have exposed cascade workers (Buildings X-333, 330, and 326) as well as maintenance/decontamination workers affiliated with

CIP/CUP (Buildings X-700, X-705, X-705E, and X-720). Inspection of Table 5.1.2.6-4 in TBD-5 identifies Np-237 activity fractions that are **several orders of magnitude lower**.

Example #2: GAT-521-75-113 (GAT 1975) and BCJ (2000):

In the interdepartmental correspondence enclosed herein as Exhibit 4.5-3A, cascade samples were analyzed for Tc-99 in the inlet lines and valves at a concentration of 20% by weight.

Even higher concentrations were cited in the BJC 2000 Report, which stated the following (pp. 63–64):

> *The Tc-99 contamination in cascade equipment has been an acknowledged problem since the mid-1970s. Some materials removed from the cascade at areas known to concentrate Tc-99 have been as high as 40% by weight Tc-99.*

At these concentrations, the relative activity of Tc-99 to uranium is orders of magnitude higher than values cited in Table 5.1.2.6-4.

In summary, credible bioassay data for TRU and FP contaminants contained in recycled uranium are not available, and the default methodology imbedded in Table 5.1.2.6-4 is inappropriate and not claimant favorable.

Example #3: From document No. 74 – *Summary of Materials on Plantsite from TRU Activities at X-705* (see Exhibit 4.5-3B)

Exhibit 4.5-3B identifies the analysis of ash in "container F861667." This ash material was assayed at 2.9% U-235 enrichment. As such, the specific activity of uranium in the ash is estimated at 1.53 µCi/g:

- Sp. Act of U_{enrich} $= (0.4 + 0.38E + 0.0034E^2)$ µCi/g
  $= (0.4 + 1.10 + 0.0286)$ µCi/g
  $= 1.53$ µCi/g U

- Thus, the Pu_{Total} of 1,970,000 dpm/g U is equal to 0.895 µCi/g U and represents a fractional activity 5.85×10^{-1} relative to uranium.

The above-derived value of 5.85×10^{-1} is more than 2,000 times the generic value of 2.64×10^{-4} recommended for the X-705 and X-705E location defined in Table 5.1.2.6-4.

EXHIBIT 4.5-3A

INTERDEPARTMENTAL CORRESPONDENCE

TO: R. D. Jackson, Supervisor

DATE: June 26, 1975

DEPT: Materials Sampling & Testing
LOCATION: X-710 Building

CODE: GAT-521-75-113

1799

SUBJECT: SAFETY PROCEDURES FOR HANDLING TECHNETIUM-CONTAINING MATERIALS

Cascade samples from Area 6 have recently been found to contain excessive amounts of technetium^{99} (Tc^{99}), a potentially hazardous element because of its beta (β^-) radiation. During June 16-20, 1975, multigram deposits removed from the inlet lines and valves to the Freon Degrader were found to consist of 20 percent Tc^{99} by weight.

While safe procedures for dealing with materials containing enriched uranium are standard practice at GAT, technetium contamination is a new and separate problem. The unfamiliar nature and unexpected presence of large quantities of Tc^{99} demand that special precautions be taken by plant personnel who must handle cascade equipment or materials contaminated with technetium.

Please initiate the necessary action to define the scope of the technetium hazards problem and specify safe procedures for sampling, storage, and disposal of technetium-bearing materials from operations involving cascade equipment maintenance to laboratory testing.

A. J. Saraceno
A. J. Saraceno
Process Technology Department

EXHIBIT 4.5-3B

SUMMARY OF MATERIAL ON PLANTSITE FROM TRU ACTIVITIES AT X-705*

Container	Source	Date	Assay	g/U	g/U-235	Net	Location	Pu Analysis
UF6 (Full)								
5SST0119	A/A	1-74	48.420	3508	1699	5205(g)	345S-JJ-021	No
5SST0403	NLO	5-76	3.562	1560	56	2308(g)	744G-7-9-42	No
12120076	NLO	1-76	2.923	7079	207	10.47(kg)	744G-7-15-14	No
12120187	NLO	5-76	3.017	114129	3443	168.88(kg)	744G-5E-Dolly	No
3A300076	NLO	5-76	2.920	517279	15105	1687(lb)	745B-7-1-B-1	No
3A300173	NLO	5-76	2.920	1215469	35492	3964(lb)	745B-7-1-C-4	No
3A300195	NLO	5-76	2.920	1371542	40049	4473(lb)	745B-7-1-B-5	No
3A300208	NLO	1-76	2.922	1491435	43580	4863(lb)	745B-7-1-A-17	No
Oxide								
265 containers	A/A	4-75	54.000	(UO_3 Material)			345N	No
ZF11116	NLO	10-75	2.956	5499	163	6526(g)	345N-P-141	No
Non UF6								
F860579(NaF)	A/A	2-74	58.300	20	12	10507(g)	744G-5C-9-5	No
F861667(Ash)	NLO	6-76	2.900	348	10	1698(g)	744G-4-17-38	1,970,000 d/m/gU
UF6 (Empty w/Heels)								
12120157	NLO	1-76	2.952	95	3	.14(kg)	744G-5E-1-A-1	No ?

* Scrap returns from Allied/Aerojet, NLO, and RGI/Germany (Jan. 1974 to Dec. 1977).

Finding 4.5-4: Generic Default Value for Uranium Enrichment is Inappropriate/Claimant Unfavorable

Worker exposure to uranium at PORTS was primarily assessed by means of urinalysis. Because exposures at PORTS may have involved depleted, natural, low-enriched, enriched, highly enriched, and very highly enriched uranium, an assessment of both chemical toxicity and radiological exposure required that two independent bioassays be performed concurrently: (1) the concentrations of elemental uranium in urine was determined by a standard fluorimetric procedure; and (2) the specific activity of the uranium isotopic mixture was determined by gross alpha activity. An essential step in the gross alpha urinalysis is the chemical isolation/extraction of uranium from a urine sample.

For dose reconstruction purposes, it is the second or gross alpha urinalysis that is critical. In a 1966 GAT Report entitled, *Determination of Uranium Alpha Activity in Urine* (GAT 1966), the authors described a new and efficient extraction method that would replace the previous/existing methods, as given in the following:

> *… Previously available procedures for determining alpha activity, however, have not been entirely satisfactory; the main problem has been the long, tedious chemical operations used to separate a sufficient quantity of uranium from urine for satisfactory alpha counting. This situation is exemplified in the previously used method of Whitson and Kwasnoski in which uranium is separated from 100 ml of urine by (1) destruction of organic matter by wet oxidation and ignition, (2) two precipitations with oxalate, and (3) electroplating on a nickel disk.*

Regardless of the time period and which method was used for gross alpha analysis of urine samples, not all urine fluorimetric bioassays were concurrently evaluated for gross alpha activity. For those instances in which fluorimetric bioassay data are **not** accompanied by gross alpha data Section 6.7 on page 24 of TBD-5 informs the dose reconstruction to assume ". . . uranium enrichment – 3.5% unless HEU of 93% is suspected."

It should be noted that a 3.5% enrichment has the specific activity of 2.2 pCi/μg of uranium (or 2.2 μCi/g of uranium).

SC&A questions the default value of 3.5% enrichment with its attendant specific activity of 2.2 μCi/g of U as generically applicable without consideration of time and location/job function.

In the 1986 GAT-S-60 Report (GAT 1986b), data are presented that identify huge fluctuations in the specific activity values over time and by job categories. Provided herein as Exhibits 4.5-4A and 4.5-4B are summary bioassay data for **all** monitored workers for years 1965 through 1985. While page 1 of Exhibit 4.5-4A identifies gross alpha data (in dpm/100 ml urine), Exhibit 4.5-4B identifies uranium concentrations (in mg/liter of urine). By comparing these data, an estimate of the specific activity can be readily determined, as illustrated below:

Example Calculations of Average Specific Activity for All Monitored Workers

For 1965:

Average α activity for routine urine = 5.67 dpm/100 ml
= 56.7 dpm/liter
= 25.7 pCi/liter

Average U conc. for routine urine = 1.87×10^{-3} mg/liter
= 1.87 μg/liter

$$\text{Specific Activity} = \frac{25.7\ pCi / liter}{1.87\ \mu g / liter}$$
= 13.8 pCi/μg

For 1969:

Average activity for routine urine = 2.46 dpm/100 ml
= 24.6 dpm/liter
= 11.2 pCi/liter

Average U conc. for routine urine = 9.65×10^{-5} mg/liter
= 9.65×10^{-2} μg/liter

$$\text{Specific Activity} = \frac{11..2\ pCi / liter}{9.65 x 10^{-2}\ \mu g / liter}$$
= 116 pCi/μg U

The empirically derived yearly average values of 13.8 pCi/μg U and 116 pCi/μg U for 1965 and 1969, respectively, are more than 6 and more than 50 times higher than NIOSH's recommended generic default value of 2.2 pCi/μg U.

When bioassay data are further segregated by job function, the disparity for select job functions is further amplified. Exhibits 4.5-4C and 4.5-4D identify bioassay data for workers classified as Material Handling and Sampling. For 1965, the **average** annual specific activity of excreted uranium is calculated at 120 pCi/μg U. Even higher values are derived for Chemical Operators, whose data are given in Exhibits 4.5-4E and 4.5-4F. For 1969, the annual average specific activity for routine urine samples is estimated at 147 pCi/μg U, which is nearly 67 times the default value of 2.2 pCi/μg U.

Derived specific activity levels of this magnitude suggest that worker exposure involved high to very highly enriched uranium. Thus, the **generic** default assumption of 3.5% enrichment must clearly be regarded as inappropriate and claimant unfavorable.

EXHIBIT 4.5-4A

HISTORICAL URINALYSIS ALPHA RESULTS SUMMARY FOR PLANTSITE
NOTE: AVERAGES IN UNITS OF ALPHA DPM PER ONE HUNDRED MILLILITERS OF SAMPLE.

