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

October 2, 2007

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

Re: Contract No. 200-2004-03805, Task Order 1: Transmittal of Draft Addendum to SCA-TR-TASK1-0003, *Status Report on the Resolution of the Savannah River Site Issues Resolution Matrix*

Dear Mr. Staudt:

SC&A is pleased to submit to NIOSH and the Advisory Board its draft addendum to SCA-TR-TASK1-0003, *Status Report on the Resolution of the Savannah River Site Issues Resolution Matrix*, dated October 2, 2007. This report has been subjected to a review for Privacy Act-related information and edited accordingly, and is now cleared for unrestricted distribution. This document has not, however, been reviewed by the Advisory Board for factual accuracy or applicability within the requirements of 42 CFR 82

If you have any comments or questions regarding this report, 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

Status Report on the Resolution of the Savannah River Site Issues Resolution Matrix

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

Addendum to SCA-TR-TASK1-0003

Prepared by

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October 2007

S. Cohen & Associates: ***Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program***	Document No. Addendum to SCA-TR-TASK1-0003
	Effective Date: Draft – October 2, 2007
	Revision No. 0 (Draft)
STATUS REPORT ON THE RESOLUTION OF THE SAVANNAH RIVER SITE ISSUES RESOLUTION MATRIX	Page 2 of 146
Task Manager: ____________________ Date: _________ Joseph Fitzgerald	Supersedes: N/A
Project Manager: __________________ Date: _________ John Mauro	

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ACRONYMS AND ABBREVIATIONS

ABRWH or Advisory Board	Advisory Board on Radiation and Worker Health
AEC	Atomic Energy Commission
Bq	Bequerel
CATI	Computer-Assisted Telephone Interview
CDC	Centers for Disease Control and Prevention
CEDE	Committed Effective Dose Equivalent
CFR	*Code of Federal Regulations*
Ci	Curie
DAF	Dose Adjustment Factor
DCF	Dose Conversion Factor
DOE	Department of Energy
dpm	Disintegrations per minute
DPSOP	DuPont Savannah River Plant Operating Procedure
DPSP	DuPont Savannah River Plant Procedure
DuPont	E.I. DuPont De Nemours and Company
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EPI	Environmental Protection Institute
EU	Enriched Uranium
FMPC	Feed Materials Production Center
GI	Gastrointestinal
HEPA	High Efficiency Particulate Air
HERB	Health Related Energy Research Branch
HPAREH	Health Protection Annual Radiation Exposure History Database
HT	Tritiated Gas
HTO	Tritiated Water
HVIS	Health Visitor Information System
ICRP	International Commission on Radiation Protection
IDR	Internal Dosimetry Registry
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiologic Program

IRF	Intake Retention Fraction
keV	Kiloelectron volts
MDA	Minimum Detectable Activity
MeV	Megaelectron volts
MT	Metal Tritide
NCRP	National Council on Radiation Protection and Measurement
NIOSH	National Institute for Occupational Safety and Health
NRC	Nuclear Regulatory Commission
NTA	Neutron Track Emulsion
OBT	Organically Bound Tritium
OCAS	Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
ORAUT	Oak Ridge Associated Universities Team
OTIB	Oak Ridge Associated Universities Team Technical Information Bulletin
pCi	Picocuries
PER	Program Evaluation Report
PNNL	Pacific Northwest National Laboratory
POC	Probability of Causation
RAC	Risk Assessment Corporation
RBA	Radiation Buffer Area
RCA	Radiological Control Area
RDZ	Radiation Danger Zone
RU	Recycled Uranium
RWP	Radiation Work Permit
SC&A	S. Cohen and Associates
SEC	Special Exposure Cohort
SHI	Special Hazards Investigation
SMT	Special Metal Tritides
SRP	Savannah River Plant
SRS	Savannah River Site
STC	Stable Tritium Compounds
TBD	Technical Basis Document

TIB	Technical Information Bulletin
TLD	Thermoluminescent Dosimeter
TLND	Thermoluminescent Neutron Dosimeter
UNH	Uranyl Hexahydrate
WSMS	Washington Safety Management Solutions
WSRC	Westinghouse Savannah River Company

1.0 EXECUTIVE SUMMARY

The original review of the Savannah River Site (SRS) site profile was conducted by S. Cohen and Associates (SC&A) and submitted to the Advisory Board on Radiation and Worker Health (ABRWH or Advisory Board) as a draft report on March 21, 2005. That review was based on Revision 02 of the SRS site profile. During its June 15, 2006, meeting in Washington, DC, the Board assigned SC&A to review a subsequent, updated version of that site profile, Revision 03, which had been issued in April 2005. The Board also empanelled a Work Group to guide discussions between the National Institute for Occupational Safety and Health (NIOSH) and SC&A on outstanding issues raised in the review of Revision 02 of the site profile [also referred to as the Technical Basis Document (TBD)]. All SC&A review material has been initially submitted to and coordinated with this Work Group, as well as with NIOSH.

This report constitutes an evaluation of the progress made in resolving issues identified in the review of Revision 02 of the SRS TBD. The review focused on the extent to which Revision 03 addressed and resolved the issues identified from the review of Revision 02 and the degree to which closure had been achieved.[1] SC&A also evaluated site-specific Technical Information Bulletins (TIBs) and procedures as a part of this review. Attachment 1 provides a complete list of documents considered during the evaluation. SC&A's team of health physicists and technical personnel conducted the review from August 2006 to April 2007. The review included a Work Group meeting, technical calls, and a visit to SRS to review incident database information. NIOSH was provided a final opportunity to review the current status of issues prior to issuance of this report; however, its comments were delayed and will be forthcoming in the context of the upcoming SRS Work Group discussions. Accordingly, this review's representation of outstanding SRS site profile issues (and NIOSH's initial response at the time) does not necessarily reflect NIOSH's position regarding these matters as of the date of the publication of this report, and it awaits NIOSH's response and further Work Group deliberation and action.

SC&A's evaluation process included a review of the revised TBD, focused interviews with site experts, and review of documents previously retrieved from or subsequently requested of SRS. SC&A evaluated the TBD for completeness, technical accuracy, adequacy of data, compliance with stated objectives, and consistency with other site profiles, as stipulated in the *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004).

SC&A is aware of NIOSH's ongoing efforts to develop and issue Revision 04-E (Scalsky 2006) of the SRS site profile and has acknowledged additional information intended for inclusion in that version. (However, issues related to those intended improvements continue to be cited as "open" pending issuance of that revised site profile.) Likewise, additional guidance documents were being issued that, while not reflected in Revision 03 (Scalsky 2005) of the SRS site profile, would serve to mitigate some of the gaps and issues raised in this report.

[1] The specific comments identified, the corresponding NIOSH response, and the subsequent Work Group actions originated with an issue resolution matrix informal draft report, entitled "Summary of Task 1 Savannah River Site Technical Basis Document Finding Matrix—Vertical Issues," circulated and used at the August 22, 2006, Work Group meeting.

1.1 OVERVIEW

Under the auspices of the Board's Work Group, NIOSH and SC&A have made progress addressing a number of significant issues arising from SC&A's original review of the SRS site profile, Revision 02. The revised TBD and/or various supporting documents have satisfactorily addressed a number of key technical issues identified in SC&A's 2005 review, including the following:

- To address the lack of guidance pertaining to shallow dose, NIOSH provided subsequent guidance in ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (Merwin 2005), that SC&A found could be applied to shallow dose and nonuniform exposure at SRS.

- Several generic issues, including assumptions related to internal dosimetry (oro-nasal breathing, solubility, and ingestion), complexity of implementing guidelines, and dose estimation for subcontractors and construction workers, are being addressed with the Board and SC&A in other venues.

As noted previously, a number of issues seem amenable to resolution through proposed changes to be reflected by NIOSH in Revision 04-E (Scalsky 2006) of the site profile, including the following:

- An expanded discussion of thorium, U-233, and Pu-242 sources and operations, now lacking in Revision 03 (Scalsky 2005), is planned for Revision 04-E (Scalsky 2006).

- Further research by NIOSH and Oak Ridge Associated Universities (ORAU) is ongoing regarding SRS source terms and worker exposures to neptunium and curium, and Revision 04-E (Scalsky 2006) will reflect the results of this research.

- Additional information has been added to draft Revision 04-E (Scalsky 2006) to better characterize the exposure potential of exotic radionuclide sources associated with the high flux program.

However, the resolution of other key issues requires documentation, evaluations, or technical closure via the Work Group:

- The adequacy of the characterization provided for the F- and H-Area Tank Farm remains a concern as both NIOSH and SC&A, through the Board's Work Group, strive to obtain further information on radionuclide source terms through the Fault Tree Data Bank and other SRS incident databases.

- The completeness of dose information for early SRS workers remains in question based on SC&A's evaluation of the Health Protection Annual Radiation Exposure History Database (HPAREH) records file that is being relied upon for dose estimation.

- Although NIOSH has proposed an approach for conservatively estimating dose to workers from special tritium compounds (STCs), it falls short of characterizing who was exposed, where they were exposed, and in what time periods, all necessary parameters for dose estimation.

- Concerns regarding the validity of neutron-to-photon ratios at SRS remain unresolved, given how neutron-to-photon ratios are assigned in Revision 04-E (Scalsky 2006).

It is anticipated that the issuance of Revision 04-E (Scalsky 2006) of the TBD will be the means to address these issues and achieve final closure on those that are currently open.

1.2 MATRIX ISSUE STATUS

Comment 1: Recycled uranium. The site profile does not contain adequate guidelines for resolving uncertainties related to recycled uranium (RU) in ways that give the benefit of the doubt to the claimants. For instance, the TBD does not consider internal dose contributions for plutonium, other transuranics, or fission products. This is mitigated somewhat by the planned inclusion of updates in Revision 04-E (Scalsky 2006) of the TBD that deal directly with exposure from these radionuclides and bridge some of the gaps found in previous versions of the TBD. Among other actions, the Work Group has requested a timeline for the Oak Ridge Associated Universities Team Technical Information Bulletin (OTIB) regarding RU (publication pending), information on the ability to link to the bioassay program and extrapolate doses back in time, and the intended approach for thorium, U-233, and Pu-242. SC&A **recommends that this issue remain OPEN with outstanding Work Group actions.**

Comment 2: Beta/Gamma correction factors. The issues associated with correction factors and uncertainties have not been satisfactorily resolved by Revision 03 or by planned additions in Revision 04-E (Scalsky 2006). The beta/gamma dosimeter adjustment factors and uncertainties applied underestimate the true exposure measured by the dosimeter. Correction factors applied to dosimeter results account for on-phantom calibration and do not consider uncertainty from field exposure conditions. The standard deviation for film dosimeters prior to 1971 is too low. SC&A agrees that ORAUT-OTIB-0017 (Merwin 2005) provides suitable guidance for the assignment of shallow dose. NIOSH provided references on the adjustment factors used in the TBD, and SC&A compared the uncertainty factors from the TBD with relevant site-specific workbooks and guides. SC&A **recommends that this issue remain OPEN pending Work Group review.**

Comment 3: Neutron/photon ratios. The geometric mean and standard deviation that describe the post-1971 neutron-to-photon ratio are neither technically defensible nor likely to be claimant favorable to a large number of claimants. The neutron doses recorded by the thermoluminescent neutron dosimeter (TLND) between 1971 and 1995, as well as the pre-1971 neutron doses (derived from neutron-to-photon ratios), suffer from a high degree of uncertainty. The use of the 95th percentile value for the TLND neutron dose of records is recommended. (Revision 04-E (Scalsky 2006) provides for the 95th percentile neutron-to-photon dose ratio to be applied to the recorded dose for likely noncompensable cases.) The Work Group requested from NIOSH a detailed description of neutron-to-photon ratio methods used at SRS. SC&A **recommends that the issue remain OPEN pending Work Group review.**

Comment 4: Adequacy of radionuclide characterization. The adequacy of the F- and H-Area Tank Farm characterization in Revision 03 of the TBD is questionable for use as dose reconstruction guidance. This is particularly true for early periods of operation, where primary records involving key operations and incidents are lacking. Moreover, no references are provided for the tank farm discussion in the TBD, and there is no analysis indicating how the conclusions were reached. The proposed Revision 04-E (Scalsky 2006) included (at the time of this review) only minimal changes that do not resolve key issues. The Work Group requested that NIOSH provide an expanded review of the adequacy of its SRS radionuclide list by reviewing additional source term information (e.g., the Fault Tree Data Bank). This led to a series of requests to the Department of Energy (DOE) for access to the Fault Tree Data Bank, culminating in a February 2007 visit to review firsthand selected files of what turned out to be the Washington Safety Management Solutions (WSMS) data file of likely different pedigree. A series of actions, provided in Section 4.13 of this report, stemmed from this onsite review. SC&A **recommends that this issue remain OPEN with additional actions identified.**

Comment 5: Use of early monitoring data. The adequacy of early worker monitoring data is questionable and requires further investigation. This issue will not be resolved by an inventory of records provided in claimant files. Additional validation of the HPAREH database as the exclusive source for external coworker dose determination is necessary to demonstrate that such early data are adequate for use. The Work Group requested that NIOSH provide a description of categories of workers and timeframes during which workers were not appropriately monitored during the early phases of beginning operations; SC&A was requested to provide an inventory of data sources used in the preparation of individual dosimetry data submitttals. SC&A **recommends that this issue remain OPEN pending further Work Group review.**

Comment 6: Validity of high-five approach. Revision 03 continues to use the "high-five" approach [by reference to ORAUT-OTIB-0001 (Brackett 2003)] and, in SC&A's view, would remain inconsistent with the methodologies recommended 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), for the calculation of internal dose. NIOSH has indicated its intention to update the high-five approach and base the revised calculations on bioassay data, rather than data in the SRS Internal Dosimetry Registry (IDR); however, this update has not yet been released and referenced in Revision 04-E (Scalsky 2006). SC&A **recommends that this issue remain OPEN pending further Work Group review.**

Comment 7: Onsite atmospheric dispersion and resuspension. The method used to reconstruct doses to unmonitored outdoor workers due to airborne emissions employs an atmospheric dispersion model, assumptions, and resuspension factors that do not appear to be claimant favorable and are not entirely appropriate for this class of problem. Revision 03 and proposed Revision 04-E (Scalsky 2006) provide no clarification on these issues. SC&A **recommends that this issue remain OPEN pending further Work Group review.**

Comment 8: Special tritium compounds. The TBD does not adequately address potential exposures of workers handling tritium and performing decontamination and decommissioning to STCs, including OBT and stable metal tritides. NIOSH did issue ORAUT-OTIB-0066, Revision 00, *Calculation of Dose from Intakes of Special Tritium Compounds* (LaBone 2006), but this guidance requires knowledge of site-specific processes. For SRS, NIOSH has provided a proposed site-specific dose estimation methodology for stable tritium compounds (STCs) at SRS; however, they have not verified the timeframe or location where STCs were handled or the types and quantities of STCs handled at SRS. An evaluation of the adequacy of the dose estimation methodology cannot be carried out without this key information, which the Work Group has requested. SC&A **recommends that this issue remain OPEN pending further Work Group review.**

Comment 9: Completeness of intake values for the high-five approach. Intake values for the high-five approach are derived from an average of the highest five intakes identified by NIOSH. The values used to derive the average were obtained from a limited review of the SRS IDR, which does not contain all intakes occurring at SRS. The Work Group requested actions on the part of both NIOSH and SC&A, including completing an update of high-five intakes using new models, reviewing the Fault Tree Data Bank for applicability, and reviewing additional information available on 3x5 visitor cards. SC&A **recommends that this issue remain OPEN with additional actions identified.**

Comment 10: Currency of International Commission on Radiation Protection (ICRP) methodology. The hypothetical intake, outlined in ORAUT-OTIB-0001 (Brackett 2003), uses the SRS intake quantities calculated using the methodology in ICRP 30, *Limits for the Intake of Radionuclides by Workers* (ICRP 1979). The average activity (nCi) is entered into the Integrated Modules for Bioassay Analysis (IMBA), and a dose is calculated based on the models in ICRP 66, *Human Respiratory Tract Model for Radiological Protection*, and ICRP 68, *Dose Coefficients for Intakes of Radionuclides by Workers*. The use of surrogate data for internal dose for unmonitored workers adopted in the high-five approach for target organs that do not concentrate the radionuclides in question is not necessarily a maximizing approach for making dose estimates, contrary to the claim in the TBD. The Work Group requested a NIOSH update of high-five intakes using the new models. SC&A **recommends that this issue remain OPEN with additional actions identified.**

Comment 11: Assignment of organ dose may not be sufficiently conservative. For internal dose calculations, the use of ICRP 30 methodology to calculate the intake with a subsequent use of ICRP 68 models to calculate the dose did not always result in the intended highest dose to an organ. Similarly, the appropriate solubility types between the two methodologies were not always paired consistently, resulting in discrepancies and a lack of claimant favorability. The dose reconstructor is directed to use surrogate radionuclides for radionuclides absent from the IMBA code. This issue has been partially resolved since the initial SC&A review with the inclusion of additional radionuclides in the latest version of the IMBA code, eliminating the need to use surrogate organs in many cases. Bioassay data rather than intake data should be used to calculate internal dose with the most current methodology available. As in Comment 10, resolution of this issue would be based on the results of a NIOSH update of its high-five intake method using bioassay data and appropriate models. SC&A **recommends that this issue remain OPEN with additional actions identified.**

Comment 12: Solubility, oro-nasal breathing, ingestion. Solubility, oro-nasal breathing, and ingestion should be carefully considered with regard to internal dose reconstruction. SC&A originally developed these points for the review in the Bethlehem Steel and Mallinckrodt Chemical Works site profile reviews, and they are applicable for all bioassay interpretations for the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA). The Work Group considered this a generic issue being addressed by NIOSH in a different venue. As a result, SC&A **recommends that this issue be CLOSED; no further action necessary.**

Comment 13: Appropriate application of incident information. Incidents and high-risk jobs are not listed in the TBD or referenced to alert dose reconstructors to unique exposure conditions. Without a thorough reconciliation of the DOE exposure files against these separate incident data banks, NIOSH cannot be assured that all significant exposures from incidents are considered in relation to individual worker claims or to high-five estimates of maximal doses. For consistency among dose reconstructions, the reviewers concluded that the TBD should alert the dose reconstructor to special conditions when a deviation from the standard dose reconstruction methodology is needed. Work Group actions for resolving this issue included NIOSH requesting copies of the Special Hazards Investigations reports from DOE/Westinghouse Savannah River Company (WSRC), obtaining the "User's Guide" for the WSMS database from DOE/WSRC, identifying other available incident databases, and researching the pedigree for the WSMS database reviewed during the February 2007 visit. SC&A **recommends that this issue remain OPEN with additional actions identified.**

Comment 14: Completeness of HPAREH database. SC&A provided NIOSH with an inventory of supplemental records that may be beneficial in dose reconstruction. These records are not always provided in the claimant file submitted for dose reconstruction. No effort has been made to evaluate the completeness of the HPAREH file used in the development of the external coworker model. The integrity of the HPAREH file for use in coworker modeling is questionable given the absence of much of the data for workers terminating employment prior to 1979. A basis for its appropriateness should be developed and discussed in either the TBD or the external coworker dose procedure. The Work Group requested that NIOSH review neutron logbooks referenced in the August 2006 meeting and evaluate the completeness of the HPAREH file used as a basis for the external dose coworker model. SC&A **recommends that this issue remain OPEN with additional actions identified.**

Comment 15: Clarity of TBD information and guidance. Many of the sections of the TBD, especially Chapter 4 related to internal dosimetry, are very difficult to understand, and, together with the large array of TIBs and other Office of Compensation Analysis and Support/ORAU procedures, create a virtually impenetrable and complex array of guidelines. This situation lends itself to inconsistencies in the way in which dose reconstructions are performed and makes it difficult to verify the reliability and reproducibility of the dose reconstructions. NIOSH responded that a large number of TIBs and other guidelines have been developed to address the issues raised. SC&A **recommends that this issued be CLOSED; no further action necessary.**

Comment 16: Treatment of construction workers. The TBD does not currently include the special exposure circumstances for subcontractors and construction workers; however, NIOSH is aware of this issue. ORAUT-OTIB-0052, *Parameters to Consider When Processing Claims for Construction Trade Workers* (Chew et al. 2006) was developed to provide dose reconstruction guidance for trade workers. SC&A reviewed this procedure and submitted a draft report addressing this issue to NIOSH and the Board on July 30, 2007, as part of Task Order 3. SC&A **recommends that this issue be CLOSED; no further action necessary.**

2.0 SCOPE AND INTRODUCTION

The original review of the Savannah River Site (SRS) site profile (SC&A 2005a) was conducted by S. Cohen and Associates (SC&A) and submitted to the Advisory Board on Radiation and Worker Health (ABRWH or Advisory Board) as a draft report on March 21, 2005. During its June 15, 2006, meeting in Washington, DC, the Board assigned SC&A to review a subsequent, updated version of that site profile, Revision 03, which had been issued in April 2005. The review was to focus on the extent to which Revision 03 addresses and resolves issues identified by SC&A from the review of Revision 02. This report presents the status of that review, which the SC&A team conducted from August 2006 to April 2007.

The Board also empanelled a Work Group to guide the review, chaired by Michael Gibson, with Board members Mark Griffon, James Lockey, and Brad Clawson (the Work Group is now chaired by Mark Griffon). The Work Group arranged discussions with the National Institute for Occupational Safety and Health (NIOSH) and SC&A on selected outstanding issues, as well as information retrieval and onsite visits. All SC&A review material compiled under this review was submitted to and coordinated with this Work Group. The Work Group chairman briefed the full Board regularly on progress achieved in the SRS site profile issue resolution review.

It is recognized that all site profiles are “works in progress” and are being revised. SC&A was aware of NIOSH’s ongoing efforts to develop and issue Revision 04-E (Scalsky 2006) of the SRS site profile and has acknowledged additional information intended for inclusion in that version. (However, issues related to those intended improvements continue to be cited as “open” pending issuance of that revised site profile.) Likewise, additional guidance documents were being issued that, while not yet reflected in Revision 03 of the SRS site profile, would serve to mitigate some of the gaps and issues raised in this report; where appropriate, these recent issuances have been so noted as well. NIOSH/Oak Ridge Associated Universities Team (ORAUT) proposed to the Work Group that Revision 04-E (Scalsky 2006) be used to close as many issues as possible and that comment resolution concentrate on issues that are still open.

2.1 REVIEW SCOPE

Under the Energy Employees Occupational Illness Compensation Program Act (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 is mandated to conduct an independent review of the methods and procedures used by NIOSH and its contractors for dose reconstruction. As a contractor to the Advisory Board, SC&A has been charged under Task Order 1 to support the 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 Revision 03 of the SRS Technical Basis Document (TBD), ORAUT-TKBS-0003, *Technical Basis Document for the Savannah River Site To Be Used for EEOICPA Dose Reconstructions*, Revision 03, dated April 5, 2005 (Scalsky 2005), which superceded Revision 02 of ORAUT-TKBS-0003 issued in 2004, and supporting Technical

Information Bulletins (TIBs) and referenced technical information and databases. These included the following:

Technical Basis Documents:

- ORAUT-TKBS-0003, *Technical Basis Document for the Savannah River Site To Be Used for EEOICPA Dose Reconstructions*, Revision 03, April 5, 2005. (Scalsky 2005)

Technical Support Documents:

- OCAS-PER-001, *Misinterpreted Dosimetry Records Resulting in an Underestimate of Missed Dose in SRS Dose Reconstruction*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, September 8, 2003. (Neton 2003a)

- OCAS-PER-002, *Error in Surrogate Organ Assignment Resulting in an Underestimate of X-ray Dose in SRS Dose Reconstructions*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, December 15, 2003. (Neton 2003b)

- OCAS-PER-019, *The Effect of Additional Neutron Dose Data from the Savannah River Site*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, May 18, 2007. (Allen 2007)

- OCAS-TIB-006, *Interpretation of External Dosimetry Records at the Savannah River Site (SRS)*, Revision 1, Office of Compensation Analysis and Support, Cincinnati, Ohio, February 20, 2004. (Neton 2004)

- OCAS-TIB-007, *Neutron Exposures at the Savannah River Site*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, September 17, 2003. (Neton 2003c)

- ORAUT-OTIB-0001, *Technical Information Bulletin: Maximum Internal Dose Estimates for Savannah River Site (SRS) Claims*, Revision 0, Oak Ridge Associated Universities, Oak Ridge, Tennessee, July 15, 2003. (Brackett 2003)

- ORAUT-OTIB-0032, *External Coworker Dosimetry Data for the Savannah River Site*, Revision 00 PC-1, Oak Ridge Associated Universities, Oak Ridge, Tennessee, November 7, 2006. (Merwin 2006)

This list contains site-specific TIBs. Attachment 1 lists other generic TIBs that are applicable to SRS.

Implementation guidance is also provided in "workbooks," which have been developed by NIOSH for selected sites to provide more definitive direction to the dose reconstructors on how to interpret and apply TBDs, as well as other available information. The SRS-specific workbooks have been evaluated by SC&A under a separate task (Task 4) and are included in evaluations being submitted under that task.

SC&A, in support of the Advisory Board, has critically evaluated the revised SRS TBD to achieve the following:

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

SC&A's review of the revised TBD, as with the prior review, focuses on the quality and completeness of the data that characterized the facility and its operations and the use of these data in dose reconstruction. SC&A conducted the review in accordance with *Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004), which was approved by the Advisory Board.

As a follow-up review, the evaluation is directed at reviewing previously identified issues and shortcomings identified by SC&A in Revision 02 of the TBD and determining whether these issues remain unresolved in Revision 03. As with other site profile reviews, this 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.

The TBD (for SRS, the term "TBD" is used interchangeably with "site profile") serves as the primary site-specific guidance document used in support of dose reconstructions. These site profiles provide the health physicists who conduct dose reconstructions on behalf of NIOSH with consistent general information and specifications to support their individual dose reconstructions. SC&A prepared this report to provide the Advisory Board with an evaluation of whether and how the TBD and supporting documents can support dose reconstruction decisions. The criteria for evaluation include whether the TBD provides a basis for scientifically supportable dose reconstructions in a manner that is adequate, complete, efficient, and claimant favorable.

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

2.2 REVIEW APPROACH

SC&A's review of the revised TBD (Revision 03) and supporting documentation concentrated on determining the completeness of data collected by NIOSH, the adequacy of existing SRS personnel and environmental monitoring data, and the evaluation of key dose reconstruction assumptions; all were performed in the context of the findings and issues identified in SC&A's previous review of Revision 02 of the TBD.

All review comments apply to Revision 03 of the SRS TBD, which is the most recent published version, although NIOSH has made available excerpts of the pending Revision 04-E (Scalsky 2006) of the SRS TBD. SC&A is also aware of and is participating in ongoing information-gathering activities on site at SRS involving various incident exposure dose databases. In its 2005 review of Revision 02 of the SRS site profile, SC&A found a number of shortcomings and needed clarifications that were identified in a series of findings and issue statements made in that draft report. SC&A highlighted these in an issue resolution matrix that was submitted to the Advisory Board shortly after the submission of the draft report. The matrix served to designate what issues (or parts of issues) were closed or open, and if open, what remained to be resolved. SC&A expanded and annotated this issue resolution matrix in detail on December 7, 2006, and July 3, 2007, as an issue status summary for comment and follow-up by the Work Group and NIOSH.

On August 22, 2006, the Board's Work Group held a meeting in Hebron, Kentucky, regarding the SRS review to identify issues and make assignments for follow-up. The Work Group addressed, in sequence, each finding identified by SC&A's issue resolution matrix for SRS, with SC&A outlining the issue and its technical basis and NIOSH providing its technical position or current progress to resolve the issues. The Work Group, in some cases, identified followup actions for either NIOSH or SC&A to pursue to further resolution.

This report is based on the July 3, 2007, issue status summary (derived from the original issue resolution matrix), updated to include actions stemming from the February 2007 visit as well as additional technical discussion regarding the open issues. NIOSH's review of the SRS issue summaries compiled by SC&A was delayed, and comments will be forthcoming as a part of upcoming SRS Work Group discussions in October/November 2007. Accordingly, this review's representation of outstanding SRS site profile issues does not necessarily reflect NIOSH's position and awaits further Work Group deliberation and action.

2.3 REPORT ORGANIZATION

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

(1) Executive Summary

(2) Scope and Introduction

(3) Assessment Criteria and Methods

(4) Issue/Comment Resolution Status

(5) Overall Adequacy of the SRS Site Profile, Revision 03, as a Basis for Dose Reconstruction

(6) References

This organization reflects the more proscribed nature of this follow-up review, and its focus on whether issues identified for Revision 02 of the TBD have been resolved in Revision 03 or has an acceptable intended treatment in draft Revision 04-E (Scalsky 2006). A number of the issues identified as "open" in this report are being addressed in Revision 04-E and appear to have satisfactory resolutions, but they cannot be closed until Revision 04-E is published. Those issues addressed by Revision 04-E are noted in this report where appropriate.

3.0 ASSESSMENT CRITERIA AND METHODS

SC&A is charged with evaluating the approach set forth in the site profiles that is used in the individual dose reconstruction process. To do this, SC&A reviews the site profile documents 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). SC&A identified 16 issues, listed in Attachment 3, in the original SC&A review from March 2005. The present review evaluates whether changes made in Revision 03 of ORAUT-TKBS-0003 (Scalsky 2005) resolved the issues identified in the review of Revision 02 of the SRS TBD. Items identified in this report may be applied to other facilities, especially facilities with similar source terms and exposure conditions. The review indicated that Revision 03 of the TBD does not resolve a majority of the 16 concerns from the review of Revision 02. NIOSH/ORAUT has proposed a number of changes in Revision 04-E (Scalsky 2006) that will resolve some of the 16 issues identified once Revision 04-E is formally released.

3.1 OBJECTIVES

SC&A reviewed the site profile with respect to the degree to which it employs technically sound judgments or assumptions. SC&A compared each of the 16 findings from the March 2005 review against Revision 03 of the TBD to determine whether the revision resolved the issue. In addition, the review identifies NIOSH assumptions that give the benefit of the doubt to the claimant.

3.1.1 Objective 1: Completeness of Data Sources

Objective 1 requires SC&A to identify principal sources of data and information that are applicable to the development of the site profile. The three 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, (2) investigating whether other relevant/ significant sources are available but were not used in the development of the site profile, and (3) evaluating data compiled as a part of the comment resolution process. Additionally, SC&A evaluated records publicly available relating to SRS and records provided by site experts.

3.1.2 Objective 2: Technical Accuracy

SC&A reviewed the site profile with respect to Objective 2, which requires SC&A to perform a critical assessment of the methods used in the site profile to develop technically defensible guidance or instruction, including evaluating field characterization data, source term data, technical reports, standards and guidance documents, and literature related to processes that occurred at SRS. The goal of this objective is to first analyze the data according to sound scientific principles, and then to evaluate this information in the context of compensation. If NIOSH/Oak Ridge Associated Universities Team (ORAUT) had analyzed available data, but SC&A found the technical approach used by NIOSH/ORAUT in the analysis of these data to be scientifically unsound or not necessarily claimant favorable, this would constitute a technical accuracy issue.

3.1.3 Objective 3: Adequacy of Data

SC&A reviewed the site profile with respect to Objective 3, which requires SC&A to determine whether the data and guidance presented in the site profile are sufficiently detailed and complete to conduct dose reconstruction, and whether a defensible approach has been developed in the absence of data. In addition, this objective requires SC&A to assess the credibility of the data used for dose reconstruction. Reviewing the adequacy of the data identifies gaps in the facility data that may influence the outcome of the dose reconstruction process. An example of an inadequacy in the data is the case of workers who appeared to have been exposed to neutrons but were not monitored for neutron exposures.

3.1.4 Objective 4: Consistency among Site Profiles

SC&A reviewed the site profile with respect to Objective 4, which requires SC&A to identify common elements within site profiles completed or reviewed to date, as appropriate. In order to accomplish this objective, SC&A reviewed the previous analysis of consistency and documented the modifications. These modifications were compared to assumptions at other sites.

3.1.5 Objective 5: Regulatory Compliance

SC&A reviewed the site profile with respect to Objective 5, which requires SC&A to evaluate the degree to which the site profile complies with stated policy and directives contained in 42 CFR Part 82. In addition, SC&A evaluated the TBD for adherence to general quality assurance policies and procedures used for the performance of dose reconstructions.

3.2 ORGANIZATION OF TBD

The Savannah River TBD, Revision 03, is divided into six sections, including an introduction, site process descriptions, and sections addressing internal dose, external dose, occupational medical occupational dose, and environmental occupational dose, as they pertain to historic occupational radiation exposure of SRS workers.

Section 1.0, "Introduction," explains the purpose and the scope of the site profile. SC&A was attentive to this section because it explains the role of the site profile in support of the dose reconstruction process. During the course of the review, SC&A was cognizant of the fact that neither EEOICPA nor 42 CFR Part 82, which implements the statute, requires the site profile. NIOSH developed site profiles as a resource for the dose reconstructors. Based on information provided by NIOSH personnel, SC&A understands that site profiles are living documents, which are revised, refined, and supplemented with TIBs as required, to help dose reconstructors. Site profiles are not intended to be prescriptive nor necessarily complete in terms of addressing every possible issue that may be relevant to a given dose reconstruction. Hence, the introduction helps in framing the scope of the site profile. As will be discussed later in this report, NIOSH may want to consider including additional qualifying information in the introduction to this and other site profiles describing the dose reconstruction issues that are not explicitly addressed by a given version of a site profile.

Section 1.0, as supplemented by Appendix A of the TBD, also includes process and activity descriptions of operations at SRS. This portion of the TBD is extremely important because it provides an overview of operations occurring on site through time and ultimately identifies sources of potential exposure.

Section 2, “Occupational Medical Dose,” describes the methodology used reconstruct the medical exposures received by workers as a requirement for employment at SRS. SC&A reviewed this section for technical adequacy and consistency with other NIOSH procedures and other site profiles.

Section 3, “Occupational Environmental Dose,” describes the methodology used to determine internal and external environmental dose at SRS. Environmental dose is assigned to unmonitored workers who were not likely to receive radiation exposure. Exposure of this type resulted from routine and episodic airborne emissions, resuspension or exposure to contaminated soil, and ingestion of contaminated food growing onsite. SC&A reviewed this section from the perspective of the source terms and atmospheric transport, deposition, and resuspension models used to derive the external and internal exposures to these workers. A section considering dose from the ingestion of contaminated foodstuffs was added to Revision 3 of the TBD and considered in the review.

Section 4, “Occupational Internal Dose,” presents background information on in vitro and in vivo monitoring and air sampling at SRS. It describes methods of dose assignments from personnel monitoring data as well as the uncertainties associated with these data. An efficiency method referred to as the high-five approach is briefly described, as are the interference and uncertainties associated the internal monitoring. The detailed description of this high-five approach, which is a hypothetical intake, is available in ORAUT-OTIB-0001, *Technical Information Bulletin: Maximum Internal Dose Estimates for Savannah River Site (SRS) Claims*, dated July 15, 2003 (Brackett 2003).

Section 5, “Occupational External Dose,” presents background information on beta, photon and neutron external monitoring at SRS. This section includes information on adjustment factors to be applied to SRS external dose values and uncertainties associated with these values.

Section 6, “Trades workers,” is reserved in Revision 3 of the TBD. NIOSH has issued ORAUT-OTIB-0052, *Parameters to Consider When Processing Claims for Construction Trade Workers* (Chew et al. 2006), for determination of dose to these workers. This procedure is under evaluation in a separate task.

3.3 REVIEW OF REVISION 03 OF TBD

The Advisory Board tasked SC&A with reviewing Revision 03 of the SRS TBD. This coincided with the comment resolution process. The Advisory Board formed a Work Group to guide discussions between NIOSH and SC&A on outstanding issues raised in the review of Revision 2 of the TBD. SC&A prepared a matrix containing a summary of 16 issues identified in its review of Revision 02 and released it shortly after issuing the review. NIOSH had an opportunity to review this matrix and provide initial responses to the 16 issues. After the Work Group and SC&A received the formal response from NIOSH, the response was evaluated by SC&A and the

Work Group to determine whether the proposed actions resolved the issues. SC&A then evaluated Revision 03 of the TBD to determine whether the issue included in the matrix for Revision 02 still applied to Revision 03. For the benefit of the Work Group, SC&A prepared subsequent responses for discussion during the Work Group meetings and technical telephone calls. The Work Group, in cooperation with SC&A and NIOSH, either decided that an issue was closed or initiated follow-up actions for each issue.

