
Draft

**ADVISORY BOARD ON
RADIATION AND WORKER HEALTH**
National Institute of Occupational Safety and Health

**Review of NIOSH Site Profile
for
Savannah River Site**

**Contract No. 200-2004-03805
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S. Cohen & Associates: <i>Technical Support for the Advisory Board on Radiation & Worker Health Review of NIOSH Dose Reconstruction Program</i>	Document No. SCA-TR-TASK1-0003
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- Attachment 1: Outstanding Records Requested of NIOSH and the Savannah River Site to Support SRS Site Profile Review
- Attachment 2: Key Questions for NIOSH/ORAU Regarding Site Profile Documents
- Attachment 3: Conference Call with NIOSH and SC&A
- Attachment 4: SRS Facility Site Expert Interview Summary — Former and Current Radiation Protection Staff
- Attachment 5: SRS Facility Site Expert Interview Summary — Production and Construction Worker Staff
- Attachment 6: Consistency Between Savannah River Site and Hanford Site Profiles
- Attachment 7: Evaluation of Intakes Derived Using ICRP 30 Versus ICRP 68 Methodologies

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ACRONYMS AND ABBREVIATIONS

ACL	Administrative Control Limit
ALARA	As Low As Reasonably Achievable
Board	Advisory Board on Radiation and Worker Health
BSRI	Bechtel Savannah River, Inc.
CAM	Continuous Air Monitor
CDC	Center for Disease Control
CEDE	Committed Effective Dose Equivalent
CWPR	Center to Protect Worker Rights
DHEC	Department of Health and Environmental Control
DOE	Department of Energy
DOELAP	Department of Energy Laboratory Accreditation Program
DPSOL	DuPont Savannah River Plant Operating Log
DPSOP	DuPont Savannah River Plant Operating Procedure
DTPA	Diethylenetriaminepentaacetate
DU	Depleted Uranium
DuPont	E.I. DuPont De Nemours and Company
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EU	Enriched Uranium
GSO	General Service Operator
HEDR	Hanford Environmental Dose Reconstruction
HEU	Highly Enriched Uranium
HP	Health Physics
HPAREH	Health Protection Annual Radiation Exposure History Database
HPRED	Health Protection Radiation Exposure Database
HVAM	High Volume Air Monitor
IARC	International Agency for Research on Cancer
ICRP	International Commission of Radiation Protection
IREP	Interactive RadioEpidemiologic Program
IRF	Intake Retention Fraction
LVAM	Low Volume Air Monitor

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MDC	Minimum Detectable Concentration
MDL	Minimum Detectable Level
MPBB	Maximum Permissible Body Burden
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
ORNL	Oak Ridge National Laboratory
PA	Posterior-Anterior
PNNL	Pacific Northwest National Laboratory
POC	Probability of Causation
PPE	Personnel Protective Equipment
QAPP	Quality Assurance Program Plan
RAC	Risk Assessment Corporation
RAS	Retrospective Air Sampler
REDI	Radiation Exposure Data Investigation
RF	Resuspension Factor
RU	Recycled Uranium
RWP	Radiation Work Permit
SC&A	S. Cohen and Associates
SMT	Special Metal Tritides
SRL	Savannah River Laboratory
SRNL	Savannah River National Laboratory
SROO	Savannah River Operations Office
SRS	Savannah River Site
STC	Special Tritium Compounds
SWP	Special Work Permit
TBD	Technical Basis Document
TIB	Technical Information Bulletin
TLD	Thermoluminescent Dosimeter

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TLND Thermoluminescent Neutron Dosimeter

WSRC Westinghouse Savannah River Company

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1.0 EXECUTIVE SUMMARY

This report presents the S. Cohen and Associates (SC&A, Inc.) evaluation of the site profile, *Technical Basis Document for the Savannah River Site To Be Used for EEOICPA Dose Reconstructions*, Revision 2 (ORAUT-TKBS-0003, Scalsky 2004), commonly called the Savannah River Site (SRS) Site Profile or the SRS Technical Basis Document (TBD). In this context, SC&A has also evaluated four other documents that relate to the SRS Site Profile:

- *Technical Information Bulletin: Maximum Internal Dose Estimates for Savannah River Site (SRS) Claims* (ORAUT-OTIB-0001, Brackett 2003)
- *Technical Information Bulletin: Savannah River Site Tritium Dose Assignment* (ORAUT-OTIB-0003, Duncan 2003)
- *Interpretation of External Dosimetry Records at the Savannah River Site* (OCAS-TIB-006, Neton 2004)
- *Neutron Exposures at the Savannah River Site* (OCAS-TIB-007, Neton 2003)