YEAR	TYPE OF SAMPLE	NUMBER OF SAMPLES SUBMITTED	NUMBER OF POSITIVE RESULTS	NUMBER OF RESULTS EXCEEDING FLAG LEVEL	NUMBER OF RESULTS EXCEEDING INVESTIGATE LEVEL	NUMBER OF RESULTS EXCEEDING RESTRICT LEVEL	AVERAGE RESULT	AVERAGE OF RESULTS EXCEEDING RESTRICT LEVEL
65	ROUTINE	1819	1034	185	88	12	5.67565	424.667
	SPECIAL	336	200	0	0	0	2.22321	0
66	ROUTINE	2185	833	30	18	2	1.24531	460
	SPECIAL	482	204	0	0	0	1.05809	0
67	ROUTINE	2105	611	8	2	1	0.660808	411
	SPECIAL	388	194	0	0	0	4.29381	0
68	ROUTINE	2460	785	20	2	0	0.538618	0
	SPECIAL	525	250	0	0	0	1.64571	0
69	ROUTINE	3072	1480	186	90	13	2.46354	126.231
	SPECIAL	433	278	0	0	0	5.79446	0
70	ROUTINE	2994	1329	83	27	4	1.26987	119
	SPECIAL	448	250	5	0	0	15.8304	0
71	ROUTINE	3359	1741	87	23	4	1.32331	128
	SPECIAL	594	380	0	0	0	3.88552	0
72	ROUTINE	2870	1304	35	13	2	0.959582	162
	SPECIAL	368	209	0	0	0	1.75	0
73	ROUTINE	3212	1530	148	77	12	2.42403	234.5
	SPECIAL	444	287	50	0	0	123.644	0
74	ROUTINE	2750	1293	23	3	0	0.770182	0
	SPECIAL	306	196	0	0	0	9.93464	0
75	ROUTINE	4895	3275	96	48	9	1.88621	[illegible]
	SPECIAL	855	679	2	0	0	10.4795	0
76	ROUTINE	3567	2493	11	2	0	1.02944	0
	SPECIAL	1499	1196	1	0	0	3.57772	0
77	ROUTINE	4002	3093	27	5	2	1.56872	[illegible]
	SPECIAL	1599	1321	0	0	0	3.31395	0
78	ROUTINE	4267	1554	33	14	1	0.540193	[illegible]
	SPECIAL	1666	858	1	1	1	7.83799	[illegible]
79	ROUTINE	2439	1124	13	4	1	0.543994	83
	SPECIAL	1255	839	0	0	0	1.39594	0
80	ROUTINE	5938	4100	19	9	2	1.40308	[illegible]
	SPECIAL	1846	1425	1	0	0	1.84297	0
81	ROUTINE	7431	4848	33	17	4	0.8972	[illegible]
	SPECIAL	772	601	0	0	0	2.425	0
82	ROUTINE	6010	3092	36	13	3	0.717554	[illegible]
	SPECIAL	561	395	0	0	0	3.618	0
83	ROUTINE	3907	2115	26	9	2	0.755696	200.5
	SPECIAL	505	340	0	0	0	2.65782	0
84	ROUTINE	2987	1498	15	3	0	0.80144	0
	SPECIAL	315	183	0	0	0	6.25746	0
85	ROUTINE	4845	2323	29	10	1	0.829515	86
	SPECIAL	263	149	1	0	0	5.69582	0

? EOF in line 1700

Time: 332.08 secs.

Ready

EXHIBIT 4.5-4B

HISTORICAL URINALYSIS URANIUM RESULTS SUMMARY FOR PLANTSITE
NOTE: AVERAGES IN UNITS OF MILLIGRAMS URANIUM PER LITER OF SAMPLE.

YEAR	TYPE OF SAMPLE	NUMBER OF SAMPLES SUBMITTED	NUMBER OF POSITIVE RESULTS	NUMBER OF RESULTS EXCEEDING FLAG LEVEL	NUMBER OF RESULTS EXCEEDING INVESTIGATE LEVEL	NUMBER OF RESULTS EXCEEDING RESTRICT LEVEL	AVERAGE RESULT	AVERAGE OF RESULTS EXCEEDING RESTRICT LEVEL
65	ROUTINE	2160	160	9	7	3	1.87037E-3	0.483333
	SPECIAL	348	68	7	3	1	4.88506E-3	0.21
66	ROUTINE	2226	32	8	4	4	3.80952E-3	1.9675
	SPECIAL	487	18	4	1	0	1.04723E-3	0
67	ROUTINE	2312	69	0	0	0	3.28720E-4	0
	SPECIAL	403	65	10	6	3	6.62531E-3	0.413333
68	ROUTINE	2860	40	0	0	0	1.64336E-4	0
	SPECIAL	537	74	13	9	0	4.22719E-3	0
69	ROUTINE	3421	26	0	0	0	9.64630E-5	0
	SPECIAL	436	45	11	5	2	8.78440E-3	1.2
70	ROUTINE	3358	83	1	1	0	3.42466E-4	0
	SPECIAL	453	56	11	7	1	4.39294E-3	0.32
71	ROUTINE	3718	93	2	0	0	3.44271E-4	0
	SPECIAL	580	49	6	4	2	6.48276E-3	1.375
72	ROUTINE	3730	34	1	0	0	1.60858E-4	0
	SPECIAL	380	30	2	0	0	1.28947E-3	0
73	ROUTINE	4081	74	6	2	1	4.43519E-4	0.55
	SPECIAL	400	38	14	7	3	0.016225	1.66333
74	ROUTINE	3170	31	6	0	0	2.27129E-4	0
	SPECIAL	382	37	9	3	0	4.[illegible]E-3	[illegible]
75	ROUTINE	5131	125	5	2	0	3.99532E-4	0
	SPECIAL	977	231	35	1[illegible]	4	7.04196E-3	0.33[illegible]
76	ROUTINE	3569	64	2	0	0	2.49370E-4	0
	SPECIAL	1500	186	20	7	2	2.80667E-3	0.[illegible]
77	ROUTINE	4015	506	5	2	2	4.33375E-3	5
	SPECIAL	2359	578	36	11	4	7.53285E-3	2.0425
78	ROUTINE	9579	196	11	6	2	9.89665E-4	0.1
	SPECIAL	2180	360	72	41	13	1.19[illegible]E-2	0.[illegible]
79	ROUTINE	4447	132	18	13	[illegible]	7.60063E-4	0.[illegible]
	SPECIAL	1499	112	9	7	3	2.46831E-3	0.576667
80	ROUTINE	6186	100	6	4	2	2.07183E-3	7.66
	SPECIAL	2128	68	7	7	4	1.57895E-3	0.525
81	ROUTINE	9450	67	1	1	0	1.19577E-4	0
	SPECIAL	1008	74	13	5	3	0.052004	15.8[illegible]
82	ROUTINE	7322	59	1	0	0	1.13357E-4	0
	SPECIAL	744	86	15	9	2	4.07258E-3	0.325
83	ROUTINE	4635	27	2	0	0	9.92449E-5	0
	SPECIAL	586	25	0	0	0	6.82594E-4	0
84	ROUTINE	3945	21	3	0	0	7.60456E-5	0
	SPECIAL	501	27	3	0	0	1.13772E-3	0
85	ROUTINE	7405	106	10	4	1	2.72789E-4	0.2
	SPECIAL	627	102	19	14	4	6.36364E-3	0.315

? EOF in line 1700

Time: 327.58 secs.

Ready

EXHIBIT 4.5-4C

HISTORICAL URINALYSIS ALPHA RESULTS SUMMARY FOR MATERIAL HANDLING AND SAMPLING
NOTE: AVERAGES IN UNITS OF ALPHA DPM PER ONE HUNDRED MILLILITERS OF SAMPLE.

YEAR	TYPE OF SAMPLE	NUMBER OF SAMPLES SUBMITTED	NUMBER OF POSITIVE RESULTS	NUMBER OF RESULTS EXCEEDING FLAG LEVEL	NUMBER OF RESULTS EXCEEDING INVESTIGATE LEVEL	NUMBER OF RESULTS EXCEEDING RESTRICT LEVEL	AVERAGE RESULT	AVERAGE OF RESULTS EXCEEDING RESTRICT LEVEL
65	ROUTINE	63	40	7	3	0	3.90476	0
	SPECIAL	3	2	0	0	0	17.3333	0
66	ROUTINE	80	30	3	2	0	1.2375	0
	SPECIAL	5	5	0	0	0	4.8	0
67	ROUTINE	90	23	0	0	0	0.5	0
	SPECIAL	6	3	0	0	0	0.666667	0
68	ROUTINE	103	36	0	0	0	0.456311	0
	SPECIAL	10	8	0	0	0	3.3	0
69	ROUTINE	94	50	9	4	0	2.44681	0
	SPECIAL	19	15	0	0	0	3.73684	0
70	ROUTINE	95	56	0	0	0	1.2	0
	SPECIAL	12	9	0	0	0	3.41667	0
71	ROUTINE	139	94	13	3	0	3	0
	SPECIAL	19	17	0	0	0	10.2632	0
72	ROUTINE	177	117	7	2	0	1.99435	0
	SPECIAL	8	7	0	0	0	7.5	0
73	ROUTINE	208	118	8	4	0	1.79808	0
	SPECIAL	13	12	0	0	0	10.7692	0
74	ROUTINE	617	288	6	1	0	0.773096	0
	SPECIAL	11	8	0	0	0	3.09091	0
75	ROUTINE	1214	820	27	13	3	2.12603	298
	SPECIAL	132	110	2	0	0	26.8485	0
76	ROUTINE	871	620	6	1	0	1.24569	0
	SPECIAL	53	42	0	0	0	10.6604	0
77	ROUTINE	615	473	5	1	0	1.42439	0
	SPECIAL	74	60	0	0	0	12.1081	0
78	ROUTINE	355	114	7	3	1	0.918873	90
	SPECIAL	87	52	0	0	0	2.70115	0
79	ROUTINE	244	142	4	2	1	1.20738	83
	SPECIAL	56	30	0	0	0	1.11607	0
80	ROUTINE	325	243	7	4	0	1.36292	0
	SPECIAL	64	49	0	0	0	2.07656	0
81	ROUTINE	318	243	14	9	1	2.2566	150
	SPECIAL	24	23	0	0	0	17.8042	0
82	ROUTINE	259	170	19	8	2	2.75907	97
	SPECIAL	23	20	0	0	0	2.23043	0
83	ROUTINE	232	149	10	6	2	3.175	200.5
	SPECIAL	41	34	0	0	0	11.8736	0
84	ROUTINE	183	105	6	1	0	1.31311	0
	SPECIAL	32	24	0	0	0	49.3781	0
85	ROUTINE	530	288	10	6	0	1.41887	0
	SPECIAL	45	36	1	0	0	24.9778	0

? EOF in line 1700

Time: 309.81 secs.

Ready

EXHIBIT 4.5-4D

HISTORICAL URINALYSIS URANIUM RESULTS SUMMARY FOR MATERIAL HANDLING AND SAMPLING
NOTE: AVERAGES IN UNITS OF MILLIGRAMS URANIUM PER LITER OF SAMPLE.

YEAR	TYPE OF SAMPLE	NUMBER OF SAMPLES SUBMITTED	NUMBER OF POSITIVE RESULTS	NUMBER OF RESULTS EXCEEDING FLAG LEVEL	NUMBER OF RESULTS EXCEEDING INVESTIGATE LEVEL	NUMBER OF RESULTS EXCEEDING RESTRICT LEVEL	AVERAGE RESULT	AVERAGE OF RESULTS EXCEEDING RESTRICT LEVEL
65	ROUTINE	68	1	0	0	0	1.47059E-4	0
	SPECIAL	3	2	0	0	0	6.66667E-3	0
66	ROUTINE	82	3	1	0	0	1.46341E-3	0
	SPECIAL	5	0	0	0	0	0	0
67	ROUTINE	91	0	0	0	0	0	0
	SPECIAL	6	1	1	1	0	1.83333E-2	0
68	ROUTINE	108	0	0	0	0	0	0
	SPECIAL	10	0	0	0	0	0	0
69	ROUTINE	96	1	0	0	0	1.04167E-4	0
	SPECIAL	19	1	0	0	0	5.26316E-4	0
70	ROUTINE	98	2	0	0	0	2.04082E-4	0
	SPECIAL	12	1	1	0	0	4.16667E-3	0
71	ROUTINE	142	6	1	0	0	9.85915E-4	0
	SPECIAL	19	7	1	1	0	1.21053E-2	0
72	ROUTINE	182	2	0	0	0	1.09890E-4	0
	SPECIAL	8	1	0	0	0	0.0025	0
73	ROUTINE	212	7	0	0	0	3.77358E-4	0
	SPECIAL	13	4	0	0	0	4.61538E-3	0
74	ROUTINE	619	8	1	0	0	2.58481E-4	0
	SPECIAL	11	6	0	0	0	9.09091E-3	0
75	ROUTINE	1215	36	0	0	0	3.86831E-4	0
	SPECIAL	133	30	5	4	1	8.12030E-3	0.2
76	ROUTINE	871	16	0	0	0	2.06659E-4	0
	SPECIAL	53	11	3	1	0	8.11321E-3	0
77	ROUTINE	610	66	1	0	0	1.16393E-3	0
	SPECIAL	81	16	4	0	0	5.30864E-3	0
78	ROUTINE	373	6	1	0	0	3.75335E-4	0
	SPECIAL	101	24	11	5	2	1.70297E-2	0.3
79	ROUTINE	265	6	2	0	0	7.16981E-4	0
	SPECIAL	59	2	0	0	0	3.38983E-4	0
80	ROUTINE	328	3	0	0	0	1.82927E-4	0
	SPECIAL	72	8	1	1	1	4.02778E-3	0.2
81	ROUTINE	317	0	0	0	0	0	0
	SPECIAL	24	13	5	3	1	4.91667E-2	0.65
82	ROUTINE	259	4	0	0	0	1.54440E-4	0
	SPECIAL	25	5	0	0	0	0.0032	0
83	ROUTINE	240	2	0	0	0	8.33333E-5	0
	SPECIAL	42	5	0	0	0	2.14286E-3	0
84	ROUTINE	224	10	3	0	0	8.92857E-4	0
	SPECIAL	36	10	2	0	0	6.66667E-3	0
85	ROUTINE	669	25	3	0	0	6.42750E-4	0
	SPECIAL	173	37	7	6	2	8.43931E-3	0.32

? EOF in line 1700

Time: 358.38 secs.