At the request of the Board's Work Group, members of SC&A, NIOSH, ORAUT, and the Advisory Board conducted an onsite visit at SRS from February 28 to March 1, 2007. The purpose of the visit was to review, firsthand, what was thought to be the tank farms Fault Tree Data Bank cited in SC&A's original findings regarding deficiencies in Revision 02 of the site profile as it pertained to the SRS tank farms. The three objectives of the onsite review were to determine the contents of the Washington Safety Management Solutions (WSMS) database, compare entries in this database to those from the tank farms Fault Tree Data Bank, and determine its usefulness in dose reconstruction. The database provided for review by SRS was actually the WSMS incident database, not the tank farms Fault Tree Data Bank. WSMS is a separate database developed by a former SRS employee for safety analysis of activities in the separations area. As a secondary task, SC&A conducted an interview in the presence of ORAUT, NIOSH, and Board members specifically dealing with special tritium compounds (STCs) at the site.

Directed interviews were conducted by SC&A with a limited number of SRS workers. The information derived from these interviews was used to obtain additional supporting documentation or to substantiate comments described in the main text. A worker interview summary is not provided; however, interview information is integrated into the report and referenced.

In July 2007, SC&A provided NIOSH, ORAUT, and the Work Group with an extended matrix including interchanges between NIOSH, ORAUT, SC&A, and the Work Group. At this time, NIOSH/ORAUT was offered an opportunity to provide additional comments prior to the issuance of this report. Although this report provides a status update, outstanding action items and issues that have not been satisfactorily resolved remain in the matrix. The process of comment resolution is ongoing until each issue has been resolved. Information from the matrices and conference calls has been provided in writing to members of the working group throughout the comment resolution process. Transcripts of Work Group meetings are published on the NIOSH Web site.

Finally, it is important to note that SC&A's review of Revision 03 of the TBD and its supporting TIBs focused on the areas identified in the matrix and is not exhaustive.

4.0 ISSUE/COMMENT RESOLUTION STATUS

4.1 COMMENT 1: INCOMPLETE ASSESSMENT AND GUIDANCE PERTAINING TO RECYCLED URANIUM AND TRANSPLUTONIUM RADIONUCLIDES

The assessment and guidance pertaining to recycled uranium (RU) and some transplutonium radionuclides are incomplete. The revised site profile (Revision 03) does not contain guidelines for resolving uncertainties related to RU in ways that give the benefit of the doubt to the claimants. For example, the revised TBD does not consider internal dose contributions for plutonium, other transuranics, or fission products (e.g., Pu-239, Np-237, Tc-99, Ru-103, Rh-106, Sb-125, Zr-95, Nb-95, U-232, U-233, U-236, and U-237). Throughout the period when SRS and all other sites were producing and processing RU, few, if any, exposure limits were set for these radionuclides and no efforts were made to measure internal exposures to these isotopes among recycling workers.

4.1.1 Issue Description

4.1.1.1 Recycled Uranium Impurities

SRS processed significant quantities of RU, both from its own reprocessing plants and from other plants (DOE 1985; ERDA 1976; DuPont 1960; McCarty 2000; DOE 2001). For the SRS reactors, the major products are summarized by Boswell (2000) and reproduced below.

Table 4-1. SRS Reactor Major Products

Radioisotope	When Produced	Amount	Application
Plutonium-239	1954–1988	1000s kg	Nuclear Weapons
Tritium	1954–1988	100s kg	Nuclear Weapons
Uranium-233	1956–1968	100s kg	Breeder Reactor Development
Plutonium-238	1959–1988	100s kg	Thermoelectric Generators for Space Exploration
Plutonium-240	1958–1984	100s kg	Target Material for Transplutonium Isotopes
Plutonium-242	1964–1984	10s kg	
Cobalt-60	1956–1970	¯ 66 mega curies	Gamma Radiation Source Heat Generation Sources
Curium-244	1962–1978	¯ 12 kg	Thermal Electric Generators Target for Production of Transplutonium Isotopes
Polonium-210	1966–1969	¯ 600 g	Intense Radiation Source
Californium-252	1965–1970	21 g	Cancer Treatment, Oil-Well Logging, etc.

It is estimated that from 1959 to 1999, some 31,355 metric tons of uranium were shipped from SRS to other U.S. Department of Energy (DOE) sites, including (but not limited to) the gaseous diffusion plants in Oak Ridge, Tennessee, Paducah, Kentucky, and Portsmouth, Ohio; the Oak Ridge Y-12 Plant in Oak Ridge, Tennessee; and the Feed Materials Production Center (FMPC) in Fernald, Ohio (McCarty 2000). During this same time period, it is estimated that SRS received 54,544 metric tons of uranium from other sites, such as FMPC, DOE's gaseous diffusion plants, and the Y-12 Plant (McCarty 2000).

From 1961 to 1999, SRS processed approximately one-third of an estimated total of 250,000 metric tons of RU in the DOE complex. SRS processed uranium metals, oxides, and solutions of various assays, including depleted uranium, natural uranium, and low-enriched and highly enriched uranium. Enriched uranium was also extracted from domestic and foreign research reactor spent fuel. In addition, from 1964 to 1969, thorium was recycled to produce U-233 (McCarty 2000). During the peak period of the Cold War, SRS generated 2,000 to 3,000 drums of RU trioxide per year. During this same period of RU production and processing, approximately 300 workers handled these materials annually at SRS (McCarty 2000).

The Revision 03 site profile review discussion of radionuclides found in RU is limited to the glossary and includes only uranium isotopes (Scalsky 2005, p. 136). Guidelines are not provided for resolving uncertainties related to RU in ways that give the benefit of the doubt to the claimants. For instance, the TBD does not consider internal dose contributions from plutonium or other transuranics, or fission products for uranium area workers. Recycled uranium is recovered from reprocessing plants after it has already been irradiated in a reactor one or more times. This creates uranium with radioisotopes that are not found in unirradiated uranium. Virgin uranium contains U-234, U-235, and U-238. Recycled uranium contains all three of these, as well as other isotopes of uranium, notably U-236, and traces of certain fission products and transuranic radionuclides. While the possible list of impurity radionuclides in RU is long, the main radionuclides potentially include Tc-99, Pu-238, Pu-239, Pu-240, Np-237, U-232, U-233, and U-236 (DOE 1985).

Throughout the period when SRS and all other sites were producing and processing RU, limited or no efforts were made to measure internal exposures from the impurities in RU. A preliminary analysis of the production, flow, and disposition of RU at SRS states the following (McCarty 2000):

> *SRS workers were not routinely monitored for exposure to plutonium, neptunium, or technetium that might have been present in the recycle uranium streams.*

To further compound the problem, DOE/OR-859, *The Report of the Joint Task Force on Uranium Recycle Materials Processing* (DOE 1985), states the following:

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

Table A-2 of the SRS TBD provides a partial listing of radionuclides of concern for the 221-F Area A-Line facility, which converted depleted uranyl nitrate solution to uranium trioxide for recycling (Scalsky 2005). However, this table does not equate with radionuclides of concern recommended by a special task force on RU convened in 1985 by DOE; these radionuclides include Pu-239, Np-237, Tc-99, Ru-103, Rh-106, Sb-125, Z-95r, Nb-95, U-232, U-233, U-236, and U-237 (DOE 1985). In fact, the presence of transuranic trace contamination was a large part of the reason that the uranium enrichment plants at Oak Ridge, Portsmouth, and Paducah were granted Special Exposure Cohort (SEC) status in EEOICPA. Such trace contamination has been shown (in the following excerpt) to have the potential for significant radiation doses, if the concentrations are high enough (DOE 2000, p. 77).

Table 4-2. Estimated Bone Surface Doses from Recycled Uranium to Workers at the Paducah Gaseous Diffusion Plant

(Committed Effective Dose Equivalent – CEDE)

Average Air Concentrations	Maximum Air Concentrations
48.06 – 188 rems	599.24 – 2,238 rems

SRS defines radionuclides of concern for the air monitoring and the bioassay program as follows (WSRC 2001):

> *Although there may be many radionuclides present in a facility, typically only a few have the potential for delivering significant doses and they are usually quite obvious: uranium in uranium facilities, plutonium in plutonium facilities, and tritium and tritium facilities for example. Also, some radionuclides are important because they are relatively easily detected and can be used as tracers for the radionuclides that deliver the dose. Americium-241 in a plutonium facility is a good example. Radionuclides that deliver most of the dose and their tracers are referred to as radionuclides of concern. Air monitoring and bioassay programs are designated to detect these radionuclides.*
>
> *Radionuclides of concern are determined in the following manner: All radionuclides in a work area to which workers could be exposed are identified from waste certification records, contamination surveys, safety analysis reports, technical reports, the open literature, personal interviews, etc. The radionuclides in the area that deliver a cumulative dose fraction of more than 90% are deemed to be the radionuclides of concern and are considered for inclusion on the RWP. All other radionuclides may be ignored unless they are suitable for use as a tracer….*

Radionuclides in a mixture resulting in less than 10% of the total dose are not considered significant in terms of air sampling and bioassay monitoring unless these radionuclides serve as a tracer for significant dose-producing radionuclides. In the case of an operational dosimetry program, this is justifiable as long as the site meets the intent of the regulations. In terms of a compensation program, the additional dose must be accounted for as different radionuclides concentrate in different organs of the body. Prior to excluding a radionuclide from analysis, it should be investigated in the context of all potential organs of interest.

Crase and LaBone (2000) evaluated the dose fractions from impurities, such as plutonium in RU, for materials processed and handled in SRS facilities. The source term was derived from a detailed assessment of the radionuclide mix in the 221-H waste stream (Elliott 1997). The relative activities of the radionuclides were normalized to an activity fraction and dose conversion factors were applied to the activity fractions to determine the committed effective dose equivalent (CEDE). The analysis used maximizing internal dose assumptions. Crase and LaBone (2000) summarized dose contributions from impurities in RU as follows:

> *Dose fractions calculated from the radioisotope mix for the SRS uranium recovery facilities indicate that impurities do not contribute a significant fraction of the total dose. For the enriched uranium recovery facility, the total dose fraction due to impurities was less than 8%, assuming intake parameters that would maximize the internal dose contribution from impurities. For intake parameters that would maximize the internal dose from all radionuclides (including uranium), the impurity dose contribution is much less than 1%. In the depleted uranium recovery facility, impurities could contribute up to a maximum of 16% of the total dose, again assuming intake parameters that would maximize the internal dose from impurities. For intake parameters that would maximize the internal dose from all impurities (including uranium), the dose contribution from all impurities is much less than 1%. In none of the cases did any single radioisotope contribute as much as 10% of the total dose. Even using these conservation assumptions, the results support the SRS internal dosimetry practice of not monitoring SRS uranium workers routinely for plutonium and other actinides.*

The site clearly recognized the presence of impurities in RU. Crase and LaBone (2000) indicate that their analysis may not have been applicable to RU that may have been shipped to other nuclear facilities for additional processing or mixing. Based on the analysis completed by Crase and LaBone (2000), McCarty (2000) concluded the following:

> *No evidence was found during the course of this study, which would indicate SRS recycled uranium presented any unusual challenge to radiation protection measures historically used at the site.*

This assertion does not inspire confidence that individual doses from trace contaminants in RU may not have been considerably higher. Data on fission product and transuranic impurities handled by workers are sparse at best (McCarty 2000):

> *No authenticated copies of procedures from the majority of the processing period* [involving the processing of recycled uranium] *exist outside of the Records Management system, if they exist there.*

Reconstruction of doses to workers processing RU is made even more difficult because most of the laboratory personnel who performed analytical work on RU prior to the 1970s have long since retired. Thus, knowledge of changes in technology and analytical techniques, particularly during the 1950s and 1960s, is scant.

A preliminary review of historical records indicates that estimates of contaminant concentrations by Crase and LaBone (2000) of transuranic and fission product contaminants in RU could be tenuous. A DOE task force on RU reported in 1985 that the following occurred since the inception of the recycling program (DOE 1985):

- There never existed "formal specifications on maximum permissible contaminant levels between reprocessing, intermediate and customer sites." Rather, "...informal specifications in the form of 'gentlemen's agreements' did evolve and have been in use since."

- Trace contaminant levels were increased without the proper review and concurrence that would have been required under formalized specifications. For example, in 1976, the maximum alpha activity specification from all transuranic elements of 1,500 disintegrations per minute (dpm) per gram adopted by SRS in 1960 of total uranium was informally raised to 3,000 dpm/g uranium for shipment to the Fernald facility because of "the difficulty being experienced at SRP in attaining the 1,500 dpm g U specification."

- "Early SRP (1964–1972) returns [toY-12] based on 144 samples," found that 10 samples exceeded the "gentlemen's agreement," with the highest at 180%. "Sample results over the most recent eight-year period (spanning 214 samples) indicate that 22 samples exceeded the informal specifications (the highest was 165%). It should be noted that SRP does not analyze for beta activity or recognize a beta specification."

The omission of transuranic and fission product isotopes from consideration in analyzing dose records of workers who handled RU may be a significant gap in internal dose for uranium facility workers, notably in those areas that were considered to have a "high potential" for worker contact with RU, including the following:

- The FA-Line facility (in the 200 Area) in which uranium from the radiochemical separations operations was converted to trioxide. Workers involved in facility cleanup and removal of uranium trioxide (UO_3) from the denitrator may have had the greatest contact with respirable RU particles.

- Building 321-M, where casting and machining of RU was performed. In addition, building exhaust high-efficiency particulate air (HEPA) filter change-out activities may have also created a high potential for high airborne concentrations of RU.

In summary, Revision 03 does not take into consideration transuranic and fission product contaminants in RU. Proposed Revision 04-E (Scalsky 2006) includes a discussion of RU and estimated impurities. NIOSH also indicated that a TIB concerning RU in the complex is in draft, but the document was not available for review.

Dose reconstructors are instructed to include exposures from uranium impurities based on activity fractions in Table 4.6, ORAU-TKBS-0003, Revision 04-E (Scalsky 2006), for individuals having no impurity-specific bioassay (e.g., plutonium, neptunium). SC&A notes that Table 4.6 of ORAU-TKBS-0003, Revision 04-E (Scalsky 2006), contains estimates for radionuclide impurities in RU based on estimates in waste streams (Crase and LaBone 2000). This table is derived from concentrations measured in waste streams and appears to be at odds with significantly higher contaminant levels set as product specifications at SRS for recycled uranyl hexahydrate (UNH) metal and low-enriched uranium, which were reported by the DOE 1985 Joint Task Force on Recycled Uranium (DOE 1985, pp. 75–76).

The Task Force found that several shipments of RU from SRS exceeded contaminant specifications by as much as 180 percent. (DOE 1985, pp. 47–48). The draft narrative in Revision 04-E (Scalsky 2006) does not address the processing of recycled UNH. It may be more claimant favorable and realistic to use impurity concentrations from shipments from SRS that exceeded product specifications to receiving sites. In view of this significant uncertainty, a further investigation of RU source term data should be completed to determine upper bounds of impurity concentrations and resulting doses. Other assays, such as metallurgical analyses, may assist in determining concentrations and relative uncertainties in these values.

4.1.1.2 Pu-242 Programs

While the primary products produced at SRS were plutonium and tritium, a variety of other isotopes were produced during the transplutonium program and for nonmilitary commercial uses. The transplutonium program started in the late 1950s and included the production and processing of Cf-252, Pu-242, Cm-244, and Am-243. The Curium I campaign produced Pu-242 from Pu-239 using plutonium-aluminum assemblies. During the Curium II campaign, the material from Curium I was separated and purified. The plutonium was then refabricated into fuel and irradiated further to ultimately produce Cm-244. Because a high neutron flux was required to produce transplutonium isotopes, the site established the High Neutron Flux program in support of the curium programs. Furthermore, the High Neutron Flux program also resulted in the production of high specific-activity Co-60. Other activities at SRS included the thorium campaigns for the production of U-233 and the heat source programs that involved Pu-238, Po-210, and Co-60. Cobalt-60 was later found to have uses in medicine and for sterilization. Special programs involved the production of other isotopes (e.g., Tm-170, Ir-192, Eu-152 and various isotopes of lanthanum) (Reed et al. 2002). Some of these radionuclides are considered in the dose reconstruction process, while others are not.

In Revision 03, many of these isotopes are mentioned only as trace radionuclides or as a part of a routine mixture of product and/or waste. For example, Pu-242 is only mentioned as a trace contaminant (Scalsky 2005, p. 66) and not a material produced in its own right. The use of Pu-242 as a radiobioassay tracer beginning in 1981 (Scalsky 2005, p. 66) may further complicate the detection of uptakes of plutonium. If a part of the recovered tracer in some cases was actually Pu-242 present in the bioassay sample, then the reported results would tend to underestimate the other plutonium isotopes present in addition to masking any intake of Pu-242. The TBD has not analyzed these campaigns to determine their potential influence on internal and external dose, the adequacy of the monitoring program with respect to these radionuclides, and the effect of these campaigns on isotopic ratios. This may be an important gap in the TBD. This gap affects bioassay as well as in vivo count interpretation for some groups of workers. The production of Pu-242 may also affect neutron dose calculations for Pu-242 production workers, as well as those in the target fabrication operations.

A brief description of the high weight percent (wt.%) Pu-242 source term was apparently added in the plutonium bioassay section of proposed Revision 04-E (Scalsky 2006). The effect of the higher wt.% Pu 242 was investigated with respect to its impact on plutonium bioassay yield calculations; however, the high Pu-242 campaigns ended in 1967, long before alpha spectrometry was used for plutonium bioassay. The impact on yield would have been less than 10%, which is adequately accounted for in the overall geometric standard deviation of 3 used for intake uncertainty.

In summary, Revision 03 does not consider potential contributions from exposure to high Pu-242 containing plutonium. Revision 3 gives no justification for the absence of such a discussion. Proposed Revision 04-E (Scalsky 2006) contains a discussion of the production of Pu-242 during curium campaigns and provides information on the activity composition of a high Pu-242 mixture. The default assumption for plutonium remains at 10-year-old 12% plutonium. SC&A will review proposed Revision 04-E (Scalsky 2006) of the TBD when it is released, if so directed by the Board.

4.1.1.3 Thorium/U-233

As a part of the breeder reactor program, U-233 was produced by the irradiation of thorium slugs. By 1956, U-233 was produced in small quantities. The production of U-232 impurities made fuel fabrication difficult because of the high gamma dose rates. This separation of fuel and target reduced the formation of U-232 impurities in 1965 to 3–6 parts per million. Several hundred kilograms of U-233 were produced through 1968. The Thorex process was used to recover U-233 and thorium from irradiation thorium targets at the F- and H-Canyon (WSRC 2000).

Revision 03 of the TBD provides a brief description of the U-233/thorium program in connection with the H-Canyon facility. Thorium was processed to recover the U-233. Attachment A identifies thorium as a radionuclide of concern at the $^{238}PuO_2$ Fuel Form Facility and the $^{238}PuO_2$ Experimental Facility. U-233 is not mentioned as a radionuclide of concern for any facility. In fact, the TBD acknowledges that the *SRS Internal Dosimetry Technical Basis Manual* lists Th-228 and Th-232 as radionuclides of concern (WSRC 2001). Information on thorium monitoring is limited. The TBD does not include guidance on the assignment of internal dose from thorium.

Substantial revisions are proposed in Revision 04-E (Scalsky 2006) of the TBD. Changes include a description of the fabrication of thorium slugs, a discussion on exposure to U-233 including impurities, and an added discussion of exposure to thorium including default intake assumptions. SC&A believes that this additional information will be helpful but provides the following cautionary notes.

With respect to thorium, it appears to SC&A that the 0.8 curie (Ci) of Th-228 per 1 Ci of Th-232 cited in this in Revision 04-E (Scalsky 2006) material may not be correct. It should be one-to-one because Th-228 is not separated from Th-232 by any of the processing. For Ra-228, the ratio of 0.3 to 1 may be reasonable, but NIOSH needs to provide a better justification for this (i.e., why not 0.5? or 0.2?). NIOSH should indicate whether it has any data on the delay between storage and reprocessing or recovery from reprocessing and fuel fabrication.

Thorium urinalysis was seldom performed in the 1950s and 1960s (Scalsky 2006, p. 74), and the method used was not reliable. In fact, urinalysis was not a good monitoring tool until recently, when inductively coupled mass spectrometry started to be used, with very low limits of detection. The SRS site profile TBD (Scalsky 2006) states that in vivo monitoring was the main monitoring method. It also mentions that, in 1990, thorium was measured through the 106 L-x-ray (abundance 0.12%). This is only possible using very pure germanium detectors. The abundance, however, is very small. This type of measurement is difficult and rarely done in most laboratories, even today. SC&A does not believe it was possible to measure Th-232 this way in 1966–1969. Thus, it may not make sense to apply the minimum detectable activity (MDA) from the 1990s for 1966–1969.

Thorium is, in general, measured through its daughters. The best daughter to measure is Ac-228 because it comes before Rn-220. Rn-220 makes equilibrium assumptions more problematic. The measurement of Ac-228 requires the knowledge of equilibrium factors between Th-232, Ra-228, and Ac-228. Another problem is that Ra-228 leaves the lung faster than Th-232; thus, the assessment of intakes based on measurements of Ac-228, or the other daughters, can be underestimated by a large and unknown amount (up to two orders of magnitude for Type M).

In summary, in vivo bioassay monitoring results for thorium should be analyzed very carefully. Errors of two orders of magnitude can be made, depending on the material type, equilibrium assumptions, and time of measurement after intake. Urinalysis results should also require a very cautious analysis, including the influence of natural thorium in the diet.

With respect to the added discussion on U-233 (including impurities in uranium), NIOSH may need to do more to check the reasonableness of the impurity ratio assumptions. Assuming U-233 is like U-234 is most likely reasonable, although the half-lives are different.

4.1.1.4 Trivalent Actinides

Bioassay for trivalent actinides (americium, curium, and californium) was not available until the mid-1960s. The exposure potential for these radionuclides predates the development of a bioassay method. For example, one of the first incidents at SRS in September 1954 involved a spread of contamination from an americium source (Nichols et al. 1954). Product and waste streams likely contained concentrations of Am-241. The lack of monitoring data from years prior to bioassay monitoring brings into question whether all intakes were captured.

Section 4.1.2 of Revision 03 provides information on trivalent actinide monitoring, including minimum detectable activity for Am-241, Np-237, Cm-244, and Cf-252 since 1994. Gross alpha bioassay results prior to 1994 are used to assign dose from trivalent actinides. In the absence of specific information on the principal radionuclide, the dose reconstructor is told to assume Am-241 (Scalsky 2005, p. 65). This provides a methodology for reconstructing dose when monitoring data are available. Potential unmonitored exposures have not been considered.

Furthermore, proposed Revision 04-E includes modifications. SC&A finds that the statement in proposed Revision 04-E (Scalsky 2006) on page 36 at the bottom and on page 37 at the top, which indicates that the majority of exposure is in the neptunium cycle, does not appear to be justified by any data. According to the TBD, the exposure it is from plutonium and cerium.

On page 74 (Scalsky 2006), the discussion of neptunium MDA includes apparently inconsistent statements. First, the MDA is cited as 0.035 dpm/liter (L), and from a "personal communication," it is cited as 0.4 dpm/L. Later in the discussion, it is indicated that it is acceptable to use numbers less than 0.4 dpm/L, as reported.

NIOSH has acknowledged that the history of potential exposure to neptunium and curium prior to the 1960s requires more research. It was further noted by NIOSH/ORAUT that the activity of these elements would have been less than in the 1960s and would have involved only a few workers associated with feasibility studies. The basis for assumptions on the neptunium cycle should be documented. Consideration also needs to be given to americium and californium.

Numerous special radionuclides were handled at SRS, ranging in quantities from fractions of a gram to kilograms. Many of the sources produced were encapsulated and therefore posed primarily an external hazard. Some of the special radionuclides handled at SRS included Po-210, Co-60, Cf-252, Tm-170, Ir-192, Eu-152, and various isotopes of lanthanum (Reed et al. 2002). Revision 03 of the SRS TBD gives inadequate or no consideration to potential exposures and missed dose from these radionuclides. It does not discuss the implementation of monitoring techniques for these radionuclides.

Revision 04-E (Scalsky 2006) of the TBD adds the following information related to special radionuclides production and the High Neutron Flux program.

> *Starting in 1955 the reactors were used to produce megacuries of ^{60}Co for industrial and medical uses. From 1965 through 1970, C and K reactors were modified to be able to produce a very high flux of neutrons for production of unusual radionuclide sources, referred to as the High Flux program. Curies to megacuries of an estimated 150 different radionuclides were produced (Bebbington 1990, DuPont 1965, Gray 2006). These sources were encapsulated prior to irradiation and shipped offsite so potential for intake was minimal; the sources were inspected and loaded into casks under water to reduce the external exposure (Gray 2006).*

This statement seems to indicate that internal dose will not be calculated unless specific situations are noted in the dosimetry file. This will be further evaluated by SC&A at the finalization of Revision 04-E (Scalsky 2006).

4.1.1.5 High Neutron Flux Programs

The High Neutron Flux program involved the irradiation of targets to produce transplutonium radionuclides, such as curium and californium. Furthermore, the High Neutron Flux program also resulted in the production of high specific-activity Co-60. Revision 03 provides no guidance on how these radionuclides should be considered in dose reconstruction.

NIOSH added a paragraph to proposed Revision 04-E of the site profile indicating that sources in the High Neutron Flux program were sealed and shipped off site after removal from the reactors. The proposed TBD (Scalsky 2006, p. 25) states the following:

> *…these sources were encapsulated prior to irradiation and shipped offsite so potential for intake was minimal:….*

This seems to indicate that there was little potential for intake from the various "exotic" radionuclides made at SRS during the High Neutron Flux program. This does not seem to take into account incidents or exposure during encapsulation. As was the case in the Y-12 SEC evaluation, records may be available that would serve to characterize how these "other radionuclides" were handled, by whom, and in what amounts. The identification and retrieval of such records would shed light on this potential source of exposure. NIOSH should look more carefully at additional sources, such as monthly progress reports, "Bebbington 1990," and "Gray 2006," which discuss such "exotics." It is clear that there were neptunium, curium, and americium exposures (Du Pont 1965).

4.1.2 Applicability to Revision 03

Revision 03 includes a new Section 4.1.2 on bioassay, but Comment 1 remains substantially the same for the updated TBD. Several updates proposed for Revision 04-E (Scalsky 2006) of the TBD deal directly with this issue. This issue remains open pending review of Revision 04-E of the TBD, when it is issued, and ORAUT-OTIB-0053, *Dose Reconstruction Considerations for Recycled Uranium Contaminants* (ORAUT 2005).

4.1.3 Summary of Overall NIOSH Response

A table providing impurities to be included with intakes of uranium has been drafted for the next revision of Chapter 4 of the SRS TBD. This table will provide the six impurities with the largest impact on dose, with different impurities for depleted uranium or natural uranium and enriched uranium. The impurity values were obtained from *Historical Generation and Flow of Recycled Uranium at the Savannah River Site* (McCarty 2000), which relied heavily on *Dose Contribution from Plutonium and Other Impurities in Uranium in Waste Streams from Savannah River Site Uranium Recovery Facilities* (Crase and LaBone 2000). In general, the SRS impurities are lower than those used for Hanford. Neither set of impurities adds much to the internal organ doses. In addition, NIOSH is developing ORAUT-OTIB-0053 (ORAUT 2005) to provide generic guidance on RU. NIOSH indicated in its initial issue resolution response that this Oak Ridge Associated Universities Team Technical Information Bulletin (OTIB) would be released in December 2006.

A brief description of the higher wt.% Pu-242 source term was added in the plutonium bioassay section. The effect of the higher wt.% Pu-242 was investigated for its impact on plutonium bioassay yield calculations; however, the high Pu-242 campaigns ended in 1967, long before alpha spectrometry was used for plutonium bioassay. The impact on yield would have been less than 10%, which is adequately accounted for in the overall geometric standard deviation of 3 used for intake uncertainty.

Bioassay for americium, curium, and californium was in place during the mid-1960s.

The first irradiation of targets to produce Co-60 appears to have occurred in August 1956. Those targets were encapsulated and after irradiation were shipped directly to Oak Ridge National Laboratory. Hence, there was little chance for intake at SRS. Work began on a urinalysis procedure for Co-60, Fe-59, Zn-65, and Cr-51 in November 1956, but the impetus appears to be normal reactor activation products, not special targets. This procedure appears not to have been used routinely until 1961; that is, its use was event driven. Whole-body counts starting in December 1960 do not list Co-60 as a frequently detected radionuclide despite its easy detection relative to other fission/activation products that were listed. The targets were encapsulated and Co-60 was not often seen in whole-body counts, leading to the conclusion that the Co-60 source program was not a source of unmonitored intakes.

A description of the thorium/U-233 operations will be added to Section 4.1.2 of the Revision 04-E (Scalsky 2006) of the SRS TBD, along with some description of the bioassay and instructions for incorporating default thorium intakes for M-Area and canyon buildings workers.

With regard to radionuclides from the High Neutron Flux program, whole-body counting would have detected intakes of most of the special radionuclides, but it is agreed that it would not have been appropriate for the beta-only or low-energy photon-emitting radionuclides, Tm-170 being most notable. The reason that specific in vitro bioassay analyses were not developed for Tm-170 and like sources is that the preirradiated source material was in welded aluminum cans; after irradiation, the cans were placed directly into a shielded shipping cask under water. The sources were shipped to customers without being opened. The potential for intakes was low.

NIOSH has added a paragraph to proposed Revision 04-E (Scalsky 2006) of the site profile indicating that sources in the High Neutron Flux program were sealed and shipped off site after removal from the reactors. Because the sources were encapsulated prior to irradiation and shipped offsite, NIOSH indicates there was minimal potential for intake. This does not seem to take into account incidents or exposure during encapsulation. As was the case in theY-12 SEC evaluation, records may be available that would serve to characterize how these “other radionuclides” were handled, by whom, and in what amounts. The identification and retrieval of such records would shed light on this potential source of exposure.

4.1.4 Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Provide a timeline for the development and release of the RU OTIB (pending)
- Provide intended approach for addressing special nuclides
- Provide additional information on thorium and U-233 and the intended approach for addressing this issue

- Provide justification for the ability to link to the bioassay program and substantiate extrapolation back to the earliest timeframe

- Provide information regarding the monitoring of uranium at SRS

- Provide an intended approached for addressing potential intakes from the Pu-242 program

- Conduct further research as indicated and appropriate regarding potential exposure to trivalent actinides

SC&A Actions:

- Provide a timeline for the review of the RU OTIB when it becomes available (pending)

- Review discussions added to the revised site profile (Revision 04E) when available

4.1.5 Closure Status

The assessment and guidance on dose assignment is incomplete pertaining to RU, Pu-242, transplutonium radionuclides, thorium, and U-233 in Revision 03 of the TBD. Section 4.1.2 on bioassay was added in Revision 03, but the issues related to these radionuclides have not been adequately resolved. Several updates proposed for Revision 04-E of the TBD deal directly with exposure from these radionuclides and bridge some of the gaps found in previous versions of the TBD. This draft revision will respond to several of the NIOSH actions items requested by the Work Group. It is recommended that this overall issue remain open pending review of yet-to-be-released Revision 04-E (Scalsky 2006) of the TBD and ORAUT-OTIB-0053 (ORAUT 2005).

4.2 COMMENT 2: BETA/GAMMA ADJUSTMENT AND UNCERTAINTY FACTORS

The assessment of the beta/gamma dosimeter adjustment factors and uncertainties within the TBD is incomplete, resulting in a probable underestimation of doses. The TBD needs to be more specific and complete in the following areas:

- The calibration of dosimeters is often not representative of incident angles encountered in the field and, depending on exposure geometry, could result in an underestimation of the true exposure that is being measured.

- The on-phantom correction factor of 1.119 may be too low for photon energies between 30 and 250 kiloelectron volt (keV).

- The TBD’s generic standard deviation value of 30% is likely to be low for film dosimeters prior to 1971. Early film dosimeters are likely to have a workplace standard deviation of at least 40%.

- Dosimeter adjustment factors for SRS are inconsistent with DOE complex-wide TIBs.

SC&A originally identified an initial lack of guidance pertaining to shallow doses, which has since been resolved. A minor deficiency of this TBD, particularly Section 5.0, is the absence of guidance pertaining to the interpretation of open window dose, shallow dose, 7 milligrams per square centimeter (mg/cm^2) dose, and/or skin dose. SC&A has reviewed ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (Merwin 2005), which was released since SC&A's original evaluation, and found that it satisfactorily addresses shallow doses.

4.2.1 Issue Description

4.2.1.1 Dose Adjustment Factors

In the mid-1980s, SRS implemented changes in the calibration of dosimeters that replaced the previous Ra-226 source with a Cs-137 source and switched from in-air calibration of dosimeters to on-phantom. The overall change in recorded dose was assumed to require a correction factor of 1.119 for dosimeter readings prior to 1986 and 1.039 for dosimeter results during 1986 (Taylor et al. 1995). These factors are recommended as dose adjustment factors (DAFs) in the TBD. These DAFs were generated for the purpose of comparing doses across timelines (i.e., normalizing recorded doses from these earlier periods in terms of 1995 dose assignment methods). The DAFs were developed by comparing thermoluminescent dosimeter (TLD) responses during calibrations under different conditions: air versus phantom, and Cs-137 versus Ra-226 as the gamma calibration source (Taylor et al. 1995, pp. 36–37). These adjustment factors were intended for comparative purposes only; the authors did not intend to use them for amending individual dose records (Taylor et al. 1995, p. 93).

In Revision 03 of the TBD, NIOSH/ORAUT acknowledged that multiple factors may contribute to an under-response of a film dosimeter or TLD. However, they conclude that the dosimeters' over-response at low photon energies was offset by under-responses caused by calibration methods, angular response, environmental factors, and other considerations. Furthermore, they indicate that the recorded dose for all types of dosimeters employed was, in fact, a reasonable estimate of the 1,000 mg/cm^2 deep dose. Since the over-response was energy dependent and was limited to the two-element film used from 1951–1959, SC&A did not agree that it would offset the many factors contributing to under-response. It appeared likely that doses recorded after 1959 and not impacted by the over-response phenomenon would be significantly underestimated.

Hine and Brownell (1956) have evaluated backscatter and concluded that it depends in a complex way on (1) the energy of the radiation, (2) the area of the field, and (3) thickness of the scattering medium. The percentage of backscatter may be as high as 50% for a large field, adequate thickness, and select photon energy. The data indicate that for photons with half-value layer between 0.6 millimeters Copper (mm Cu) and 1.0 mm Cu (or approximately 60–80 keV), the backscatter factor for a dosimeter worn on the upper torso of an adult could reach a value of about 1.5. Such a backscatter factor would apply to dose conversion factors (DCFs) with photon energies between 30 keV and 250 keV, which is commonly assumed for SRS workers.

SC&A reviewed the available references relevant to beta/photon dosimetry at SRS. SC&A also reviewed Revision 03 and proposed Revision 04-E of the SRS TBD (Scalsky 2005, Scalsky

2006) and a comparison study of DOE external dosimetry (Alvarez 2003). The following concerns are noted regarding the use of these DAFs within the TBD:

- Application to TLDs Over Time: While the methods and data used to derive the DAFs may be sound for the study's intended purpose, the resulting factors do not necessarily represent comprehensive dose adjustment factors valid for the entire period to which they are being applied. TLDs were first used for official records in April 1970; it has not been shown that one DAF of 1.119 is appropriate for the entire time span from 1970 through 1985.

- Application to Film Badges: It is feasible that a detailed analysis might produce year- and location-specific DAFs in the 5–15% range for TLDs. However, the TBD has not addressed the validity of applying these DAFs for film badges. One DAF of 1.119 (derived from TLD data) would not appear appropriate for film-based dosimetry used at SRS from 1952 through 1969. This issue is not addressed in Revision 03 or the draft Revision 04-E (Scalsky 2005, Scalsky 2006) of the SRS TBD.