These documents are used by NIOSH, along with individual dose data provided by the Department of Energy (DOE) and information gathered in interviews with claimants, to reconstruct doses for SRS employees (including contractor and sub-contractor employees). This review is designed to fulfill the objectives set by the Advisory Board on Radiation and Worker Health (Advisory Board) for assessing the accuracy and adequacy of the SRS Site Profile to serve as one of the main documents that informs dose reconstruction for claimants. For instance, it provides the data on the limits of detection of radiation monitoring methods, as well as descriptions of facilities and processes that resulted in the worker exposures. The site profile also provides direction for assigning internal and external dose to monitored and unmonitored workers.

Savannah River Site was a complex operation involved in numerous missions, each of which had its own unique exposure hazards. These facilities included the following:

- Five heavy water reactors for plutonium and tritium production, where radiological hazards included external photon and beta exposure from fission products, internal exposure from tritium, and neutron exposure in some areas
- Two chemical separation plants and associated facilities, where radiological hazards included potential for internal and external exposure to a variety of radionuclides
- Uranium processing and fuel fabrication facilities, where natural, enriched, and depleted uranium were processed, including recycled uranium – the latter involved the potential for exposure to transuranic contaminants
- Other materials production facilities, including heavy water and lithium-6 production facilities

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- Waste facilities, including two high-level waste Tank Farms, a high-level waste vitrification plant, a burning ground for radioactive waste, seepage basins for liquid discharges, and low-level waste burial areas, with potential for external and internal exposure, as well as exposures via environmental transport pathways
- Laboratories and other support facilities for the site

It has not been possible within the time and resources available for this review to examine all aspects of the Site Profile in detail due to the immense complexity and long history of the SRS facilities and the many changes that have occurred over the decades. We have selected certain issues for detailed discussion because they may affect dose reconstruction significantly or because they are methodologically important for this and other sites, or both.

This review has been hindered by delayed access to Savannah River Site and NIOSH information, including site technical reports, audit reports, and critical data. Some critical information was not received in time to address in this evaluation; the outstanding data requested is listed in Attachment 1. Recognizing that site profiles are “living documents,” which may be further revised or supplemented in the future, SC&A chose to issue this review despite the incompleteness of some of the planned inquiries.

The SC&A review procedures, as approved by the Advisory Board, require that each site profile be evaluated against five measures of adequacy (also referred to as review criteria), including (1) completeness of data sources, (2) technical accuracy, (3) adequacy of data, (4) site profile consistency, and (5) regulatory compliance. **The SC&A review of the Savannah River Site profile finds that the profile generally satisfies these objectives, although shortcomings and potential issues of varying significance exist that will need to be addressed.** Many of these involve lack of sufficient conservatism in key assumptions or estimation approaches, incomplete site data or incomplete analysis of that data, or incomplete reflection of operational or dosimetry history.

Following the introduction and a description of the criteria and methods employed to perform the review, the report describes the strengths of the site profile, followed by a discussion of the issues our review has uncovered. The strengths of the site profile and each issue are described and discussed with respect to the five major review criteria.

The issues were carefully reviewed and categorized as either Findings or Observations. The Findings and Observations are related to the technical accuracy and scientific and statistical validity of the site profile for estimating the various categories of doses that NIOSH uses in its assessment of claimant records. Findings represent deficiencies in the TBD that need to be corrected and which have the potential to substantially impact at least some dose reconstructions. Observations simply raise questions, which, if addressed, would further improve the TBD and may reveal deficiencies that will need to be addressed in future revisions of the TBD.

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Strengths

The Savannah River Site TBD has clear strengths, including its compilation and summary of site operations, summary of the site internal and external dosimetry programs, and use of personnel monitoring data as a basis for dose reconstruction. In many cases, the TBD has likely overestimated the dose to unmonitored or nonradiological workers with use of the missed external dose methodology. NIOSH/ORAU provides background information guidance within the TBD that allows the dose reconstructor to adjust evaluations in a claimant-favorable direction for increased breathing rate and extended work-weeks. To provide further direction on the interpretation of SRS data and dose reconstruction methodologies, a number of technical information bulletins have been developed to assist the dose reconstructor. In the case of SRS, internal, external, and environmental monitoring data are plentiful, as the site benefited from the experience of other Radiological Control programs within the DOE complex.