Ready

EXHIBIT 4.5-4E

HISTORICAL URINALYSIS ALPHA RESULTS SUMMARY FOR CHEMICAL OPERATIONS
NOTE: AVERAGES IN UNITS OF ALPHA DPM PER ONE HUNDRED MILLILITERS OF SAMPLE.

YEAR	TYPE OF SAMPLE	NUMBER OF SAMPLES SUBMITTED	NUMBER OF POSITIVE RESULTS	NUMBER OF RESULTS EXCEEDING FLAG LEVEL	NUMBER OF RESULTS EXCEEDING INVESTIGATE LEVEL	NUMBER OF RESULTS EXCEEDING RESTRICT LEVEL	AVERAGE RESULT	AVERAGE OF RESULTS EXCEEDING RESTRICT LEVEL
65	ROUTINE	523	405	114	64	7	10.0382	299
	SPECIAL	68	49	0	0	0	3.91176	0
66	ROUTINE	540	267	11	5	0	1.1037	0
	SPECIAL	82	39	0	0	0	0.890244	0
67	ROUTINE	393	129	1	0	0	0.524173	0
	SPECIAL	34	17	0	0	0	2.94118	0
68	ROUTINE	462	177	10	2	0	0.904762	0
	SPECIAL	145	89	0	0	0	2.02759	0
69	ROUTINE	591	352	55	23	3	3.27073	96.3333
	SPECIAL	108	80	0	0	0	8.87037	0
70	ROUTINE	402	178	2	2	0	0.828358	0
	SPECIAL	55	32	0	0	0	1.63636	0
71	ROUTINE	505	287	12	4	0	1.20594	0
	SPECIAL	192	145	0	0	0	4.33854	0
72	ROUTINE	609	296	8	4	0	0.878489	0
	SPECIAL	97	58	0	0	0	3.06186	0
73	ROUTINE	710	374	45	27	6	4.25352	270.833
	SPECIAL	115	65	2	0	0	33.3217	0
74	ROUTINE	473	228	5	1	0	0.849894	0
	SPECIAL	44	31	0	0	0	37.3864	0
75	ROUTINE	714	482	11	7	2	[illegible]	[illegible]
	SPECIAL	172	146	0	0	0	6.57558	0
76	ROUTINE	321	245	1	0	0	1.16199	0
	SPECIAL	185	145	0	0	0	5.01622	0
77	ROUTINE	175	141	2	0	0	1.38571	0
	SPECIAL	148	123	0	0	0	2.88514	0
78	ROUTINE	353	106	7	1	0	0.58102	0
	SPECIAL	114	67	0	0	0	1.72368	0
79	ROUTINE	194	104	2	0	0	[illegible]	0
	SPECIAL	101	74	0	0	0	1.31287	0
80	ROUTINE	485	355	3	3	1	1.45876	[illegible]
	SPECIAL	132	103	0	0	0	1.70909	0
81	ROUTINE	502	382	2	1	0	1.04004	0
	SPECIAL	60	51	0	0	0	1.45833	0
82	ROUTINE	344	216	4	1	0	1.05145	0
	SPECIAL	42	34	0	0	0	6.04524	0
83	ROUTINE	287	203	6	0	0	1.18223	0
	SPECIAL	26	18	0	0	0	0.988462	0
84	ROUTINE	269	171	3	1	0	1.27546	0
	SPECIAL	19	13	0	0	0	1.57895	0
85	ROUTINE	441	245	8	1	0	1.12472	0
	SPECIAL	7	3	0	0	0	1.14286	0

? EOF in line 1700

Time: 382.78 secs.

Ready

EXHIBIT 4.5-4F

HISTORICAL URINALYSIS URANIUM RESULTS SUMMARY FOR CHEMICAL OPERATIONS
NOTE: AVERAGES IN UNITS OF MILLIGRAMS URANIUM PER LITER OF SAMPLE.

YEAR	TYPE OF SAMPLE	NUMBER OF SAMPLES SUBMITTED	NUMBER OF POSITIVE RESULTS	NUMBER OF RESULTS EXCEEDING FLAG LEVEL	NUMBER OF RESULTS EXCEEDING INVESTIGATE LEVEL	NUMBER OF RESULTS EXCEEDING RESTRICT LEVEL	AVERAGE RESULT	AVERAGE OF RESULTS EXCEEDING RESTRICT LEVEL
65	ROUTINE	528	97	4	3	1	→ 3.44697E-3	0.35
	SPECIAL	68	9	0	0	0	2.35294E-3	0
66	ROUTINE	540	9	0	0	0	1.85185E-4	0
	SPECIAL	83	1	0	0	0	1.20482E-4	0
67	ROUTINE	402	21	0	0	0	5.97015E-4	0
	SPECIAL	36	8	2	1	1	0.01	0.2
68	ROUTINE	470	15	0	0	0	3.61702E-4	0
	SPECIAL	145	22	4	3	0	4.75862E-3	0
69	ROUTINE	594	5	0	0	0	1.01010E-4	0
	SPECIAL	108	4	0	0	0	9.25926E-4	0
70	ROUTINE	406	5	0	0	0	2.21675E-4	0
	SPECIAL	55	6	0	0	0	1.63636E-3	0
71	ROUTINE	508	8	0	0	0	1.57480E-4	0
	SPECIAL	183	11	0	0	0	8.74317E-4	0
72	ROUTINE	619	9	0	0	0	3.23102E-4	0
	SPECIAL	98	14	1	0	0	2.24490E-3	0
73	ROUTINE	714	25	5	1	0	8.96359E-4	0
	SPECIAL	115	11	6	2	0	6.34783E-3	0
74	ROUTINE	475	8	4	0	0	6.52632E-4	0
	SPECIAL	44	15	2	0	0	6.59091E-3	0
75	ROUTINE	719	19	0	0	0	3.61613E-4	0
	SPECIAL	181	56	11	4	1	9.28177E-3	0.24
76	ROUTINE	321	14	0	0	0	4.67290E-4	0
	SPECIAL	185	23	2	1	1	2.75676E-3	0.2
77	ROUTINE	173	39	0	0	0	2.42775E-3	0
	SPECIAL	182	38	0	0	0	2.41758E-3	[illegible]
78	ROUTINE	367	22	1	0	0	1.00817E-3	0
	SPECIAL	170	52	6	3	0	5.05882E-3	0
79	ROUTINE	199	25	6	4	0	4.02010E-3	0
	SPECIAL	112	8	0	0	0	7.14286E-4	0
80	ROUTINE	497	22	2	1	1	1.28773E-3	[illegible]
	SPECIAL	142	9	0	0	0	9.85915E-4	0
81	ROUTINE	512	8	0	0	0	2.14844E-4	0
	SPECIAL	66	6	3	2	2	0.759242	[illegible]
82	ROUTINE	348	11	0	0	0	4.02299E-4	0
	SPECIAL	51	5	1	1	1	7.64706E-3	0.33
83	ROUTINE	291	6	0	0	0	3.43643E-4	0
	SPECIAL	27	1	0	0	0	7.40741E-4	0
84	ROUTINE	309	2	0	0	0	9.70874E-5	0
	~~SPECIAL~~	~~34~~	~~6~~	~~0~~	~~0~~	~~0~~	~~2.05882E-3~~	~~0~~
85	ROUTINE	530	11	1	1	0	3.96226E-4	0
	SPECIAL	22	0	0	0	0	0	0

? EOF in line 1700

Time: 385.88 secs.

Ready

Finding 4.5-5: Inappropriate Use of MDC Values for Deriving Missed Dose of Monitored Workers

Section 5.7.2 of TBD-5 contains the following guidance for dose reconstruction of "Monitored Workers with Nothing Detected in Bioassay:"

> *For monitored workers with no positive results, a triangular distribution is used, with the mode determined using half of the MDC value and the maximum using the MDC as input into the dosimetry codes.*

Table 5.1.1-1 of TBD-5 identifies that for 1954–1995, the MDC (as well as the "Minimum Recorded Level") for gross alpha counting was 10 dpm/liter (or 1 dpm/100 ml). In Section 5.1.2.1, (In Vitro Bioassay for Uranium) of TBD-5, however, the following statements are cited in behalf of gross alpha counting of urine samples:

> *… the alpha activity was measured on a proportional counter in α counts/min/100 ml urine {1 dpm/100 ml minimum detectable concentration (MDC)}. 100 ml of urine was needed for this technique. A result of 5 dpm/100 ml was considered the* ***reporting level*** *for the time-period of 1954–1993 (GAT, 1955; GAT, 1985a; Hill and Strom, 1993). Review of claim records reveals that* ***sometimes*** *values less than 5 dpm were recorded,* ***probably*** *down to the MDC of 1 dpm/100 ml.*

SC&A reviewed the above-cited reference (GAT 1985a), which verified the fact that for gross alpha counting of urine sampling, the **minimum recorded concentration/level** at 5 dpm/100 ml (or 50 dpm/liter) is 5 times higher than the MDC level of 1 dpm/100 ml (or 10 dpm/liter) (see Exhibit 4.5-5A).

SC&A concludes that TBD-5 contains contradictory/erroneous data, and guidance that instructs the use of MDC values instead of minimum recorded values is inappropriate and claimant unfavorable.