- Impact on Coworker Data: ORAUT-OTIB-0032, *External Coworker Dosimetry Data for the Savannah River Site* (Merwin 2006), lists the recommended coworker doses to be assigned to unmonitored or under-monitored workers at SRS. These data were taken from the Health Protection Annual Radiation Exposure History (HPAREH) database and adjusted by the correction factor of 1.119 prior to 1986 (Merwin 2006). Apparently, the 1.039 adjustment factor for 1986 was not taken into account. The use of a single adjustment factor for film and TLD dosimeters from 1952 through 1985 will affect the dose assigned to unmonitored workers, as well as the monitored workers, and could result in an underestimate of their doses.

- Limitations of DAFs: Section E.4.1.2 of the TBD (Scalsky 2005, Scalsky 2006) states that adjustments are needed to account for uncertainties associated with complex workplace radiation fields and exposure orientations. The DAFs of 1.119 and 1.039 are assumed to address this need. However, it is important to realize that a DAF mainly corrects for bias in the calibration system/processing and not for other variables and uncertainties. Section 4.2.1.2 of this report addresses these uncertainties.

The uncertainty identified for adjustment factors to the beta/gamma dosimeter-measured dose does not entirely estimate uncertainty from all potential field exposure conditions. For this reason, other compensating sources of uncertainty must be considered, including laboratory uncertainty for measured dose, missed dose uncertainty as defined in OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2002), and, for some workplaces, uncertainty for assigned neutron dose (measured, missed, and unmonitored), as well as the Interactive RadioEpidemiologic Program (IREP) method to calculate the target organ dose [and thus the probability of causation (POC)]. Based on the overall effect of the uncertainty incorporated in this analysis, the assigned organ dose used in compensability determination does not underestimate the actual dose.

4.2.1.2 Uncertainty Factors

SC&A's searches and reviews have not yielded further qualification for the uncertainty factors described in the TBD (Scalsky 2005, Scalsky 2006). The SRS documents that SC&A has been able to find do not deal specifically with uncertainties or adjustment factors, and SC&A could not locate the "3/29/04 guide" identified in NIOSH's reference list. Additional information would be needed in order for SC&A to locate this document and evaluate its applicability to this issue.

The revised TBD addresses recorded dose uncertainties in Section 5.3.5, Table 5.3.5-1 of Revision 03 (Scalsky 2005) and Section 5.3.5, Table 5-20 of Revision 04-E (Scalsky 2006). Table 5.3.5-1 (Revision 03) is reproduced below:

Dosimeter	**SRS**	**Laboratory uncertainty**[a]	**Workplace uncertainty**[b]		
			Reactor	**Plutonium**	**Reprocessing**
Beta/gamma dosimeters					
Two-element film	Used 1951–1959	+/- 25%	+/- 50%	+/- 75%	+/- 50%
Multi-element film	Used 1960–1969	+/- 20%	+/- 40%	+/- 60%	+/- 40%
TLD	Used 1970–present	+/- 10%	+/- 20%	+/- 30%	+/- 20%
Neutron dosimeters					
NTA	Used 1951–1970	+/- 50%	+/- 100% (need to use another method)		
TLND	Used 1971–present	+/- 25%	+/- 50%	+/- 75%	+/- 75%

a. In relation to Hp(10) response of dosimeter.
b. 95% confidence interval.

These tables list laboratory uncertainties (excluding neutron track emulsion (NTA)) ranging from 10–25% and workplace uncertainties ranging from 20–75%, but they do not state what variables are addressed (i.e., energy response, mixed fields, geometry). There is no further quantitative discussion of laboratory uncertainties (e.g., calibration, chemical processing, reading); radiological uncertainties (changes in energy spectra, geometry of exposure, and other field variables); and environmental uncertainties (e.g., fading, moisture, light, temperature, chemical exposures).

The TBD continues to recommend a generic standard deviation of 30% for best estimate dose reconstruction or another appropriate value from Section 5 [Revision 03, Section 5.7.2, p. 117 (Scalsky 2005); Revision 04-E, Section 5.10.6, p. 121 (Scalsky 2006)]. It would appear that the combined uncertainties from the large range of uncertainties listed in Table 5.3.5-1 and Table 5-20 would be more in the range of 40–50% under actual working conditions, especially during the earlier years of film dosimetry, such as before 1971. For example, combining the laboratory and workplace uncertainties using simple quadratic calculations gives a range of 22% to 79% (excluding NTA), with an average of around 55%.

If an uncertainty value of 30% is to be retained in the TBD, detailed mathematical derivation of this value should be provided as it applies to SRS. Additionally, the statement that "measured doses are to be treated as a normal distribution with a standard deviation of 30% (or other appropriate value that may be provided in Section 5)" lacks clarity and guidance for the dose reconstructor. Does "other appropriate value" refer to a combination of uncertainties listed in Table 5.3.5-1 (Revision 03, Scalsky 2005) or Table 5-20 (Revision 04-E, Scalsky 2006)? If so, how are they to be combined? The uncertainty factors provided in Section 5.3.5 of the TBD are similar to those found in some of the documents from NIOSH's reference list (Attachment B of 8 November 2006), but a final combination value or mechanism is needed in order for dose reconstructors to consistently apply all of the relevant uncertainties to the dose of record.

4.2.1.3 Shallow Dose

SC&A originally identified an initial lack of guidance pertaining to shallow doses, which has since been resolved. A minor deficiency of this TBD is the absence of guidance pertaining to the interpretation of open window dose, shallow dose, 7 mg/cm^2 dose, and/or skin dose. Although these terms are defined in the glossary and mentioned in the Executive Summary of Revision 03 (see page 17, which states, "**… Section 5 presents the occupational dosimeter program for measuring skin and whole-body doses to workers**" [Emphasis added.]), there is neither a discussion of shallow dose interpretation nor a reference to ORAUT-OTIB-0017 (Merwin 2005). SC&A reviewed ORAUT-OTIB-0017 (Merwin 2005), which was released since SC&A's original evaluation, and found that it satisfactorily addresses shallow doses.

4.2.2 Applicability to Revision 03

Comments regarding adjustment factors are still applicable. ORAUT-OTIB-0017 (Merwin 2005) provides guidance for shallow dose assignment; hence, this is no longer an issue.

4.2.3 Summary of NIOSH Response

The information in the TBD regarding correction factors for beta/gamma dosimetry is based on SRS historical evaluations of dosimeter performance. NIOSH provided the following references as the historic basis for dosimeter performance:

- Atomic Energy Commission, 1955, "Intercomparison of Film Badge Interpretations," *Isotopics*, Volume 2, number 5, pp. 8–23.

- Brackenbush, L.W., G.W.R. Endres, J. M. Selby, and E.J. Vallario, 1980, "Personnel Neutron Dosimetry at Department of Energy Facilities," PNL-3213, Pacific Northwest Laboratory, Richland, Washington. 99352.

- Brackenbush, L.W., K.L. Soldat, D. L. Haggard, L. G. Faust and P. L. Tomeraasen, 1987, "Neutron Dose and Energy Spectra Measurements at Savannah River Plant," PNL-6301, Pacific Northwest Laboratory, Richland, Washington. 99352.

- Brodsky, A., and R.L. Kathren, 1963, "Accuracy and Sensitivity of Film Measurements of Gamma Radiation—Part I: Comparison of Multiple-Film and Single-Quarterly-Film Measurements of Gamma Dose at Several Environmental Conditions," *Health Phys* 9(4):453–461.

- Brodsky, A., and R.L. Kathren, 1963, "Accuracy and Sensitivity of Film Measurements of Gamma Radiation—Part II: Limits of Sensitivity and Precision," *Health Phys* 9(5):463–471.

- Brodsky, A., A.A. Spritzer, F.E. Feagin, F.J. Bradley, G. Karches and H.I. Mandelberg, 1963, "Accuracy and Sensitivity of Film Measurements of Gamma Radiation—Part IV: Intrinsic and Extrinsic Errors," *Health Phys* 11(10):1071–1082.

- Gorson, R.O., N. Suntharalingam, and J.W. Thomas, 1963, "Results of a Film-Badge Reliability Study," Presented at the 49th Annual Meeting of the Radiological Society of North America, Chicago, Illinois. November 17–22, 1963.

- Hoy, J.E., 1972, "Personnel Albedo Neutron Dosimeter with Thermoluminescent 7Li and 6Li," DP-1277, E.I. du Pont de Nemours and Company, Inc., Savannah River Plant, Aiken, South Carolina.

- Hoy, J.E., 1980, "Thermoluminescent Dosimeters for Personnel Neutron Monitoring," DPST-70-533, E.I. du Pont de Nemours and Company, Inc., Savannah River Plant, Aiken, South Carolina.

- Morgan, K.Z., 1961, "Dosimetry Requirements for Protection from Ionizing Radiation," *Selected Topics in Radiation Dosimetry, Proceedings of the Symposium on Selected Topics in Radiation Dosimetry, Sponsored by and Held in Vienna 7–11 June 1960*, International Atomic Energy Agency, Vienna, Austria, pp. 3–23.

- Parker, H.M., 1945, "Comparison of Badge Film Readings at the Metallurgical Laboratories, Clinton Laboratories and the Hanford Engineering Works," 7-3090, Hanford Atomic Products Operation, Richland, Washington. (SRS used Clinton dosimetry services and then implemented systems similar to other laboratories.)

- Plato, P., 1978, "Testing and Evaluating Personal Dosimetry Services in 1976," *Health Phys* 34(3):219–223.

- Savannah River Site, 1993, "External Dosimetry Technical Basis Manual," WSRC-IM-92-101, Westinghouse Savannah River Company, Aiken, South Carolina.

- Taylor, G.A., K.W. Crase, T.R. LaBone, and W.H. Wilkie, 1995, "A History of Personnel Radiation Dosimetry at the Savannah River Site," WSRC-RP-95-234, Westinghouse Savannah River Company, Aiken, South Carolina.

- Thierry-Chef, I., F. Pernicka, M. Marshall, E. Cardis and P. Andreo, 2002, "Study of a Selection of 10 Historical Types of Dosemeter: Variation of the Response to Hp(10) with Photon Energy and Geometry of Exposure," *Radiat Prot Dos*, 102(2):101–113. (Includes SRS Panasonic 802 dosimeter.)

- Vallario, E.J., D.E. Hankins, and C.M. Unruh, 1969, "AEC Workshop on Personnel Neutron Dosimetry," BNWL-1340, Battelle, Pacific Northwest Laboratory, Richland, Washington.

- Vallario, E.J., D.E. Hankins, and C.M. Unruh, 1971, "Second AEC Workshop on Personnel Neutron Dosimetry," BNWL-1616, Battelle, Pacific Northwest Laboratory, Richland, Washington.

- Wilson, R.H., J.J. Fix, W.V. Baumgartner, and L.L. Nichols, 1990, "Description and Evaluation of the Hanford Personnel Dosimeter Program from 1944 Through 1989," PNL-7447, Pacific Northwest Laboratory, Richland, Washington. (See Section 5.3 on intercomparison programs.)

The TBD excludes beta and nonpenetrating doses for the extremity, skin, gonads, and breast exposure in Section 5.1.

ORAUT-OTIB-0017 (Merwin 2005) provides guidance for assessing the shallow dose and nonuniform exposures. This guidance was present in the draft TBD (Revision 04-E, Scalsky 2006) at the time of this review.

4.2.4 Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Provide references on adjustment factors used. (NIOSH supplied a list of references on November 8, 2006)

SC&A Actions:

- Compare the uncertainty factors in the TBD against relevant site-specific workbooks as well as available dose reconstruction instructions (e.g., March 29, 2004, guide).

4.2.5 Closure Status

The issues associated with correction factors and uncertainties have not been satisfactorily resolved. It has not been demonstrated that the application of a DAF of 1.119 or 1.039 for both TLDs and film for 1952–1986, and an uncertainty of 30% without full consideration of laboratory, radiological, and environmental factors, is claimant favorable. The dosimeter calibration is based on an incident angle of zero degrees, which underestimates the actual field dose where incident angle is greater than zero. The correction factor applied to recorded dosimeter results is too low for photon energies from 30 to 250 keV, which is the default photon energy used.

SC&A recommended the following specific actions to achieve closure:

- Provide more detailed period/location film-specific DAFs for the period 1952–1970.

- Provide period/location TLD-specific DAFs for the period 1971–1986.
- Assess the impact of new DAFs on coworker data.
- Account for differences in incident angles between calibration and field use.
- Account for photon energies between 30 and 250 keV (the default photon energy used in calibration).
- Clarify the basis for applying a generic 30% uncertainty factor and/or provide clear instructions for applying other appropriate uncertainty factors.

Revision 03 does not provide addition information to resolve this issue. The recent draft of Revision 04-E (Scalsky 2006), which has not yet been officially issued, contains a few changes to external dose reconstruction, but these additions do not satisfactorily address the original issues; therefore, the issues are still applicable. None of the SRS site-specific workbooks and guides that SC&A has been able to locate provide further qualification for using the DAFs as recommended in the TBD (Scalsky 2006). The SRS documents that SC&A has been able to find do not deal specifically with uncertainties or adjustment factors, and SC&A needs more information on the title or other bibliographic data to locate the guide of 3/29/04 (as listed in the NIOSH response) and evaluate its applicability to this issue. SC&A recommends that this issue remain open for these reasons. SC&A concurs that ORAUT-OTIB-0017 (Merwin 2005) provides suitable guidance for the assignment of shallow dose.

4.3 COMMENT 3: INADEQUATE NEUTRON-TO-PHOTON RATIOS

Questions remain unanswered regarding the validity of neutron-to-photon ratios at SRS. The geometric mean and standard deviation that describe the post-1971 neutron-to-photon ratio are not technically sound or likely to be claimant favorable for a large number of claimants. The TLND-recorded neutron doses between 1971 and 1995, as well as the pre-1971 neutron doses (derived from neutron-to-photon ratios), suffer from a high degree of uncertainty. The use of the 95th percentile value for the TLND neutron dose of record is recommended.

4.3.1 Issue Description

Neutron dosimetry is considerably more complex and difficult to assess than beta/photon dosimetry. Difficulties in assessing neutron dose are principally the result of design limitations of past dosimeters used at SRS, and the highly variable and complex neutron spectra that workers may have encountered. The four main areas at SRS with potential for neutron exposure include the plutonium facilities in the 200 Area; the Calibration Facility (736-A) and the Cf-252 Facility (773-A) in the 700 Area; reactors in the 100 Area; and Building 321 (plutonium-aluminum alloys) in the 300 Area. These facilities differ not only in neutron energy spectra but also in terms of their neutron-to-photon dose rate ratios. The significance of the latter is highly relevant to the time period at SRS when neutron exposure was assessed by neutron track emulsion (NTA) film.

Neutron dosimeters used to monitor individual workers at SRS involved three different designs. The first involved the NTA Type A film dosimeter, which was used from August 3, 1953, through the end of 1970. This dosimeter relied on the interaction of neutrons with sensitive elements of the film to produce visible tracks. When manually counted, the number of tracks per film provides an estimate of the total neutron fluence to which a worker was exposed and from which an estimate of neutron dose is derived.

Due to the insensitivity of NTA film to neutrons with energies below 500 keV (or even 1 megaelectron volt (MeV), as reported by others), as well as other factors contributing to the dosimeter's uncertainty, NIOSH/ORAUT concluded that NTA monitoring data used from 1953 through the end of 1970 were insufficiently reliable and therefore could not be used for dose reconstruction. It was concluded that a suitable substitute for NTA neutron data was the use of facility-specific neutron-to-photon ratio data.

The SRS TLD neutron dosimeter was introduced on January 1, 1971, and was used until December 31, 1994. The TLND employed two polyethylene spheres covered with a layer of cadmium. Proper interpretation of this dosimeter requires matching the neutron energy spectrum of the calibration source with that of the workplace spectrum.

On January 1, 1995, SRS began using the commercial Panasonic neutron TLD, which detects albedo neutrons. Albedo neutrons are those reflected backwards out of the worker's body into the TLD's phosphor, where the neutron interacts with Li-6 to give an alpha particle and tritium (i.e., n + Li-6 → α + H-3). A combination of Li-7 and Li-6 phosphors, along with multiple filters, and an empirically derived algorithm allows this dosimeter to quantify exposure to betas, low-energy photons, high-energy photons, and neutrons.

In order to assign neutron doses to workers who had been monitored by means of NTA film prior to 1971, the surrogate use of the neutron-to-photon ratio method required NIOSH/ORAUT to assess the neutron-to-photon dose rate ratios for each major location that posed the potential for neutron exposure between 1953 and the end of 1970. [One location where the use of NTA film dosimeters was considered useable is the Fuel Fabrication Area (321 M Area).] NIOSH/ORAUT employed empirical, location-specific neutron-to-photon ratios that were found to represent a lognormal distribution. These data could then be used to estimate neutron exposures on the basis of (1) recorded photon doses and (2) photon doses recorded as zero (i.e., missed photon doses).

Starting in January 1971, neutron doses were monitored and recorded by means of the SRS Hoy TLND and the Panasonic TLD. NIOSH/ORAUT regards monitoring data for these dosimeters as "reasonably accurate" and therefore, useable for dose reconstruction, but not without "adjustment." Since 1971, neutron doses recorded by TLDs were based on neutron quality factors in National Council on Radiation Protection and Measurement (NCRP) 38, *Protection Against Neutron Radiation* (NCRP 1971), which assigned specific values to discrete neutron energy intervals. Neutron quality factors defined in NCRP 38, however, have been updated by weighting factors in International Commission on Radiation Protection (ICRP) 60, *Recommendations of the International Commission on Radiological Protection* (ICRP 1990). In compliance with 42 CFR Part 82, NIOSH/ORAUT evaluated the neutron energy spectra at each of the major locations and provided corresponding location-specific neutron correction factors that account for revised neutron quality factors for post-1971 recorded neutron doses.

4.3.1.1 Neutron-to-Photon Ratio Method

The TBD prescribes two very different protocols for neutron dose reconstruction that correspond to pre- and post-1971 time periods. SC&A's review comments are therefore directed to each of these methods separately.

SC&A reviewed the scientific literature regarding the use of NTA film dosimeters and agrees with the decision not to use NTA film data in dose reconstruction. SC&A further agrees with the use of the neutron-to-photon ratio method as a reasonable surrogate, but only on a conditional basis, as explained below.

Of concern are the limited data that were used and the interpretation of such data for defining location-specific neutron-to-photon ratios. Table 4-3 below summarizes surrogate post-1971 data that are to be used for neutron dose reconstruction prior to 1971. Values in bold indicate changes proposed for Revision 04-E (Scalsky 2006).

Table 4-3. Neutron-to-Photon Ratio Values Used as Surrogate Data for NTA Film Dosimeters

Areas/Process	Neutron/Photon Ratio		Neutron/Photon Ratio		
	Avg.	Range	Geometric Mean	Geometric Standard Deviation	95th %
100 Area—Reactors	0.26	(0.05–0.62)	0.18	2.52	0.82
Plutonium Production:	0.52	(0.09–1.23)	0.91	2.84	5.05
- HB-Line			**1.0**	**2.0**	**3.1**
- FB-Line	1.29	(0.05–3.10)	0.36	2.52	1.65
			1.0	**2.0**	**3.1**
Radionuclide Production and Calibration	0.85	(0.10–3.83)	0.62	2.29	2.41

Source: Scalsky 2005

Table 4-2 shows that not only are there large differences in neutron-to-photon ratios among the four general areas, but even larger differences exist within a given area, as indicated by the wide range of ratio values. For example, at the FB-Line, observed ratio values, which range from a low of 0.05 to a high of 3.1, differ 62-fold. The observed wide range of neutron-to-photon ratios is clearly the aggregate of the following independent uncertainties of the post-1971 TLND neutron dosimeter:

- The uncertainty of the post-1971 TLD photon dosimeter
- The variability of the neutron-to-photon ratios among locations within a given area, such as the FB-Line

In addition to these uncertainties are two more uncertainties that contribute to the actual pre-1971 neutron dose. The third uncertainty is the pre-1971 photon dose (which must be multiplied with the post-1971 neutron-to-photon ratio), and the fourth uncertainty is the unfounded assumption that a post-1971 neutron-to-photon ratio at any of the four general areas is representative of the pre-1971 neutron-to-photon ratios. This assumption would only hold true if all processes, production quantities, engineering controls, radiological practices, and other factors during the assessed post-1971 era were, in fact, identical/comparable to those that existed between 1953 and 1970. Table 4-4 summarizes the uncertainties that collectively define the overall uncertainty of pre-1971 neutron doses that are derived by the photon-to-neutron ratio method.

Table 4-4. Uncertainties Contributing to the Derivation of Neutron Dose by the Neutron-to-Photon Ratio Method

Source of Uncertainty	Workplace Uncertainty
(1) Pre-1971 photon dose: - Two-element film	±50% to ±75% [a]
- Multi-element film	±40% to ±60% [a]
(2) Post-1971 TLND dose	±50% to ±75% [a]
(3) Post-1971 photon dose	±20% to ±30% [a]
(4) Neutron-to-photon ratio By locations within area	Unknown
(5) Neutron-to-photon ratio before 1971 Versus measured neutron-to-photon Ratios post-1971	Unknown

[a] Source: Table 5.3.5-1 in ORAUT-TKBS-0003

SC&A concludes that the surrogate use of the neutron-to-photon ratio method encompasses three large/quantifiable and two nonquantifiable uncertainties. Together, these uncertainties preclude the use of guidance, as given in Section E.4.1.6 of Attachment E of the TBD, which states the following:

> *Prior to 1971, . . . using a ratio of the potential neutron dose to the measured photon dose is done as a claimant-favorable option to reconstruct an individual worker neutron dose . . . As can be determined from [Table] E-9, the recommended method to apply the ratio is as a lognormal distribution using the* ***geometric mean*** *and* ***geometric standard deviation****.* [Emphasis added.]

SC&A believes that the use of the geometric mean and geometric standard deviation that describe the post-1971 neutron-to-photon ratio is neither technically defensible nor likely to be claimant favorable for a large fraction of potential claimants. A claimant-favorable alternative is to use the 95th percentile neutron-to-photon ratio as a point estimate for all claimants regardless of compensability of the claim.

4.3.1.2 Performance Characteristics of the TLND

Closely linked to the issues identified above is SC&A's second concern about the use of the TLND in its other role as the neutron dosimeter of record between 1971 and 1995. In part, Section 5.3.4.1.2 of the TBD explains the decision to accept the TLND data:

Trends in the annual SRS and Hanford neutron collective dose (Taylor et al. 1995; Buschborn and Gilbert 1993, respectively), normalized to the annual plutonium production (DOE 1996), are illustrated in Figure 5.3.4.2-2. ***It is evident in this figure that the collective neutron dose was under-recorded prior to implementation on 1 January 1971 of the SRS TLND and 1 January 1972 for the Hanford TLD. The extent of the under-estimate is difficult to estimate. SRS and Hanford showed a significant increase in the ratio of the annual collective neutron dose to the annual plutonium production when the TLD neutron dosimeters were implemented.*** [Emphasis added.]

Figure 5.3.4.2-2 referenced above is reproduced below as Figure 4.1. Data shown in Figure 4.1 do not support either statement emphasized in the quotation above:

- At both SRS and Hanford, the rise in collective dose (supposedly standardized to plutonium production) began well before the advent of the TLND; and, in both cases, the standardized collective dose dropped precipitously after the implementation of the TLND with subsequent fluctuations.
- Because the collective neutron doses were standardized (i.e., defined in person-rem per unit quantity of plutonium), the observed oscillations clearly indicate that the collective dose is not correlated with or linked to plutonium production but may very well be the result of variations in neutron fields that surround work conditions in a given area and the variable response of the TLND.

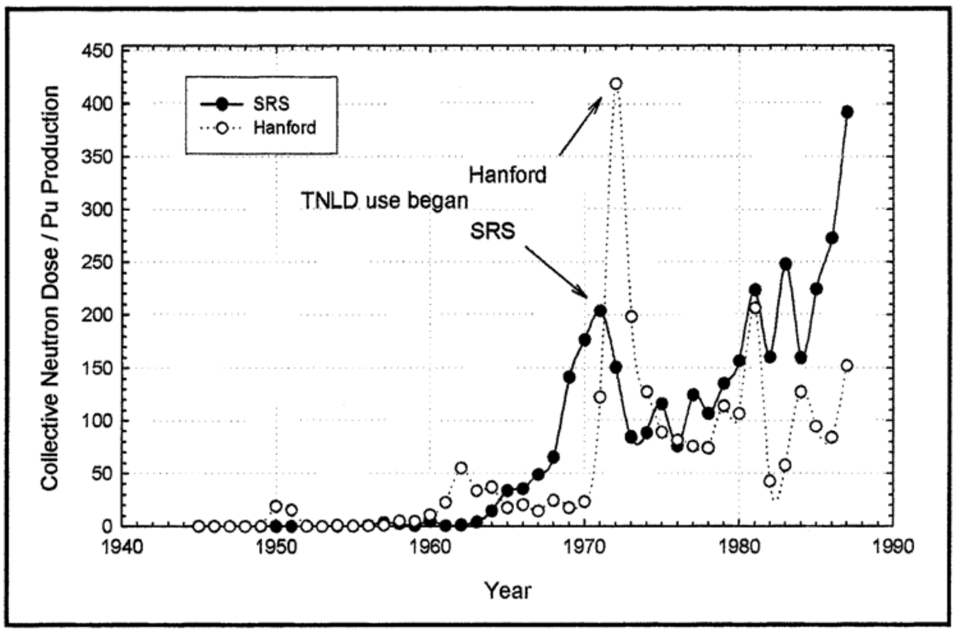

Figure 4-1. Trends in SRS and Hanford Collective Neutron Dose Normalized to Plutonium Production

Section 5.3.5 of the TBD discusses the uncertainty of the TLND's response and includes the following statements:

> *As reported in the PNNL report, measurements with the TEPC, multisphere system and ^{3}HE spectrometer were in general agreement. The TLND agreed within about 30% for most measurement locations along the plutonium production lines and storage areas. The TLND was within a factor of 3 (i.e., 0.3 to 3) for the extremes in neutron energy spectra encountered at the K-reactor door (i.e., highly thermalized field) and for a californium shipping cast (i.e., where most lower energy neutrons had been removed). Over long time periods, workers would generally be expected to be involved in several different exposure profiles that will serve to minimize the extremes identified. These results are indicative of the technical difficulties to accurately measure neutron dose in the workplace. Table 5.3.5-2 presents a summary of common workplace neutron dosimeter performance characteristics . . . Measurements of TLND performance at SRS in 1987 (Brackenbush et al. 1987) indicate that the SRS measured neutron dose with the TLND (beginning 1 January 1971) is reasonably correct. For dose reconstruction under EEOICPA a claimant favorable standard error estimate of 50% should be made for neutron dosimetry between 1971 and 1985.*

Based on data provided above, the TBD provides no compelling evidence to suggest that the TLND dosimeter offered significant improvements over NTA film. From statements made in Section 5.3.5 of the TBD, it is also unclear whether the recommended "claimant-favorable" standard error of ±50% for the TLND represents a time-average value, as stated above (i.e., "… over long time periods, workers would generally be expected to be involved in several different exposure profiles that will serve to minimize the extremes identified").

In brief, this suggests that both the TLND-recorded neutron doses between 1971 and 1995 and the pre-1971 neutron doses (derived by neutron-to-photon ratios) suffer from a high degree of uncertainty and must be viewed with caution. SC&A recommends the use of a 95th percentile value for the TLND neutron dose of record.

Specific concerns regarding the neutron-to-photon ratios at SRS are applicable to Revision 03 and proposed Revision 04-E (Scalsky 2006) of the TBD; therefore, SC&A recommends the use of the 95th percentile neutron-to-photon value for all SRS neutron dose reconstruction cases. SC&A's evaluation of Revision 03 (Scalsky 2005) did not reveal any changes related to this issue.

Proposed Revision 04-E of the TBD (Scalsky 2006) provides some changes and clarifications to the applications of neutron-to-photon values for dose reconstruction. The site profile recommends that in likely noncompensible cases, the 95th percentile neutron-to-photon dose ratio be applied to the recorded dose. For plutonium facilities, it sets the neutron-to-photon geometric mean equal to 1.0, with the geometric standard deviation equal to 2.0 and the 95th percentile equal to 3.1, and for reactors it sets the neutron-to-photon geometric mean equal to 0.18, with the geometric standard deviation equal to 2.52 and the 95th percentile equal to 0.82. However, there are still some areas of concern in view of the following facts:

- Table 5-14, page 102, of the TBD (Revision 04-E) (Scalsky 2006) lists the average neutron-to-photon value for the HB-Line as 1.29, with a range up to 3.10.

- Section 5.3.4.2.3.2 of the TBD (Revisions 03 and 04-E) states that Brackenbush et al. (1987) reported the neutron-to-photon dose rate value for the F-Areas as 2.

- According to the TBD (Revisions 03 and 04-E) and NIOSH's original matrix response, likely noncompensable cases will be assigned the 95th percentile, but the likely compensable cases will be assigned the geometric mean, and for best estimates it is only stated that the dose reconstructor provides a claimant-favorable analysis of dose.

In view of the first two points listed above, a neutron-to-photon value of 1.0 would appear to be an average value rather than a claimant-favorable value. Therefore, as per SC&A's original comment and considering the third point listed above, it is recommended that the 95th percentile neutron-to-photon values be used in all SRS dose reconstruction cases, not in just the likely noncompensable cases.

4.3.2 Applicability to Revision 03

The application of geometric mean and 95% values to dose reconstruction is a programmatic issue. Specifics regarding the neutron-to-photon ratios at SRS are applicable to the current revision of the TBD.

4.3.3 Summary of Overall NIOSH Response

Extensive and detailed claimant data are now available that could be examined to verify the recommended neutron-to-photon dose ratio statistical parameters contained in the site profile.

Additional claimants were selected to expand the analysis to more workers, but determining the actual work location of SRS employees proved difficult. NIOSH/ORAUT discussed this with SRS Health Physics, and they considered the approach too uncertain to provide confidence in the analyses.

The site profile recommends applying the 95th percentile neutron-to-photon dose ratio to the recorded and missed photon dose for workers routinely present in work areas prior to 1972 with a potential for neutron exposure. This is applied to all likely noncompensable claims. This value is based on the SRS post-1971 TLND measurements, as well as the results of Pacific Northwest National Laboratory (PNNL) dose measurements in SRS facilities and PNNL dose measurements at Hanford, because of similar workplace source terms and activities. The magnitude of the recommended neutron-to-photon dose ratios is consistent with the results of an Atomic Energy Commission technical investigation of neutron doses (using neutron-to-photon dose ratios) in 1972 at Hanford. For likely compensable claims, the geometric mean value of the neutron-to-photon dose ratio is applied and, if necessary, a Monte Carlo analysis performed, taking into consideration the 95th percentile value as part of a lognormal distribution. The overall assessment of the neutron dose component in a best estimate dose reconstruction for a SRS claimant provides a claimant-favorable analysis of the neutron organ dose.

4.3.4 Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Provide a detailed description of neutron-to-photon ratio methods at SRS.

4.3.5 Closure Status

The TBD prescribes two very different protocols for neutron dose reconstruction that correspond to pre- and post-1971 time periods. Prior to 1971, the uncertainty factors associated with the neutron-to-photon ratio are neither technically defensible nor likely to be claimant favorable. The TBD provides no compelling evidence to suggest that the TLND dosimeter offered significant improvements over NTA film. In brief, this suggests that both the TLND recorded neutron doses between 1971 and 1995 and the pre-1971 neutron doses (derived by neutron-to-photon ratios) suffer from a high degree of uncertainty and must be viewed with caution.

SC&A's evaluation of Revision 03 (Scalsky 2005) did not reveal any changes concerning this issue since the review of Revision 02. SC&A believes that the use of the geometric mean and geometric standard deviation that describe the post-1971 neutron-to-photon ratio is neither technically defensible nor likely to be claimant favorable for a large fraction of potential claimants. Proposed Revision 04-E of the TBD (Scalsky 2006) provides some changes and clarifications to the applications of neutron-to-photon values for dose reconstruction. The site profile recommends that, in likely noncompensible cases, the 95th percentile neutron-to-photon dose ratio be applied to the recorded dose. SC&A recommends the 95th percentile neutron-to-photon values be used in all SRS dose reconstruction cases, not in just the likely noncompensable cases. No Work Group action is currently pending, but it is recommended that this issue remain open pending the release of proposed Revision 04-E (Scalsky 2006) of the SRS TBD.

4.4 COMMENT 4: ADEQUACY OF THE F- AND H-AREA TANK FARM CHARACTERIZATION

The adequacy of the F- and H-Area Tank Farm characterization in the TBD is questionable for use as dose reconstruction guidance. Data evaluation appears to be incomplete with regard to exposure conditions and uncertainty. This is particularly true for early periods of operation, where primary records involving key operations and incidents are lacking. Moreover, the TBD provides no references for the tank farm discussion and includes no analysis indicating how the conclusions were reached.

4.4.1 Issue Description

The TBD guidance needs to be more specific and complete in the following areas:

- Radionuclide lists are incomplete for both internal and external radiation.

- Early worker incident and contamination records may be seriously incomplete.
- Raw data on incidents and high-radiation areas indicate that geometry of exposure may be a problem.
- The potential for internal and external exposure to unmonitored workers in areas not designated as radiological control areas needs to be investigated.
- The completeness and adequacy of tank farm data used in the TBD are in question.

In this section, the term "tank farm workers" refers to all personnel who performed work around tanks in the F- and H-Area Tank Farms.

4.4.1.1 Radionuclide Lists

The TBD (Scalsky 2005, p. 31) gives the radionuclide lists for the F- and H-Area Tank Farms as follows:

> *Internal exposure. The majority of the annual internal effective dose equivalent in the F Area combined waste tank is delivered by ^{90}Sr, ^{144}Ce, and ^{244}Cm. The majority of the annual internal effective dose equivalent in the H Area combined waste tank is delivered by ^{90}Sr, ^{144}Ce, and ^{238}Pu.*
>
> *External exposure. The majority of the external dose in the F Area Combined Tank Waste is delivered by ^{90}Sr, ^{144}Ce, ^{137}Cs, and ^{106}Ru. The majority of the external dose in the H Area Combined Tank Waste is delivered by ^{90}Sr, ^{144}Ce, and ^{238}Pu.*

Discrepancies exist between radionuclides for the tank farms on page 31 as compared to those in Table A-14. Revision 03, Table A-14, includes Pu-241 and Am-241, both of which are listed as "[s]ignificant to external exposure." Yet neither radionuclide appears in the list in the main text of the TBD. Attachment A of the TBD extracts source term data from ESH-HPT-960197, *Facility Description* (LaBone 1996), but does not discuss the basis and method of determination of key internal and external radionuclides in this document. NIOSH determined which radionuclides delivered "the majority" of the internal and external doses in the respective areas to the exclusion of others present in those areas. SC&A finds that the lists are incomplete.

Revision 03 does not include Cs-137 and Ru-106 as radionuclides of concern for internal dose. As abundant fission products, Cs-137 and Ru-106 are both of concern for internal exposure and are readily soluble in liquid. It is unclear why they are not included in the internal exposure list of radionuclides for both the F- and H-Area Tank Farms, despite evidence of their importance. For example, a body burden of 2% of the maximum permissible limit of 30 microcuries (i.e., 600 nCi) of Cs-137 was estimated for a mechanic in the H-Area Tank Farm who was accidentally exposed to high-level waste on February 28, 1974. This is higher than all but one of the high-five Cs-137 intakes listed for SRS in ORAUT-OTIB-0001 (Brackett 2003), Table 1.

Similarly, the tank farm data bank contains records of internal Ru-106 exposure. Furthermore, it is unclear why Ru-106 is not listed as a radionuclide of importance for external exposure in the H-Area Tank Farm, since that set of tanks also contains fission products. Rh-106, the short-lived decay product of Ru-106, is a gamma emitter. Both Ru-106 and Rh-106 are also sources of beta radiation. Therefore, Ru-106 (including its decay product, Rh-106) should have been flagged as important to internal and external exposure in the tank farm. Proposed Revision 04-E (Scalsky 2006) includes Cs-137 and Ru-106 in Table A-14, resolving some of the issue.