EXHIBIT 4.5-5A
(Source: GAT 1985a)

TABLE II

URINALYSIS ANALYTICAL PROCEDURES

Contaminant	Method	Reference	Minimum Sample Volume (ml)	Minimum Detectable Concentration (mg/l)	Minimum Recorded Concentration (mg/l)	Precision and Accuracy: Actual Concentration (mg/l)	Precision and Accuracy: Standard Deviation (mg/l)
Uranium (Total)	Fluorimetry	GAT-2-J-22-0	1	0.005	0.01	0.01 0.04 0.32	0.0009 0.0028 0.0130
Alpha (Total)	Ion-Exchange w/Subsequent Proportional Count	GAT-2-J-10-0	100	1 $\frac{dpm}{100\ ml}$	5 $\frac{dpm}{100\ ml}$	6 $\frac{dpm}{100\ ml}$ 12 $\frac{dpm}{100\ ml}$ 48 $\frac{dpm}{100\ ml}$	1.5 $\frac{dpm}{100\ ml}$ 3.5 $\frac{dpm}{100\ ml}$ 7.7 $\frac{dpm}{100\ ml}$
Technetium-99 (Total)	Liquid Scintillation w/ Proportional Count	GAT-2-G-16-2	5	1000 $\frac{dpm}{100\ ml}$	1000 $\frac{dpm}{100\ ml}$	5000 $\frac{dpm}{100\ ml}$	500 $\frac{dpm}{100\ ml}$
Fluoride (Total)	Selective Ion Electrode	GAT-2-J-12-1	40	0.1	0.1	0.9 4.5	0.027 0.190
Mercury (Total)	AAS Cold Vapor Method	GAT-2-J-14-0	10	0.01	0.01	0.05 0.29	0.0021 0.0107
Lead (Total)	AAS Furnace/ Atomization	GAT-2-J-18-1 (modified)	10	0.01	0.01	N/A	N/A
Chromium (Total)	AAS Furnace/ Atomization	BC-4	10	0.01	0.01	N/A	N/A
Nickel (Total)	AAS Furnace/ Atomization	GAT-2-J-18-1	10	0.005	0.02	0.005 0.060	0.0005 0.0018

Finding 4.5-6: MIVRML Chest Counts for the Detection of Insoluble Isotopes of Uranium, TRUs, and Fission Products are Subject to Significant Limitations and Uncertainties

Relevant Background Information

Fourteen years after startup, an in-vivo body counting program was initiated in 1965 to monitor for **insoluble enriched uranium**. Initially, until November 1967, all counting was done at the in-vivo facility of the Y-12 plant in Oak Ridge. For 1965 and 1966, only 27 and 30 chest counts, respectively, were performed on PORTS workers. Since November 1967, chest counts were performed by the Mobile In-Vivo Radiation Monitoring Laboratory (MIVRML). Although some PORTS workers were provided chest counts at the MIVRML while the mobile unit was at the National Lead of Ohio or FMPC facility, most chest counts of PORTS workers were done at the plant during the semi-annual visits.

The MIVRML employed two very large sodium iodide crystal detectors measuring 9 inches in diameter by **4 inches thick.** While such large/thick crystals offer maximum counting efficiency that is highly desirable for whole-body counting of high-energy photons of fission/activation products (e.g., Cs-137, Co-60), their use in detecting low-energy photons is severely compromised by their 4-inch thickness. This limitation in sensitivity is due to the fact that for large/thick crystals, the low-energy photons fall into that region of Compton-scattered photons that is maximal.

For the MIVRML, detection of **U-235** was based on the 186 keV photon, while detection of **U-238** relied on the 63 keV and 93 keV photons emitted by the Th-234 daughter product. Important to note is that these two photons also have low yields only 3.5% and 4%, respectively. The detection of Np-237 was based on multiple photon emissions (Np-237 and its radioactive daughter Pa-233); and detection of Tc-99 (a pure beta emitter) was based on the production of Brehmstrahlung. This system was used until 1991, when it was replaced by the Helgesson Phoswich detector system. With its 0.5-inch thick crystals, detection limits significantly improved. However, by this time, the potential for exposure to insoluble uranium compounds that largely involved Buildings X-705 and Building X-705E had been reduced/ eliminated.

In May 2000, DOE's Office of Oversight issued a report entitled *Independent Investigation of the Portsmouth Gaseous Diffusion Plant*. In behalf of in-vivo chest counting, the report stated the following:

Page 4:

> *… In 1965, an in-vivo body counting program was initiated to monitor for insoluble enriched uranium, a material for which the urinalysis program was not sufficiently sensitive or reliable. Studies performed in 1990 indicated that the in-vivo counter's capability for analyzing transuranics was* ***questionable****, making it difficult to demonstrate that all internal exposures have been accurately detected and assessed.*

Pages 38–39:

> *In a report prepared for Martin Marietta Utility Services in 1990, the effectiveness of the mobile whole body counter was evaluated for analysis of uranium, neptunium, plutonium, and americium. Additionally, a review was conducted of historical lung counting data from Martin Marietta Utility Services sites, with particular emphasis on neptunium-237. A summary of the findings indicated that the counter's capability for analysis of those radionuclides, with the* ***exception of uranium-235****, was* ***somewhat questionable due to system hardware limitations (i.e., use of sodium iodide detectors, resolution of spectra insufficient to identify peaks in the presence of background radiation****,* ***efficiency calibrations did not use multiple source*** *strength measurements for isotopes other than uranium-235). The studies of historical data indicated difficulties, including the inability to retrieve the appropriate data, lack of system access, and insufficient documentation.* ***The root cause for most of the problems could be attributed to physical limitations of the system, lack of understanding of these limitations, and the lack of adequate training.*** *Incomplete isotopic and uranium solubility characterization, coupled with design and analytical limitations, has impacted the Plant's ability to demonstrate that all internal exposures have been accurately detected and assessed.* [Emphasis added.]

SC&A not only agrees with these conclusions that place root cause on the physical limitations of the MIVRML, but has previously voiced these same concerns in behalf of other DOE facilities that provided chest counting by means of the MIVRML (see draft report, *Review of the Feed Materials Production Center (FMPC) Special Exposure Cohort (SEC) Petition-00046 and the NIOSH SEC Petition Evaluation Report* (SCA 2007).

Furthermore, the system's failure to accurately assess U-238 renders the only "reliable" measurements of U-235 useless for the following reasons:

(1) In the absence of an accurate U-238 measurement, the activity of U-235 provides no information regarding the enrichment of uranium and, therefore, the variable activity and dose contributions of **U-234** and **U-238**.

(2) For even modestly enriched uranium, it is U-234 that dominates activity (and therefore dose), as shown for the following levels of enrichment:

U-235 Enrichment	U-234	% Activity U-235	U-238
Depleted U	15.5	1.1	83.4
U Natural	48.9	2.3	48.9
2.0%	64.8	4.1	31.1
3.5%	81.8	3.4	14.7
93%	96.8	3	0.03

Exhibit 4.5-6A provides annual summary in vivo data in behalf of all monitored workers for the years 1965 through 1985. A visual inspection that compares columns #3 and #4 clearly demonstrates the questionable values of "total U results" relative to U-235 measurements (GAT 1986a).

In conclusion, SC&A regards in-vivo bioassay data of limited value.

EXHIBIT 4.5-6A

3

ANNUAL AVERAGES OF IN-VIVO RESULTS FOR PLANTSITE

YEAR	NUMBER OF COUNTS	AVERAGE U-235 RESULT IN UGM	AVERAGE TOTAL U RESULT IN MGM	AVERAGE TC-99 RESULT IN UCI	AVERAGE PERCENT OF U-235 RPG (240 UGM)	AVERAGE PERCENT OF TOTAL U RPG (27 MGM)	AVERAGE PERCENT OF TC-99 RPG (9 UCI)	AVERAGE PERCENT OF COMBINED RPG
65	27	320	0	0	133.333	0	0	133.333
66	30	117.2	0	0	48.8333	0	0	48.8333
67	236	31.0339	0	0	12.9308	0	0	12.9308
68	364	31.3407	0.922747	0	13.0586	3.41758	0	16.4762
69	393	60.4478	2.10532	0	25.1866	7.79747	0	32.9841
70	147	66.4898	4.36803	0	27.7041	16.1779	0	43.882
71	179	78.5698	1.61402	0	32.7374	5.97786	0	38.7153
72	157	80.7516	4.37745	0	33.6465	16.2128	0	49.8593
73	392	25.5102	-1.66974	0	10.6293	0	0	10.6293
74	521	52.4645	-0.854415	0	21.8602	0	0	21.8602
75	684	52.5175	-0.711901	0	21.8823	0	0	21.8823
76	411	51.382	-1.00455	0	21.4092	0	0	21.4092
77	971	32.4851	-0.819773	2.82904E-2	13.5354	0	0.314338	13.8498
78	542	40.679	0.464631	-0.686458	16.9496	1.72086	0	18.6704
79	497	31.66	2.50374	0.355714	13.1917	9.27312	3.95238	26.4172
80	924	32.3918	1.6326	0.118323	13.4966	6.04666	1.31469	20.8579
81	868	7.58986	-0.800622	9.42972E-2	3.16244	0	1.04775	4.21019
82	910	8.82967	-0.841582	0.100242	3.67903	0	1.1138	4.79283
83	632	4.11867	-1.01253	0.115427	1.71611	0	1.28252	2.99864
84	613	8.89396	-0.622463	0.144861	3.70582	0	1.60957	5.31539
85	798	6.70927	-0.291704	0.180476	2.79553	0	2.00529	4.80082

TIME: 36.14 SECS.

4.6 REVIEW OF TBD-6 (ORAUT-TKBS-0015-6) PORTSMOUTH GASEOUS DIFFUSION PLANT – OCCUPATIONAL EXTERNAL DOSE

Relevant Summary Data

Between 1954 and 1980, exposure to external photon and beta radiation at PORTS was performed by means of a 2-element film dosimeter—an open window and a shielded element. This film badge dosimeter contained Kodak Type-2 film with two emulsions for normal conditions. The more sensitive emulsion responded to doses between 30 mR and 2,000 mR, while the less sensitive emulsion had a range of 5 R to 300 R and was reserved for accidental/criticality exposures. Important to note was that the film dosimeters were processed in-house.

Starting in 1981, PORTS replaced the film dosimeters with a series of TLDs that included a 4-element Harshaw TLD, a 4-element Panasonic TLD, and a 4-element International Chemical and Nuclear Corporation (ICN) TLD.

Personnel monitoring for neutron exposures at PORTS did not begin until 1992, and routine personnel monitoring for neutrons did not occur until 1997. Personnel neutron dosimeters employed Li-6 phosphor in a design that measured albedo neutrons.

Specific dates for use of individual dosimeters, their exchange frequency, limits of detection, and selective use are summarized in Table 6-3 of TBD-6. For convenience, Table 6-3 is reproduced herein as Exhibit 4.6-1A.

A substantial portion of TBD-6 is committed to supportive background information regarding dosimeter performance specifications/uncertainties, gaps in monitoring data, and deficiencies in monitoring practices. Data gaps and monitoring deficiencies acknowledged in the TBD include the following:

(1) Prior to 1960, select persons/groups with the highest exposure potential were monitored; this implies that other persons (albeit with lower exposure potential) were not monitored.

(2) The presence of substantial quantities of beta emitters (e.g., Tc-99, Th-234, and Pa-234) have the potential for high external shallow doses/skin doses, as well as from skin contamination; however, extremity monitoring was not performed.

(3) Neutron exposures may have occurred throughout the years of PORTS operations, but routine monitoring was not provided until the 1990s.