The internal radionuclide list is incomplete in other ways; for example, Tc-99 is not included. A number of other radionuclides, such as Zr-95 (and its decay product Nb-95), should also be evaluated for inclusion in the list of radionuclides of concern in both the F- and H-Area Tank Farms. Finally, the tank farm radionuclide list does not include several radionuclides that were produced, processed, or used as target material, specifically Th-232, Np-237, Pu-242, and U-233. Since the TBD presents no analysis regarding the tank farm radionuclide lists, it is unclear whether these radionuclides were evaluated for inclusion and then excluded because they did not contribute significant dose, or whether they were simply omitted. In the case of Th-232, Np-237, and U-233, the TBD discusses their use but does not include them in the tank farm radionuclide list for reasons that are not explained. If these radionuclides have been evaluated, the analysis should be presented. If not, they should be evaluated. NIOSH/ORAUT also needs to be aware of the differences in the constituents of the tanks based on the processes that fed them.

4.4.1.2 Early Tank Farm Workers

The tank farm data bank is incomplete. The F- and H-Area Tank Farm data bank entry of August 24, 1965, states the following:

> *Prior to 1965, information on instrument failure, pump failure, leaks in the waste tank system are not recorded unless the individual occurrence is of particular interest.* (Makhijani et al. 1986)

The tank farm data bank did not identify any criteria by which an occurrence would be judged to be "of particular interest." But it is clear that the data bank is incomplete in a number of different ways. For example, the data bank contains no entry that explicitly shows the amount of worker exposure prior to 1960, although there are many entries after that date. The change in the frequency of entries per year in the tank farm data bank is another indication that the vast majority of incidents, maintenance problems, cleanup activities, and similar events associated with the tank farms were not recorded during the 1950s, 1960s, and at least part of the 1970s. Table 4-5, reproduced from Makhijani et al. 1986, p. 30, shows the increasing frequency of tank farm data bank entries:[2]

[2] The Environmental Policy Institute (EPI) obtained the data bank in about 1983 as a result of a Freedom of Information Act request. EPI no longer exists due to a merger, and the document is no longer available. It covered the period from late 1953 to 1982. SC&A requested it from NOISH as part of the SRS document request, but it was not available. A visit to SRS was conducted in February 2007 to provide for a firsthand review.

Table 4-5. Annual Average Number of Entries into the F- and H-Area Tank Farm Data Bank in Various Periods

Period	Average number of entries per year	Comments (added in this review, not part of the table in the reference)
1953–1959	4	Spills and other incidents not recorded; no entries showing worker exposures in this period, though some incidents and conditions with high radiation rates are reported.
1960–1965	32	First explicit worker radiation dose estimate entries are from this period.
1966–1969	85	
1970–1976	290	
1977–1982	1,800	Increase is mostly in items such as instrument maintenance and other entries not containing worker dose data. Many entries contain worker dose data.

It is clear that there were substantial changes in the frequency of entries into the data bank. This does not necessarily indicate a corresponding increase in the frequency of incidents. Rather, it appears likely that more inclusive criteria for making entries into the data bank were adopted over the decades. Since many early incidents, including spills of high-level radioactive waste, were not recorded in the data bank, and since the Special Hazards Investigation (SHI) index is also incomplete, as acknowledged by SRS, it raises the question of how complete the record of incidents might be in terms of individual worker dose records, at least for tank farm workers. This is a crucial issue, since the NIOSH dose reconstruction procedure for SRS relies heavily on the essential completeness of DOE dose records and looks to the computer-assisted telephone interview (CATI) as a supplement. At least in the case of the F- and H-Area Tank Farms, this assumption needs to be verified.

Two steps are necessary to verify the completeness of incident information provided to NIOSH/ORAUT for dose reconstruction. The record of known incidents in various data banks, worker records, SHI reports, and incident reports should be compared to the master incident list. Second, the master incident list needs to be scrutinized for completeness through review of records, interviews with site experts, and statistical analysis. This appears essential, since it is clear that outside workers, such as those in the high-level tank farm areas, repeatedly and frequently encountered conditions with high radiation rates of several roentgens per hour, tens of roentgens per hour, and sometimes even hundreds of roentgens per hour (Makhijani et al. 1986, Tables 1 through 11).

Given the paucity of entries in the F- and H-Area Tank Farm data bank, the problem of inadequate or missing data regarding incidents may be especially acute in the early years. In this context, and pending further investigation, it would be reasonable to apply the term "early years" to mean the period from the inception of tank farm operations to at least 1965 and probably to the end of the 1960s. An evaluation is needed as to whether the term

should be extended to the mid-1970s with regard to missing incidents. To SC&A's knowledge, NIOSH has not made a master list of incidents available to dose reconstructors for reference.

4.4.1.3 External Exposure Geometry Issues Related to Tank Farm Workers

Employees working in the F- and H-Area Tank Farms experienced a large variety of exposure geometries. Determination of the location of the badge in relation to geometry of exposure is especially important in the tank farm area due to the highly nonuniform nature of the maintenance work done there, frequent high radiation rates, areas with spills, and other contamination having very site-specific contamination geometry. For instance, in repair and maintenance work performed on piping, in junction boxes, and on other tank farm equipment, as well as during cleanup after spills of high-level waste, multiple badges would be essential to a sound estimate of organ dose.

Dose reconstruction for tank farm workers would therefore appear to face significant issues of technical accuracy and possibly data adequacy with regard to external dose because of the following factors, none of which are discussed in the TBD:

- The location of the exposed organ relative to the source of radiation compared to the location of the badge(s)
- Whether multiple badges were used
- What entries were made into the records when multiple badges were used

NIOSH proposed an exposure scenario analysis using the ATTILA code for a situation where the source of radiation originates underneath the workers, with selected examination of various geometry parameters and source terms. SC&A notes that the Mallinckrodt ATTILLA run for a similar situation showed that the correction factor was significant. The completion of this work should be expedited to reduce the reworking of claimant dose reconstruction cases.

Although SRS had an established multiple badging program, it is unclear whether multiple badging was used during tank farm work. NIOSH/ORAUT should investigate the specific exposure conditions of the tank farm workers and evaluate the incident exposure versus the badge location.

4.4.1.4 Radiological Zone Designation

In addition to these issues, questions remain regarding how the various radiological control areas were designated and how such designations were changed over time. SC&A understands from site expert interviews that some parts of the F- and H-Area Tank Farms were designated as radiation zones, but that the entire F- and H-Area Tank Farm was not designated as such (SC&A 2005a). Given that incidents may have been missed due to lack of recording, the potential for the significant exposure of workers who were in radiologically contaminated areas that were not designated as such needs to be investigated. The TBD does not discuss this issue. The importance of this and similar issues arises from the fact that the TBD assumes that unmonitored workers were those unlikely to encounter radiation areas. The validity of this assumption needs to be checked against actual historical practices of contamination of outdoor areas, as well as changing definitions of radiological control areas over time.

4.4.1.5 Comments on Completeness and Adequacy of Data Relating to F- and H-Area Tank Farm Workers

The above discussion indicates that NIOSH had not evaluated site data, including the crucial tank farm data bank and the master list of incidents, when preparing the original TBD and Revision 03. Hence, NIOSH's data evaluation is incomplete with regard to tank farm exposure conditions.

SC&A's evaluation of the tank farm data bank (based on summaries of entries in Makhijani et al. 1986) indicates the waste management Fault Tree Data Bank is critical to addressing exposures associated with the tank farm operations. This compilation of data is unique to the DOE complex and provides valuable references for nonroutine exposures to SRS workers (Minnick and Wellmaker 1995). Unlike other sites where incident data are scattered in various sources, SRS has assembled a centralized databank that can aid claimants and NIOSH. Some 35,000 incidents are documented from "60 types of sources, 47 Westinghouse Savannah River Company (WSRC) publications and 6 unpublished sources such as log books" (Minnick and Wellmaker 1995, p. 4) spanning the period 1951 to 1994. The database includes "…incidents with significant potential for injury or contamination of personnel" (Baughman et al. 1993, p. 6). For instance, in 1993 a status report and description of this database contained search results for incidents of interest to NIOSH as reproduced in the following table:

Table 4-6. Statistical Data from the tank farms Fault Tree Data Bank by Category.

Category	Number of Incidents	Date Range	Trend	Best Fit
Fires	79	1956–1992	Increasing	Weibull
Skin Contamination	142	1980–1992	Increasing	Weibull
Transfer Errors	73	1961–1992	No Change	Weibull
Overflows	101	1961–1992	Increasing	Lognormal
Uncontrolled Reactions	10	1977–1992	Increasing	Weibull

Source: Baughman et al. 1993, Table 2.

The data bank has been used for several purposes at SRS, including radiation incident analyses. It has been used to determine the frequency and severity of radiation releases involving potential exposure to workers. In 1995, an analysis of tritium releases and fires was performed on 13,000 incidents that took place at SRS 232-, 233-, and 234-H facilities from the 1950s to the early 1990s. The study identified 37 fires at these facilities involving releases greater than 1,000 curies of tritium (Wellmaker 1995). This data bank has been made available to SRS site contractors and well as to other groups, including Risk Assessment Corporation (RAC), which conducted radiation dose reconstruction for the Centers for Disease Control and Prevention (CDC). A user's guide for the data bank also has been developed (WSRC 1995). Default dose reconstruction models for tank farm workers, as proposed by NIOSH/ORAUT in response to the SC&A finding, should not be developed in a vacuum that does not account for incident data providing radiological exposure data. Although NIOSH/ORAUT's position is that "…significant dose to workers from incidents are already contained in the dose of record information…" (Minnick and Wellmaker 1995), this does not necessarily mean that such records are accurate or complete.

The Work Group, through NIOSH, made arrangements for an onsite review on February 28–March 1, 2007, by representatives from NIOSH, ORAUT, SC&A, and the Work Group of what was thought to be the tank farm Fault Tree Data Bank. The database reviewed is referred to as the WSMS incident database, which was developed by Bill Durant for safety analysis of activities in the separations area. In general, the information in the database appears to be primarily from the 200-F and 200-H Areas with minimal incident information from the reactor areas, heavy water area, the Savannah River Technical Center, the raw materials area, and other site operations. Over 90% of the entries were related to the separations area operations, including tritium operations. The discussion under Comment 13 provides a more detailed description of the database.

The entries in the WSMS incident database were compared to those abstracted from the tank farms Fault Tree Data Bank documented in Makhijani et al. (1986). Most of the entries reviewed from the tank farms Fault Tree Data Bank were not located in the WSMS incident database. This may reflect a difference in operational scope between the two collections of incident data. SRS requested Central Records to search the records database for "Fault Tree." However, the observations made to this point imply that multiple Fault Tree Data Banks have been maintained, corresponding to the various areas of the site. For example, the raw materials area has its own data bank. SRS did not originally provide the results of the search to NIOSH/ORAUT or SC&A, but NIOSH/ORAUT has recently requested this information.

The tank farms database can serve to determine the assumptions that would be suitable for giving claimants who worked in the tank farms the benefit of the doubt in the face of considerable uncertainties. The lack of evaluation of primary data sources has left the TBD without a realistic way to estimate uncertainties. These problems are likely to be especially acute for the early years. The TBD radionuclide list is not complete for reasons that are not clear. The lack of clarity arises from the absence of any references or analysis in relation to the radionuclide lists that were chosen for the F- and H-Area Tank Farms.

It is unclear whether adequate data are available to reconstruct any but the minimum doses for tank farm workers because of the various issues, data gaps, and uncertainties discussed above. The situation regarding early workers is especially unclear. A judgment on this question will be difficult or impossible without a careful evaluation of the available literature and an accompanying analysis of radiological conditions, exposure potential, and issues related to whether multiple badging was prevalent in tank farm work and, if so, how the data were generated and entered into dose records.

These tank farm findings may also have implications for other areas of outdoor work, such as the burning ground and seepage basins. The TBD has no discussion of the former and no analysis relating to dose reconstruction of the latter.

4.4.2 Applicability to Revision 03

Comments related to the tank farms are still applicable to Revision 03.

4.4.3 Summary of Overall NIOSH Response

Internal Dose

For internal dose estimation, the radionuclides of concern were identified on the basis of those that would deliver the majority of the dose and those that could be used as tracers. Both the annual and the committed effective dose equivalents were used. The source of the list of radionuclides of concern for the tank farms, as well as most of the radionuclide lists for the other facilities, is found in *Facility Descriptions* (LaBone 1996), as is mentioned in Section 1.3 of the site profile following the list of tables. During the period of active reprocessing of fuel, the tanks would have contained many different fission products.

The significance of the radionuclides listed in *Facilities Descriptions* was determined as follows. "The radionuclide contents of the waste tanks shown in Table 17.3 through 17.6 [references are to tables in *Facilities Descriptions*] were reviewed and the content of the combined tanks (Table 17.3) was chosen as being a representative sample for design purposes. The radionuclides of concern were identified on the basis of those that would deliver the majority of the dose and those that could be used as tracers."

Table A-14 of the TBD lists Cs-137 and Ru-106 as significant in terms of total activity but these radionuclides do not produce as much dose as Sr-90, Ce-144, and Cm-244. However, NIOSH/ORAUT indicates that Cs-137 and Ru-106 can easily be mentioned in the brief description in Chapter 1 of the TBD. NIOSH will add Ru-106 to the list of H-Area radionuclides important to external dose, and remove Pu-238. U-233, Tc-99, and Th-232 are not listed in Table A-14 of Revision 03 of the TBD as significant in terms of either total activity or dose, nor would they be expected to be significant relative to other high-level waste products or actinide wastes. Zr-95/Nb-95 are quite abundant in freshly irradiated fuel but tend to be less abundant in the waste tanks because of the shorter half-lives relative to Ce-144, Ru-106, Cs-137, and others. They also have smaller organ dose conversion factors.

Attachment A from the June 5, 2006, matrix provides a response regarding the inadequacy of incident records for tank farms in the 1950s and 1960s.

External Dose

Section 5.6 of Revision 03 of the TBD, "Organ Dose," is a summary section of recommendations to the dose reconstructor to be used in dose reconstruction. Since shallow dose is not addressed in this Site Profile, tank farm operations do not present unique sources of radiation exposure that are not already identified in the existing Site Profile.

The dose of record information already documents significant doses to workers from incidents in the dose of record information. In addition, Form OCAS-INT-004 requests DOE to submit relevant incident information for consideration in dose reconstruction.

4.4.4 Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Action Items:

- Provide the Board with an expanded review of the adequacy of its SRS radionuclide list by referencing additional source term information [e.g., Fault Tree Data Bank compilation (NIOSH to obtain from DOE)].

- Provide the Board with its specific comments related to issues raised regarding Attachment A, as identified in SC&A comments (and elaborated on in a separate SC&A note). This review should address gaps in tank farm radionuclide characterization and incidents (including scenarios).

- Provide the Board with its evaluation of exposure geometry in tank farm work, encompassing common tasks such as cleaning up spills and work on pipes and valves where the source was at a lower level than the badge.

- Pull data (e.g., from the tank farm Fault Tree Data Bank registry) and verify representativeness and claimant favorability.

4.4.5 Closure Status

The adequacy of the F- and H-Area Tank Farm characterization in the TBD is questionable for use as dose reconstruction guidance. Data evaluation appears to be incomplete with regard to exposure conditions, radionuclides of concern, and uncertainty. This is particularly true for early periods of operation, where primary records involving key operations and incidents are lacking. The tank farms database, not currently evaluated by the TBD, can serve to determine the assumptions that would be suitable for giving claimants who worked in the tank farms the benefit of the doubt in the face of considerable uncertainties. The lack of evaluation of primary data sources has left the TBD without a realistic way to estimate uncertainties. The potential for internal and external exposure for unmonitored workers in areas not designated as radiological control areas needs to be investigated. Revision 03 of the site profile included no changes in the discussion regarding the tank farms. Proposed Revision 04-E (Scalsky 2006) included only minimal changes, such as the addition of Cs-137 and Ru-106 to Table A-14. Default intakes for tank farm workers for different periods of time were added, including those for actinides. This revision does not resolve all issues associated with the tank farms. It is recommended that this issue remains open pending completion of Work Group action items and the release of proposed Revision 04-E (Scalsky 2006).

4.5 COMMENT 5: LACK OF COMPREHENSIVE EVALUATION OF EARLY MONITORING PROGRAM

For early SRS workers, the site profile lacks a comprehensive evaluation of the early monitoring program with respect to its consistent application, adherence to procedures, recordkeeping, and adequacy, all of which hold significant implications for reconstructing doses for unmonitored workers during the early years. Although early procedures clearly define when beta/gamma dosimeters were to be worn, the requirements for the use of neutron dosimeters and bioassay collection were left up to the field support organization. Gaps in data availability were noted for individual neutron exposure data, tritium exposure data, internal and external exposure data from special campaigns, and the radionuclide source term lists (and attendant concentrations and activity levels) used in the TBD, including those for the tank farms, RU, and environmental releases. As the basic approach for dose assessment for early workers, NIOSH proposed to use a maximizing dose for noncompensable claims and coworker dose for unmonitored, potentially exposed workers. Assignment of missed neutron dose is based on the application to the photon dose of a neutron-to-photon ratio prior to 1971.

4.5.1 Issue Description

The Radiological Control Organization was not centralized at SRS until more recent times. DPSOP-40, *Operating Procedure for Radiation and Contamination Control*, outlined the basic requirements that were to be followed with respect to personnel monitoring. In essence, field support personnel determined the requirements for routine and special bioassay and dosimetry using the following guidelines, as set forth in DPSOP-40 (DPSOP 1959 and 1960):

> *Film badge dosimeters are to be worn at all times by all personnel in exclusion areas, Regulated Zones, or Radiation Danger Zones (RDZs).*
>
> *Pocket meters are to be worn by all personnel where exposure rate is 25 mr/hour or greater, or when specified on the Special Work Permit.*
>
> *Neutron film badges or TLNDs are worn when specified by Health Physics on jobs where personnel are exposed to neutron radiation.*
>
> *Neutron pencils are worn when specified by Health Physics on jobs where personnel are exposed to slow neutron radiation.*
>
> *All personnel working in Regulated Zones or Radiation Danger Zones are periodically checked for assimilation of radioactive material. In buildings in which tritium is present, bioassay samples are submitted as directed by Health Physics.*
>
> *Special bioassay samples may be requested by Health Physics through the employees' supervision, when a suspected assimilation of radioactive material occurs.*

Prior to 1959, the Operating and Health Physics departments had to approve bioassay and dosimeter requirements (DPSOP 1953 and 1956). Work permits and facility-specific procedures were used to supplement the requirements of DPSOP-40. These requirements were documented on a Special Work Permit for nonroutine jobs. In 1971, requirements for in vivo and in vitro bioassay were outlined in DPSOL-193, *Health Protection Procedures*, or by specific request from Health Physics (DPSOL 1971). The requirements for external monitoring were the same as defined above (DPSOP 1971, 1974, and 1976). By the late 1980s, the dosimeter and bioassay requirements were clearly outlined in DPSOL-193. A beta/gamma dosimeter was required for all personnel who handled radioactive materials or entered facilities where radioactive material was handled or stored. TLNDs were required when the neutron dose rate was equal to or greater than 1 millirem (mrem)/hour (DPSOL 1989b). Routine and special bioassay requirements were also outlined for both in vitro and in vivo counting (DPSOL 1987, 1988, and 1989a). As is noted in the monitoring requirements listed above, facility personnel and not a central organization initially determined neutron, tritium, and special monitoring. This raises questions of consistency in monitoring for the early years.

In February 1999, SRS underwent an independent assessment of its internal dosimetry program. One of the findings is stated, as follows (WSRC 1999):

> *Facility personnel did not consistently adhere to the WSRC procedural requirements for initiating special bioassay sampling. In addition there was no mechanism in place for ensuring subcontractors submitted termination bioassay samples.*

One of the corrective actions implemented by SRS as a result of the audit was to make the Internal Dosimetry Group responsible for determining bioassay requirements.

Without a single organization determining requirements for neutron and bioassay requirements, there may have been inconsistencies from area to area in who was monitored and who was not monitored. Some consideration should be given to the following:

- Early monthly progress reports from the Works Technical Department include statistics by area of who was monitored for tritium. Initially, according to the reports, all the tritium monitoring was listed under F-Area, with no monitoring indicated for the reactor areas. Starting in 1957, monitoring was conducted for F-Area and H-Area workers. It was noted that no tritium analysis was documented in these reports for reactor workers during the 1956–1960 timeframe (DuPont 1957, 1958, 1959, 1960, and 1961). Based on SRS dose reconstructions reviewed by SC&A, it is evident that bioassay samples were collected from some reactor workers in this time period. The adequacy of the tritium monitoring in the reactor areas and heavy water rework area should be evaluated and the potential for exposures compared to the missed dose assignments outlined in ORAUT-OTIB-0011, *Technical Information Bulletin: Tritium Calculated and Missed Dose Estimates* (Siebert 2004).

- Review of early handwritten monitoring records (1954–1957) indicated that neutron monitoring was intermittent rather than routine in some areas.

- The progress reports indicate that uranium bioassay was initiated in 1953, plutonium bioassay was initiated in 1954, and tritium and fission product bioassay were initiated in 1955. Some thorium bioassay appears to begin in 1956. In 1960, mention of “Special Product” bioassay begins to appear in the reports. Americium-curium statistics on a number of monitored workers appear starting in early 1964. This information should be reviewed thoroughly with respect to identifying those exposed versus those monitored in these earlier periods of time.

Furthermore, monthly reports from the Works Technical Department contain information indicating that the tritium-monitoring program may not have been comprehensive. These reports track the number of tritium urinalysis samples by area. ORAUT-OTIB-0011 (Siebert 2004) provides a process for calculating missed dose based on the detection limits in bioassay sampling, but this may not be bounding for unmonitored individuals exposed to tritium.

Although early procedures clearly defined when beta/gamma dosimeters were to be worn, the requirements for neutron dosimetry were determined by the field support organization and may not have been consistent across the plant site. Figure 5.3.4.2-2 of the SRS TBD shows the trends in SRS and Hanford collective neutron dose normalized to plutonium production. The table clearly shows that there was minimal collective neutron dose prior to 1964 (Scalsky 2005). Review of the monthly reports from the Works Technical Department for December 1956, 1957, 1958, 1959, and 1960 indicates that 1,776, 1,001, 1,805, 2,023, and 2,778 NTA badges were processed for each year, respectively (DuPont 1957, 1958, 1959, 1960, and 1961). During this period of time, dosimeters were exchanged on a weekly schedule. The number of badges processed likely included multiple badges for an individual. From this information, it appears that minimal numbers of individuals at SRS were monitored for neutron exposure. A majority of these individuals were likely from the FB-Line and HB-Line where the plutonium finishing process occurred. NIOSH should investigate the completeness of the neutron monitoring at each facility for which there was a neutron potential. Such an investigation would be helpful to dose reconstructors in determining when an individual may have experienced neutron exposures but were not monitored.

For noncompensable claims, Revision 04-E (Scalsky 2006) of the site profile recommends applying the 95th percentile neutron-to-photon dose ratio to the recorded and missed photon dose for workers routinely present in work areas prior to 1972 with a potential for neutron exposure. For likely compensable claims, the geometric mean value of the neutron-to-photon dose ratio is applied. The four main areas at SRS with potential for neutron exposure include the plutonium facilities in the 200 Area; the Calibration Facility (736-A) and the Cf-252 Facility (773-A) in the 700 Area; reactors in the 100 Area; and Building 321 (plutonium-aluminum alloys) in the 300 Area.

It is not clear whether missed neutron dose is assigned to all job classifications, or how NIOSH determined whether an individual worked in an area with potential neutron exposure. This is especially concerning for the mobile segment of the work force. Any reliance on previous neutron monitoring as an indicator of potential exposure would require the availability of all neutron data.

Database Gaps

ORAUT has developed ORAUT-OTIB-0032, *External Coworker Dosimetry Data for the Savannah River Site* (Merwin 2006) for the assignment of external dose to workers without monitoring data but with the potential for exposure. This would include workers who would have been monitored under current standards but were not, or workers who have incomplete dosimetry data. These external beta/gamma coworker doses are proposed for use in cases were early worker monitoring data are incomplete. NIOSH used dosimetry data for monitored SRS workers from the HPAREH database. The annual data in HPAREH were prorated for partial years of exposure.

External radiation exposure data are available in three electronic files commonly referred to as the Fayerweather file, HPAREH and SRPABST. The three files do not contain radiation exposure information for all individuals employed at SRS. In fact, the Fayerweather and SRPABST files were compiled for particular studies conducted by DuPont and ORAU, respectively, and include only individuals from specific cohorts. HPAREH, described below, contains only those workers who terminated in 1979 and after (Richardson et al. 2006, Robertson-DeMers 2003). For all remaining individuals, dose information has to be obtained from the hardcopy records (e.g., logbooks). As SRS researches individuals terminating prior to 1979, they add this exposure information to the HPAREH file. This is why HPAREH contains data regarding some individuals terminating prior to 1979, although it does not include a majority of the monitored population terminating prior to 1979. The HPAREH and successive electronic dosimetry files are the only sanctioned radiation exposure files at SRS. The hardcopy records contain the most complete set of dosimetry data available. For the cohort of 18,883, 15,752 annual dose records were identified from a review of dosimetry logbooks, which were not in a computerized data system (Richardson et al. 2006).

NIOSH is currently using only the HPAREH file to evaluate external coworker doses. HPAREH consolidated data from the personnel radiation exposure files, logbooks, and magnetic tapes. It was designed to provide annual radiation exposure reports to plant workers and DOE. There was no requirement to go back and reconstruct doses for terminated workers. As a result, SRS focused their computerization of data on individuals who were actively working at the site in 1979 and thereafter. For those individuals included in HPAREH, exposure for all years of employment is available. In some cases, a cumulative dose is recorded in the first year of data in HPAREH, rather than the annual dose by year up to that point. Each year, the Health Physics Master System is used to update the HPAREH database with additional exposure information, terminations, and other information (Taylor et al. 1995). As individuals or other entities requested exposure records, SRS compiled the individual's dose history and entered the information into HPAREH. During the implementation of HPAREH, SRS made a concerted effort to resolve radiation exposure history gaps and inconsistencies.

ORAUT-OTIB-0032 (Merwin 2006) uses a version of the HPAREH file as a basis for determining coworker dose by year. NIOSH described the validation of the data selected for coworker dose development as follows (Merwin 2006):

> *The validity of the data selected for coworker dose development was confirmed by selecting a sampling of HPAREH summary data submitted by the site as a part of the EEOICPA Subtitle B program and comparing it to the data described above. A review of the data for two claimants with complex and extensive dosimetry records covering the years 1953 to 1999 indicated good agreement between the two data sets. Importantly, when the data did not match exactly, there was no apparent bias toward an under or overestimate of dose for either data set, suggesting that relying on the HPAREH annual data for coworker dose reconstruction would not result in a negative bias against the claimants.*

However, SC&A contends that the review of hardcopy results verses HPAREH results for two individuals is not an adequate sampling for validation of the database. The systematic gaps in HPAREH occur for individuals terminating prior to 1979. The two individuals worked at SRS through 1999 and would not fit into this category of workers.

NIOSH needs to evaluate whether using HPAREH exclusively for determining coworker dose is appropriate. The basis for considering the HPAREH data to be sufficient for the assignment of coworker dose prior to 1979 requires further explanation since this file includes only a portion of the population. NIOSH should demonstrate that data for the assignment of coworker dose are representative of all time periods and work groups.

4.5.2 Applicability to Revision 03

NIOSH/ORAUT has established methods to assign dose to unmonitored workers in Revision 03 of the site profile and the following supplemental documents, which were reviewed in preparation of this report:

- ORAUT-OTIB-0032, *External Coworker Dosimetry Data for the Savannah River Site* (November 7, 2006)—Provides information to allow dose reconstructors to assign doses to SRS workers who have no or limited monitoring data, based on site coworker data. (Merwin 2006)

- ORAUT-OTIB-0001, *Maximum Internal Dose Estimates for Savannah River Site Claims* (July 15, 2003)—Describes the maximum internal dose estimate for SRS claims. (Brackett 2003)

- ORAUT-OTIB-0011, *Technical Information Bulletin: Tritium Calculated and Missed Dose Estimates* (June 29, 2004)—Documents the method for estimating tritium missed and calculated doses from urine. (Siebert 2004)

4.5.3 Summary of Overall NIOSH Response

NIOSH has established methods to provide doses to unmonitored workers, and new guidance is being developed for unmonitored internal dose that clarifies intake radionuclides and levels at the various facilities.

4.5.4 Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Provide the Board with a description of classes of workers who were not appropriate monitored or time periods when workers were not appropriately monitored during the early phases (e.g., no early H-3 bioassay in reactors, early exposure gaps in HPARAH database, tank farms).

SC&A Actions:

- Provide an inventory of data sources used in the preparation of the individual dosimetry data submittals. This inventory will assist in questions related to the monitoring of early workers.

4.5.5 Closure Status

An evaluation of the comprehensiveness of the early monitoring program should be completed for early workers to determine whether existing site profile methodologies bound their dose. This is especially important in the case of workers who were not monitored but were exposed to a radiological hazard. Without a single organization determining neutron dosimeter and bioassay requirements, actual practices may have been inconsistent. The inconsistencies may also include the request for special interpretations of film badges for workers in the plutonium areas. The adequacy of the early monitoring program (i.e., who received monitoring) will not be resolved exclusively by an inventory of records provided in claimant files. References such as incident databases, monthly reports, and dosimetry logs also need to be considered. Additional validation of the HPAREH database as the exclusive source for external coworker dose determination, given the incompleteness of early data, is necessary to demonstrate this data is adequate for this use. It is recommended that this finding be considered open pending completion of Work Group actions provided to NIOSH. The discussion for Comment 14 includes additional information on data sources available for early workers, and Comment 9 discusses the application of a maximizing internal dose to unmonitored workers.

4.6 COMMENT 6: HIGH-FIVE APPROACH INCONSISTENT WITH THE METHODOLOGIES RECOMMENDED IN 42 CFR PART 82 FOR CALCULATION OF INTERNAL DOSE

The high-five approach, as currently explained in ORAUT-OTIB-0001 (Brackett 2003), is not consistent with the methodologies recommended in 42 CFR Part 82 for the calculation of internal dose. The dose reconstruction process must comply with the requirements of 42 CFR Part 82. To provide a method for effectively implementing these requirements, NIOSH has written technical guidance documents on external and internal dosimetry. While ORAU has committed to the use of these guidance documents in its quality assurance program plan, the use of the high-five approach to assign internal dose is not consistent with this guidance.

The method outlined in ORAUT-OTIB-0001 (Brackett 2003) to assign internal doses is based upon a hypothetical intake with the following characteristics (Brackett 2003, p. 3):

> *All radionuclides for which internal deposition by inhalation was calculated by the Savannah River Site were reviewed, except for tritium, which is addressed separately.*
>
> *The amount of the inhalation intake for each radionuclide is the average (mean) of the five largest documented intakes, or the average of all intakes if there were fewer than five intakes reported for a radionuclide.*

> *An acute inhalation intake was assumed to have occurred on January 1 in the first year of employment.*
>
> *ICRP 66 and 68 modeling and default parameter values were used to determine dose.*
>
> *The material type resulting in the largest dose to the organ or tissue of interest was used. This was typically the most soluble form of the material because it would clear from the lung more rapidly than insoluble material, thus depositing in the organ or tissue sooner.*
>
> *…Intakes and doses at SRS were calculated using regulatory-prescribed ICRP 30 methodologies rather than the newer ICRP methodology prescribed for this dose reconstruction effort. The material classes used in the calculations were based on workplace source term information or the class that provided the best fit to the bioassay data; the most claimant favorable class was not necessarily selected.*

As clearly stated above, SRS uses ICRP 30, *Limits for the Intake of Radionuclides by Workers* (ICRP 1979), to calculate the intakes used in the high-five approach. The organ dose is then calculated using modeling and default values from ICRP 66, *Human Respiratory Tract Model for Radiological Protection* (ICRP 1994), and ICRP 68, *Dose Coefficients for Intakes of Radionuclides by Workers* (ICRP 1995). The regulations in 42 CFR 82.18(b) direct NIOSH to do the following:

> *…calculate the dose to the organ or tissue using the appropriate current metabolic models published by the ICRP*.

Furthermore, in the question-and-answer section accompanying 42 CFR Part 82, NIOSH discusses the use of ICRP models:

> *As explained in the interim final rule and above, NIOSH is using current ICRP models because they represent improvements in the science of internal and external radiation dosimetry compared to older ICRP models.*

The intake quantity is the basis for determining final organ or tissue dose. In the case of SRS, NIOSH has decided to use intake information calculated using ICRP 30 methodology. Since the issuance of ICRP 30, ICRP has developed the new lung model outlined in ICRP 66 and the revised dose coefficients in ICRP 68. Therefore, ICRP 30 is not the most current metabolic model and is not consistent with the direction provided by 42 CFR 82.18(b).

4.6.1 Applicability to Revision 03

This issue is applicable to Revision 03 and ORAUT-OTIB-0001 (Brackett 2003).

4.6.2 Summary of Overall NIOSH Response

NIOSH has indicated that it expects to complete the process of completing the update to the high-five approach in December 2006. Information in the revised TBD refers the dose reconstructor to ORAUT-OTIB-0001 (Brackett 2003), which describes the high-five approach. The updated approach along with any modifications to the TIB will be reviewed by SC&A when provided by NIOSH.

4.6.3 Work Group Action

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Complete the update of high-five intakes using new models and provide this to the Board.

4.6.4 Closure Status

As noted above, information in the revised TBD refers the dose reconstructor to ORAUT-OTIB-0001 (Brackett 2003), which describes the high-five approach. NIOSH indicated that it would update the high-five approach and base revised calculations on bioassay data rather than data in the SRS Internal Dosimetry Record. This update was expected in December 2006 but has not been completed as of August 2007. Until ORAUT-OTIB-0001 (Brackett 2003) is revised and the TBD references the correct TIB, regulatory issues will continue to be associated with the application of the high-five approach. Using urinalysis as a basis for dose calculation will eliminate the use of intake data derived with ICRP 30 and allow for the exclusive use of ICRP 60 methodology or more current methods. Revision 03 (Scalsky 2005) and Revision 04-E (Scalsky 2006) of the TBD still reference the 2003 procedure on maximizing internal dose. The updated approach along with any modifications to the TIB will be reviewed when provided by NIOSH. It is recommended that this issue remains open pending the release of the revision to ORAUT-OTIB-0001 (Brackett 2003) and subsequent reflection in Revision 04-E (Scalsky 2006) of the TBD.

4.7 COMMENT 7: LIMITATIONS ASSOCIATED WITH THE ASSIGNMENT OF OCCUPATIONAL ENVIRONMENTAL DOSE

The method used to reconstruct doses to unmonitored outdoor workers due to airborne emissions employs an atmospheric dispersion model, assumptions, and a resuspension factor that do not appear to be claimant favorable and are not entirely appropriate for this class of problem. For modeling of airborne radionuclide releases, one potentially significant issue is the nonconservatism of the standard Gaussian model used in the TBD where it pertains to “nonstandardized” short-term releases occurring during stable atmospheric conditions. Based on an SC&A review of the literature, it also appears that the TBD resuspension factor of $1{\times}10^{-9}$ per meter may not be claimant favorable.

4.7.1 Issue Description

The fundamental approach employed in the environmental dose section (Chapter 3) of the site profile for deriving occupational environmental doses uses (1) a sector-averaged Gaussian plume model, (2) source terms and a radionuclide list originally intended for offsite dose estimation, and (3) a resuspension factor of 1×10^{-9} per meter for estimating air dust loading due to radionuclides in the soil. For the purpose of deriving outdoor doses to unmonitored workers from airborne emissions, NIOSH employed the source terms reported in the summary report prepared by Cummins et al. (1991) and the dose reconstruction report prepared by RAC, RAC Report No. 1-CDC-SRS-1999, *Savannah River Site Dose Reconstruction Project Phase II: Source Term Calculation and Ingestion Pathway Data Retrieval, Evaluation of Materials Released from the Savannah River Site* (CDC 2001). The atmospheric source terms reported in these documents appear to be limited to monitored releases, and they are reported in terms of total annual releases by year for the purpose of deriving historical offsite doses. This strategy may not be entirely applicable to reconstructing the doses to onsite workers for a number of reasons. Unmonitored and episodic releases that occur over a relative short period of time (e.g., days) may deliver relatively high doses to nearby outdoor workers that may have been missed. In addition, the application of average annual atmospheric dispersion factors based on standard Gaussian models may not be appropriate for exposures occurring close to the source term, especially to ground-level and/or episodic releases.