In summary, these data gaps and monitoring deficiencies fall into the categories of "missed and unmonitored" dose and were addressed in TBD-6 in the following subsections:

- Section 6.5.1: Missed and Unmonitored Shallow Dose
- Section 6.5.2: Missed and Unmonitored Photon Dose
- Section 6.5.3: Missed and Unmonitored Neutron Dose

EXHIBIT 4.6-1A

Table 6-3. PORTS Dosimeter Type, Period of Use, Exchange Frequency, LOD, and Potential Annual Missed DE (rem)

Dosimeter	Period of Use	Exchange Frequency	Laboratory LOD	Maximum Annual Missed Dose[a]
Beta/photon dosimeters				
PORTS film 2-element[b]	9/22/54–7/16/57	Weekly (n=52) {selected groups}	0.03[c]	0.78
	7/17/57–9/30/58	Biweekly (n=26) {selected groups}	0.03[c]	0.39
	10/01/58–4/8/59	Weekly (n=52) {chemical operators and material handlers}	0.03[c]	0.78
		Monthly (n=12) {remainder of selected groups}	0.03[c]	0.18
	4/9/59–7/31/60	Every 4 wk (n=13) {all selected groups}	0.03[c]	0.195
	8/1/60–7/5/64	Monthly (n=12) {all selected groups}	0.03[c]	0.18
		Quarterly (n=4) {all other employees}	0.03[c]	0.06
	7/6/64–12/28/69	Quarterly (n=4) {all employees}	0.03[c]	0.06
	12/29/69–12/30/73	Quarterly (n=4) {selected employees}	0.03[c]	0.06
	12/31/73–6/29/75	Quarterly (n=4) {selected employees}	0.03[c]	0.06
		Semiannual (n=2) {unselected employees}	0.03[c]	0.06
	6/30/75–12/31/80	Quarterly (n=4) {selected employees}	0.03[c]	0.06
		Monthly (n=12) {selected female employees only}	0.03[c]	0.18
PORTS Harshaw 2276 4-element TLD without window	1/1/81–12/31/82	Monthly (n=12) {all monitored} Quarterly (n=4) {all monitored}	0.015[d] 0.015	0.09 0.03
PORTS Harshaw 2276, 8000, 8800 4-element TLD with window	1/1/83–12/31/98 {1/1/93–12/31/96 for BJC employees}	Quarterly (n=4)	0.010[e](0.04 SDE)	0.04 (0.08 SDE)
ICN TLD 760	1/1/99–present {USEC employees}	Quarterly (n=4)	0.01[f](0.03 SDE)[f]	0.02 (0.06 SDE)
ORNL Panasonic 8805/8806 4-element TLD with window	1/1/99–present {BJC employees}	Quarterly (n=4)	0.01[h](.03 SDE)[h]	0.02 (0.06 SDE)
Neutron dosimeters				
PORTS TLD albedo dosimeter {USEC and BJC}	1/1/1992–12/31/94 {unmoderated Cf-252 calibrated}	Quarterly (n=4)	0.02[i]	0.04
	1/1/95–12/31/96 {moderated Cf-252 calibrated}	Quarterly (n=4)	0.02[i]	0.04
ICN TLD 760 {USEC}	1/1/97–present {moderated Cf-252 calibrated}	Quarterly (n=4)	0.01[f]	0.02
Y-12 Panasonic TLND {BJC employees}	1/1/97–12/31/98	Quarterly (n=4)	0.01[g]	0.02
ORNL Panasonic TLND 8806 4-element TLD {BJC employees}	1/1/1999–present	Quarterly (n=4)	0.01[h]	0.02

a. Maximum annual missed dose (NIOSH 2002): For photon/beta missed dose = LOD/2 × n (frequency (p. 18); for neutron missed dose = LOD/2 × n(frequency) (p. 29).
b. Kodak personnel Type 2 film with gold sandwiched with cadmium for high-energy gamma, OW with aluminum for low-energy gamma and beta. LOD for SDE and DDE are the same (0.03 rem), as the reporting level.
. GAT (1963a), 30 mrem reporting level for gamma and beta.
d. (Wagner 2003).
e. Bassett (1986a, p. 3).
f. ICN (2003).
g. Souleyrette (2003).
h. McMahon (2003).
i. Cardarelli (1997).

Missed and Unmonitored Shallow Dose

For **missed** shallow dose (SDE), Section 6.5.1 of the TBD recommends the use of *n*(LOD/2) and resultant values shown in Table 6-29. For the 2-element film dosimeter, the LOD for shallow dose of 30 mrem is identified.

For **unmonitored** shallow dose, the TBD states that ". . . the dose reconstructor should consult ORAUT-OTIB-0040 for instructions." The TBD further states that the **empirical** PORTS SDE data provided in Table 6-31 of the TBD ". . . are for informational purposes only."

Finding 4.6-1: The Assumed LOD Value for SDE Associated with the Two-Element Film Dosimeter used Between 1954 and 1980 Lacks Technical Support and is Not Claimant Favorable

The following information is provided in Section 6.3.2.1.1 of TBD-6:

> *The film dosimetry program began in 1954. The dosimeter description from the documentation obtained is **cryptic**. A **Description of Co-Operative Work Assignments in Industrial Hygiene and Health Physics** (Wooldridge 1964) describes the dosimeter as a "film badge with Kodak Type-2 personal monitoring film combined with aluminum, cadmium, and gold filters for beta-gamma, low energy gamma, and high energy gamma radiation. There are also sulfur and gold filters for neutron exposures." This document indicates a **detection range** from 30 to 2,000 mrem…* [Emphasis added.]

Exhibit 4.6-1B identifies the source of information used by NIOSH to base their decision to employ the LOD value of 30 mrem for the shallow dose.

SC&A interprets the statement in Exhibit 4.6-1B ". . . The present film will detect from approximately 30 millirem…" to mean that 30 mrem was the minimum reportable dose (as opposed to LOD). Moreover, the **detection** level of 30 milliroentgen is likely to refer to the deep dose equivalent (DDE). In fact, NIOSH shares this interpretation as indicated in footnote b of Table 6.3 shown in Exhibit 4.6-1A, where NIOSH states the following:

> *Kodak personnel type 2 film with gold sandwiched with cadmium for high-energy gamma, **OW with aluminum for low-energy gamma and beta**. LOD for SDE and DDE are the **same** (0.03 rem), as the **reporting level**.* [Emphasis added.]

EXHIBIT 4.6-1B

06 FR34 067 265

PROCEDURE FOR EXTENDING FILM BADGE RANGE

Limits of the film badge dosimetric program are established by (1) the type of film used, and (2) the sensitivity of the densitometer. The film is Kodak Type-2 personal monitoring film. This contains two emulsions: a slow emulsion indicated by a highly polished surface or an indenture in one corner, and a fast emulsion indicated by a dull finish or a protuberance in one corner. The Weston densitometer has a response to transmitted light up to a density of three. (The number three is derived from a negative logarithmic ratio of the light transmission intensity of the film before and after exposure).

The present film will detect from approximately 30 milliroentgens to 2 roentgens gamma or X radiation under normal conditions. During an emergency, though, it is possible to have much more intense exposures. Removal of the fast emulsion enables evaluation with proper calibration of exposures from 5 roentgens to 300 roentgens gamma or X radiation. A blank film should be prepared for calibration purposes. There are three standard methods for removing the fast emulsion:

a. Coat slow emulsion with grease and apply Farmer's Reducer to fast emulsion for 30 minutes and scrape. Wash in running water for 10 minutes and allow to dry for 1-1/2 hours.

b. Apply a small quantity of Kodak Photo-Flo Solution and scrape.

c. Heat water to 164° F. Suspend film in bath for 4 minutes. (Air bubbles will appear on the surface and should be removed). Place film on smooth surface covered with water. A small lucite block has been used and is found to work well. The layer of water should be

SC&A disagrees with NIOSH's unsupported assumptions regarding the LOD value of 30 mrem for the following reasons:

(1) No documentation or empirical data exist that defines the SDE LOD value for the PORTS 2-element film dosimeter.

(2) The minimum reporting dose (MRD) is not equivalent to the LOD.

(3) Footnote b of Table 6-3 of the PORTS TBD-6 (see Exhibit 4.6-1A) makes reference to "…**OW** with **aluminum** for low-energy gamma and beta," which suggests that the open window (OW) of the 2-element dosimeter may have used an aluminum filter, which would limit its sensitivity to the low-energy (i.e., 294 keV Tc-99) betas.

(4) Barring the presence of an aluminum filter, the standard Kodak Type-2 film dosimeter used at other DOE facilities is consistently characterized as having an SDE LOD value of 40 mrem.

(5) The ability to assess shallow dose from beta radiation may have further been compromised by a practice of wearing the badge "inside the anti-Cs," as suggested in handwritten notes by a cascade process operator in 1986 that coincided with the CIP/CUP program. Enclosed as Exhibit 4.6-1C are pages 1, 2, 5, and 6 of the handwritten notes. (Note: Select passages are highlighted by arrows on the left-hand margin.)

EXHIBIT 4.6-1C (Handwritten Notes - Page 1)

APPROVED FOR RELEASE
H. H. Thomas
Date: 5/30/07

(1)

ACR Control Operator (Process Opr) 6/1986

X-333 (3 yrs), X-330 Cascades
Low
Intermediate (~6% at top stages)
0.716% → Normal Feed (NU)
1.1 Range - PP Product - Paducah feed

LAW - Low Assay Withdrawal
Tails withdrawal Bldg.

Steady progression in safety improvement - as a result of the NRC

- Treatment Phase → DOE had shut the site down → Deposit Remediation

USEC
- 5-year Cold Standby Project - Buffering of the cells → dry air or nitrogen. Maintain cells in an operating state.

333 size equipment @ Paducah
Agreement w/ the Russians → providing the feed material for Paducah.

710 Mass Spec Technician - worked with all the enrichments.
U-234, U-235, U-236 & U-238 → ~~A~~ Actually

EXHIBIT 4.6-1C (Handwritten Notes - Page 2)

(2)

analyzed UF_5 (state)

Additional Isotopes:
U-234, U-236 (Don't want these)

Ford Administration - Reactor Material was feed into the cascades.
Tc-99
IHHP - Small organization
Certain place
2904-2906 → Tc concern in this area. Cut a converter out.
Column B2 → stored at this location

[redacted] (Bldg Supv) → 330, 333
Coordinator for stored material.

Minor releases → backed away from the product /no evacuation. - Occurred on a weekly basis. (frequently)
More of a chemical processing plant, People got complacent.

Badges were historical worn inside the Anti-C's.

Routine Readings → efficiency of the separative material. when cell was not

EXHIBIT 4.6-1C (Handwritten Notes - Page 5)

5

be larger release. – Supposed to have a respirator on. Full mask more for escape. Protection from intial release. You were supposed to flee. When he first got to Ports → Prior to ~~flee~~ fleeing mitigate the leak.

Three converters in the 330 Bldg. On a boundaried area (general RWP) → Brushed up against a ~~boundary~~ tygon piece. Changed the tygon tubing.
Pancake GM → for frisking from boundared area.

24-hour samples are taken for special bioassay.
Don't remember the kinds of Rad.

The cascades were ~~very~~ hot (440°F).
Turbo frisk due to the heat.

No optional upgrade of respiratory protection.

Significant → skin contamination event w/ Tc.
Told that things were okay by some FLMs.

EXHIBIT 4.6-1C (Handwritten Notes - Page 6)

6

Maintenance / Operations

Maintenance Mechanic → During the CIP/CUP program – ~~tut~~ Mechanical Work

X-333 feed here → series of cells (80) in this Bldg. 2

X-330 – Series of cells (~108) in this Bldg.

X-326 – High Enrichment Bldg 97% for the Nuclear Navy.

Each cell has 10 stages ~~5 Con~~ (10 x 80 = 800 Converters)

Tore all equipment out of cascade → converters, coolers, control valves, compressors, motors, etc.
↳ Rewound

~~Rebuilt this~~ – retube & take out all the insides.

Significant increase in contamination. UF6 in its solid form. There would be an outgassing from the converters while on the cranes.