For example, Chapter 4 of the RAC report (CDC 2001) refers to a report by Miller (1956), which presents information on releases due to incidents and accidents. The RAC report explains that these releases appeared to have been captured in the annual estimates of the source terms. However, it may be instructive to review these incidents from the perspective of the potential doses to onsite workers. The site profile would benefit from a more in-depth analysis of these issues, or at least a demonstration that the doses to workers from episodic and ground-level releases could not have contributed significantly to the doses to onsite workers, as compared to the doses derived in Chapter 3 of the site profile.

SC&A has also noted that NIOSH/ORAUT has not made comprehensive use of information available relating to environmental releases at SRS. SRS published a series of reports providing a summary of releases from SRS facilities, including atmospheric and liquid releases, transport mechanisms, and concentration on and in the vicinity of SRS. These environmental reports include information on releases of activation products, americium, cesium, curium, fission products, neptunium, noble gases, plutonium, radiocarbon, radioiodine, strontium, technecium, tritium, and uranium.

The assessment of environmental dose did not mention a number of radionuclides that are known to have been released from SRS facilities. These radionuclides included Am-241/243, Br-82, C-14, Ce-141/144, Cm-242/244, Co-60, Cr-51, Cs-137, Eu-154/155, I-133, I-135, Kr-85/85m, Kr-87, Kr-88, Nb-95, Np-239, P-32, Ru-103, Ru-106, S-35, Sr-89/90, technecium, Th-232, Xe-131m, Xe-133, Xe-135, Y-91, Zn-65, and Zr-95 (Carlton et al. 1995; Jannik 1997; WSRC 1996 and 1997; Carlton et al. 1992a, 1996, 1993a, 1992b, and 1993b; Kantelo et al. 1993). Although the TBD indicates that the radionuclides selected for evaluation deliver 95% of the potential missed dose, it does not discuss the methodology used to determine which radionuclides and source pathways are important to onsite dose assessment. Screening calculations used for developing an offsite radionuclide list may not be appropriate for determining an onsite radionuclide list. Some discussion is needed regarding the methodology used to determine significant radionuclides and pathways that are included in the determination of environmental dose.

4.7.1.1 Dispersion model

Chapter 3 of the site profile makes extensive use of the RAC report (CDC 2001) for deriving occupational environmental doses. In so doing, NIOSH has adopted the sector-average Gaussian plume model (using site-specific meteorological joint frequency data) to derive onsite exposures from both elevated and ground-level releases. This approach to reconstructing historical offsite doses, as performed by RAC in support of CDC, is generally scientifically sound for offsite dose reconstructions at sites that have generally flat or rolling terrain, and where the releases are relatively uniform over time. In addition, this approach can also be used for deriving offsite doses from episodic releases if the episodic releases were numerous during a given year and random over time. However, as discussed in the following paragraphs, this approach may not be appropriate for reconstructing onsite doses to workers for a number of reasons.

SC&A recognizes that, within the framework of the approach it chose, NIOSH used some conservative assumptions for deriving occupational doses. For instance, NIOSH selected the highest sector-average annual atmospheric dispersion factor, which assumes that the worker is located year round downwind in the most prevalent wind direction at the site. However, certain fundamental issues associated with this approach could result in a substantial underestimate of the dose. Specifically, the Gaussian model breaks down in the near field for ground-level releases (i.e., those emissions that are released at a height that is less than about 2.5 times the height of the adjacent buildings). Under these circumstances, building wake effects cause turbulence that cannot be easily modeled by Gaussian methods.

Another concern is that some workers may have been located downwind at the time of episodic ground-level or close to ground-level releases at a time when the meteorology may have been highly stable (i.e., very little dispersion). The estimated annual intakes in Attachment C of the TBD are based on average annual atmospheric dispersion factors, and the actual instantaneous meteorology could vary from these averages by several orders of magnitude on any given day. Under these circumstances, it may be more appropriate to employ the upper 95th percentile atmospheric dispersion factors for deriving doses, as opposed to the average annual atmospheric dispersion factors. This is the approach recommended for use by the U.S. Nuclear Regulatory Commission (NRC) for deriving doses associated with accidental releases from commercial nuclear power plants (NRC 1974).

If large short-term releases occurred during stable conditions, such as during low windspeeds and stable atmospheric stability conditions (e.g., stability class E or F), the approach employed in the TBD could result in substantial underestimates of the doses to outdoor workers downwind from releases. The Hanford TBD acknowledges and explicitly addresses this issue and uses the RATCHET atmospheric dispersion code instead of the standard Gaussian model employed in the SRS site profile. SC&A suggests that the TBD revisit this issue and confirm that doses from episodic releases were not, in fact, significantly underestimated because of the use of conventional Gaussian models.

NIOSH's response provided on June 5, 2006, did not resolve the short-term episodic exposure issues raised in that it did not address ground-level plumes from the open-pan burning of large amounts of contaminated reprocessing solvents. Based on a review of DuPont Savannah River Plant Procedure (DPSP) documents spanning 1956 to 1962, large amounts of spent solvent from the F- and H-Canyons were transported and stored in underground tanks. Tens of thousands of gallons containing significant concentrations of fission products and transuranics were stored and then piped to an open-pan incinerator, which burned approximately 1,000 gallons at a time. Pan-burning of spent reprocessing solvent appeared to occur every few months in the 1950s and 1960s. In addition, several episodes involved environmental spills and leaks resulting in high-body dose rates as well as stack releases resulting in contamination of personal vehicles and large occupied areas on site. Modeling should address these episodic releases, particularly with respect to large particles. Offsite dispersion models used in the TBD (Scalsky 2006) are not viable for particle sizes above 0.5 micron and do not address the onsite deposition trajectory for large particles. Finally, ambient gamma dose rates to employees working outside and near the chemical separations operations were high in the 1950s and 1960s; and it is not clear if exposed workers wore badges.

4.7.1.2 Annual Intake from Resuspension

The methods available for deriving inhalation exposure from resuspended radionuclides include the resuspension-factor approach and the dust-loading approach. The dust-loading approach is used for those scenarios where information is available on the radionuclide concentration in surface soil dust (e.g., picocuries per gram (pCi/g)) and the airborne dust loading (g/m^3) of respirable-size particles. Using this approach, the product of the radionuclide concentration in the surface soil (pCi/g) with the dust loading of respirable-size particles (g/m^3) yields the airborne radionuclide concentration (pCi/m^3). This may be a suitable approach when dust-loading data are available, because the radionuclide concentrations in soil are reported in terms of pCi/g.

The resuspension-factor approach is used when information regarding the scenario is limited to surface contamination levels (e.g., pCi/m^2). Resuspension factors are empirically determined values expressed in units of pCi/m^3 per pCi/m^2 (which reduces to units of 1/m) for a given exposure setting. The product of the surface contamination level with the resuspension factor yields the equilibrium airborne radionuclide concentration (i.e., pCi/m^3). This is the approach employed in the site profile.

NIOSH considers the resuspension factor of 1×10^{-9}, chosen by Till (1983, p. 5-32), as the value most representative of undisturbed field conditions at SRS. The TBD does not define the basis for this assumption. The resuspension factor of 1×10^{-9}, derived for offsite dose reconstruction, however, may not be representative of workers on site. These workers would be in closer proximity to contaminated soils and liquids than members of the public would be. In some cases, active work may be occurring at these sites that will generate more airborne dust and suspended particles. Consideration should be given to the determination of an appropriate resuspension factor differentiating between a member of the public and an onsite, unmonitored worker. NIOSH should demonstrate that 1×10^{-9} is a claimant-favorable resuspension factor.

Measured resuspension factors vary over very wide ranges. Kennedy and Strenge (1992) reported resuspension factors from approximately 1×10^{-11} to 1×10^{-2} m^{-1}, which suggests that resuspension is a complex process of several parameters and that the specific conditions present at the time of measurement are critical. For modeling purposes, a resuspension factor is a lumped parameter that is used to account for a complex combination of mechanisms that are not very well understood but whose net effect is observed in the real world.

The representative indoor resuspension data compiled by Beyeler et al. (1999) from experimental data and available information for indoor resuspension factors ranges from 2×10^{-8} m^{-1} (Jones and Pond 1964) to 1×10^{-3} m^{-1} (Healy 1971). The reported data are generally from experiments that examined the resuspension of liquid or powder contaminated material that had been uniformly applied to clean surfaces in a laboratory-like setting. The highest values are typically associated with inefficient ventilation, excessive mechanical disturbance, or dusty conditions. Typically, the purpose of these studies was to help determine radiation protection safety guidelines for loose, residual, surface radioactivity. The original SRS TBD review (SC&A 2005a) provides a more detailed analysis of Beyeler's summary. Indoor resuspension factor applicability to outdoor resuspension factors is questionable, but intuitively, one might expect outdoor resuspension factors to be generally higher due to wind and anthropomorphic activities that are likely to be greater outdoors than indoors.

Several reports by Sehmel (1977 and 1980) revealed that there is enormous uncertainty in outdoor resuspension factors, as there is for indoor resuspension factors. In an investigation of resuspension factors at the Hanford reservation, Sehmel (1977) found outdoor resuspension factors ranging from 1×10^{-11} to 1×10^{-5} per meter. In a review of the literature on outdoor resuspension factors, Sehmel (1980) cites experimental studies where the values ranged from 9×10^{-11} to 3×10^{-4} per meter for wind resuspension, and 1×10^{-10} to 4×10^{-2} per meter for mechanical stresses from human activities. He explains that there are many reasons for this variability, many of which relate to sampling and experimental techniques and the depth and nature of the contamination. For these reasons, the dust-loading approach is probably preferable when it can be employed. Stewart (1964) concludes that a representative value for a resuspension factor outdoors under quiescent conditions is 1×10^{-6} m^{-1} and under conditions of moderate activity is about 10 times greater.

Based on this review, it would seem that a resuspension factor of 1×10^{-9} per meter, as used in the TBD, may not be claimant favorable. An average value closer to 1×10^{-5} to 1×10^{-6} per meter would seem more appropriate for use in worker dose reconstruction, resulting in worker inhalation doses from resuspension that are 3 to 4 orders of magnitude greater than those derived in the site profile.

4.7.1.3 Dust Loading

A report prepared for the NRC by Battelle Pacific Northwest Labs (Sutter 1982) provides a review of this subject. Table 4.7 summarizes some of the relevant information contained in that report.

Table 4-7. Some Airborne Particulate Mass Concentrations

Aerosol	Mass Concentration	Reference
Dust storm	0.5 to 10 g/m^3	First 1952
Uranium dioxide powder	10 g/m^3 to 0.1 to 0.01 g/m^3	Schwendiman 1977
Dust devil	5 g/m^3	Sinclair 1947
Steam generating station	50 to 3000 mg/m^3	Bond 1972, p. 62
Mine working face (no controls)	500 mg/m^3	First 1952
Mine air	0.05 to 0.5 g/m^3	First 1952
Foundry workroom	2 to 30 mg/m^3	First 1952
Los Angeles smog	0.5 to 50 mg/m^3	Bond 1972, p. 62
Nuisance dust	10 mg/m^3	United Power Assoc. 1974
Industrial atmosphere	0.1 to 50 mg/m^3	Dennis 1976, p. 9
Cigarette smoke (steady-state cocktail party)	5 mg/m^3	Stern 1976, p. 157
Ambient atmosphere	0.05 to 1 mg/m^3	Dennis 1976, p. 9
Air conditioned building	0.3 mg/m^3	First 1952
Cigarette smoke (average)	40 to 400 μg/m^3	Stern 1976, p. 157

Source: Sutter 1982, Table 2.1-2

Yu et al. (1993) also presents a review of the literature on dust loadings. Table 4.8 summarizes those studies.

Table 4-8. Summary of Dust Loading Studies Cites by Yu et al. (1993) (g/m^3)

Setting	Dust Loading	Author
Urban outdoors	3.3E-5 to 2.54E-4	Gilbert et al. 1983
Nonurban outdoors	9E-6 to 7.9E-5	Gilbert et al. 1983
Construction activities	6E-4	Oztunali et al. 1981
Construction traffic on unpaved roads	4E-4	Oztunali et al. 1981
Agricultural-generated dust	3E-4	Oztunali et al. 1981
Maximum dust loading in a cab of heavy construction equipment during a coal mining operation	1.8E-3	Oztunali et al. 1981
Upper bound values report	1.3	Yu et al. 1993

In addition, experience gained in various industries involved in the handling of bulk material, such as sand, coal, coke, alumina, borax, phosphate ore, and vermiculite, reported average dust loadings ranging from about 0.3 to 4 mg/m^3 outdoors, with peak dust loadings of up to 80 mg/m^3

(Rando et al. 2001, Heederik et al. 1994). For workers in the concrete industry (blasting, drilling, grinding), the 8-hour time-weighted average dust loading of respirable particles was measured at between 0.26 to 14 mg/m^3 (Linch 2002).

NIOSH should evaluate the dust-loading approach, using an average work-year dust loading on the order of perhaps 1 mg/m^3. Furthermore, Carlton (1999) used this dust-loading factor in an evaluation of dose from the Par Pond dam repair. When using this approach, the average soil concentration should be determined over a large work area on the order of several acres.

4.7.2 Applicability to Revision 03

Changes in the environmental section in Revision 03 are limited to the inclusion of dose from the consumption of foodstuffs. They did not resolve issues associated with dispersion models and resuspension factors.

4.7.3 Summary of Overall NIOSH Response

NIOSH believes that the dispersion model has been adequately validated by comparison to measured values of tritium and iodine. It should also be noted that the measured data represent integrated measurements that include inversion conditions when they existed.

The comparisons made in Sections 5.1 and 5.2 of Revision 03 of the TBD also demonstrate that the dispersion model is claimant favorable when compared to the measured data for all areas modeled. In addition, because the simplified Gaussian model does not include losses due to wet or dry deposition, the calculated air concentrations for plutonium and uranium should also be claimant favorable.

With regard to the resuspension factor, Till and Meyer 1983 chose 1×10^{-9} as the value most representative of undisturbed field conditions at SRS. NIOSH does not agree that this value is too low for the majority of the area over which the contamination was deposited in 1955 (F-Area) and 1969 (H-Area), which is highly vegetated and undisturbed.

4.7.4 Work Group Action

The following actions were identified in Work Group deliberations:

NIOSH Actions

- Provide the Board with a written evaluation of the existing dispersion model, focusing on episodic releases, particularly at tank farms (e.g., contaminated solvent burns).

4.7.5 Closure Status

The method used to reconstruct doses to unmonitored outdoor workers due to airborne emissions employs an atmospheric dispersion model, assumptions, and a resuspension factor that does not appear to be claimant favorable and are not entirely appropriate for this class of problem. Revision 03 (Scalsky 2005) and proposed Revision 04-E (Scalsky 2006) provide no clarification on these issues. In its response to SC&A's finding, NIOSH maintains that the dispersion model has been adequately validated and is sufficiently conservative. However, the TBD has not specifically addressed ground-level plumes from open-pan burning of contaminated solvents or from environmental spills and leaks. Furthermore, NIOSH has not provided a written evaluation of the existing dispersion model with a focus on episodic releases, as proposed by the Work Group.

The TBD does not provide the basis for the use of a resuspension factor of 1×10^{-9} for resuspension of contaminated soil. Based on an SC&A review of the literature, it also appears that the TBD resuspension factor of 1×10^{-9} per meter may not be claimant favorable. Kennedy and Strenge (1992) reported resuspension factors from approximately 1×10^{-11} to 1×10^{-2} m^{-1}, which suggests that resuspension is a complex process involving several parameters and that the specific conditions present at the time of measurement are critical. Based on recommended resuspension factors presented in the literature, an average value closer to 1×10^{-5} to 1×10^{-6} per meter would seem more appropriate for use in worker dose reconstruction, resulting in worker inhalation doses from resuspension that are 3 to 4 orders of magnitude greater than those derived in the site profile. The dust-loading approach should also be considered, using an average work-year dust loading on the order of perhaps 1 mg/m^3. It is recommended that due to the remaining issues associated with the atmospheric dispersion model, assumptions, and a resuspension factor, this overall finding remains open.

4.8 COMMENT 8: INADEQUATE EVALUATION OF SPECIAL TRITIUM COMPOUNDS

SRS has been one of the key facilities for the production of tritium for the U.S. defense program. WSRC is currently the only source of tritium in the DOE complex. The key processes at SRS leading to occupational tritium exposure included the following:

- Production reactor operations
- Product recovery in separations
- Tritium recovery and processing
- Laboratory research
- Heavy water rework

The exact forms of tritium encountered at SRS are not available; however, over the course of operations, SRS has handled tritium in the form of tritiated water (HTO), tritiated gas (HT), organically bound tritium (OBT), and metal tritides (MTs). A majority of the tritium handled was in the form of HT or HTO. HTO was also produced by exposure of HT and some STCs to air. Tritium-handling operations, research and development activities, and tritium facility decontamination and decommissioning may have exposed workers to other forms of tritium. Tritium-handling operations can form other tritium compounds, such as OBT and stable MTs.

Organically Bound Tritium

In a special study evaluating effluents from the tritium-handling facilities, less than 1% of the total tritium released was OBT in six cases out of seven. Carbon sources in the tritium processing area can contribute to the production of organic tritium forms by hydrogenation or exchange reactions. The processes where this might have occurred include the following (Milham and Boni 1976):

- Graphite crucibles used in lithium-aluminum target preparation
- Polyethylene film from wrapping extraction crucibles
- Ink marking for identifying targets
- Carbon dioxide in the extraction furnace
- Carbon dioxide in the recovery system
- Neoprene o-ring seals throughout the process
- Vacuum pump oil

Milham and Boni (1976) indicated that the process associated with by-product helium purification produced a gaseous form of tritium that was mostly OBT (Milham and Boni 1976). Other tritium compounds such as rust, pump oil droplets, and tritiated solvents are known to form during operations with significant amounts of HTO and HT present. Observed OBT in the environment may be present as a result of conversion of HTO to OBT or HT to OBT (via HTO) in soil, plants, and animals, or from direct releases of OBT from the site. Sweet and Murphy (1982) documented that tritium in the soil and leaf litter near the chemical separations facility formed "bound" tritium as a result of the update of molecular tritium (HT) by living pine needles. Depending on the compound formed, concentrations of OBTs can be equivalent to or greater than that of HTO.

The TBD does not adequately address potential exposures of workers to STCs including OBT and stable MTs during production and decommissioning activities. With documentation existing indicating that OBT was released to the environment, it may be reasonable to assume that it was present in some working conditions. ORAUT-OTIB-0011 (Siebert 2004) provides a basis for estimating calculated and missed dose from urine data assuming an uptake of tritiated water and excludes potential intakes of STCs. While the assumption of tritium as tritiated water is generally claimant favorable, it is not so in specialized situations. ORAUT-OTIB-0066, *Calculation of Dose from Intakes of Special Tritium Compounds* (LaBone 2007), presents guidance on how to calculate the best estimate dose from urine bioassay data for OBT and stable MTs. This procedure does not address unmonitored workers who were exposed to these compounds and does not specify which processes involved potential exposure to OBTs. NIOSH indicates that the methodology in ORAUT-OTIB-0011 (Siebert 2004) can be used without modification to calculate intakes to all organs or tissues, yet it does not specifically discuss the difference in effective half-life between HTO and OBTs (LaBone 2007). In its response to SC&A on June 6, 2006, NIOSH indicates that exposure to organic forms of tritium in the tritium facilities was possible. Furthermore, the response indicates that significant uptakes of OBTs are compensated for by assuming that all tritium is HTO rather than that a fraction of the total tritium is HT.

The effective dose per becquerel (Bq) intake of OBT is more than twice the effective dose per Bq intake of HTO. The urinary excretion rate is almost the same after the second day of exposure. One day after exposure, the activity concentration in urinary excretion for OBT is 57% of the HTO activity concentration in urine. As a consequence, for the same amount excreted in urine in the first day, the intake of OBT would be 77% higher than for HTO. Thus the effective dose calculated for each Bq excreted in urine is 4 times higher since it is due to OBT instead of HTO.

Stable Metal Tritides

Metal hydrides are formed when hydrogen gas reacts with metal. SRS actively produced these chemical compounds during tritium processing or as a byproduct material from research activities (Moxley 2002). In some cases, this was done intentionally, as described by Reed et al. (2002).

> *The production and storage of tritium, an isotope of hydrogen gas, was particularly tricky, and new methods were always explored to make this task easier and more efficient. Since most metals will react with hydrogen under certain conditions, the Savannah River Laboratory explored using metals to manipulate isotopes of hydrogen more efficiently. This led to the development of metal hydrides for the processing and storage of hydrogen. Metals that react with hydrogen to both absorb and release the gas under the right conditions, similar to a sponge that can absorb and release water, are called reversible metal hydrides, and this class of hydrides is important for hydrogen storage and processing. Effective reversible metal hydrides can be made from pure palladium, titanium, or zirconium; or from alloys of two or more metals, such as iron and titanium, or lanthanum and nickel. By the late 1970s, metal hydrides were used in tritium operations at Savannah River. This use expanded in the 1980s, and played an important part in the development of the Tritium Replacement Facility that began operations in 1994.*

The radiological concern associated with the metal hydrides occurs when tritium reacts with metal and produces tritium compounds, which pose challenges in personnel and radiological monitoring.

Stable MTs are encountered in tritium processing facilities and in legacy materials where substantial amounts of tritium were handled. These compounds are solid substances containing tritium that does not readily react with air or aqueous solution because the tritium is tightly bound to the matrix. These fine particles can easily be spread by work activities and suspended as airborne particulates. Particles are taken into the lungs where they dissolve and release the tritium from the metal. SMTs may be soluble or insoluble depending on metal substrate and particle size and shape. The lighter MTs (e.g., lithium tritide) degrade to tritiated water, which can be detected through bioassay. Heavy-metal tritides behave similar to particulates. Stable MTs documented in unclassified sources include titanium tritide, lithium tritide, lanthanum-aluminum-nickel tritide, and tritiated mercury. Zirconium tritide was under considered for use at SRS (Moxley 2002).

During the February 28, 2007, visit to SRS, SC&A interviewed two individuals knowlegable of tritium operations at the site regarding MTs and OBT encountered on the site. The Savannah River Operations Office limited the interview to an unclassified discussion but indicated that additional questions could be submitted for consideration. The types, quantities, locations, and time periods where tritides were handled could not be discussed. This information is critical to the evaluation of the proposed tritide dose assignment methodology. The site experts referred SC&A to a report discussing tritides at SRS that would inform this review. SC&A formally requested this document and received it on July 30, 2007. Although the document provided some information regarding which tritides were handled, it primarily concentrated on the implementation of radiological controls in areas with tritides. SC&A subsequently requested a second document that may provide additional detail.

Because SMTs are relatively insoluble and the retention of this type of tritium is longer than the retention of HTO, the internal dose delivered to the body is higher for some of these compounds. For particles of these tritides, the primary organ of concern is the lung. Some of the tritium may leach out in the lung fluids and then be incorporated into the body water. These particles may also produce OBT from contact with lung tissue, which would further complicate the metabolic process (DOE 2004).

McConville and Woods (1995) discuss the challenges with determining internal dose from MTs:

> *Tritium in the form of metal tritides particles presents a peculiar problem for the calculation of internal dose. Standard calculations indicate that just a few 3 to 5 micron sized particles appears to lead to a very large dose. There are very few data on which calculations can be based.*

Cheng et al. (1997) determined the dissolution time for fine and coarse titanium tritide particles to be 33 days and 361 days, respectively. McConville and Woods (1995) demonstrated, with individual excretion data following tritide uptakes, that tritium excretion curves for particulate tritides do not follow a simple exponential curve, as is the case with HTO. In the case of the individuals evaluated, tritides built up for a few days followed by a more traditional elimination curve.

Furthermore, SMTs present unique challenges to radiological protection programs. Routine workplace monitoring techniques make it difficult to differentiate between STCs and more common forms of tritium, such as HTO. Due to the physical and chemical behavior of STCs, common bioassay and dose calculation models can be ineffective. For select STCs, air monitoring is preferable to bioassay (DOE 2004). *The ICRP Database of Dose Coefficients: Workers and Members of the Public* (ICRP 2001) provides information on tritium in particulate forms (Types F, M, and S). In these cases, the default parameters of lung clearance and absorption are applied and the biokinetic model for tritiated water is used. Thus, the dose coefficients from the specific MTs should be equal to the generic Types F, M, and S if the ICRP recommendations are followed.

NIOSH has included in Revision 04-E of the SRS Site Profile (Scalsky 2006) a discussion on exposures to MTs in tritium facilities, including derivation of default intakes. In addition, NIOSH has included a discussion of exposure to OBT, which requires further consideration. In the draft version of the TBD, NIOSH has limited its application of tritide dose assessments to 1975 and after and has exclusively applied it to workers in the tritium facilities. According to SRS staff interviewed, OBTs and SMTs potentially extend back to the 1950s.

NIOSH must be familiar with the STCs handled, the quantities of material, the locations and time periods of potential exposure, and the physical behaviors of tritium compounds in the environment (e.g., conversion to HTO, formation of rust) to correctly characterize tritium exposure. Bounding techniques, as proposed for SRS, cannot be effectively developed and applied without some basic understanding of the compounds handled and the extent to which individuals were exposed. This includes exposure to tritium compounds other than HTO and HT outside the tritium facilities.

With these concerns in mind, SC&A does not feel that the TBD revision adequately justifies the upper bound approach for exposure to tritides and OBT. A bounding dose reconstruction methodology must consider what tritium compounds were handled or inadvertently produced, what percentage of tritium exists in nontraditional forms for various operations, releases of tritium to the work environment, and radiological controls put in place to prevent exposures. NIOSH should also be cognizant of the fact that STCs are not specific to SRS but may affect other DOE sites (e.g., Lawrence Livermore National Laboratory, Mound).

4.8.1 Applicability to Revision 03

Revision 03 handles the general dosimetry of exposure from STCs as a generic issue. The details of potential exposure to tritides and OBTs at SRS are still applicable and were not resolved in Revision 03.

4.8.2 Summary of Overall NIOSH Comments

The nature of exposure to MTs and OBT was investigated. Tritium was stored as metal hydrides, also called MTs, starting in the mid 1970s in the tritium facilities 232-H, 233-H, 234-H, and 238-H. These would have been minor sources of intake relative to the total activity as HTO or HT. However, intake of MTs cannot be completely ruled out, and bioassay sampling for particulate forms of tritium was not performed. The surface contamination limit for tritium in accessible spaces was 10^6 dpm/100 cm^2. By far the most contamination would have been HTO, but a conservative assumption that 50% was MTs and application of a resuspension factor of 10^{-4} m^{-1}, which would apply to aggressive disturbance of the surfaces, would result in the following:

$$\text{air concentration}_{mt} = (10^6\ \text{dpm}/100\ \text{cm}^2)(0.5)(10^4\ \text{cm}^2/\text{m}^2)(10^{-4}\ \text{m}^{-1})$$

$$\text{air conc.} = 5\times10^3\ \text{dpm}/\text{m}^3 = 2.3\times10^3\ \text{pCi}/\text{m}^3\ .$$

The daily intake is determined by the air concentration times the breathing rate per year divided by 365 days/year.

$$\text{intake}_{mt} = (2.3\times10^3\ \text{pCi}/\text{m}^3)(2{,}400\ \text{m}^3/\text{yr})/365\ \text{d/yr} = 1.5\times10^4\ \text{pCi/d}.$$

This intake should be assigned to workers in the tritium facilities who are monitored for tritium from 1975 to present; it is in addition to tritium intakes determined as HTO from urinalysis. Either absorption Type M or S can be assigned to be claimant favorable. (In IMBA, the user should choose the inorganic form of tritium, toggle the inhalation mode, and assign the absorption type.) This intake will typically produce annual dose to most organs of a few mrem and about 10 mrem to the lung.

Exposure to organic forms of tritium (commonly called OBT) in the same tritium facilities was possible. Milham and Boni (1976) describe measurements of tritium species in effluents from the tritium processing areas, noting that, "The concentration of tritium in the organic form is generally 1% of the tritium released to the atmosphere." Even if workers inside the facilities were exposed to several times this concentration relative to HTO, the increase to the organ doses calculated from urine concentrations assuming 100% HTO was insignificant. When the molecular sieve beds were purged, a much higher percentage of OBT versus HTO and HT was measured. However, this was infrequent and the effect on organ doses is compensated by assuming all tritium is HTO instead of accounting for the percentage of HT in the air. Unless an energy employee's file provides indications of a significant intake of OBT, assuming that all the tritium is HTO is considered appropriate.

4.8.3 Work Group Action

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Verify the historic timeframe, location, and quantities of tritide compounds at SRS.
- Provide the Board with the planned disposition of the issue in terms of dose estimation methodology.
- Determine the types and quantities of STCs handled at SRS to allow for a more informed evaluation of the tritide dose bounding methodology.

4.8.4 Closure Status

Revision 03 contains inadequate information regarding the assessment of dose from STCs. In June 2006, NIOSH proposed a methodology for the assignment of dose from STCs, which is provided above. Bounding techniques, as proposed for SRS, cannot be effectively developed and applied without some basic understanding of the STCs handled, the quantities of material, the locations and time periods of potential exposure, and the physical behaviors of tritium compounds in the environment (e.g., conversion to HTO, formation of rust) to correctly characterize tritium exposure. Furthermore, NIOSH limited the application of this technique to the period from 1975 to present. This is in conflict with site expert statements that indicate that potential exposures occurred back to the late 1950s. Given the large amount of tritium handled at SRS in various areas and the propensity of tritium to bind with organics and metals, it is reasonable to assume that STCs were present at SRS in some form prior to 1975.

In April 2007, NIOSH released ORAUT-OTIB-0066 (LaBone 2007), which provided guidance on the assignment of dose for OBTs and SMTs using urinalysis data. Application of this procedure is based on process knowledge. Proposed SRS-specific guidance assigns dose from tritides based on surface contamination limits rather than production information and surveillance data, making the basis for assumptions weak, particularly in years when engineering controls were not as advanced as they are today. NIOSH has provided a dose estimation methodology for STCs; however, it has not verified the timeframe and location where STCs were handled or the types and quantities of STCs handled at SRS. An evaluation of the adequacy of the dose estimation methodology cannot be completed without this key information; thus it is recommended that the issue remains open.

4.9 COMMENT 9: HIGH-FIVE APPROACH IS BASED ON INCOMPLETE DATA

The high-five approach is an efficiency process used to assign internal dose to employees who were monitored but had results less than the limit of detection, or to employees who were not included in the bioassay program. The technique is primarily for the assignment of dose to nonmetabolic organs (Brackett 2003). Intake values are derived from an average of the highest five intakes identified by NIOSH. Although NIOSH has indicated the contrary, it has not identified the highest five intakes for all radionuclides, as its search focused on the SRS Internal Dosimetry Registry (IDR), which is not complete.

4.9.1 Issue Description

The SRS TBD (Revision 03) recommends the use of a maximizing approach for likely noncompensable claims for nonmetabolic or digestive system cancers (Siebert 2004, p. 85). ORAUT-OTIB-0001 (Brackett 2003) describes an efficiency process for estimating internal doses for unmonitored workers for organs that do not concentrate the radionuclides in question—that is, for digestive tract organs and nonmetabolic organs. The approach is also applied to "employees who were monitored but had no detectable activity ('positive') in their samples and to employees who were not included in the bioassay program." This is an attempt to create an efficiency procedure to estimate a worst-case internal dose (except for tritium) in noncompensable cases (Brackett 2003, p. 2):

> *To facilitate timely processing of Savannah River Site claims under the Energy Employee Occupational Illness Compensation Program Act (EEOICPA), cases were reviewed to identify those with 1) little or no apparent internal dose and 2) cancer of an organ that does not concentrate internally deposited radionuclides that might be associated with work at the Savannah River Site. The cases were further screened to find those that met the following criteria:*
>
> - *No detectable activity in vitro bioassay samples, other than H-3.*
> - *No detectable activity in chest counts.*
> - *No detectable activity in whole body counts other than Cs-137, Co-60, or Eu-152.*

When this technique is applied to nonradiological workers and minimally exposed workers, the resulting internal dose is an overestimate of the actual internal dose received by these individuals. However, the question of whether one or more groups of unmonitored workers were not in either category remains to be investigated. For instance, if trade workers were unmonitored even when they were in hazardous job locations, the issue of onsite doses becomes far more complex. For those workers who were on a monitoring program and/or had the potential to receive internal dose, it is unclear whether the high-five approach bounds the internal dose.

The high-five approach is based on SRS intake data taken from the SRS IDR, as follows:

> *These hypothetical intakes were based on recorded internal doses at SRS and were assumed to be composed of the radionuclides contributing the majority of the recorded internal dose at the Savannah River Site, except for tritium (assignment of tritium dose is discussed at the end of this paper). All recorded inhalation intakes in the history of the site were reviewed and the largest intakes for each radionuclide were selected. Hypothetical inhalation intakes, based on the recorded intakes, were used to calculate organ doses for each calendar year from the start of employment at the Savannah River Site through the year of cancer diagnosis* (Brackett 2003).

The registry includes individuals at the site who had uptakes of radioactive material and met the criteria applied for inclusion. The purpose of the IDR is to ensure appropriate followup bioassay of individuals and to ensure that workers with significant intakes are informed about the U.S. Transuranium and Uranium Registries (WSRC 2001). From 1951–1983, the site implemented the methodology from ICRP 2, *Report of Committee II on Permissible Dose for Internal Radiation* (ICRP 1960), for intake determination. An action level concentration was defined for each radionuclide, which was based on a fraction of a body burden. If the individual's urine had a concentration in excess of the action level for that particular radionuclide, a follow-up bioassay sample was requested. If the second bioassay sample was positive, the individual was identified as having a confirmed assimilation. The SRS IDR is supposed to include these individuals. In about 1984, the site implemented the ICRP 30 methodology for dose calculation for radionuclides other than tritium. At that time, the criterion for inclusion in the IDR was changed to include those individuals who received 100 mrem during the first year following intake (DPSOP 1987). With the release of the DOE *Radiological Control Manual* (DOE 1994), the criterion was changed to 100 mrem CEDE. Most recently, the criterion for inclusion in the SRS IDR was established at 10 mrem CEDE. The IDR includes the date of intake, the radionuclide, the intake quantity (nCi), annual doses for 2002 and 2003, CEDE, and personal information about the individual. The IDR includes intakes of Am-241, Cm-244, Co-60, Cs-137, Np-237, Pu-238, Pu-239, Pu-241, Sr-90, U-234, U-235, U-238, Ce-144, Cf-252, Cm-242, Nb-95, Ru-106, Zn-65, and Zr-95. Approximately 1,100 individuals are included in the registry.

ORAUT-OTIB-0001 (Brackett 2003) used the average intake quantity for each radionuclide listed in the registry of the five largest documented intakes, or the average of all intakes if there were fewer than five intakes reported for each radionuclide (Brackett 2003). Table 4-9 summarizes the radionuclides, intake quantities, and IREP radiation types.

Table 4-9. Average Hypothetical Intake Quantities for the High-Five Approach

Radionuclide	No. of Intakes	Years of Intakes	Average Intake (nCi)	Assumed Radiation Type for IREP
Am-241	5	1961–1984	14.2	Alpha
Ce-144	5	1967–1986	623.96	
Cf-252	3	1972–1998	7.28	
Cm-242	2	1972–1979	13.35	
Cm-244	5	1964–1971	91.8	Alpha
Co-60	1	1985	430	Electron >15 keV
Cs-137	5	1967–1979	361.5*	Electron >15 keV
Nb-95	1	1983	650	
Np-237	5	1961–1999	1.17	Alpha
Pu-238	5	1967–1981	250	Alpha
Pu-239	5	1956–1979	129.1	Alpha
Pu-241	5	1956–1979	1863.9	Alpha
Ru-106	5	1967–1983	306.6	
Sr-90	5	1959–1986	158.18	Electron >15 keV
U-234	5	1959–1971	105.4	Alpha
U-235	2	1985–1990	0.14	
U-238	5	1953–1971	20.95	Alpha
Zn-65	1	1989	700	
Zr-95	5	1967–1983	359.72	

* The intake used for this dose calculation was slightly larger than the calculated average. Because it was claimant favorable and had a small impact on overall dose, the calculations were not redone.