Seal Changes → 1-2 everyday. Material was on these. (Rad)

Finding 4.6-2: Unmonitored SDE Estimates Derived from Co-Worker Data Suffer Deficiencies that are likely the Result of Dosimeter Design Limitations and/or Dosimeter Process Policies

ORAUT-OTIB-0040 (External Coworker Dosimetry Data for the Portsmouth Gaseous Diffusion Plant) is to be used for assignment of unmonitored dose. Key to the assignment of unmonitored dose is the use of Portsmouth monitored coworker data for which 50th and 95th percentile dose estimates have been derived on a annual basis. In addition to **recorded** dosimeter doses, coworker doses were modified to account for **missed** doses (assumedly all individual doses recorded as zero were replaced by LOD/w or 30 mrem/2 or 15 mrem for both SDE and DDE. Table 8-2 of OTIB-0040 provides the adjusted Portsmouth coworker dose data and is herein reproduced as Exhibit 4.6-2A.

Inspection of Exhibit 4.6-2A shows that with few exceptions, the **non**-pentrating (or beta) component that would be expected to have registered on the OW portion of the film dosimeter was on **average** essentially zero (or nonexistent). Due to the presence of substantial amounts of Tc-99, as well as several radioactive daughter products of uranium that moreover concentrated, the near absence of measurable beta radiation up until 1980 is unexpected and must, therefore, be viewed with some suspicion. Possible explanations may include issues raised in Finding 4.6-1.

Another issue of concern that affects the ability to account for both missed and unmonitored exposures involves a statement contained in the 2000 report issued by the Bechtel Jacobs Company LLC (BJC 2000). On page 68 of the report entitled *Recycled Uranium Mass Balance Project Portsmouth, Ohio Site Report*, the following statement appears:

> *Worker monitoring began in 1954 with the Film Badge and Bioassay Programs. Workers with the potential for external radiation were provided film badges for monitoring. However, not all workers were provided film badges, and **not all badges issued to workers were read**. This changed in the mid-70s when the film badges were replaced with TLD badges. . . . Some badges are not read unless there is cause to believe a significant dose may have been recorded.* [Emphasis added.]

Page 68 of the BJC (2000) report is enclosed as Exhibit 4.6-2B. It should be noted that this exhibit also provides a relevant discussion about Tc-99.

EXHIBIT 4.6-2A

Year	Gamma 95th%	Gamma 50th%	Non-pen 95th%	Non-pen 50th%
1954	1.736	0.780	0.055	0.000
1955	1.104	0.874	0.048	0.000
1956	0.945	0.799	0.016	0.000
1957	0.714	0.615	0.335	0.000
1958	0.857	0.574	0.434	0.031
1959	1.164	0.591	0.579	0.040
1960	0.283	0.195	0.123	0.000
1961	0.240	0.180	0.086	0.000
1962	0.332	0.180	0.094	0.000
1963	0.360	0.180	0.051	0.000
1964	0.262	0.120	0.021	0.000
1965	0.140	0.060	0.050	0.000
1966	0.136	0.060	0.010	0.000
1967	0.122	0.060	0.005	0.000
1968	0.338	0.060	0.007	0.000
1969	0.281	0.085	0.011	0.000
1970	0.459	0.158	0.047	0.005
1971	0.281	0.060	0.065	0.000
1972	0.367	0.078	0.173	0.000
1973	0.407	0.077	0.087	0.000
1974	0.337	0.060	0.033	0.000
1975	0.474	0.078	0.114	0.000
1976	0.415	0.060	0.050	0.000
1977	0.365	0.078	0.050	0.000
1978	0.414	0.077	0.223	0.016
1979	0.181	0.060	0.174	0.000
1980	0.307	0.060	0.001	0.000
1981	0.120	0.090	N/A[1]	N/A1
1982	0.112	0.090	N/A[1]	N/A1
1983	0.053	0.020	0.112	0.060
1984	0.053	0.020	0.080	0.060
1985	0.045	0.020	0.083	0.060
1986	0.058	0.020	0.082	0.053
1987	0.063	0.020	0.083	0.046
1988	0.057	0.020	0.087	0.060
1989	0.063	0.020	0.091	0.047
1990	0.075	0.022	0.098	0.055
1991	0.057	0.020	0.101	0.050
1992	0.054	0.020	0.109	0.060

EXHIBIT 4.6-2B

The amount of Np in inventory annually is estimated to peak at 76.01g in FY 1973, with 44.3g Np in inventory as of March 31, 1999.. The estimated annual Np in inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.4 Annual Mass Balance of Technetium in Recycled Uranium

This section discusses the annual mass balance for ^{99}Tc that was introduced to the PORTS site with either RU or ^{99}Tc-contaminated PPF. This is not an annual mass balance of ^{99}Tc in RU (See rationale for Pu mass balance in section 5.2).

The ^{99}Tc mass balance is developed using the same campaigns discussed earlier for Pu and Np to model the constituent movement after arrival at PORTS.

^{99}Tc was first introduced at PORTS in FY 1955 with feed manufactured by Paducah from HRT/SRT oxide and Paducah or Oak Ridge product feed. Upon receipt, the material was fed and the cylinders returned to Paducah or Oak Ridge. Beginning in FY 1968, cylinders were sometimes held for a period of time before feeding. Any ^{99}Tc contained in the cylinders that were stored is included in the year-end inventory. During periods when a cylinder is fed and returned to Paducah, the ^{99}Tc in the cylinder is not included in the year-end inventory.

During cascade feeding, it is estimated that 90% of the ^{99}Tc enters the cascade with 10% remaining in the cylinder. The ^{99}Tc that enters the cascade initially absorbs on the metal surfaces as it moves up the cascade. While ^{99}Tc is highly mobile and moves quickly to the top of the cascade once equilibrium has been established, it was not unequivocally identified until 1974. This 19-year lag is assumed to be at least in part due to the time it took the ^{99}Tc to reach equilibrium (Ref. 19). Once at equilibrium, additional ^{99}Tc in the feed rapidly traveled from the feed point to the top of the cascade. The migration of ^{99}Tc in the cascade was slowed by the equipment change-out in FY 1958 - FY 1960 when much of the equipment contaminated with ^{99}Tc was removed and decontaminated

The process equipment was decontaminated with essentially all of the ^{99}Tc going into solution through uranium recovery. The oxide produced was stored for oxide conversion at a later date. All of the ^{99}Tc processed through uranium recovery is assumed to end up at the X-701B.

The model (Campaigns 2 & 3) includes ^{99}Tc releases to the environment as identified in Table 2.5-1.

The amount of ^{99}Tc in inventory annually is estimated to peak at 65.26 kg in FY 1975. The ^{99}Tc in inventory as of March 31, 1999 is estimated to be 35.11 kg. The estimated annual ^{99}Tc in inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.5 Potential for Worker Exposure from Recycled Uranium

Worker monitoring began in 1954 with the Film Badge and Bioassay Programs. Workers with the potential for external radiation exposure were provided film badges for monitoring. However, not all workers were provided film badges, and not all badges issued to workers were read. This changed in the mid 70's when the film badges were replaced with TLD badges. All workers, regardless of exposure potential since that time, have been provided TLD badges. Some badges are not read unless there is cause to believe a significant dose may have been recorded. Records of badge readings obtained since 1954 are retained by USEC.

The bioassay program began with urine sampling for uranium or gross alpha. Uranium sampling was used to monitor intake of workers with the potential for exposure to low assay soluble uranium. Workers with the potential for exposure to high assay uranium were monitored by gross alpha. In the mid-1990s, both methods were replaced with Inductively Coupled Plasma/Mass Spectroscopy (ICP/MS) methods. Results of urine bioassay monitoring since 1954 are retained by USEC.

Finding 4.6-3: External Exposures to Localized Skin and to Extremities were Inadequately Monitored at PORTS, and Guidance to Dose Reconstructors is too Subjective and Arbitrary

Numerous activities at PORTS subjected workers to external radiation fields in which extremities (i.e., hands and feet) would have received exposures that were substantially higher than shallow doses recorded by a personnel dosimeter worn on the anterior portion of the chest. Activities with potential for high extremity doses would have included those involving oxide conversion, ash handling, routine maintenance and CIP/CUP efforts, and equipment decontamination. A substantial component of extremity doses would have been contributed by beta radiation, as noted in DOE 2000 (page 36):

> *Routine whole body beta exposures in excess of PORTS investigative levels existed primarily in areas where uranium daughter products tended to concentrate. …Exposure evaluations during the mid to late 1950s indicated numerous instances of workers being placed on work restrictions based on whole body exposures that were determined to be in excess of PALs* [Plant Allowable Limits]. *Documents also indicated that before the mid-1980s, Goodyear Atomic Corporation* ***had never performed extremity monitoring for any operation or work activity****. Documents indicated that various valves associated with pigtail operations had recorded beta readings as high as 1 rad/hour. Feed production plant ash receiver areas had* ***floor*** *readings of 5 rad/hour beta. Operators routinely handled these valves and equipment in X-705 and other locations where significant hand exposures could occur.* [Emphasis added]

In addition to unmonitored extremity exposures from external sources, unrecorded skin and extremity exposures would also have resulted from personnel contamination involving bare skin, as well as from clothing. (It should be noted that select radionuclides, inclusive of Tc-99, are not easily removed from the skin by ordinary washing. Failure to detect localized contamination would, therefore, yield long residence half-times and with time, produce substantial localized skin doses.)

To accommodate extremity monitoring deficiencies and potential exposures resulting from skin contamination, TBD-6 (pages 41–43) provides several hypothetical "examples," including source term models.

SC&A believes that for dose reconstruction of skin/extremities, the current guidance is too subjective and arbitrary.

Finding 4.6-4: Before 1992, PORTS failed to monitor workers for neutron exposures. Current guidance to account for unmonitored neutron exposures is incomplete.

Based on the large quantities of uranium compounds at PORTS, it is axiomatic that neutron exposures have existed throughout PORTS as a result of spontaneous fission and, more importantly, from the alpha/neutron reactions of UF_6. Neutron exposure rates were likely highest in areas were uranium was stored (i.e., cylinder yards), handled (i.e., feed and withdrawal areas), or deposited (i.e., within the cascade). Important variables affecting neutron production/ exposure include the level of uranium enrichment, geometry of the source term, and neutron moderation.

Concern for neutron exposures associated with unplanned accumulation of fissile material sufficient to initiate criticality was recognized from the beginning. On October 5, 1954, Goodyear Atomic Corporation issued a report entitled, *Significant Incident Report: Condensation of UF_6 in the Cascade during Initial Start-Up Operations* (GAT 1954). This report noted that on September 11, 1954, conditions were favorable to condensation of UF_6 gas in surge drums that were used as a reservoir during startup operations. The report also noted, "…that one of the chief critical mass hazards in a gaseous diffusion plant is the unplanned accumulation of a mass of fissionable material sufficient to initiate a critical reaction."

In a 1957 Committee Report (FR-434-06-0035 Box-1), a total of 105 "false alarms" were recorded in a single year by 29 criticality instruments at PORTS. At PORTS, criticality instruments were **Argon Gamma Graphs**, which respond to transient spikes in gamma radiation and, therefore, only indirectly record potential criticality events. Furthermore, these instruments had five different alarm settings. An investigation of causes for the 105 alarms provided answers to 82 false alarms. However, for the remaining 23 criticality alarms in 1957, no plausible explanation could be identified. **SC&A believes that near-criticality conditions cannot be ruled out as a cause.**

Sub-critical accumulations of UF_6 were routinely observed at PORTS. For example, in a September 9, 1990, report entitled, *Significant Non-Critical Incident No. 49: Deposit of Uranium Material in Cells X-27-3-9 and X-27-3-11* (GAT 1990), the following quantities of accumulated materials were discovered by means of portable Ludlum gamma spectrometer, as given in the following reported findings:

- *… the deposits in X-27-3-9 in the upper part of the cell were reduced from **22** pounds to 12 pounds UO_2F_2 in the compressor and from **38** pounds to less than 13 pounds UO_2F_2 in the coller. At an assay of **15 percent U-235** (assumed to be the assay of the deposit) **the ordinary safe mass is 10 pounds UO_2F_2 and the minimum critical mass is 27 pounds UO_2F_2**.* [Emphasis added.]

- *The deposit in the X-27-3-11 converter was reduced from 30 pounds UO_2F_2 to less than 2 pounds UO_2F_2 ….*

Of relevance here is the fact that dosimetry programs at PORTS from 1954 to 1992 failed to monitor workers by means of calibrated personnel neutron dosimeters. Reference to the term "calibrated" requires additional explanation. When film dosimeters were replaced by a 4-element TLD in 1981, three of the chips consisted of Li-7 (which responds to betas and photons), while the 4th chip of Li-6 responds to betas, photons, and neutrons. However, this 4-element TLD was not calibrated or evaluated for neutron exposures. Routine processing of these personnel dosimeters for beta/gamma exposures in some instances yielded highly abnormal chip ratios that either reflected a damaged dosimeter or neutron exposures as high as 4.8 rem (DOE 2000).

Unfortunately, these abnormal/inexplicable chip ratios were dismissed by PORTS as damaged chips. The combined failure to monitor workers for neutrons and dismissal of unexplained chip ratios resulted in a 1996 request for NIOSH to conduct a Health Hazard Evaluation (HHE) at PORTS. The principal objectives of the evaluation were two-fold; (1) assess historical neutron exposures at PORTS, and (2) resolve the anomalous chip ratio values. In 1997, NIOSH released its HHE (Cardarelli 1997). With regard to Objective #2, the HHE provided the following explanation:

<u>From page 7 on the nature of slow cookers</u>:

> *A slow buildup of uranium material within the cascade causes a slight increase in the production of neutrons . . . This phenomenon of a slow build-up of uranium material that* ***approaches criticality*** *has been termed by the gaseous diffusion industry as a "slow cooker." In essence, a* ***slow cooker*** *is a mass of uranium in which there is a multiplication of neutrons but at a rate below the critical threshold.* ***Slow cookers*** *are directly associated with uranium deposits which have* ***routinely*** *occurred at the PORTS since the plant's* ***inception****. . . .* [Emphasis added.]

Neutron doses specifically attributable to this phenomenon could **not** be addressed in this evaluation for the following reasons:

<u>From page 11 on the inability to retrieve past data</u>.

> *. . . regarding abnormal chip ratios, a request was made to obtain historical computerized data containing chip readings from 1981 to the present. These were to be viewed to determine if certain worker groups had higher rates of abnormal ratios than other workers groups. This would provide additional information about the existence and relative degree of past neutron exposures to the workforce. However, problems were encountered in retrieving the data. First, the storage format and type of dosimetry reading equipment changed through time and would have required substantial time and resources to recover. Secondly, the older dosimeter equipment (Harshaw 2276) was a direct-reading machine and did not create a date field when the results were stored. Finally, many archive tapes had been reused, and most of the historical data was overwritten with more recent data. Consequently, data available for this purpose was limited to only the most recent measurements (1992–1995).* ***Thus, it was not feasible to reconstruct neutron exposures from previous TLD chip readings before 1992****.* [Emphasis added.]

In summary, NIOSH's HHE of the PORTS facility concluded the following:

- The observed anomalous chip ratios were not the result of defective TLDs, but of neutron exposures to slow cookers
- Uranium deposits that produced slow cookers existed since the startup of PORTS
- Although TLD badges used since 1981 had the potential for assessing neutron exposures, the dosimeters were neither calibrated nor processed to assess neutron dose
- For the time period of 1981 through 1992, historical dosimetry records of anomalous chip ratios are either irretrievable or no longer exist

Although the current technical basis document (ORAUT-TKBS-0015-6) acknowledges the HHE findings (and recommends use of the study's area neutron dose rates and neutron-to-photon ratios for dose reconstruction), the TBD ignores the issue of past unaccounted neutron doses associated with slow cookers.

5.0 WORKER INTERVIEWS, DATA COMPLETENESS, AND DATA INTEGRITY

The PORTS Site Profile does not adequately address outstanding concerns that have been raised regarding the completeness and integrity of dosimetry records representing time periods within and outside the SEC.

5.1 RELEVANT BACKGROUND INFORMATION

In addressing questions relating to the credibility and adequacy of monitoring data for use in dose reconstruction, it is important to recognize that data quality and data completeness are subjective terms. The PORTS facility has operated for more than 50 years, during which time industry-wide standards, monitoring practices, and technologies have steadily changed/improved.

Regulations defined in behalf of EEOICPA fully anticipated some limitations and deficiencies regarding past methods for monitoring workers, the availability of monitoring records, etc. To compensate for these deficiencies, regulatory guidance defined in 42 CFR 82 specifies the use of surrogate data in dose reconstruction on the condition that such data is technically defensible and claimant favorable.

The ability to reconstruct doses may, nevertheless, be challenged if select conditions exist that are defined in §83.9 of 42 CFR 83. Conditions that may preclude a credible dose reconstruction are the following:

> *(i) Documentation or statements provided by affidavit indicating that radiation exposures and doses to members of the proposed class were not monitored, either through personal or area monitoring; or*
>
> *(ii) Documentation or statements provided by affidavit indicating that radiation monitoring records for members of the proposed class have been lost, falsified, or destroyed; or*
>
> *(iii) A report from a health physicist or other individual with expertise in dose reconstruction documenting the limitations of existing DOE or AWE records on radiation exposures at the facility, as relevant to the petition. This report should specify the basis for believing these documented limitations might prevent the completion of dose reconstructions for members of the class under 42 CFR Part 82 and related NIOSH technical implementation guidelines; or*
>
> *(iv) A scientific or technical report, published or issued by a government agency of the Executive Branch of government or the General Accounting Office, the Nuclear Regulatory Commission, or the Defense Nuclear Facilities Safety Board, or published in a peer-reviewed journal, that identifies dosimetry and related information that are unavailable (due to either a lack of monitoring or the destruction or loss of records) for estimating the radiation doses of employees covered by the petition.*

To determine if any of the above-cited conditions existed to a level that would preclude the dose reconstruction for **non-presumptive** cancers, SC&A conducted interviews with PORTS workers/site experts and reviewed a variety of relevant documents/reports.

5.2 SITE INTERVIEWS

An integral part of SC&A's review of any site profile is to conduct interviews with site expert(s) and site workers/union representatives. Thus, during a site visit to the PORTS facility on May 20–26, 2007, several current and former employees expressed concerns regarding the quality, completeness, and potential corruption of personnel monitoring records. These very concerns have previously been raised by others, including members of Local #66 of the Security, Police, and Fire Professionals of America (SPFPA) Union and members of the United Steel Workers (USW) Union. SC&A was told that these allegations had been shared with the Department of Labor in a letter dated July 12, 2005 (Bowe 2005), as well as with NIOSH during a worker outreach meeting between SPFPA and NIOSH in Piketon, Ohio, on November 30, 2005. At the time of the worker outreach meeting, NIOSH was provided with copies of two documents that addressed these allegations in detail (i.e., the Butler 1996 Report and the Cardarelli 1997 Report), and was asked to review an evaluation of the PORTS facility by the DOE (DOE 2000). In summary, members of the SPFPA and USW had requested NIOSH to review these documents in context with their allegations and assess their potential impact on the feasibility of worker dose reconstruction.

Most relevant to the feasibility of dose reconstruction among allegations raised by PORTS workers are the following:

(1) <u>Generic Concerns</u>

- Workers with long-term employment witnessed the many changes/improvements in worker protection and worker monitoring that only occurred in the more recent years. Generic concern among these workers was that during peak production periods, worker protection was not a top priority; health physics personnel were grossly understaffed and lacked experience; and engineering controls were inadequate. Thus, the time periods with the highest potential for worker exposures coincided with the lowest standards for protecting and monitoring workers.

- Radiological incidents involving spills, fires, gaseous leaks were many and routine. However, the threshold for documenting such incidents was arbitrary. For example, equipment failures such as those involving pigtails to cylinders in which UO_2F_2 was released were not always documented/investigated and not all workers who may have been exposed were monitored.

(2) Concerns for Select Workers

Among the worker groups interviewed, special concerns were voiced by security personnel. For obvious reasons, site security was an integral part of all PORTS activities. Thus, security personnel were not only granted unfettered access to **all** buildings that processed radioactive materials, but were also responsible for around-the-clock security of SNM, LEU, and HEU during onsite transit and storage. The latter assignments placed security personnel in close proximity to these sources for lengthy periods of time.

Unique to security personnel was the absence or reduced radiological protection afforded to them with regard to worker training in radiological protection, use of anti-contamination clothing, use of respirators, and routine bioassay sampling during the 1970s and 1980s.

(3) Concerns Pertaining to Unmonitored Neutron Exposures

- Since the introduction of TLDs at PORTS in 1981, "abnormal" chip ratios have been routinely observed, but were never investigated by health physics personnel who dismissed them as artifacts of defective TLDs
- The "abnormal" chip ratios are suggestive of significant exposure to neutrons
- However, prior to 1992, the personnel TLDs were neither calibrated for neutron exposures nor processed for potential neutron exposures
- Dosimetry records for this time period that might have documented these abnormal chip ratios, however, are no longer available

(4) Concerns Regarding Dosimeter Data and the Integrity of Records

- Personnel dosimeters were frequently processed in behalf of a "generic bar code" that subsequently could **not** be linked to the individual to whom the dosimeter had been assigned. These dosimeters were referred to as the "bucket dosimeters."
- Documented allegations suggest that official dosimetry records were altered. In one specific case, the recorded shallow and deep dose of 26 mrem was replaced with a dose of 0, but was subsequently restored to their original value of 26 mrem.

A comprehensive assessment of workers' concerns is enclosed herein as Appendix 1. Appendix 1 represents comments provided during interviews of 46 current and former workers at PORTS whose employment periods range from 1954 to the present. The experience among those interviewed range from production, laboratory support, facility maintenance, environmental monitoring, medical security, and health physics.

5.3 DOCUMENT REVIEW

In response to concerns raised during the interview sessions, SC&A carefully assessed the following documents:

(1) Memorandum from Don Butler (Lockheed Martin) to Dan Hupp (Lockheed Martin). *Internal Investigation into Health Physics Management Practices*, POEF-150-96-0088, February 16, 1996.

(2) Cardarelli, J., 1996. HETA 96-0198-2651, *Portsmouth Gaseous Diffusion Plant, Piketon, Ohio*, National Institute for Occupational Safety and Health, Cincinnati, Ohio.

(3) DOE (Department of Energy) 2000. *Independent Investigation of the Portsmouth Gaseous Diffusion Plant – Volume 1: Past Environment, Safety, and Health Practices*, DOE Office of Oversight, Environment, Safety and Health.

(4) Public comments provided by SPFPA at the June 14, 2006, Advisory Board Meeting in Washington, DC (ABRWH 2006).