Source: Brackett 2003

The data used as the basis for the high-five approach are incomplete and do not capture all intakes that have occurred at SRS. ORAUT-OTIB-0001 (Brackett 2003) incorrectly states, "**All recorded inhalation intakes in the history of the site were reviewed and the largest intakes for each radionuclide were selected**" [Emphasis Added]. As mentioned above, the criterion for inclusion in the IDR has changed over time. While the IDR contains many of the intakes that occurred at SRS, neither the TBD or in ORAUT-OTIB-0001 (Brackett 2003) evaluated the completeness of the registry. The site profile and procedure failed to consider other sources of information on internal dose monitoring and incidents. Table 4-10 summarizes examples of intakes that potentially meet the criterion for inclusion in the high-five approach but are not included. Radionuclides not considered in the high-five approach identified in other source documents are also included.

Table 4-10. In-Vivo or Bioassay Information of Relevance Not Included in the High-Five Data

Radionuclide	Date	Result
Source: Visitor Card (i.e., 3" x 5" cards) in vivo count data		
Cm-244	8/11/72	30 nCi
Cm-244	1/15/71	~43 nCi
Cm-244	2/18/72	33 nCi
Cm-244	11/6/72	44 [31] nCi
I-131	6/30/72	2.6 nCi
Pu-239	2/29/72	64 nCi
Pu-239	8/24/72	63 nCi
Pu-239	8/23/72	117 [103] nCi
Pu-239	6/21/72	57 [49] nCi
Cs-134	11/9/70	85 nCi
Cs-134	2/22/71	1.34 nCi
Cs-134	11/9/70	1.9 nCi
Sb-122	11/9/70	206 nCi
Sb-124	11/9/70	78 nCi
Sb-124	1/15/71	283 nCi
Source: Bioassay Results in DPSP Monthly Reports*		
Pu-239	May-66	5.0 dpm/1.5 L
Pu-239	Jul-57	8 dpm/1.5L
Pu-239	Feb-62	9.38 dpm/1.5L
Pu-239	Mar-61	9.6 dpm/1.5L
Pu-239	Jun-60	9.8 dpm/1.5 L
FP	Aug-56	1220 dpm/750mL
FP	Oct-56	3530 dpm/750mL
FP	Jan-57	1628 dpm/750m
FP	Oct-59	567 dpm/750mL
FP	Jun-59	413 dpm/750mL
U-235	Sep-57	320 dpm/1.5L
U-235	Oct-57	146 dpm/1.5L
U-235	Dec-57	232 dpm/1.5L
U-235	May-57	15 dpm/100mL (225 dpm/1.5 L)
U-235	Feb-61	85 dpm/1.5L
U-235	Nov-60	24 dpm/1.5L
Np-237	Feb-61	0.24 dpm/1.5L
Np-237	Oct-61	4.13 dpm/1.5L
Np-237	Feb-63	0.5 dpm/1.5L
Np-237	May-63	4.0 dpm/1.5L
Np-237	Aug-65	0.3 dpm/1.5L
Total U	Nov-59	33 ug/L
Total U	Apr-61	3 ug/L
Total U	May-66	41 ug/L
Total U	Apr-64	48 ug/L
Total U	Dec-56	52 ug/l

*Note: The dates listed represent the date of the report, not the bioassay sample.

The above examples are only a fraction of the data identified from the DPSP monthly progress reports. Figure 4-2 depicts a synopsis of the bioassay data in the monthly reports versus those from individuals included in the high-five analysis for Pu-239 and Np-237. The monthly reports generally provide bioassay information up through 1965 that is higher than that observed for individuals represented in the high-five analysis. The monthly progress reports accessible to SC&A were available only through 1965 for this analysis.

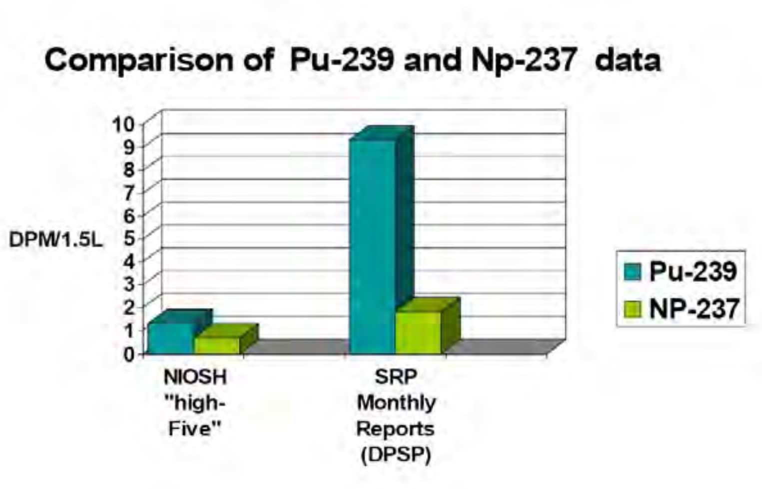

Figure 4-2. Pu-239 and Np-237 Bioassay Comparison between Data in the Monthly Reports and Data Used for the High-Five Approach

Progress reports also included generic statements of relevance that warrant further investigation. The progress report from the Works Technical Department for December 1960 (DPSP 1960k) indicates that other intakes of Nb-95 occurred prior to 1983:

> *Routine examination of manufacturing area personnel began early in December. The first employees scheduled were from the Health Physics Section in the Separations Areas. Body burdens of 58 men were measured. Ce-Pr^{144} was detected in 66% of the personnel, Ru-Rh^{106} in 29% and Zr-Nb^{95} in 22%. The maximum body burdens found in these individuals were 49 nanocuries of Ce-Pr^{144}, 36 nanocuries of Ru-Rh^{106}, 7.2 nanocuries of Zr-Nb^{95} and 24 nanocuries of Cs-Ba^{137}.*

To further bolster the assertion that uranium posed unimportant risks, McCarty (2000) states the following:

> *These protection measures, not withstanding, records indicate that 99 workers received internal doses of uranium over the history of the plant, which are well documented in site incident reports.*

There is concern that this number of uranium uptakes is based on data currently being used by ORAU for dose reconstruction purposes. However, a preliminary review of an incomplete set of

Works Technical Department reports indicates that there were 155 positive bioassays for uranium between 1953 and 1959 alone.[3]

There are also other indications that high intakes, possibly higher than those listed, may have been missed. Only one of the five entries for Pu-239 in Table 1 of ORAUT-OTIB-0001 (Brackett 2003) regarding the largest intakes assigned at SRS (based on radionuclides available in IMBA) is from the 1950s; one is from 1962 and the rest are from the 1970s (Brackett 2003, p. 4). This is surprising since the highest exposures in the nuclear weapons complex typically occurred in the 1960s and earlier. None of the high Cs-137 intakes in Table 1 are from the 1950s. Three of the high Sr-90 intakes are reported on the same day in the 1980s (November 5, 1986) and are very close in value. As another example, the production of Pu-238 from Np-237 targets started in the late 1950s and ended in 1986 (Reed et al. 2002, p. 429), but four of the five Np-237 entries in Table 1 are from the 1990s. At the same time, all of the Pu-238 entries are from the period of production. These are among the indicators that the records used to compile the high-five analysis may be inadequate to determine the highest five intakes for the listed radionuclides. Table 2 of ORAUT-OTIB-0001 (Brackett 2003) does not contain any entries for the 1950s for fission products.

SC&A's review also revealed that incidents during the early years may have been under-reported. For example, at the time of a significant incident, one would expect a follow-up that included a detailed review of the circumstances and special bioassay monitoring of the exposed individual. These types of follow-up activities likely occurred, because SC&A noted more than 400 cases of post incident chelation therapy. Based on these findings, SC&A suggests that the site profile provide direction to dose reconstructors on how to identify the occurrence of incidents, investigate incidents, and obtain records and data sources that can be useful in reconstructing doses from incidents when bioassay data or personnel dosimetry are lacking or suspect.

Several radionuclides had no bioassay technique available in the early years during peak production. For example, the TBD indicates that analytical methods for Np-237 were not available until about 1959, and that analytical methods for americium, curium, and californium were not available until the mid-1960s. One of the first incidents investigated at the site in September 1954 involved the spread of contamination from an americium source (Nichols et al. 1954), demonstrating that americium was present on the site prior to the development of a bioassay technique. The lack of monitoring data from this era of operations also brings into question whether the highest intakes for each radionuclide have actually been captured.

The TBD does not mention Am-243, Eu-152, Eu-154, M-54, Sb-125, Th-228, Th-232, and U-236, although the SRS *Internal Dosimetry Technical Basis Manual* (WSRC 2001) lists them as radionuclides of concern. ORAUT-OTIB-0001 (Brackett 2003) outlines the default radiation type for radionuclides in the high-five analysis. It is unclear why NIOSH restricts radionuclides

[3] DPSP 1956, DPSP 1956a, DPSP 1956b, DPSP 1956c,, DPSP 1956e, DPSP 1956f, DPSP 1957a, DPSP 1957b, DPSP 1957c, DPSP 1957d, DPSP 1957e, DPSP 1957f, DPSP 1957g, DPSP 1957h, DPSP 1957i, DPSP 1957j, DPSP 1957k, DPSP 1958a,, DPSP 1958b, DPSP 1958c, DPSP 1958d, DPSP 1958e, DPSP 1958f, DPSP 1959a, DPSP 1959b, DPSP1959c, DPSP 1959d, DPSP 1959e, DPSP 1959f, DPSP 1959g, DPSP 1959h, DPSP 1959i, DPSP 1959j, DPSP 1959k, DPSP1959l.

to only one radiation type when multiple high-yield radiation types are associated with a radionuclide. For example, NIOSH does not consider photon exposures from Cs-137 or Co-60. Radiation types should represent all high-yield forms emitted by the particular radionuclide.

In summary, the completeness of the IDR should be evaluated against incident and technical reports, as well as against operations history. NIOSH should also evaluate the adequacy of the bioassay program during the period of operation and the radionuclides in the source terms with regard to the complex history of radionuclide production. Furthermore, NIOSH should consider the impact of the bioassay program's adequacy on internal dose reconstructions. The high-five approach cannot be used to cover all omissions of data for internal dose since its use is limited to organs that do not concentrate internally deposited radionuclides and to workers with little or no apparent internal dose.

4.9.1.1 3"x 5" Visitor Cards

In the past, the 3"x5" cards were used to record internal (i.e., in vivo) and external monitoring data for visitors, subcontractors, construction workers, and employees who were assigned a temporary badge. The 3"x5" cards are sometimes referred to as visitor cards. Historically, these cards were not typically included in the Health Physics files, but were stored separately. SRS scanned the data from all the 3"x5" cards into pdf format (Morgan 2006).

The NIOSH Health Related Energy Research Branch (HERB) obtained imaged copies of at least some of the visitor cards for the leukemia case-control study. OCAS was referred to NIOSH-HERB for a copy of these 3"x5" cards, but was unable to locate them or obtain any copies. SC&A provided NIOSH-OCAS with a few examples of these cards on June 29, 2007, to assist with their search. In lieu of these card images, NIOSH reviewed 20 SRS claims with early work histories for these cards and located scanned data cards within claimant files. NIOSH further noted that the information on the cards only related to external dose. However, in a review of a fraction of 3" x 5" cards, SC&A noted that a few cards also contained internal monitoring data. NIOSH should verify that all 3"x5" card data is included in the claimant files submitted by SRS, and that both internal and external monitoring information from these cards are considered during dose reconstruction.

SRS entered data after 1970 into the Health Visitor Information System (HVIS) database. It is uncertain how many of these data were entered into HPAREH or HVIS in the version used by NIOSH. SC&A could provide NIOSH with a limited subset of this data; however, it is more beneficial for NIOSH to request the complete set of data from SRS.

4.9.2 Applicability to Revision 03

This issue is applicable to Revision 03 and ORAUT-OTIB-0001 (Brackett 2003). Revision 03 of the site profile provides the same direction to dose reconstructors as the previously reviewed version. ORAUT-OTIB-0001 (Brackett 2003) describes the high-five methodology in detail.

4.9.3 Summary of Overall NIOSH Response

It is extremely unlikely that data for larger intakes exist yet were not included in the database used. The specific purpose of the database was to capture all significant intakes at SRS. The high-five approach includes only the largest five intakes for any given nuclide.

SC&A provided an email regarding the 3"x5" cards. NIOSH asked ORAUT on October 10, 2006, to determine whether these 3"x5" cards already existed in the HERB data folder on the o-drive. No 3"x5" cards exist in the HERB data folder on the o-drive, nor does NIOSH have 3"x5" cards in any of its records collections. In July 2007, NIOSH made a special data capture trip to SRS to attempt to discover more information about intakes. The trip yielded standard bioassay cards, not 3"x5" cards but instead 6"x9" cards. NIOSH therefore asked SC&A to clarify the issue.

NIOSH indicated that the original text file of bioassay data for those individuals included in the high-five approach analysis was located on the O-drive at O\AB document review\SRS TBD review\srs.text.

4.9.4 Work Group Action

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Identify the location of confirmed intake registry and bioassay data on the o-drive. (completed)
- Complete the update of high-five intakes using new models and provide this to the Board.
- Review the incident database (i.e., Fault Tree Data Bank) for applicability.
- Make the copy of the incident databases available for Work Group review when the copy is received.

SC&A Actions:

- Assist in providing additional information on the SRS 3"x5" visitor cards and identify the intake information discussed during Work Group meetings. (Completed)

4.9.5 Closure Status

Revision 03 includes the same information on the high-five approach as the previous revision reviewed by SC&A. Proposed Revision 04-E (Scalsky 2006) clarifies the purpose and application of the high-five approach but does not resolve the issues related to the basis for the high-five approach. The completeness of the data used as the basis for the high-five approach is questionable. NIOSH has not reviewed all recorded inhalation intakes in the history of the site despite its indication to the contrary. The criteria for inclusion of individuals in the SRS intake file have changed over time such that some intakes are excluded. Other sources of information such as the DPSP monthly reports, data banks, incident files, and visitor cards were not considered when identifying intakes. SC&A has presented numerous examples from these sources that likely meet the criteria for inclusion in the high-five approach and deserve further investigation. The general bioassay trend for Np-237 and Pu-239 from the individual high-five data and the DPSP monthly report data indicates that monthly report bioassay results were higher. In addition, some radionuclides were present at the site prior to the availability of bioassay techniques. It is recommended that several Work Group actions remain open, including clarification of the location of the SRS high-five bioassay data, review of the tank farms Fault Tree Data Bank, and completion of the revised high-five approach.

4.10 COMMENT 10: TECHNICAL ISSUES ASSOCIATED WITH THE HIGH-FIVE APPROACH

Technical issues are associated with the high-five approach. The use of surrogate data for internal dose for unmonitored workers and for target organs that do not concentrate the radionuclides in question is not necessarily a maximizing approach for making dose estimates, contrary to the claim in the TBD.

4.10.1 Issue Description

The hypothetical intake outlined in ORAUT-OTIB-0001 (Brackett 2003) uses the SRS IDR to identify the highest five intake quantities (nCi) for each radionuclide in the IDR, or all available intakes if the reported intakes for a given radionuclide are less than five. The intake quantities calculated using ICRP 30 methodology is then averaged. The average activity (nCi) is entered into IBMA and a dose is calculated based on the ICRP 66 and ICRP 68 models. For each, the dose reconstructor is instructed to assume an acute inhalation occurred on January 1 in the first year of employment (Brackett 2003).

NIOSH justified the use of intakes calculated with the ICRP 30 methodology rather than the newer ICRP methodology (e.g., ICRP 66, ICRP 68) prescribed for the dose reconstruction effort by comparing intake retention fractions (IRFs) from ICRP 30 and ICRP 68. This justification is not necessarily claimant favorable. The use of ICRP 30 models does not produce intake values that are higher than those derived by the new ICRP models for a majority of the relevant radionuclides included in the hypothetical intake.

Plutonium and americium are not significantly overestimated as stated in ORAUT-OTIB-0001 (Brackett 2003, pg. 8). In fact, intakes from Zr-95 (Types M and F), Zn-65 (Type S), Ru-106 (Type S), Nb-95 (Type M), Cf-252 (Type M), Ce-144 (Type M), Cs-137 (Type M), Co-60 (Type M), Sr-90 (Type S), uranium (Types F, M, and S), plutonium (Types M and S), and Am-241 (Type S) for all reasonable times of collecting samples, after an intake occurred, are underestimated using ICRP 30 methodology instead of the ICRP 68 biokinetic model. Types M and F of Ru-106 are underestimated most of the time using ICRP 30 methodology. For Type M of Am-241, ICRP 30 methodology may or may not underestimate the intakes, depending on the time samples are taken after the intake. Attachment 6 of SC&A's previous review (SC&A 2005a) presents calculations demonstrating that the approach used in ORAUT-OTIB-0001 (Brackett 2003) is not claimant favorable for many radionuclides and that ICRP 68 models would have been more claimant favorable.

4.10.2 Summary of Overall NIOSH Response

This approach is used as an overestimate for people who were not monitored or had no positive bioassay results. It assigns all possible nuclides for which there are recorded intakes on the site and treats isotopes of the same nuclide independently. For example, an individual is assigned the largest U-238 intake, the largest U-234 intake, and the largest U-235 intake, even though these would have been from different source terms (depleted versus enriched uranium). It is not plausible that an unmonitored individual would have a larger intake of every radionuclide than the largest recorded in the history of the site without some kind of indicator.

The data for the individual workers upon which the high-five doses are based have been retrieved from SRS. These data are currently being entered into an electronic database. The reevaluation of the cases using the newer ICRP (60, 66–69, 71–72) models is expected to be completed by the end of August 2006.

4.10.3 Work Group Action

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Complete the update of high-five intakes using new models and provide this to the Board.

4.10.4 Closure Status

Revision 03 (Scalsky 2005) and Revision 04-E (Scalsky 2006) reference ORAUT-OTIB-0001 (Brackett 2003), which continues to use both ICRP 30 and subsequently ICRP 66 as the method for assignment of internal dose. ORAUT-OTIB-0001 (Brackett 2003) justifies the use of intakes calculated with the ICRP 30 methodology rather than the most current ICRP methodology by comparing IRFs from ICRP 30 and ICRP 68. The ICRP 30 model does not produce intake values that are higher than those derived by the new ICRP models for a majority of the relevant radionuclides included in the hypothetical intake. The use of the ICRP 30 methodology to calculate the intake, with a subsequent use of ICRP 68 models to calculate the dose, did not always result in the intended highest dose to an organ. Similarly, the appropriate solubility types between the two methodologies were not always paired consistently, resulting in discrepancies that were not claimant favorable. It is recommended that this issue remain open pending the update of the high-five method using bioassay data and appropriate models.

4.11 COMMENT 11: ICRP 30 WITH SUBSEQUENT USE OF ICRP 68

For internal dose calculations, the use of ICRP 30 methodology to calculate the intake, with a subsequent use of ICRP 68 models to calculate the dose, did not always result in the intended highest dose to an organ. Similarly, the appropriate solubility types between the two methodologies were not always paired consistently, resulting in discrepancies that were not claimant favorable.

4.11.1 Issue Description

The hypothetical intake outlined in ORAUT-OTIB-0001 (Brackett 2003) uses an average of the highest five intakes in the SRS IDR quantities calculated using the ICRP 30 methodology. The average activity (nCi) is entered into IBMA and a dose is calculated based on the ICRP 66 and ICRP 68 models. For each dose reconstruction where bioassay data are lacking (and which meet certain other criteria that are described below), the dose reconstructor is instructed to assume that an acute inhalation occurred on January 1 in the first year of employment (Brackett 2003). The use of two different ICRP methodologies requires assumptions of two solubility types. The first solubility assumption was made by SRS when they calculated the intake using ICRP 30. The second solubility assumption was made by NIOSH when the intake was entered into IMBA.

The appropriate solubility types applied in the comparison were not always paired between the two methodologies (i.e., Type F corresponding to Class D, Type M corresponding to Class W, and Type S corresponding to Class Y). Instead, the ICRP 68 solubility types are chosen "as the most soluble form of the material because it would clear from the lung more rapidly than insoluble material, thus depositing in the organ or tissue sooner" (Brackett 2003). The ICRP 30 classes, on the other hand, are chosen using "the material class(es) applied for the SRS calculated intakes in Tables 1 and 2" (Brackett 2003, pp. 5–8). There is a fundamental problem with the comparisons of these IRFs from ICRP 30 and ICRP 68. In general, when intakes are used to calculate organ doses, the choice of the most soluble type is claimant favorable for doses calculated to systemic organs. When bioassay results are used to calculate organ doses, many times the assignment of the most insoluble material type results in a higher dose for systemic organs, as illustrated by the following example (SC&A 2005a, p. 152):

- A 24-hour urine sample is collected five days after a single inhalation intake of Pu-238. The bioassay result is 1 Bq of Pu-238. Using the model in *Age-dependent Doses to Members of the Public from Intake of Radionculides: Part 2, Ingestion Dose Coefficients* (ICRP 67) for plutonium, the calculated intakes are as follows:

    - For Pu-238, Type S: intake of 2.2×10^6 Bq [50-year committed bone surface dose is 75 sieverts (Sv), 50-year committed dose to colon is 0.053 Sv, 1-year committed dose to the colon is 0.006 Sv]

    - For Pu-238, Type M: intake of 2.6×10^4 Bq (50-year committed bone surface dose is 24 Sv, 50-year committed dose to colon is 0.042 Sv, 1-year committed dose to the colon is 0.002 Sv)

Thus, the use of Pu-238, Type S, results in a higher intake than the use of Type M (and in higher doses to systemic organs).

- Using the ICRP 30 IRF from Table 3 of ORAUT-OTIB-0001 (Brackett 2003) , the same bioassay result of 1 Bq of Pu-238 in a 24-hour urine sample taken 5 days after a single intake corresponds to the following intakes:

- For Class Y: intake of 3.5×10^5 Bq (ICRP 30) (50-year committed bone surface dose is 12 Sv, 50-year committed dose to colon is 0.33 Sv, 1-year committed dose to the colon is 0.037 Sv)

- For Class W: intake of 1.9E4 Bq (ICRP 30) (50-year committed bone surface dose is 17.5 Sv, 50-year committed dose to colon is 0.03 Sv, 1-year committed dose to the colon is 0.0015 Sv)

Thus, the more claimant-favorable approach to choosing solubility type should be initiated with the intake calculation and not limited to the internal dose calculation.

ORAUT-OTIB-0001 (Brackett 2003) directs the dose reconstructor to use surrogate radionuclides for radionuclides included in the high-five approach, which are not available in IMBA. Since the initial review of the SRS site profile, the IMBA code was updated to include radionuclides previously not available, eliminating the need to use surrogate organs. ORAUT-OTIB-0001 (Brackett 2003) should be updated to reflect this change.

4.11.2 Applicability to Revision 03

This comment is applicable to Revision 03 and ORAUT-OTIB-0001 (Brackett 2003). The proposed use of bioassay data will resolve this issue.

4.11.3 Summary of Overall NIOSH Response

ORAUT-OTIB-0001 (Brackett 2003) contains information demonstrating that intakes calculated using ICRP 30 IRFS are in most cases larger than those obtained from ICRP 68 IRFs. The subsequent dose would have been smaller had ICRP 30 DCFs been used, but ICRP 68 DCFs were used. As noted in the response to Comment 10, NIOSH did not have access to the original bioassay results at the time ORAUT-OTIB-0001 (Brackett 2003) was written; SC&A had the intakes that had already been calculated by SRS. Solubility types were not chosen to match the SRS-calculated intakes; they were selected to maximize dose to the systemic organs once an intake had already been determined.

It should also be noted that the latest version of IMBA contains radionuclides previously not available, eliminating the need for surrogate radionuclides in many cases.

4.11.4 Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Complete the update of high-five intakes using new models and provide this to the Board.

4.11.5 Closure Status

Revisions 03 and 04-E of the TBD reference ORAUT-OTIB-0001 (Brackett 2003), which continues to use two internal dose assessment methodologies with at inappropriate pairing of solubilities. Furthermore, the procedure includes recommendations for the use of surrogate organs. For internal dose calculations, the use of ICRP 30 methodology to calculate the intake, with a subsequent use of ICRP 68 models to calculate the dose, did not always result in the intended highest dose to an organ. Similarly, the appropriate solubility types between the two methodologies were not always paired consistently, resulting in discrepancies that were not claimant favorable. The dose reconstructor is directed to use surrogate radionuclides for radionuclides absent from the IMBA code. The optimum solution is to add these radionuclides to the code. In lieu of this, a more prudent approach to address the absence of radionuclides is to use the dose coefficients provided in ICRP (2001) and employ a linear interpolation for the radionuclides that are not explicitly given. This issue has been partially resolved since the initial SC&A review with the inclusion of additional radionuclides in the latest version of the IMBA code, eliminating the need to use surrogate organs in many cases. ORAUT-OTIB-0001 (Brackett 2003) should be updated to reflect these and other changes to the high-five approach. The closure of this issue is pending the update of the high-five method using bioassay data and appropriate models.

4.12 COMMENT 12: SOLUBILITY, ORO-NASAL BREATHING, AND INGESTION ASSUMPTIONS

Internal dosimetry needs to be improved with regard to radionuclide solubility, oro-nasal breathing, and the ingestion pathway. These factors should be carefully considered with regard to internal dose reconstruction. SC&A originally developed the points described below for the review of the Bethlehem Steel and Mallinckrodt Chemical Works site profiles, and they are applicable for all bioassay interpretations for EEOICPA.

4.12.1 Solubility Assumptions

The solubility assumptions that are used to estimate organ dose from urine need to be discussed in the TBD. For example, an assumption of Type S or Type M (and Type F in the case of UNH) must be considered more carefully when deriving doses to organs based on urinalysis data, since a Type S assumption in this case may yield a higher dose for nonrespiratory tract organs than a Type M assumption. The analysis of organ dose from urine data can be complex, and more specific analysis is needed in any future revisions of the TBD.

Oro-nasal Breathing

SC&A has addressed oro-nasal breathing in detail in Attachment 5 of SCA-TR-TASK1-0002, *Review of NIOSH Site Profile for Mallinckrodt Chemical Company, St. Louis Downtown Site St. Louis, Missouri* (SC&A 2005c). That finding is also applicable to SRS workers. Oro-nasal breathing affects intakes for light as well as heavy work. The assumption of oro-nasal breathing should be used in a manner similar to solubility assumptions—that is, uncertainty as to whether a worker breathed through the mouth should be addressed and a determination made whether NIOSH should continue to strictly follow ICRP models, which do not address oro-nasal breathing, or whether oro-nasal breathing should be included in dose reconstructions as a more claimant-favorable assumption. Oro-nasal breathing needs to be taken into account when air concentration data are used in estimating intakes and doses, as for instance in estimating environmental occupational dose (Section 3 of the TBD) or missed dose due to fission products (p. 80 of the TBD).

Ingestion

NIOSH/ORAUT has assumed that inhalation is the only pathway for internal exposure at SRS. During discussions between SC&A and NIOSH/ORAUT, the following issue arose with regard to ingestion doses:

> *Question:*
>
> *For purposes of internal dose calculations; are airborne release levels well documented; are potentials for ingestion and inhalation sufficiently documented; are bioassay techniques well documented and is each bioassay technique's uncertainty and accuracy well understood?*
>
> *Answer:*
>
> *"Potentials for inhalation" are accounted for by estimating missed dose or accounting for unmonitored periods. Ingestion is not usually considered at major DOE sites* [however, it is important at Atomic Weapons Employers], *but uptake from the GI tract is accounted for in the bioassay, although the default intake mode is inhalation unless a worker's records have information indicating otherwise.*

SC&A believes that ingestion cannot be ignored a priori by assuming a default value. Furthermore, in order to take ingestion into account using bioassay data, the inhalation component must be known. In other words, a single bioassay result gives one data point, but there are two unknowns: how much was inhaled and how much was ingested. One cannot solve this problem accurately without one more data point.

There are several ways to approach this problem. The first, of course, is to look for an additional data point. This could be provided by an in vivo count, for instance. The problem would also be solvable if fecal analysis and urinalysis data were both available for estimating the intake in question. Although it currently does not, the TBD would have to then specify a procedure for solving for the inhalation and ingestion intakes. Revision 03 of the TBD adds the annual dose from ingestion of foodstuffs to the environmental discussion in Section 3.0. Doses are calculated for the ingestion of 60 kilograms (wet mass) of food per year for Sr-90, Cs-137, Pu-238, Pu-239, and tritium.

This gap creates a potential issue at all DOE sites. It should also be noted that this issue does not affect only the early period; it will not appear on **any** worker records. The potential maximum effect on organ doses for each important radionuclide can be assessed by a simple screening technique (not to be misunderstood as a dose estimate). If one assumes inhalation is equal to zero in interpreting the bioassay data, then one can obtain a theoretical maximum ingestion value and corresponding organ doses. This puts an upper limit on potential errors. NIOSH may wish to perform these types of screening analyses in order to close out this issue.

4.12.2 Applicability to Revision 03

Revision 03 includes a discussion of dose from the ingestion of foodstuffs that was not included in previous revisions.

4.12.3 Summary of Overall NIOSH Response

Although ORAUT believes that the assumptions for internal dosimetry are reasonable, it understands this to be a generic issue outside the scope of the SRS site profile that is being addressed by NIOSH and SC&A.

4.12.4 Work Group Actions

There are no Work Group Actions at this time.

4.12.5 Closure Status

Section 3.0, Revision 03 of the TBD, now contains an annual dose from the ingestion of foodstuffs. Doses are calculated for the ingestion of 60 kilograms (wet mass) of food per year for Sr-90, Cs-137, Pu-238, Pu-239, and tritium. Proposed changes in Revision 04-E (Scalsky 2006) do not include any additional information on ingestion dose. While inhalation is likely to be the predominant intake mode the vast majority of the time, the TBD needs to provide an overall analysis to sustain the default assumption that inhalation is always the predominant intake mode. The importance of such an evaluation is increased because doses are being estimated for individuals rather than populations.

Onsite atmospheric dispersion and resuspension. The method used to reconstruct doses to unmonitored outdoor workers due to airborne emissions employs an atmospheric dispersion model, assumptions, and a resuspension factor that do not appear to be claimant favorable and are not entirely appropriate for this class of problem. Revision 03 and proposed Revision 04-E (Scalsky 2006) do not provide any additional clarification of these issues. SC&A **recommends that this issue remain OPEN pending further Work Group review.**

4.13 COMMENT 13: INCIDENTS AND HIGH-RISK JOBS NOT ADEQUATELY ADDRESSED

The TBD does not list or reference incidents and high-risk jobs to alert dose reconstructors to unique exposure conditions.

4.13.1 Issue Description

The TBD does not address exposure conditions that may present themselves during an incident or occurrence. SC&A maintains that some workers have sustained significant exposures that are relevant to their dose reconstruction but may not be captured by reviewing the DOE individual dosimetry file and the CATI. In site expert interviews conducted during the initial review, SRS staff indicated that the individual dosimetry file contains many, but not all, incident investigations. Furthermore, SRS implemented several methods for documenting incidents and occurrences. Other sources of incident information include SHI reports, the WSMS incident database, the tank farms data bank, Works Technical Department progress reports, and multiple other incident files corresponding to the main areas of the site (e.g., tritium facility incidents, raw materials incidents, and reactor incidents). These reports provide brief descriptions of the incidents and were maintained separately from the individual radiation exposure records. Although recommended in the original SC&A review, NIOSH only recently (following the February 2007 site visit) obtained a copy of the SHIs. Prior to this, these reports were not considered in the dose reconstructions or development of efficiency methods. In addition, the tank farm data bank entries have not been evaluated to identify significant exposure situations and environmental releases, which may be significant in dose calculations. SC&A was unable to obtain a copy of the tank farm data bank for this review.

Per the telephone conversation held between SC&A and NIOSH during the previous review, the DOE exposure file and the CATI provide the mechanisms for identifying incidents, and the SRS personnel radiation exposure file includes many incident reports. The CATI interview is used as a secondary source of information on incidents or occurrences. While the CATI provides some potential to identify incidents that may be missing from the personnel radiation exposure file, reliance on the CATI for this purpose places an inappropriate burden upon the worker or survivor to recall events and to recognize potential implications for the dose reconstruction. This concern particularly impacts family member claimants, as they are far less likely to have knowledge of incidents and their relevance to the claim. As a result, there is a disadvantage for dose reconstructions that must rely on the CATI to determine the possibility that the worker was involved in an incident. The CATI should be used as a positive indicator of an incident; however, it should not be used to rule out the existence of incidents.

Although individuals involved in incidents are usually monitored, the incident itself may pose special exposure conditions that need to be considered in the dose reconstruction (e.g., injection versus inhalation, partial body exposure to an external beam, cleanup of a spill involving nontraditional radionuclides). Incident databases maintained by SRS provide valuable information on the nature of incidents and source documents and reduce the disadvantage to family member as opposed to energy employee claimants. A redundant system for incident identification is necessary for an effective evaluation of incidents and accidents. The high-five approach cannot be used to cover all omissions of data for internal dose since its use is limited to organs that do not concentrate internally deposited radionuclides and to workers with little or no apparent internal dose. Without a thorough reconciliation of the DOE individual dosimetry files against these separate incident data banks, NIOSH cannot be assured that all significant exposures from incidents are submitted by SRS and considered in relation to individual worker claims or to high-five estimates of maximal doses.

One source of incident information mentioned in the original SC&A review is the SHIs. DPSOP-40, Special Hazard Bulletin 2, *Investigating Radiation and Contamination Incidents* (DPSOP 1981), contains the standard operating procedure for SHIs. The following are among the incidents required to be investigated under this procedure (DPSOP 1981):

- *Acts or situations which caused or could have caused hazardous radiation or contamination conditions.*

- *Contamination incidents which can lead to significant loss of containment of radioactivity, require costly clean up, or concern to Health Protection.*

- *Incidents that result in body contamination or radiation exposure of concern to Health Protection or Medical.*

SC&A's previous review (SC&A 2005a) included several examples of incidents from the tank farm Fault Tree Data Bank that meet the criteria for SHIs (DPSOP 1981). Only three of the 12 incidents from the databank were not found in DuPont's log of SHIs (DuPont 1990). Incidents selected from the tank farm data bank contained three estimated internal exposure entries (one in the F-Area in 1972 and two during the same incident in the H-Area in 1974) that were larger than the lowest two values listed in ORAUT-OTIB-0001 (Brackett 2003), Table 1, for the high-five Cs-137 intakes. The average for the NIOSH high-five Cs-137 intake is about 361 nanocuries, while the SC&A average, using multiple sources of data, is about 475 nanocuries, or about 31% higher. Furthermore, it is unclear how the standard operating procedure for SHIs has changed over time.

SRS staff obtained a copy of a database (containing major and minor incidents through 1999) from WSMS. Because this database was lengthy and contained classified information, a review team assembled on site to review its content. Sam Glover (NIOSH), Elyse Thomas (ORAUT), Brad Clawson (Advisory Board), Mark Griffin (Advisory Board), and Kathryn Robertson-DeMers (SC&A) visited SRS from February 28 through March 1, 2007, to review an incident database thought to be the tank farm Fault Tree Data Bank, and talk to individuals knowledgeable about this particular database. The three objectives of the visit were to determine the contents of the database, compare entries in this database to those from the tank farm Fault Tree Data Bank, and determine its usefulness in dose reconstruction. As a secondary task, the group conducted an interview specifically dealing with STCs at the site.

A former SRS employee developed the WSMS database to provide data for safety analysis. It incorporates information from progress reports, incident investigation reports, field logbooks, and other sources containing safety incident information. The main table of the database, referred to as "incidents," includes as core information an incident number, incident date, database entry date, and a text description of the event. Five definition tables identify the source of information, area, facility, operation, and equipment. Five intermediate tables contain information necessary to link the definition tables to particular incidents. The incidents table does not identify individuals involved in the incident, but this information may be contained in the referenced source documents. The review team was not able to interpret the codes used to identify source documents in the database and was therefore unable to pull source documents to examine their content. SRS was asked to provide a database manual to define the database codes more clearly. SRS has determined that the originator of the database is alive and has agreed to attempt to contact him. It would be beneficial for the reviewers to interview this individual to better understand the content and objectives of the database.

The team reviewed a representative collection of incidents from the WSMS database to evaluate the scope and potential utility of the information. The data appear to cover a limited scope of site operations. Most incidents relate to the 200-F and 200-H Areas, with minimal data from the reactor areas, heavy water area, the Savannah River Technical Center, the raw materials area, and other site operations. The vast majority of entries reviewed (more than 90%) relate to the separations area operations, including tritium operations. As expected, the early years (1950s/1960s) have substantially fewer entries compared to later years. Statistics of the number of entries by year, facility code, and area code were compiled into a Microsoft Excel spreadsheet and are shown in Attachment 2 to this report. Several generic examples of incidents were also compiled in a spreadsheet to inform the Work Group more effectively of the types of incidents related to radiological exposure. Attachment 3 to this report provides the recommended selection criteria for the retrieval of pertinent incidents in the WSMS database.

The WSMS incident database contains thousands of entries relating to worker uptakes, high airborne concentrations, wounds, high dose rate exposure situations, skin contaminations, and bioassays. These data are relevant to the identification of incidents by dose reconstructors and provide valuable information for the development and/or verification of efficiency methods. In order to link the incidents in this file with individual workers, it is essential to identify and review the source data. Some of these source documents are readily available to NIOSH. However, the database also contains incidents that are not of interest in dose reconstruction, such as basketball, skiing, and false fire alarms. Therefore, information from this database should be requested based on the information compiled in the spreadsheet of generic examples and available documentation to isolate radiological incidents. Since it does not adequately represent all areas of the site, the WSMS incident database cannot be used as the sole source of SRS incidents. However, a combination of the incident information in the database and the source document information would provide one mechanism for determining the completeness of incidents in the individual Health Physics file. The entries in the WSMS incident database were compared to those abstracted from the tank farm Fault Tree Data Bank in Makhijani et al. (1986), as previously discussed in Comment 4.

Most of the entries reviewed from the tank farm Fault Tree Data Bank were not located in the WSMS incident database. As previously mentioned, this may reflect a difference in operational scope between the two collections of incident data. SRS requested Central Records to search the records database for “Fault Tree.” However, the observations made to this point imply that multiple Fault Tree Data Banks have been maintained, corresponding to the various areas of the site. For example, the raw materials area has its own data bank.

Additional data may also be available for this purpose for the period 1956 to 1965 in Works Technical Department progress reports (DPSP). These reports provide insights into facility-specific high-risk job exposures relative to maintenance and repair of reactors; entries into very high dose rate areas in the F- and H-Canyons; repair and maintenance of contaminated equipment, suck-backs, and outside ambient exposures from process buildings; and waste tank maintenance, repair and surveillance of high-level waste tanks. During this period, SRS management clearly recognized that these activities posed potentially high risks.

The database also refers the reader back to the source document, which may contain additional information on the incident. Having incident information available will also lift the burden of providing this information during the CATIs and bring some equality to this process.

The data review described above appears to underscore the validity of this concern. Multiple databases from various areas of SRS contain a wealth of data regarding incidents and exposures throughout the site's history, and it is clear that all incidents are not reflected in worker exposure records or other radiological records. In order to ensure consistently claimant-favorable dose reconstructions, NIOSH should ensure that incidents involving significant internal or external exposure are either captured in worker exposure records or made accessible to dose reconstructors, with appropriate guidance for when to use such records.

Without a thorough reconciliation of the DOE exposure files against these separate incident data banks, NIOSH cannot be assured that all significant exposures from incidents are considered in relation to individual worker claims or to high-five estimates of maximal doses. These tools should be made available to dose reconstructors as an additional reference.

High-Risk Exposure

In its previous review, SC&A recommended including guidance within the TBD on how to address special exposure conditions. SC&A listed several groups of workers (e.g., construction, subcontract, decontamination/decommissioning) and authorized or unauthorized practices (e.g., recovering/processing U-233 and thorium, producing transplutonium radionuclides, burning tributyl phosphate, opening high-level waste tank risers, and eating contaminated foodstuffs) that may have experienced or caused significant unrecognized or unreported exposures (SC&A 2005a).

Additional data may also be available related to high-risk job exposures in Works Technical Department progress reports (DPSP). These reports provide insights relative to maintenance and repair of reactors; entries into very high dose rate areas in the F- and H-Canyons; repair and maintenance of contaminated equipment, suck-backs, and outside ambient exposures from process buildings; and waste tank maintenance, repair, and surveillance of high-level waste tanks. SRS management clearly recognized that these activities posed potentially high risks.

For consistency among dose reconstructions, the reviewers concluded that the TBD should alert the dose reconstructor to special conditions when a deviation from the standard dose reconstruction methodology is needed.

4.13.2 Applicability to Revision 03

The lack of consideration of incidents and high-risk jobs continues to be applicable to Revision 03. Some progress has been made towards resolution in this area; however, this occurred after the release of Revision 03.

4.13.3 Summary of Overall NIOSH Response

The dose of record information already contains significant dose to workers from incidents. In many cases, the workers will state that no incident occurred, but the records do, in fact, identify incidents. It is possible for a claimant to say that an incident occurred but the records do not identify any such incident. Not many cases exist in which a worker states that an incident and the dose record does not include it. The high-five approach is used as a means to ensure that missing an incident during the performance of a dose reconstruction will not result in an underestimate of the reconstructed doses. In addition, Form OCAS-INT-004 requests DOE to submit relevant incident information for consideration in dose reconstruction.

NIOSH provided information in the OCAS response to SC&A on June 6, 2006, and via comments during the Work Group call on August 22, 2006, that it obtained from staff at SRS, who told NIOSH that the WSMS incident database is no longer maintained at SRS, and that it is maintained by a contractor, who would provide a redacted database for a charge. NIOSH has since contacted both WSMS as owner of the database and SRS as owner of the content. NIOSH spoke with the Nuclear Safety Manager at SRS, who promised to work with the DOE point of contact at SRS to work out details for providing the database to ORAUT, through the assistance of WSMS.

Work Group Actions

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Request of copy of the SHI reports from DOE/WSRC in writing.
- Obtain the user's guide for the WSMS database from DOE/WSRC.
- Research the pedigree of the WSMS database reviewed during the February 28–March 1, 2007, visit to SRS and determine the source database for Makhijani et al. (1986).
    a. Follow up to determine if there is a record of the Freedom of Information Act request associated with Makhijani et al. (1986).
    b. Follow up with the original author of the database reviewed during the site visit.
- Check on the availability of other incident databases and reports (e.g., Waste Management Incident, Raw Materials Incident, Tank Farms Incident, Separations Incident, Reactor Incident)

SC&A Action Items:

- Have the excess spreadsheet (queries) reviewed for release by DOE and forward them to the Work Group, NIOSH, and ORAUT. (Completed)
- In conjunction with the Work Group, provide queries isolating records of interest in the WSMS database. (Completed; see Attachment 3 to this report)

4.13.4 Closure Status

SC&A maintains that some workers have sustained significant exposures that are relevant to their dose reconstruction but may not be captured by reviewing the DOE exposure file and the CATI. Sources other than dosimetry files exist at SRS that summarize incidents and provide supporting documentation. Databases identified may be specific to an area or operation, so a compilation of several sources may be necessary to capture sitewide incidents. The databases vary in the criteria for inclusion, giving a broader perspective on what constitutes an incident. Without a thorough reconciliation of the DOE exposure files against these separate incident data banks, NIOSH cannot be assured that all significant exposures from incidents are considered in relation to individual worker claims or in efficiency methods (e.g., high-five approach, coworker dose assignments). Dose reconstructors should be alerted to these situations. Based on record storage practices, redundant systems are necessary to develop a complete list of incidents. While the CATI provides some potential to identify incidents that may be missing from an individual's dose record, reliance on the CATI for this purpose places an inappropriate burden upon the worker or survivor to recall events and to recognize potential implications for the dose reconstruction.

Revision 03 gave no additional consideration to many potentially unrecognized or unreported exposures from high-risk work activities. Revision 03 did include a discussion of dose from the ingestion of foodstuffs (fruits and vegetables) containing radioactive material. Proposed Revision 04-E (Scalsky 2006) has discussions on potential dose from special campaigns involving U-233, thorium, and transplutonium radionuclides. ORAUT-OTIB-0052 (Chew et al. 2006) addresses dose to construction workers. Neither Revision 03 (Scalsky 2005) or the proposed Revision 04-E (Scalsky 2006) of the TBD include information on decontamination and decommissioning, open burning of solvents, and other high-risk or unusual jobs. It is recommended that this issue remains open pending the completion of Work Group actions and the release of proposed Revision 04-E (Scalsky 2006) of the site profile.

4.14 COMMENT 14: AVAILABILITY OF ADDITIONAL SOURCE DOCUMENTS AND DATA

Based on searches performed for records pertaining to dose reconstruction, SC&A has identified additional sources of dosimetry data, primarily neutron dosimetry data that are not currently provided to NIOSH/ORAUT by SRS, but that may be important to dose reconstruction.

4.14.1 Issue Description

4.14.1.1 Neutron Data

SC&A pointed out in the original review that logbooks containing neutron dose data exist but the information was not being provided to NIOSH in the claimant dosimetry file because SRS had not retrieved and scanned these logbooks (as of February 28, 2007) (Robertson-DeMers 2007). As previously stated, the most complete records at SRS from a population standpoint are the hardcopy records. Prior to the last Work Group meeting, SC&A asked the Radiological Records Manager at SRS the following question:

> *Have you retrieved the neutron exposure logbooks for 1963–1972? If so, is this data provided to NIOSH when requesting claimant information?*

He responded as follows:

> *No. We do not have time allocated for this search.*

Specifically, for 1963–1972, three sets of logbooks contain data for composite (beta/gamma + tritium + neutron) doses, tritium dose only, and neutron dose only. These logbooks would be necessary to decompose the composite data to perform best estimate dose reconstruction. The availability of NTA data not provided in the claimant files was reinforced by the Radiological Records Manager during a meeting with Sam Glover (NIOSH), Elyse Thomas (ORAUT), Mark Griffin (Advisory Board), Brad Clawson (Advisory Board), and Kathryn Robertson-DeMers (SC&A).

SC&A provided a list of pertinent dosimetry records to NIOSH for consideration. In addition to the data sources identified in the site profile, History Associated Incorporated prepared site-specific guides to epidemiologic and health-related records at a number of sites including SRS. A list of records available at SRS is available through the DOE Office of Epidemiologic Studies Web site. This inventory of records includes the following (DOE 1995):

- *Densitometry Records (Personnel Monitoring Film Badge Data), 1952–1976, 1979, 1981–1984, Health Physics Section and Health Protection Department, E.I. du Pont de Nemours and Company….*

  Series Description: This record series contains personnel monitoring data for employees and visitors at the Savannah River Site. Data sheets and computer-generated reports provide exposure data collected from film badges, neutron pencils, and range ring dosimeters. Information includes employee name, payroll number, dosimeter number, health protection area, cycle or pull date, and readings. X-ray and gamma exposures, slow neutron exposures, and current and cumulative open window and shield dosimeter readings are given, usually in millirem. Calibration data includes densitometer, emulsion, code number, developer and reader name, developing temperature, and beta and gamma results in millirem.

- *Neutron Exposure Reports, 1971, Health Physics Section, E.I. du Pont de Nemours and Company….*

  Series Description: This series consists of neutron exposure reports and related correspondence. Information includes employee name, payroll number, thermoluminescent badge number, work area, cycle number, date badge was issued and returned, code number, and dose in millirems.

- *Dosimetry Logs, 1954–1978, Health Physics Section, E.I. du Pont de Nemours and Company…*

  Series Description: This series consists of logs that document the distribution of dosimeters and associated results. Dosimeters include finger rings, film badges, thermoluminescent dosimeters, thermoluminescent neutron dosimeters, and

"NTA" badges. Information includes frequency of dosimeter changes and readings, number distributed to certain areas, calibration data, and results for individual employees.

- *Neutron Dosimetry Data Logs, 1972–1982, 1986–1992, Health Physics Section, E.I. du Pont de Nemours and Company…*

  Series Description: This series consists of logs which record the assignment of thermoluminescent dosimeter (TLD) neutron badges to Savannah River Site employees. Information includes TLD badge number, name, social security or payroll number, supervisor's name and telephone number, date annealed and processed, and cycle number. There are separate sheets for temporary badges. Dates and locations are listed on each sheet. Readings are usually given in millirems. The series also includes Neutron Badges Indices listing the total number of badges issued, extra issues, badges read for cycle, controls, late badges, badges not used, and badges not returned.

- *Neutron Pencil Results (1952), Health Physics Section, E.I. du Pont de Nemours and Company…*

  Series Description: This series consists of lists recording neutron pencil dosimeter results. Information includes employee name, payroll number, neutron dose in millirems, pencil identification number, and reason why doses could not be determined from some pencils.

- *NTA Film Badge Inventory and Calibration Sheets, (1950–1962, 1964–1969)…*

  Series Description: This series consists of "NTA" film badge inventory and calibration sheets and memoranda regarding the "NTA" monitoring program. Information concerns the collection and processing of "NTA" film badges, procedures, suggested revisions to the program, areas where badges were collected, and the date received. "NTA" badge and payroll numbers are included. Some records include employee names, as well as other personnel information.

A review of several records storage transfer request forms from the Health Physics Section was conducted by Kathryn Robertson-DeMers (SC&A). Documents identified as containing neutron exposure information for the period of 1963–1972 are listed below:

- Transmittal dated December 29, 1971:

  NTA Film, 1962 through Cycle 8, 1967, Document #T2092
  NTA Film, Cycle 9, 1967 thru Cycle 6A, 1970, Document #T2091
  NTA Film, Cycle 6B, 1970 thru Cycle 9B 1971, Document #T2101

- Transmittal dated March 14, 1973:

  Neutron Data 1970 and 1971, Document #M440

The combination of sources above should provide a complete history for neutron dose from 1963–1972. NIOSH may want to consider performing a search of the site records database, especially as it pertains to information related to neutron exposure data for individuals. This additional source of data may be relevant to the assignment of neutron dose.

The Web site also lists other pertinent documents related to the SRS evaluation:

- Film Processing and Dosimeter Calibration Procedures (1954–1960)
- Daily Badge Processing Reports (1954–1957)
- Employee Radiation Exposure Record Cards (1964–1992)
- Construction Worker Exposures (1958–1959)
- Bioassay Analysis Reports (1979, 1981, 1989–1992)
- Bioassay and Dosimeter Data Sheets (1961–1972, 1975–1977, 1986–1988)
- Bioassay Logs (1953–1960, 1962–1989, 1991–1992, 1995)
- Bioassay Master Report (1975)
- Bioassay Monthly Reports (1982)
- Bioassay Program Records (1953–1954, 1992)
- Bioassay Results—Fission Products Induced Activity Reports (1965, 1967–1977, 1987–1989)
- Bioassay Sampling Cards (1954–1987)
- Contamination Cases Logbooks (ca. 1977–1992)
- Contamination Incident Reports (1951–1955, 1958, 1971–Current)
- In Vivo Count Results (1982–1983, 1985)
- Tritium Doses (1958–1972)
- Tritium Dosimetry Monthly Reports (1972–1976, 1979, 1981–1989, 1992–1993)
- Tritium Dosimetry Reports (1977–1979, 1983–1987, 1992–1993)
- Type A Occurrence Report (1979)
- Type C Occurrence Reports (1972–1981)
- Uranium Concentration Reports (1989–1990)
- Urinalysis Reports Strontium and Plutonium (1984–1990)

These records may assist NIOSH in its evaluation of the high-five approach, tritium, and incidents.

Completeness of HPAREH

Another issue relates to the fact that the HPAREH database does not contain data for all the workers for 1952 onward. It contains data for those that were monitored beginning in 1979 and afterwards. It also contains data for some (but not most) workers who worked during 1952–1978. This is important to realize when constructing a coworker model for assigning doses, such as OTIB-0032 (Merwin 2006) for beta/gamma doses and if a neutron coworker dose model is constructed in the future. The dose data from the earlier years (1952–1978) for a large population of workers should be included to create realistic year-by-year dose assignment tables for unmonitored workers. It is especially important to include these early years of dose data because doses during these early years most likely were greater than later years as radiation protection practices/knowledge improved.

NIOSH indicates that it presented the issue of what records from prior to 1979 were included in the HPAREH database to an SRS staff member, who replied in an email dated August 24, 2006. NIOSH has not provided the specific questions asked, making it difficult to understand the answer. An SRS staff member did, however, state the following in his response to NIOSH:

> *The latest version of HPAREH at SRS includes the complete dose history for more than 70% of the individuals ever monitored at SRS, and more than 70% of the cumulative collective worker doses delivered at SRS. It is important to note that SRS staff has never envisioned (or used) HPAREH as the only component of the official dose of record for any worker at SRS. The official dose of record includes all the forms of dose records available to us.*

As previously stated above, SRS believes that the most complete records at SRS from a population standpoint are the hardcopy records. Comment 5 above provides a detailed discussion of the limitations of HPAREH.

Multiple Dosimetry Data

A number of conditions can result in partial body exposures or portions of the body being exposed unevenly. SRS recognized this early in its operations and implemented a multiple dosimetry program. Multiple dosimeters were required where measurements showed nonuniform radiation fields (e.g., specific areas within the reactors required head monitoring, as the dose to the head was greater than that to the chest). The term "multiple badging" in this context refers to instances where workers wore more than one badge at the same time in order to capture doses to various parts of the body at risk of greater exposure more accurately than would be indicated by a single dosimeter worn at the pocket level. The TBD has not mentioned the multiple badging programs and how this information could be applied to assigning organ dose.

During site expert interviews, SC&A learned that multiple badge results are not routinely included in the personnel radiation exposure record and are not currently provided to NIOSH/ORAUT. The whole-body dose for an individual wearing multiple badges was assumed to be the highest recorded result on any of the badges. For example, if the results from four badges positioned over the body were 100 mrem, 200 mrem, 300 mrem and 400 mrem, the whole-body dose would be recorded as 400 mrem regardless of the position of the badge. In about 1992, the methodology for assigning whole-body dose from multiple badges changed. Each dosimeter was assigned an effective whole-body dose equivalent, and the resulting values were added to obtain the whole-body dose. In this case, weighting factors from ICRP 26, *Recommendations of the International on Radiation Protection*, are used (ICRP 1977). The whole-body dose is the only number that is documented in the individual's dosimetry file. All dosimeters worn were processed and results were recorded by area of the body. NIOSH/ORAUT should evaluate the results of multiple dosimeter processing by body position to determine whether partial-body exposures are an issue for specific organs and to evaluate whether the dose values provided by the multiple dosimetry are more claimant favorable. As individual files do not routinely include this information, a special request will be required.

4.14.2 Applicability to Revision 03

This comment is applicable to Revision 03 of the SRS TBD. Draft Revision 04-E (Scalsky 2006) provides no additional information to resolve this issue.

4.14.3 Summary of Overall NIOSH Response

Initially, NIOSH stated that the nature of this comment was unclear. It referred SC&A to the additional guidance in OCAS-TIB-006, *Interpretation of External Dosimetry Records at the Savannah River Site (SRS)* (Neton 2004), and OCAS-TIB-007, *Neutron Exposures at the Savannah River Site* (Neton 2003c), prepared following preparation of Section 5.0 in the SRS site profile in 2003. NIOSH examined the original SC&A review for clarification regarding logbooks.

NIOSH issued OCAS-PER-019, *The Effect of Additional Neutron Data from the Savannah River Site* (OCAS 2007). This program evaluation report indicated that SRS provided additional neutron data. NIOSH reevaluated 17 claims, including four noncompensable claims. No changes occurred in the POC. Previously unevaluated claims were completed with the new data.

4.14.4 Work Group Action

The following actions were identified in Work Group deliberations:

NIOSH Actions:

- Review neutron logbooks referenced in the August 22, 2006, meeting. The meeting minutes are available on the NIOSH Web site.

- Provide a response to the Board regarding the completeness of the HPAREH file used for the development of the external coworker model.

- Investigate the sources of information provided by SC&A.

4.14.5 Closure Status

Revisions 03 and draft Revision 04-E (Scalsky 2006) of the TBD lack an explanation of why the additional records are not being considered for dose reconstruction. SC&A provided NIOSH with an inventory of supplemental records that may be beneficial in dose reconstruction. Based on a review of OCAS-PER-019 (Allen 2007), SRS provided some additional neutron data. Whether these are the neutron logbook data mentioned above is unknown. To date, NIOSH has not discussed the exact nature of these data. NIOSH has made no effort to evaluate the completeness of the HPAREH file used in the development of the external coworker model. The integrity of the HPAREH file for use in coworker modeling is questionable given the absence of data for most workers terminating prior to 1979. Either the TBD or the external coworker dose procedure should develop and discuss a basis for its appropriateness. Furthermore, consideration should be given to multiple dosimetry results, which may provide valuable information on nonuniform exposures and may be of assistance in determining organ dose for some workers. For the above reasons, it is recommended that this overall finding remains open pending review of the NIOSH Work Group action items.

4.15 COMMENT 15: AMBIGUOUS DOSE RECONSTRUCTION DIRECTION

Well-developed technical documents are necessary to ensure that methods are effective and consistent. Applying consistent methodologies can provide continuity of assessment over time and across multiple facilities. The SRS TBD contains ambiguous instructions, inconsistencies, and unwarranted precision.

4.15.1 Issue Description

Dose reconstruction is a complex process even under the best circumstances. It is, therefore, imperative that supportive background information/data and specific instructions are presented in a logical manner that ensures understanding, process efficiency, and consistency among dose reconstructors. Many of the sections of the TBD, especially Chapter 4 related to internal dosimetry, are very difficult to understand and, together with the large array of TIBs and other OCAS/ORAUT procedures, create a virtually impenetrable, complex array of guidelines. This situation lends itself to inconsistencies in the way in which dose reconstructions are performed and makes it difficult to verify the reliability and reproducibility of the dose reconstructions. A major factor that limits the readability and, therefore, comprehensibility of Section 5.0 of the TBD is the mingling presentation of data that alternates between beta/photon and neutron dosimeters/dosimetry. Since reconstruction of beta/photon and neutron exposures requires two different methods, as well as IREP inputs, a more logical and comprehensible format would have separated these two major topics.

Quality assurance is an important part of maintaining a consistent and defensible dose reconstruction program. NIOSH/ORAUT should make the TBD transparent to the user and ensure that the various portions of the TBD are consistent with one another. Inconsistencies in the TBD and between the TBD and other procedures result in confusion and a potential misapplication of available dose reconstruction methods, and they should be corrected or explained. The issuance of complexwide TIBs designed to address specific issues such as glovebox work, tritium exposure, ambient environmental dose, and medical x-ray exposure among others has provided clarification. Procedures are used in conjunction with TIBs and site profiles to guide the dose reconstruction process. This provides more consistency in the process overall.

4.15.2 Applicability to Revision 03

These comments are applicable to Revision 03 of the TBD.

4.15.3 Summary of Overall NIOSH Response

A large number of TIBs have indeed been developed to address unique situations or very general, complexwide situations, all designed to address issues not easily covered in site profiles or to add efficiency to the POC evaluation process. Procedures are written, in part, to help ensure that the dose reconstructors consider all the various guidance documents. Site profiles, TIBs, and procedures are all essential aids to assist the dose reconstructors in performing their POC evaluations. Indeed, the lack of these documents would likely lead to less consistency and efficiency.

4.15.4 Work Group Action

There are no Work Group actions pending.

4.15.5 Closure Status

This issue is currently being addressed under a separate review task. The additions and updates to TIBs and procedures have clarified some of the ambiguity in the site profile. The use of site workbooks by dose reconstructors ensures that methods are consistent. The mingling of data causes confusion in the identification of dose reconstruction methodologies. Some improvement has been made in this area with the later revisions of the TBD. It is recommended that this issue be considered closed.

4.16 COMMENT 16: SPECIAL EXPOSURE CIRCUMSTANCES FOR SUBCONTRACTORS AND CONSTRUCTION WORKERS

The TBD does not currently include the special exposure circumstances for subcontractors and construction workers; however, NIOSH is aware of this issue. ORAUT-OTIB-0052 (Chew et al. 2006) was developed to provide dose reconstruction guidance for trade workers. SC&A is currently reviewing this procedure under Task 3.

4.16.1 Applicability to Revision 03

The trade worker section in Revision 03 of the TBD is reserved. NIOSH issued a generic TIB, ORAUT-OTIB-0052 (Chew et al. 2006), providing dose estimate parameters for construction workers, which includes consideration of SRS construction workers.

4.16.2 Summary of Overall NIOSH Response

No response was provided.

4.16.3 Work Group Action

This is no longer an issue and there is no Work Group action pending; therefore this issue is closed.

4.16.4 Summary

The issuance of ORAUT-OTIB-0052 (Chew et al. 2006) provides a mechanism for estimating construction worker doses. It has been included in the Task 3 procedure reviews and will not be discussed in this review. It is recommended that with the issuance of the construction worker procedure, this issue be considered closed and requires no further action. SC&A recommends that NIOSH finish the trade worker section in the SRS TBD for completeness.

5.0 OVERALL ADEQUACY OF THE SRS SITE PROFILE, REVISION 03, AS A BASIS FOR DOSE RECONSTRUCTION

The SC&A procedures call for both a "vertical" assessment of a site profile in terms of the adequacy and completeness of each particular element of the profile as well as a "horizontal" assessment of the extent to which the profile as a whole satisfies its intended purpose and scope. This section addresses the latter objective by evaluating (1) how, and to what extent, the site profile satisfies the five objectives defined by the Advisory Board for determining adequacy; (2) the usability of the site profile for its intended purpose (i.e., to provide a generalized technical resource for the dose reconstructor when individual dose records are unavailable), and (3) generic technical or policy issues that transcend any single site profile that need to be addressed by the Advisory Board and NIOSH.

5.1 SITE PROFILE IMPROVEMENTS

In general, Revision 03 of the TBD included minimal changes, including an analysis of the maximum plausible annual dose from the ingestion of foodstuffs at SRS. This issue was originally raised in a worker outreach meeting and integrated to respond to this comment. Draft Revision 04-E (Scalsky 2006) proposes additional improvements.

5.2 SATISFYING THE FIVE OBJECTIVES

The SC&A review procedures, as approved by the Advisory Board, require that each site profile be evaluated against five measures of adequacy—completeness of data sources, technical accuracy, adequacy of data, site profile consistency, and regulatory compliance. The SC&A finds that Revision 03 of the SRS site profile represents an adequate accounting of the primary internal issues related to plutonium, uranium, and tritium, as well as main external hazards. Revision 03 is not substantially different than Revision 02 and therefore only minimally resolved the 16 issues identified in the Revision 02 review. The primary changes made to Revision 03 included incorporation of internal review comments and inclusion of a model for the ingestion of foodstuffs. Therefore, Revision 03 of the SRS site profile falls short in fully characterizing a number of key underlying issues that are fundamental to guiding dose reconstruction. In some cases, these issues may impact other site profiles. Many of the issues involve a lack of sufficient conservatism in key assumptions or estimation approaches, or incomplete site data or analyses of these data.

SC&A is aware of NIOSH's ongoing efforts to develop and issue Revision 04-E (Scalsky 2006) of the SRS site profile and has acknowledged additional information intended for inclusion in that version. Likewise, additional guidance documents were being issued that, while not yet reflected in Revision 03 of the SRS site profile, would serve to mitigate some of the gaps and issues raised in this report. While Revision 04-E (Scalsky 2006) incorporates changes that will close a number of the comments, it remains a draft and has not been formally released. Therefore, this report continues to cite issues related to those intended improvements as "open" pending issuance of Revision 04-E (Scalsky 2006).

5.2.1 Objective 1: Completeness of Data Sources

Revision 03 of the TBD contains incomplete assessment and guidance on dose assignment pertaining to RU, Pu-242, transplutonium radionuclides, thorium, and U-233. The calculation of internal dose does not explicitly consider transuranic and fission product impurities in RU. Impurity concentrations are based on estimates made from waste stream data and appear to be at odds with the DOE 1985 Task Force review on RU (DOE 1985). A further investigation of the RU source term data should be completed to determine the upper bounds of impurity concentrations and resulting doses. Other assays, such as metallurgical analyses, may assist in determining concentrations and relative uncertainties in these values. Revision 03 does not address potential internal dose from U-233 (including impurities in uranium). Revision 04-E (Scalsky 2006) adds discussions on the assignment of dose from impurities in RU and from U-233; however, the document has not been formally released.

Revision 03 does not consider potential contributions from exposure plutonium containing higher levels of Pu-242, nor does it justify the absence of such a discussion. Proposed Revision 04-E (Scalsky 2006) discusses the production of Pu-242 during curium campaigns and provides information on the activity composition of the high Pu-242 mixture. The default assumption for plutonium remains at 10-year-old 12% plutonium.

SRS handled exotic radionuclides as a result of special production campaigns and source production ranging in quantities from fractions of a gram to kilograms during special campaigns. These radionuclides included transplutonium elements, Po-210, Co-60, Cf-252, Tm-170, Ir-192, Eu-152, and various isotopes of lanthanum (Reed et al. 2002). Many of these sources produced were encapsulated and therefore posed primarily an external hazard. Neptunium and curium were also processed for periods of time. Revision 03 of the TBD does not explicitly cover the potential dose from these radionuclides. The document gives inadequate or no consideration to potential exposures and missed dose from these radionuclides and does not discuss the implementation of monitoring techniques for these radionuclides.

Section 4.1.2 on bioassay was added in Revision 03, but the issues related to RU, Pu-242, transplutonium radionuclides, thorium and U-233 have not been adequately resolved. Several updates proposed for Revision 04-E (Scalsky 2006) of the TBD deal directly with exposure from these radionuclide sources. This will bridge some of the gaps found in previous versions of the TBD. It is recommended that issues related to exposure from RU, Pu-242, transplutonium radionuclides, U-233 and other exotic radionuclides remain open pending review of the yet-to-be-released Revision 04-E (Scalsky 2006) of the TBD and ORAUT-OTIB-0053 (ORAUT 2005).

SC&A maintains that some workers have sustained significant exposures that are relevant to their dose reconstruction but may not be captured by relying on the DOE exposure file and the CATI. Other sources besides dosimetry files exist at SRS that summarize incidents and provide supporting documentation. Databases identified may be specific to an area or operation, so a compilation of several sources may be necessary to capture site-wide incidents. The databases vary in the criteria for inclusion, giving a broader perspective on what constitutes an incident. Without a thorough reconciliation of the DOE exposure files against these separate incident data banks, NIOSH cannot be assured that all significant exposures from incidents are considered in relation to individual worker claims or in efficiency methods (e.g., high-five approach, coworker dose assignments). Dose reconstructors should be alerted to these situations. Based on record storage practices, redundant systems are necessary to develop a complete list of incidents. While the CATI provides some potential to identify incidents that may be missing from an individual's dose record, reliance on the CATI for this purpose places an inappropriate burden upon the worker or survivor to recall events and to recognize potential implications for the dose reconstruction.

Revision 03 gave no additional consideration to many potential unrecognized or unreported exposures from high-risk work activities. Proposed Revision 04-E (Scalsky 2006) includes discussions on potential dose from special campaigns involving U-233, thorium, and transplutonium radionuclides, and ORAUT-OTIB-0052 (Chew et al. 2006) addresses dose to construction workers. However, neither Revision 03 nor proposed Revision 04-E (Scalsky 2006) of the TBD includes information on decontamination and decommissioning, open burning of solvents, and other high-risk or unusual jobs. It is recommended that this issue remains open pending the completion of Work Group actions and release of proposed Revision 04-E (Scalsky 2006) of the site profile.

The completeness of the data used as a basis for the high-five approach is questionable. NIOSH has not reviewed all recorded inhalation intakes over the history of the site. The criteria for inclusion of individuals in the SRS intake file have changed over time, excluding some intakes from the file. Other sources of information such as the DPSP monthly reports, databanks and incident files, and visitor cards were not considered when identifying intakes. SC&A has presented numerous examples from these sources that likely meet the criteria for inclusion in the high-five approach and deserve further investigation. The general bioassay trend for Np-237 and Pu-239 from the individual high-five data and the DPSP monthly report data indicate that monthly report bioassay results were higher. In addition, some radionuclides were present at the site prior to the availability of bioassay techniques. It is recommended that several Work Group actions remain open, including clarification of the location of the SRS high-five bioassay data, review of the tank farm Fault Tree Data Bank, and completion of the revised high-five approach.

The issuance of ORAUT-OTIB-0052 (Chew et al. 2006) provides a mechanism for the estimation of construction worker doses. It has been included in the Task 3 procedure reviews and will not be discussed in this review. It is recommended that with the issuance of the construction worker procedure, this issue be considered closed and requires no further action. SC&A recommends that NIOSH complete the trade worker section in the SRS TBD for completeness.

5.2.2 Objective 2: Technical Accuracy

The method used to reconstruct doses to unmonitored outdoor workers due to airborne emissions employs an atmospheric dispersion model, assumptions, and a resuspension factor that do not appear to be claimant favorable and are not entirely appropriate for this class of problem. Revision 03 (Scalsky 2005) and proposed Revision 04-E (Scalsky 2006) provide no clarification on these issues. In its response to SC&A's finding, NIOSH maintains that the dispersion model has been adequately validated and is sufficiently conservative. It has not specifically addressed ground-level plumes from open-pan burning of contaminated solvents or from environmental spills and leaks in the TBD. Furthermore, it has not provided a written evaluation of the existing dispersion model focusing on episodic releases, as proposed by the Work Group.

The TBD lacks a basis for the use of a resuspension factor of 1×10^{-9} for resuspension of contaminated soil. Based on an SC&A review of the literature, it also appears that the TBD resuspension factor of 1×10^{-9} per meter may not be claimant favorable. Kennedy and Strenge (1992) reported resuspension factors from approximately 1×10^{-11} to 1×10^{-2} m^{-1}, which suggests that resuspension is a complex process of several parameters and that the specific conditions present at the time of measurement are critical. Based on recommended resuspension factors presented in the literature, an average value closer to 1×10^{-5} to 1×10^{-6} per meter would seem more appropriate for use in worker dose reconstruction, resulting in worker inhalation doses from resuspension that are 3 to 4 orders of magnitude greater than those derived in the site profile. The dust-loading approach should also be considered, using an average work-year dust loading on the order of perhaps 1 mg/m^3. It is recommended that, due to remaining issues associated with the atmospheric dispersion model, assumptions, and a resuspension factor, this overall finding remains open.

Revision 03 contains inadequate information regarding the assessment of dose from STCs. In June 2006, NIOSH proposed a methodology for the assignment of dose from STCs. Proposed SRS-specific guidance assigns dose from tritides based on surface contamination limits rather than production information and surveillance data, making the basis for assumptions weak, particularly in years when engineering controls were not as advanced as they are today. In April 2007, NIOSH released ORAUT-OTIB-0066 (LaBone 2007), which provides a bounding technique for the assignment of dose from intakes of OBTs and SMTs. NIOSH has provided a dose estimation methodology for STCs; however, it has not verified the timeframe and location where STCs were handled or the types and quantities of STCs handled at SRS. The bounding techniques, as proposed for SRS, cannot be effectively developed and applied without some basic understanding of the STCs handled, the quantities of material, the locations and time periods of potential exposure, and the physical behaviors of tritium compounds in the environment (e.g., conversion to HTO, formation of rust) to correctly characterize tritium exposure. Furthermore, NIOSH limited the application of this technique to 1975 to present. This conflicts with site expert statements that indicate that potential exposures occurred as far back as late 1950s. Given the large amount of tritium handled at SRS in various areas and the propensity of tritium to bind with organics and metals, it is reasonable to assume that STCs were present at SRS in some form prior to 1975. An evaluation of the adequacy of the dose estimation methodology cannot be completed without this key information; thus it is recommended that the issue remains open.