(5) Letter from J.W. Neton (NIOSH) to D. Bowe (President, SPFPA Local #66), dated December 19, 2005, Subject: Discussion of Portsmouth's External Dosimetry Program. National Institute for Occupational Safety and Health, Office of Compensation Analysis and Support.

(6) E-mail correspondence from J.W. Neton (NIOSH) to D.C. Bowe (President, SPFPA Local #66), dated December 16, 2005, RE: Discussion, National Institute for Occupational Safety and Health, Office of Compensation Analysis and Support.

(7) Notes prepared by SPFPA regarding dosimetry practices and worker safety.

Of primary importance to the issue of data integrity are the first three documents. Presented below are summary statements, background information, and tentative conclusions cited in each of the reports.

5.4 THE BUTLER 1996 REPORT

This 41-page report describes an internal investigation of the PORTS health physics practices that resulted from specific allegations of improper conduct by a member of the health physics staff. The investigation was performed by a three-member team. Allegations relevant to data integrity included the following:

(1) The improper changing of the recorded dose of 26 mrem assigned to a worker (who was a security guard at PORTS) to a 0 dose. The 0 dose was subsequently changed back to the previously recorded dose of 26 mrem.

(2) The corruption in the maintenance of the DOELAP TLD database.

(3) Misrepresentations of personnel training records to outside auditors in behalf of the National Voluntary Laboratory Accreditation Program (NVLAP).

In response to these allegations, the investigation team from Lockheed Martin offered the following opinions and conclusions:

- <u>On the allegations of improper changing of recorded dose</u>:

  It is difficult to understand why such an improper action would have been taken upon the negligible figure of 26 millirem. If true, such an action represents a case of gross impropriety on the part of [name deleted]. *The fact that the figures were subsequently changed back to the previous format further complicates the issue, as it raised the possibility of "cold feet" for the individual responsible for the change.*

  Although we believe the allegation to be true, we cannot definitively prove it.

- <u>On the allegation of corrupting the TLD database</u>:

  [Names withheld] *describe the DOELAP TLD database as being basically reliable, erring on the conservative side, if anything. While a number of unassignables obviously exist, several experienced Health Physics personnel believe that the data is quite sound ([Name withheld] and [Name withheld], however, maintained strong reservations about the validity of the DOELAP TLD Database).*

  The function of this database is now relegated to that of a historical file. Thus, while this allegation does contain some truth, the original assessment of large numbers of misassignments has, to say the least, not been proven. . . .

  In conclusion, the absolute validity of the DOELAP TLD Database cannot, at this time, be verified.

- <u>On the allegation that health physics supervision misrepresented training records/ qualifications of a technician during a NVLAP audit</u>:

  By [name withheld] *own admission,* [name withheld] *was not fully qualified to process dosimeters. It seems probable, however, that the auditor was persuaded otherwise, either by* [name withheld] *or by* [name withheld]. *The auditor could have, however, simply been shown the limited training records, and allowed to draw his own conclusions.* [Name withheld] *stated that he could understand why they passed the audit in regards to training, because, after all, no TLDS were yet being processed through the system, and the auditor appeared to be pleased with the planned training program.*

This allegation thus appears to have, at least, a limited amount of support, but is neither confirmed or denied.

5.5 THE CARDARELLI (1997) REPORT

On June 12, 1996, NIOSH received a request for a Health Hazard Evaluation (HHE) that was to focus on worker exposures to neutrons at PORTS. In response to this request, NIOSH conducted two site visits between November 1996 and February 1997 (Cardarelli 1997), with the following six objectives:

(1) Determine if potential neutron exposures exist at the site
(2) Identify neutron sources
(3) Identify work areas or job titles having the greatest potential for neutron exposures
(4) Quantify neutron doses by work area or job title
(5) Determine past reporting and recording practices regarding neutron doses
(6) Assess the feasibility of reconstructing past neutron doses

The HHE clearly showed that (1) neutron radiation has been present over the entire period of the PORTS facility operation, (2) significant neutron radiation sources are/were present in areas where uranium compounds are present in appreciable amounts, and (3) at the time of the HHE, neutron dose rates varied, based on work location and job function.

With regard to objectives #5 and #6, the HHE stated the following (Cardarelli 1997):

> *… personnel neutron dosimetry was not conducted . . . in the past . . .*
>
> *The early film badges (1950s–1980) and the first TLD badges (1981–1990) were not calibrated and could not measure neutron exposures. Despite that limitation, an effort to reconstruct past neutron exposures was attempted by requesting historical computerized data of TLD chip readings from 1981 to the present. Data available for this purpose was limited to only the most recent data (1992–1995). Thus it was not feasible to reconstruct potential neutron* [doses] *before 1992.*

5.6 DOE (2000) REPORT

A significant portion of this report assessed the quality of worker health and safety programs at PORTS, and documented program deficiencies, worker allegations of unsafe conditions, and management's relationship to union workers. The following statements cited in the DOE (2000) not only summarize noted deficiencies and allegations, but also reveal circumstances that may impact their interpretation.

- <u>On Programmatic Deficiencies</u>:

> *…appraisal of ES&H* [Environmental Safety and Health]*, called "contractor health protection program reviewers," were performed as early as 1957, and the*

AEC manual required annual ES&H assessments starting in 1961. These assessments were generally performed by two persons over three days and addressed radiation protection, criticality safety, industrial hygiene, environmental programs… Although important deficiencies and issues were identified by these reviews, the size and complexity of Plant operations and the nature of the industrial hazards… warranted longer and more frequent assessments.

…A more in-depth, two-week assessment conducted in 1973 by OR [DOE Oak Ridge Operations Office] *included field observations of Plant conditions, work performance, and interviews with workers and first-line supervisors; it concluded that the health protection program at PORTS was inadequate. However, there was no further evidence of more rigorous assessments, and the limited annual appraisals resumed until the 1980s.*

The AEC and its successor organizations also investigated worker allegations of unsafe conditions and practices, but with inconsistent rigor and results. … From 1979 through 1982, another major DOE investigation of worker complaints, conducted at the direction of Congress, identified performance problems in a variety of ES&H areas.

The 1989 DOE Tiger Team assessment identified numerous health, safety, and environmental deficiencies…

Historical weaknesses in DOE investigations of workers allegations have continued to the present program. One case in particular, raised during the transition from DOE to NRC oversight of USEC, still remains unresolved. That case involves allegations by a Plant guard who maintains that his radiation exposure records were falsified. Internal investigations by Lockheed Martin Utility Services found some merit to the allegation [Note: this is a reference to the Butler 1996 Report], and the allegation was forwarded to the Oak Ridge Operations Office Inspector General in 1996. That case remains inadequately investigated and unresolved by DOE.

- On Labor Relations (pp. 77–80):

  Established in 1954, the Oil, Chemical, and Atomic Workers Union (OCAW) was aggressive in its efforts to protect and improve employee welfare. This aggressiveness sometimes caused friction between Plant management and labor. On numerous occasions, the positions of management and labor differed widely, and resolution was accompanied by extreme measures, as evidenced by one unauthorized and six authorized strikes that occurred from 1954 to 1993. Furthermore, the severity of management and labor disagreements appears to have increased beginning in 1974, as suggested by the frequency and duration of strikes. While economic issues were common to most strikes, safety and health were an important element in three of these seven actions…

Collectively, the number of grievances filed, worker compensation claims submitted, and alleged acts of retaliation committed provide further support that management and labor relations were strained. From 1954 through 1993, it is estimated that more than 17,000 union worker grievances were filed addressing a variety of issues in addition to safety and health, including work jurisdiction, discipline, overtime, work rules, and benefits.. …

… there are records suggesting that labor grievances were filed to be confrontational, management appeared to have been acting appropriately and in the interest of its employees' safety and health.

From pp. 5-6:

The other major union at the Plant, the United Plant Guard Workers of America (UPGWA), has had no strikes since its formation in 1955. …

Relations between… UPGWA union and Plant management were much less confrontational. Although protective forces have been an integral part of Plant activities due to security considerations, the ES&H protection provided to production workers (such as respirators and shoe covers) were not always considered or provided to security personnel when they worked in close proximity to hazardous operations or were stationed, ate lunch, and took breaks in contaminated areas. In addition, in the late 1980s and early 1990s, protective forces performed extensive training drills in radioactively contaminated buildings without appropriate protective clothing or monitoring. Hazard communications and ES&H training have not always been provided on a timely and consistent basis for protective force workers.

5.7 CONCLUSIONS PERTAINING TO DATA INTEGRITY AND THE ADEQUACY OF DATA FROM DOSE RECONSTRUCTION

On the basis of available information gained from the review of the PORTS Site Profile, site expert interviews, and the previously cited documents, SC&A concludes the following:

- On programmatic deficiencies and gaps in personnel monitoring:

  Noted deficiencies in monitoring personnel for internal and external exposures are real, but are not unique to PORTS workers. To date, NIOSH and its contractors have encountered similar deficiencies at other facilities that have been resolved by various means (i.e., use of surrogate data, coworker models, etc.).

- On the falsification of records:

  SC&A agrees that the falsification of dosimetry data is a serious matter that should be thoroughly investigated. Documented evidence of altering personnel doses at PORTS, however, appears to be restricted to a single individual and for a single dosimeter wear period. Moreover, the initial dose of 26 mrem that was changed to a 0 dose was subsequently restored to its original value.

Although no explanation has been offered that would justify the described events, SC&A concludes that this isolated event does **not** adversely impact the reconstruction of worker doses at PORTS.

- On the "corruption" of the DOELAP TLD Database:

  SC&A believes that persistent perceptions regarding data integrity issues have been exacerbated by (1) personality conflicts among select individuals within the health physics group, and (2) a history of poor relations and mistrust between labor and management at PORTS.

  Over the years, many improvements were introduced in personnel monitoring. Key changes included period replacement of dosimetry systems that started with manually processed film dosimeters and evolved to increasingly automated TLD systems. The transition from one system to the next undoubtedly may have introduced problems, such as the alleged problem associated with the DOELAP dosimeters' bar codes.

  SC&A concludes that while the bar code problem with the DOELAP TLD may have resulted in some unassignable doses, the likely number of affected persons were small. More importantly, any discrepancy between recorded and actual doses does **not** reflect a deliberate attempt by health physics personnel to reduce/falsify worker exposures.

- On unmonitored neutron doses:

  Based on the physical nature and material quantities processed at PORTS, neutron exposures should have been anticipated for the entire period of facility operation. Therefore, the failure to properly monitor PORTS workers for neutron exposure prior to 1992 must be regarded as a deficiency that has to be accounted for in dose reconstruction.

  ORAUT-TKBS-0015-6 references multiple studies inclusive of the 1997 Cardarelli study. In combination, these studies provide surrogate data that may reasonably reflect **routine** unmonitored neutron exposures. A major limitation of the surrogate neutron dose models is their failure to account for "unusual" neutron exposures, as documented by the observed abnormal chip ratios of the DOELAP TLDs. This limitation was acknowledged by Cardarelli (1997), who concluded that ". . . it was not feasible to reconstruct neutron exposures from previous [abnormal] TLD chip readings before 1992."

  Currently, the unresolved failure to account for neutron exposures associated with abnormal chip ratios that may reflect "slow cookers" is discussed as Finding 4.6-4 in this report and must be resolved.

6.0 REFERENCES

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ATTACHMENT 1: SUMMARY OF SITE EXPERT INTERVIEWS

This attachment will be provided at a later date.