In-vivo bioassay monitoring results for thorium should be analyzed very carefully. Errors of two orders of magnitude can be made, depending on the material type, equilibrium assumptions, and time of measurement after intake. Urinalysis results should also require a very cautious analysis, including the influence of natural thorium in the diet.

Information in the revised TBD refers the dose reconstructor to ORAUT-OTIB-0001 (Brackett 2003), which describes the high-five approach. NIOSH indicated that it intends to update the high-five approach and base revised calculations on bioassay data rather than data in the SRS

IDR. This update was expected in December 2006 but has not been completed as of August 2007. Until the revision of this procedure is implemented and the TBD references the correct TIB, regulatory issues continue to be associated with the application of the high-five approach. Using the urinalysis as a basis for dose calculation will eliminate the use of intake data derived with ICRP 30 and allow for the exclusive use of ICRP 60 methodology or more current methods. Revision 03 (Scalsky 2005) and Revision 04-E (Scalsky 2006) still reference the 2003 version of ORAUT-OTIB-0001 (Brackett 2003) on maximizing internal dose. The updated approach along with any modifications to the TIB will be reviewed when provided by NIOSH.

Revisions 03 and 04-E reference ORAUT-OTIB-0001 (Brackett 2003), which continues to use both ICRP 30 and subsequently ICRP 66 as the method for assignment of internal dose. ORAUT-OTIB-0001 (Brackett 2003) justifies the use of intakes calculated with ICRP 30 methodologies rather than the most current ICRP methodology by comparing IRFs from ICRP 30 and ICRP 68. The ICRP 30 model does not produce intake values that are higher than those derived by the new ICRP models for a majority of the relevant radionuclides included in the hypothetical intake as maintained by NIOSH. The use of ICRP 30 methodology to calculate the intake, with a subsequent use of ICRP 68 models to calculate the dose, did not always result in the intended highest dose to an organ. Similarly, the appropriate solubility types between the two methodologies were not always paired consistently, resulting in discrepancies that were not claimant favorable.

The dose reconstructor is directed to use surrogate radionuclides for radionuclides absent from the IMBA code. In lieu of this, a more prudent approach to the absence of radionuclides is to use the dose coefficients provided in ICRP 2001 and employ a linear interpolation for the radionuclides that are not explicitly given. ORAUT-OTIB-0001 (Brackett 2003) should be updated to reflect these and other changes made to the high-five approach. The closure of this issue is pending the update of the high-five method using bioassay data and appropriate models. It is recommended that issues associated with the high-five approach remain open pending the release of the revised ORAUT-OTIB-0001 (Brackett 2003) and subsequent reflection in Revision 04-E (Scalsky 2006) of the TBD.

The issues associated with correction factors and uncertainties have not been satisfactorily resolved. It has not been demonstrated that the application of a DAF of 1.119 or 1.039 for both TLDs and film for 1952–1986, and an uncertainty of 30% without full consideration of laboratory, radiological, and environmental factors, is claimant favorable. The dosimeter calibration is based on an incident angle of zero degrees, which underestimates the actual field dose where incident angle is greater than zero. The correction factor applied to recorded dosimeter results is too low for photon energies from 30 to 250 keV, which is the default photon energy used.

Specific actions recommended by SC&A to achieve closure include the following:

- Provide detailed period/location film-specific DAFs for the period 1952–1970.
- Provide period/location TLD-specific DAFs for the period 1971–1986.
- Assess the impact of new DAFs on coworker data.

- Account for differences in incident angles between calibration and field use.
- Account for photon energies between 30 and 250 keV (the default photon energy used in calibration).
- Clarify the basis for applying a generic 30% uncertainty factor, and/or provide clear instructions for applying other appropriate uncertainty factors.

Revision 03 does not provide additional information to satisfactorily resolve issues associated with correction factors and uncertainties. The recent draft of Revision 04E (Scalsky 2006), which is not yet officially issued, contains a few changes to external dose reconstruction, but these additions do not satisfactorily address issues associated with correction factors and uncertainties; therefore, they are still applicable. None of the SRS site-specific workbooks and guides that SC&A has been able to locate provides further qualification for using the DAFs as recommended in the TBD (Scalsky 2006). The SRS documents that SC&A has been able to find do not deal specifically with uncertainties or adjustment factors. In addition, SC&A needs more bibliographic information to locate the "guide of 3/29/04" (as listed in the NIOSH response) and evaluate its applicability to this issue. It is recommended that for these reasons this issue remains open.

SC&A concurs that ORAUT-OTIB-0017 (Merwin 2005) provides suitable guidance for the assignment of shallow dose. This issue is closed.

The TBD prescribes two very different protocols for neutron dose reconstruction that correspond to pre- and post-1971 time periods. Prior to 1971, the uncertainty factors associated with the neutron-to-photon ratio are neither technically defensible nor likely to be claimant favorable. The TBD provides no compelling evidence to suggest that the TLND dosimeter offered significant improvements over NTA film. In brief, this suggests that both the TLND recorded neutron doses between 1971 and 1995 as well as the pre-1971 neutron doses (derived by neutron-to-photon ratios) suffer from a high degree of uncertainty and must be viewed with caution.

SC&A's evaluation of Revision 03 (Scalsky 2005) did not reveal any changes concerning neutron dose reconstruction methodologies. SC&A believes that the use of the geometric mean and geometric standard deviation that describe the post-1971 neutron-to-photon ratio is neither technically defensible nor likely to be claimant favorable for a large fraction of potential claimants. Proposed Revision 04-E of the TBD (Scalsky 2006) provides some changes and clarifications to the applications of neutron-to-photon values for dose reconstruction. For likely noncompensible cases, the site profile recommends applying the 95th percentile neutron-to-photon dose ratio to the recorded dose. SC&A recommends that the 95th percentile neutron-to-photon values be used in all SRS dose reconstruction cases, not in just the likely noncompensable cases. There is currently no Work Group action pending, but it is recommended that this issue remain open pending the release of proposed Revision 04-E (Scalsky 2006).

5.2.3 Objective 3: Adequacy of Data

The adequacy of the F- and H-Area Tank Farm characterization in the TBD is questionable for use as dose reconstruction guidance. Data evaluation appears to be incomplete with regard to exposure conditions, radionuclides of concern, and uncertainty. This is particularly true for early periods of operation, where primary records involving key operations and incidents are lacking. The tank farm database, not currently evaluated by the TBD, can serve to determine what assumptions would be suitable in giving claimants who worked in the tank farms the benefit of the doubt in the face of considerable uncertainties. The lack of evaluation of primary data sources has left the TBD without a realistic way to estimate uncertainties. The potential for internal and external exposure to unmonitored workers in areas not designated as radiological control areas needs to be investigated. Revision 03 of the site profile made no changes in the discussion on the tank farms. Proposed Revision 04-E (Scalsky 2006) made only minimal changes, such as the inclusion of Cs-137 and Ru-106 in Table A-14. Default intakes for tank farms workers for different periods of time, including actinides, were added. This revision does not resolve all issues associated with the tank farms. It is recommended that this issue remain open pending completion of Work Group action items and release of the proposed Revision 04-E (Scalsky 2006).

An evaluation of the comprehensiveness of the early monitoring program should be completed for early workers to determine whether existing site profile methodologies bound their dose. This is especially important in the case of workers who were not monitored but were exposed to a radiological hazard. Without a single organization determining neutron dosimeter, bioassay requirements, and when special interpretations of film badges are required, there may have been inconsistencies in actual practices. The adequacy of the early monitoring program (i.e., who they monitored) will not be resolved by an inventory of records provided in claimant files. Furthermore, additional validation of the HPAREH database as the exclusive source for external coworker dose determination, given the incompleteness of early data, is necessary to demonstrate that these data are adequate for this use. It is recommended that this issue remain open pending NIOSH's completion of Work Group action items.

Revisions 03 and 04-E lack an explanation of why the additional records are not being considered for dose reconstruction. SC&A provided NIOSH with an inventory of supplemental records that may be beneficial in dose reconstruction. Based on a review of OCAS-PER-019 (OCAS 2007), NIOSH/ORAUT has received some additional neutron data from SRS. The exact nature of this data has not been discussed with SC&A to date. To SC&A's knowledge, no effort has been made by NIOSH/ORAUT to evaluate the completeness of the HPAREH file used in the development of the external coworker. The integrity of the HPAREH file for use in coworker modeling is questionable, since most of the workers terminating prior to 1979 are not included in the HPAREH. A basis for its appropriateness should be developed and discussed in either the TBD or the ORAUT-OTIB-0032 (Merwin 2006). Furthermore, consideration should be given to multiple dosimetry results, which may provide valuable information on nonuniform exposures and may be of assistance in determining organ dose for some workers. For the above reasons, it is recommended that this overall finding remain open pending review of the NIOSH Work Group action items.

5.2.4 Objective 4: Consistency among Site Profiles

The key assumptions in Revision 03 for medical, environmental, internal, and external dose remained the same as those from Revision 02. Section 3.0 of Revision 03 did include an annual dose from the ingestion of foodstuffs. Doses are calculated for the ingestion of 60 kilograms (wet mass) of food per year for Sr-90, Cs-137, Pu-238, Pu-239, and tritium. NIOSH/ORAUT has stated that ingestion is not usually considered at major DOE sites; however, it was considered for SRS and for worst-case internal dose assumptions at Hanford. Ingestion dose should not be applied selectively at one facility and ignored at another.

5.2.5 Objective 5: Regulatory Compliance

The TBD has effectively complied with the hierarchy of data required under 42 CFR Part 82 and its implementation guides with one notable exception. SC&A notes that SRS used ICRP 30 to determine the relative intakes used in the high-five hypothetical intake. This appears to conflict with 42 CFR 82.18(b), which states "NIOSH will calculate the dose to the organ or issue using the appropriate current metabolic models published by the ICRP."

5.3 USABILITY OF THE SITE PROFILE FOR ITS INTENDED PURPOSES

SC&A identified a number of issues in the SRS and other site profiles reviewed to date that, in some cases, represent potential generic policy issues that transcend any individual site profile. These issues may involve the interpretation of existing standards (e.g., oro-nasal breathing), how certain critical worker populations (e.g., construction workers and early workers) should be profiled for historic radiation exposure, and how exposure itself should be analyzed (e.g., the treatment of incidents and statistical treatment of dose distributions). SC&A previously defined these issues in its evaluation of Revision 02 of the SRS site profile. NIOSH has issued TIBs to address generic issues such as shallow dose assignment, dose to construction workers, exposure from highly insoluble plutonium, and exposure from STCs, all of which apply to dose reconstruction at SRS. Other common issues still remain, such as the dose from impurities in RU, dose to decontamination and decommissioning workers, and quality assurance of records provided for claimants by SRS.

5.3.1 Ambiguous Dose Reconstruction Direction

The ambiguity of the TBD is currently being addressed under a separate review task. The additions and updates to TIBs and procedures have clarified some of the ambiguity in the site profile. The use of site workbooks by dose reconstructors ensures that methods are consistent. The mingling of data causes confusion in the identification of dose reconstruction methodologies. The later revisions of the TBD have made some improvement in this area; therefore, it is recommended that this issue be considered closed.

5.3.2 Inconsistencies and Editorial Errors in the Site Profiles

NIOSH uses Equations 3-2 and 3-3 on pages 52–53 of the TBD to derive the atmospheric dispersion factors (i.e., X/Q values expressed in units of seconds per cubic meter) for ground-level and elevated releases, respectively. These equations appear to be in error because they result in large X/Q values. For example, using Equation 3-2, the ground-level X/Q at 1,000 meters downwind from the release point is derived as follows:

$$Y = 1.0146X - 1.8809$$

where Y = Atmospheric dispersion factor (s/m^3) and X = Distance from the source (meters).

Hence, at 1,000 meters, the X/Q value is as follows:

$$Y = 1.0146(1000) - 1.8809 = 1013$$

Since X/Q values are typically a small fraction of 1 (e.g., on the order of 0.001), it appears that the equation contains a typographical error. Perhaps the equation should be inverted, giving a value of 1/1013 or about 0.001. This is also the case for Equation 3-3.

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ATTACHMENT 1: NIOSH TECHNICAL DOCUMENTS CONSIDERED DURING THE REVIEW PROCESS

Technical Basis Documents:

- ORAUT-TKBS-0003, Savannah Rive Site, Revision 03, April 5, 2005 (Scalsky 2005).

Technical Support Documents:

- OCAS-PER-001, *Misinterpreted Dosimetry Records Resulting in an Underestimate of Missed Dose in SRS Dose Reconstruction*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, September 8, 2003. (Neton 2003a)

- OCAS-PER-002, *Error in Surrogate Organ Assignment Resulting in an Underestimate of X-ray Dose in SRS Dose Reconstructions*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, December 15, 2003. (Neton 2003b)

- OCAS-PER-0019, *The Effect of Additional Neutron Dose Data from the Savannah River Site*, Revision 0, Office of Compensation Analysis and Support, Cincinnati, Ohio, May 18, 2007. (Allen 2007)

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- ORAUT-OTIB-0032, *External Coworker Dosimetry Data for the Savannah River Site*, Revision 0 PC-1, Oak Ridge Associated Universities, Oak Ridge, Tennessee, November 7, 2006. (Merwin 2006)

- ORAUT-OTIB-0006, Revision 02 (2003), *Technical Information Bulletin: Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures*, Oak Ridge Associated Universities, Oak Ridge, Tennessee. December 29, 2003. (Kathren et al. 2003)

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- ORAUT-OTIB-0066 (2007), Revision 00, *Calculation of Dose from Intakes of Special Tritium Compounds*, Oak Ridge Associated Universities, Oak Ridge, Tennessee, April 26, 2007. (LaBone 2007)

ATTACHMENT 2: WSMS DATABANK STATISTICS

SRS staff obtained a copy of the WSMS incident database. Because this database was lengthy and contained classified information, a review team was assembled to review the database at SRS from February 28 through March 1, 2007. The three objectives were to determine the contents of the database, compare entries in this database to those from the tank farm Fault Tree Databank, and determine its usefulness in dose reconstruction. The WSMS database contains 464,092 incidents, including many that do not relate to radiation exposure. SRS has expressed concern about having to review the entire database for classified material and has requested that the Work Group identify the particular incidents of interest. To accommodate uncleared members of SC&A, NIOSH, ORAUT, and the Work Group, incident counts by year, area code, and facility code have been provided. Counts for key equipment codes and operations codes associated with sample incidents retrieved from the database have also been provided. However, without the user manual, SC&A was unable to define many of the codes listed below.

Table A2-1. WSMS Incident Statistics by Year

Year	# of Incidents	Year	# of Incidents
1953	1	1980	18708
1954	107	1981	18600
1955	301	1982	18689
1956	415	1983	16879
1957	426	1984	18571
1958	390	1985	19745
1959	653	1986	25366
1960	597	1987	23944
1961	624	1988	20402
1962	752	1989	24538
1963	804	1990	27456
1964	1308	1991	36531
1965	1368	1992	31525
1966	1918	1993	31124
1967	2102	1994	32240
1968	1633	1995	28057
1969	1203	1996	5292
1970	1454	1997	38
1971	1971	1998	47
1972	2783	1999	80
1973	2578	2000	28
1974	3018	2001	10
1975	3571	2002	5
1976	5163	2003	19
1977	8831	2004	35
1978	9856	2005	5
1979	12324	2006	1

Table A2-2. WSMS Incident Statistics by Area Code

Area ID	Count	Area Code
19	6	V (ON SITE OTHER THAN IDENTIFIED AREAS)
15	22	R
20	22	W
17	33	T
10	52	L
13	64	P
21	76	Z
3	90	C
7	94	G
2	105	B
12	168	N
9	185	K
4	347	D
18	384	V
5	613	E
14	800	Q
16	1867	S
1	22131	A
11	27816	M
8	157644	H
6	251710	F

Table A2-3. WSMS Incident Statistics by Facility Code

Facility ID	Facility Code	Count
1	SA	1
2	A	3991
3	B	381
4	C	856
5	D	207
6	E	725
7	F	109
8	H	197
9	I	667
10	K	47
11	LA	45
12	LB	1335
13	LC	19
14	LD	330
15	LE	18855
16	LF	5
17	LG	292
18	LH	1
19	LI	36
20	LJ	1
21	LK	125
22	LL	1213
23	LM	241
24	LS	73
25	LT	4
26	LU	24
27	LV	21913
28	LW	46
29	LY	3
30	M	3
31	MC	49
32	MF	27447
33	O	1
34	Q	303
35	S	233
36	S0	25
37	SA	170555
38	SB	120668
39	SC	36473
40	SD	15970
41	SE	5503

Facility ID	Facility Code	Count
42	SG	2
43	SH	93
44	SI	7443
45	SJ	495
46	SK	119
47	SL	19454
48	SM	3365
49	SN	88
50	SO	183
51	SP	1048
52	SQ	28
53	SS	19
54	ST	13
55	SW	784
56	SX	486
57	SZ	355
58	T	151
59	W	43
60	WA	285
61	WB	27
62	WC	69
63	WD	3
64	WE	39
65	WG	117
66	WH	2256
67	WJ	302
68	WK	6
69	WL	13
70	WM	180
71	WO	17
72	WQ	2257
73	WR	631
74	WS	1589
75	WT	21964
76	WU	335
77	WV	18
78	WW	781
79	WX	13
80	WY	2
81	WZ	79
82	X	1

Table A2-4. WSMS Incident Statistics by Key Equipment Codes Related to Incidents Involving Radiation

Equipment ID	Equipment Code	Count
8	313-M (FABRICATION)	13352
62	ACCEPTANCE CRITERIA	4
63	ACCOUNTABILITY	667
70	ACTIVITY	2386
74	AIR EMMISSIONS	35
76	AIR REVERSAL	740
80	AIRBORNE ACTIVITY (772-F)	597
83	ALARM/HORN	33979
90	AMERICIUM	29
91	AMERICIUM, CURIUM, CALIFORNIUM	62
94	ANALYSIS	165
95	ANALYTICAL	1893
96	ANALYTICAL CELL	867
97	ANALYTICAL, 320-M	130
108	ASSAY/MONITOR ENRICHED URANIUM	7
163	BETA-GAMMA INCINERATOR	118
182	BIOASSAY AND/OR CHEST COUNT	263
183	BIOASSAY AND/OR CHEST COUNT (772-F)	33
190	BLOWER, FANS	11380
192	BODY EXPOSURE>2RAD/HR OR>1R/HR	186
205	BREATHING AIR	3237
212	BUILDING 232-H	67
213	BUILDING 234-H	43
214	BUILDING 235-H	3
215	BUILDING 238-H	2
216	BUILDING 249-H	2
219	BULDING 233-H	29
222	BURIAL GROUND	1600
225	CABINET	147
226	CABINET, ENTRY/EXIT OF MATERIAL	4386
230	CALIBRATION	3084
231	CALIBRATION SOURCE	495
232	CALIFIRNIUM FACILITY	53
297	CLEANING	541
302	CLOTHING CONTAMINATION (772-F)	738
337	CONDUCT OF OPERATIONS	344
345	CONTAMINATION	2123
346	CONTAMINATION PERSONAL CLOTHING/EFFECTS	395
347	CONTAMINATION, AIRBORNE	19160
348	CONTAMINATION, CLOTHING	4977
349	CONTAMINATION, FACILITY OR EQUIPMENT	29421
350	CONTAMINATION, NASAL	706
351	CONTAMINATION, SKIN	2475

Table A2-4. WSMS Incident Statistics by Key Equipment Codes Related to Incidents Involving Radiation

Equipment ID	Equipment Code	Count
355	CONTROL ROOM/CONTROL BOARD/PANEL BOARD	2190
368	CORE MACHINING	99
374	CORE, URANIUM	35
380	CRITICALITY POTENTIAL	2099
386	CURIE (0 TO 1E2)	75
387	CURIE (1E2 TO 1E3)	871
388	CURIE (1E3 TO 1E4)	79
389	CURIE (1E4 TO 1E5)	77
390	CURIE (1E5 TO 1E6)	75
391	CURIUM 242	8
392	CURIUM 244	10
398	DAMPER	925
407	DECONTAMINATION	1
418	DEMOLITION/DECOMMISION	8
419	DENITRATOR	54
466	DRUMS, CANS	979
469	DUCT	1220
476	DUST BAG	23
477	DUST COLLECTION	244
480	EFFLUENT TREATMENT FACILITY	377
496	ENRICHED URANIUM METAL RECEIVING, 321-M	68
497	ENRICHED URANIUM STORAGE	162
498	ENVIRONMENTAL PROTECTION AGENCY	1269
499	ENVIRONMENTAL RELEASE	2476
500	ENVIRONMETAL PROGRAMS, MISC. 300-AREA	1250
504	EQUIPMENT SEALS	416
506	ERRORS	1797
507	ERRORS, SUPERVISOR	882
509	ESTIMATED TOTAL EXPOSURE (772-F)	158
510	EU CONCENTRATE TANK	36
511	EU LOADOUT FACILITY	231
513	EVACUATION	482
517	EVAPORATOR (GENERAL)	176
535	FACILITY CONTAMINATION (772-F)	1762
548	FILTERS (GENERAL), SCREENS, STRAINER	9652
560	FIRE	2116
561	FIRE DETECTION AND SUPPRESSION	18059
562	FIRE EXTINGUISHER	13
563	FIRE WATCH	114
564	FIRE WATER SYSTEM	158
567	FIXED CONTAMINATION	184
582	FLUOROSCOPE	119
583	FLUOROSCOPE INSPECTION, 321-M	119
603	FULL BODY AND RESPIRATORY PROTECTION (772-F)	1291

Table A2-4. WSMS Incident Statistics by Key Equipment Codes Related to Incidents Involving Radiation

Equipment ID	Equipment Code	Count
604	FUMES	605
614	GANG VALVE	6713
614	GANG VALVE	6713
631	GLOVES	11839
632	GLOVES (772-F)	355
638	GROUND WATER	281
639	GROUNDWATER CLEANUP	684
654	HAZARDOUS WASTE	865
656	HEAT EXCHANGER/COOLER	819
662	HEPA FILTER	1163
675	HOOD OR RADIOBENCH (772-F)	552
676	HOOD/GLOVE BOX	541
681	HOSE	1693
691	HUT	1403
702	IDENTIFICATION, URANIUM SLUGS	34
709	IMPROPER STORAGE	799
710	IMPURITIES	782
714	INADEQUATE ADMINISTRATIVE CONTROL	392
715	INADEQUATE COMMUNICATION	1818
718	INADEQUATE MONITORING	37
719	INADEQUATE PROTECTIVE CLOTHING	805
726	INDUSTRIAL HYGIENE	102
730	INHALATION	6767
731	INJURY	36
734	INJURY, MEDICAL TREATMENT CASE	33
743	INSTALL, REPLACE (AND/OR REMOVE)	7841
744	INSTRUMENT AIR	3468
745	INSTRUMENT MALFUNCTION	68255
752	INTERLOCK	1316
760	INVENTORY	186
761	IODINE	155
762	IODINE REACTOR	1317
763	ION CHAMBER	8
764	ISOTOPE SEPARTION (GAS)	37
772	KANNE MONITOR	483
775	LABORATORY	631
786	LEAKS	37354
795	LIMIT EXCEEDED	1221
800	LIQUID EFFLUENT TREATMENT FACILITY (LETF)	674
801	LIQUID LEVEL	2485
819	LOSS	854
823	LOW LEVEL WASTE	190
844	MAINTENANCE ROOM/AREA	69
850	MASS SPECTROMETER	83

Table A2-4. WSMS Incident Statistics by Key Equipment Codes Related to Incidents Involving Radiation

Equipment ID	Equipment Code	Count
853	MATERIAL LOST	49
868	METER	186
872	MISC 300 AREA FACILITIES	5350
874	MISLABELED	974
877	MIXED WASTE	46
882	MONITORING	3591
890	NEPTUNIUM	261
891	NEPTUNIUM TUBE FABRICATION, 321-M	16
892	NEUTRON MONITOR	1578
893	NEUTRON TEST GAGE, 321-M	262
919	OPERATING ERROR	3255
920	OPERATIONAL SAFETY REQUIREMENT/TECH SAFETY REQUIREMENT	436
954	PERSONAL EXPOSURE	1656
955	PERSONNEL	4513
958	PERSONNEL EXPOSURE>20MR (772-F)	500
960	PERSONNEL RADIATION MONITOR	29
969	PIPE/LINES	6860
976	PLUGGAGE	42235
977	PLUTONIUM	1268
978	PLUTONIUM 238	13
979	PLUTONIUM STORAGE FACILITY	2284
997	PRECIPITATOR	6
992	PRE-EXTRUSION (U-AL LOGS), 321-M	148
1012	PROCEDURAL DIFFICULTY	3118
1013	PROCEDURAL VIOLATION	2986
1014	PROCEDURE, INADEQUATE	868
1016	PROCESS CONTAMINATION MONITOR (CAM, STORM WATER)	2315
1020	PROCESS MONITORS	80
1021	PROCESS RADIAITON MONITOR (VAMPS, ETC.)	627
1026	PROTECTIVE CLOTHING	2316
1029	PRSONNEL CONTAMINATION MONITOR	627
1036	PUNCTURE	321
1041	QUALITY CONTROL	427
1052	RADIATION	5
1053	RADIO BENCH	45
1054	RADIOACTIVE	1155
1055	RAIL/TRACK	305
1060	RDZ, RCA, RBA	137
1069	RECEIVING PU 238	9348
1093	RELEASE GUIDE	3
1105	RESPIRATORY PROTECTION	415
1116	RISER	16
1133	RUPTURED	1374
1134	RUTHENIUM	68

Table A2-4. WSMS Incident Statistics by Key Equipment Codes Related to Incidents Involving Radiation

Equipment ID	Equipment Code	Count
1139	SAFETY EQUIPMENT	270
1150	SAMPLING/SAMPLER	13369
1180	SEWER	795
1186	SHIELDING	76
1201	SKIN CONTAMINATION (772-F)	224
1209	SLUGS, URANIUM	37
1213	SMEARABLE CONTAMINATION	542
1225	SOLID BURNING	9
1228	SOLVENT BURNING	19
1243	SPILL	2850
1251	STACK	1936
1252	STACK MONITOR	365
1266	STEP OFF PAD	16
1280	STRUCTURAL COMPONENT	2173
1285	SUMP	18872
1288	SURVEILLANCE	750
1364	TARGET FABRICATION FACILITY	23
1368	TECHNICAL DIVISION (SRTC)	1849
1380	THERMAL PROCESSING PU 238	12
1385	THORIUM	137
1400	TRAINING	660
1417	TRANSPLUTONIUM	138
1426	TRITIUM OXIDE	2
1427	TRITIUM, D20	251
1428	TROUBLE ALARM/TROUBLE LIGHT	656
1454	UNCONTROLLED REACTION	483
1463	UPTAKE/ABSORPTION	15
1464	UPTAKE/INGESTION	13
1465	UPTAKE/INHALATION	397
1466	UPTAKE/INJECTION	74
1467	URANIUM	766
1476	VALVE FAILURE	26181
1479	VAULT	1555
1483	VENTILATION	4578
1484	VENTILATION (772-F)	2408
1486	VENTILATION, TANK	802
1489	VESSEL/TANK/CONTAINER	9386
1505	WASTE	2459
1508	WASTE DISPOSAL	2152
1509	WASTE DISPOSAL (772-F)	680
1528	WELL WATER	475
1529	WELLS	630
1535	X-RAY	74

Table A2-5. WSMS Incident Statistics by Key Operations Codes Related to Incidents Involving Radiation

Operation ID	Operation Description	Count
4	313-M SLUG FABRICATION OPERATIONS – 300 FACILITY	12209
5	320-M ROD ASSEMBLY OPERATIONS – 300 FACILITY	1702
6	321-M TUBULAR ASSEMBLY OPERATIONS – 300 FACILITY	4942
7	322-M SLUG & TUBULAR ASSEMBLY OPERATIONS – 300 FACILITY	7807
10	ACID RECOVERY	3171
11	ACTINIDE TARGET FABRICATION	791
12	ADJUSTMENT/ION EXCHANGE (NP-237)	2367
13	ADJUSTMENT/PRECIPITATION/FILTRATION (PU-38)	2898
15	ANION EXCHANGE COLUMN	8869
17	BASIN WATER CLEANUP/WATER OPERATIONS	476
18	BETA-GAMMA INCINERATOR	7
23	CABINETS	856
24	CASK OPERATIONS	612
28	CHEMICAL PREPARATION AND STORAGE	635
29	CHEMICAL STORAGE	3354
30	CHEMICAL TRANSFER FACILITY (CTF)	645
35	COMPRESSED GASES	7998
37	CONSTRUCTION	1513
39	CRANE AND HOIST OPERATIONS	18681
40	CRITICALITY	808
44	DENITRATOR, A-LINE	4861
51	ELECTRICAL	13830
53	ENVIRONMENTAL	4218
55	EVAPORATOR	3178
57	EXTENDED SLUDGE PROCESSING (ESP)	183
58	F.P. REMOVAL FROM EVAPORATOR CONDENSATE	18
59	FILTRATION	7722
61	FINISHING/PACKAGING/PU-238 OXIDE	418
64	FIRES	17567
67	FUEL AND TARGET OPERATIONS	529
69	FUEL STORAGE	230
71	GANG VALVE CORRIDOR	10306
73	GAS-TRITIUM DATABANK	2
79	HEALTH PROTECTION	4893
83	INSPECTIONS/TESTS	867
85	LABORATORY (TRITIUM FACILITY)	25
86	LAUNDRY	324
87	MECHANICAL OPERATIONS	573
88	MECHANICAL PROCESSES (IN CELL)	75
89	MISCELLANEOUS	34045
90	MPPF FINISHING AND HANDLING	27
91	MPPF PROCESS CONTROL	55
92	MPPF SEPARATION PROCESSES	92

Table A2-5. WSMS Incident Statistics by Key Operations Codes Related to Incidents Involving Radiation

Operation ID	Operation Description	Count
93	MPPF SOLUTION PREPARATION	51
95	NEW PRODUCTION REACTOR (NPR)	15
96	NEW SPECIAL RECOVERY	8351
104	OVERFLOWS	1048
110	PERSONNEL SAFETY	1314
114	PRECIPITATION/FILTRATION (NP-237)	3406
116	PROCESS CONTROL	1929
118	PROCESS OPER (TRITIUM)	81
119	PROCESS SAFETY	978
120	PRODUCT STORAGE AND ACCOUNTABILITY	737
123	QUALITY ASSURANCE	13
132	SAMPLE AISLE	11235
133	SAMPLING	1276
135	SECOND PRODUCT CYCLE	5810
136	SECOND URANIUM CYCLE	3186
142	SHIPPING/RECEIVING/STORAGE (NP-237)	333
144	SLAG AND CRUCIBLE	155
145	SOLIDS HANDLING, A-LINE	1654
147	SOLVENT WASHING	4378
148	SPECIAL RECOVERY	8577
150	STORAGE/BASIN OPERATIONS	659
153	TANK	5176
154	TANKS	353
155	THERMAL PROCESSES (NP-237)	221
169	UNCONTROLLED REACTIONS	311
176	VENTILATION	22447
180	WASTE – HAZARDOUS	831
181	WASTE – INTERMEDIATE LEVEL	110
182	WASTE – LOW LEVEL	188
183	WASTE – MIXED	64
185	WASTE – TRU	97
187	WASTE HANDLING	10002
188	WATER HANDLING FACILITIES	1083
189	WATER SYSTEMS	15884
190	WET CHEMISTRY (SCRAP RECOVERY, HBLINE)	3351

ATTACHMENT 3: RECOMMENDED CRITERIA FOR RETRIEVAL OF INCIDENTS FROM WSMS

As a result of the February 28–March 1, 2007, review of the data bank, the Work Group asked SC&A to provide the results of several queries run during the database review to obtain some general statistics on the database. Attachment 2 of this report summarizes the results. SC&A was asked to recommend criteria to the working group that would isolate the incidents of interest in the WSMS database. These criteria will be reviewed in a future Work Group meeting and modified by NIOSH, ORAUT, and the Work Group as necessary. Incidents are categorized by year, area, facility, source of information, equipment, and operations. It is recommended that the incidents retrieved first be limited by year and include incidents occurring prior to 1990. By 1990, incidents should have been filed regularly in the individual dose record. It is recommended that no restriction be placed on the area identification (ID) and the facility ID. Source ID, equipment ID, and operations ID, which are numerical values, can be used to focus the search to incidents involving radioactive material. The numerical values for each ID of interest are outlined below. Since the equipment ID and operations ID descriptions for an incident often contain text, it is suggested that these fields be searched for keywords. Suggested keywords are listed below. While visiting SRS, the review team printed more than 100 incidents from the database and cleared for release. SC&A chose the specific source ID, equipment ID, operations ID, and keywords based on the occurrence of the values and keywords in the 100 example incidents. In addition, if the ID did not occur in the incident record and it was clearly of interest (e.g., uptake/ingestion), the ID has been included below. These criteria constitute a preliminary recommendation for consideration by the Work Group in further meetings.

The following search terms are recommended to obtain records representative of the types of incidents of interest from the WSMS database.

Incidents occurring in 1990 or before

And,

Incidents where the equipment field contains the words uptake, intake, inhalation, ingestion, health protection, environment, contamination, uranium, neptunium, bioassay, chest count, americium, californium, curie, curium, fume, fire, limit exceeded, neutron, exposure, plutonium, thorium, tritium, stack, alarm, or x-ray.

Or,

Incidents where the operations field contains the words miscellaneous, health protection, special recovery, fires, or ventilation.

Or,

Incidents where Source ID equals 2, 12, 46, 47, 55, 61, 62, 63, 64, 69, 74, 76, 101, 112, 114, 121, 122, 123, 124, 126, 127, 130, 131, 145, 150, 156, 157, 160, 174, 176, 179, 182, 190, 191, 214, 225, 233, 248, 262, 263, 271, 277, 296, 309, 310, 321, 322, 323, 331, 341, 376, 380, 381, or 386.

Or,

Incidents where equipment ID equals 8, 62, 63, 70, 74, 76, 80, 83, 90, 91, 94–97, 108, 163, 182, 183, 190, 192, 205, 212, 213, 214, 215, 216, 219, 222, 225, 226, 230, 231, 232, 297, 302, 337, 345–351, 355, 368, 374, 380, 386–392, 398, 407, 418, 419, 466, 469, 476, 477, 480, 496–500, 504, 506, 507, 509, 510, 511, 513, 517, 535, 548, 560–564, 567, 582, 583, 603, 604, 614, 631, 632, 638, 639, 654, 656, 662, 675, 681, 691, 702, 709, 710, 714, 715, 718, 719, 726, 730, 731, 734, 743, 744, 745, 752, 760, 761, 762, 763, 764, 772, 775, 786, 795, 800, 801, 819, 823, 844, 850, 853, 868, 872, 874, 877, 882, 890, 891, 892, 893, 919, 920, 954, 955, 958, 960, 969, 976, 977, 978, 979, 992, 997, 1012, 1013, 1014, 1016, 1020, 1021, 1026, 1029, 1036, 1041, 1052, 1053, 1054, 1055, 1060, 1069, 1093, 1105, 1116, 1133, 1134, 1139, 1150, 1180, 1186, 1201, 1209, 1213, 1225, 1228, 1243, 1251, 1252, 1228, 1243, 1251, 1252, 1266, 1280, 1285, 1288, 1364, 1368, 1380, 1385, 1400, 1417, 1426, 1427, 1428, 1454, 1463, 1464, 1463, 1464, 1465, 1466, 1467, 1476, 1479, 1483, 1484, 1486, 1489, 1505, 1508, 1509, 1528, 1529, or 1535.

Or,

Incidents where operations ID equals 4, 5, 6, 7, 10, 11, 12, 13, 15, 17, 18, 23, 24, 28, 29, 30, 35, 37, 39, 40, 44, 51, 53, 55, 57, 58, 59, 61, 64, 67, 69, 71, 73, 79, 83, 85, 86, 87, 88, 89, 90, 91, 92, 93, 95, 96, 104, 110, 114, 116, 118, 119, 120, 123, 132, 133, 135, 136, 142, 144, 145, 147, 148, 150, 153, 154, 155, 169, 176, 180, 181, 182, 183, 184, 185, 187, 188, 189, or 190.