Issues

The SC&A review found that the use of the “high-five” approach as surrogate data for internal dose for unmonitored workers and for target organs that do not concentrate the radionuclide in question is not necessarily a maximizing approach for making dose estimates as claimed in the TBD. The method is in conflict with the 42 CFR 82-recommended methodologies for the calculation of internal dose. The completeness of the database from which the intakes were derived is likewise questionable. SC&A was not able to independently validate whether this approach considered chronic intakes (as well as acute intakes), because access was not provided to individual bioassay data that could corroborate such intakes.

For modeling of airborne radionuclide releases, one potentially significant issue is the non-conservatism of the standard Gaussian model used in the TBD where it pertains to “non-standardized” 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 1×10^{-9} per meter may not be claimant favorable by 3 to 4 orders of magnitude.

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 non-claimant favorability. The assumption that inhalation is the only pathway for internal exposure at SRS is questionable, given evidence that work practices and large particle sizes may have had a role in making ingestion a contributing pathway.

For external dose, the indicated correction factors do not take into account to a sufficient degree the uncertainty related to dosimeter use and processing, leading to likely underestimations. Another overarching issue is the use of the geometric mean when using surrogate data. NIOSH should consider using the 95th percentile values when using surrogate data to support dose reconstruction for workers that were not monitored.

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The site profile, while comprehensive in scope, does not sufficiently address several key potential sources of onsite radiation exposure at SRS. For example, for Tank Farm workers, NIOSH's site data evaluation appears to be incomplete with regard to exposure conditions and uncertainty estimates because primary data sources were not reviewed. The SRS TBD does not include transuranics and fission products in its definition of recycled uranium even though the presence of these contaminants in RU is well-established at SRS, as is their potential importance in dose reconstruction.

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, and recordkeeping, all of which hold significant implications for reconstructing doses for unmonitored workers during the early years. Similar gaps in data availability were noted for individual neutron 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, recycled uranium, and environmental releases.

Specific Findings, Observations, And Areas For Improvements

Findings

Finding 1: The use of the "high-five" approach as 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. The method is not consistent with the 42 CFR 82-recommended methodologies for the calculation of internal dose. The completeness of the database from which the intakes were derived is questionable.

Finding 2: 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 is not entirely appropriate for this class of problem.

Finding 3: The site profile does not contain 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.

Finding 4: The beta/gamma dosimeter adjustment factors and uncertainties applied underestimated 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.

Finding 5: 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 TLND recorded neutron doses between 1971 and 1995, as well as the

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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 for use.

Finding 6: The adequacy of the F- and H-area Tank Farm characterization in 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.

Finding 7: Solubility, oro-nasal breathing, and ingestion should be carefully considered in 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 EEOICPA.

Observations

Observation 1: The TBD does not adequately address potential exposures of workers handling tritium and performing decontamination and decommissioning to special tritium compounds including organically bound tritium and stable metal tritides.

Observation 2: The TBD has not completely evaluated the potential dose consequences related to the transplutonium program and non-military isotope production.

Observation 3: Incidents and high-risk jobs are not listed in the TBD or referenced to alert dose reconstructors to unique exposure conditions.

Observation 4: The adequacy of early worker monitoring data is questionable and requires further investigation.

Observation 5: Additional sources of external dosimetry data, primarily neutron dosimetry data, exist which are not currently being used in the dose reconstruction process.

Observation 6: 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/ORAU 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.

Observation 7: The special exposure circumstances for subcontractors and construction workers are currently not included in the TBD; however, NIOSH is aware of this issue and has developed a path forward for resolving it.

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Opportunities for Improvement

As a living document, the SRS Site Profile, Revision 02, may be improved by addressing specific issues raised in the main body of this review and briefly summarized below.

High-five Approach. A method to determine internal dose based on in vivo and in vitro bioassay data needs to be developed to determine internal dose to unmonitored or incompletely monitored individuals. This would alleviate many of the issues associated with the high-five approach and provide consistency among DOE sites.

Occupational Environmental Doses. NIOSH should use a consistent methodology for the calculation of occupational environmental dose that is appropriate for application to onsite workers. The components of environmental dose should be consistent between DOE facilities.

Recycled Uranium. The dose contribution for trace radionuclides in recycled uranium should be evaluated in terms of dose to particular organs of concern and the relative impact on internal dose reconstruction. NIOSH should evaluate the lack of formal policies for trace radionuclides in recycled uranium and develop bounding conditions that can be applied to DOE facilities including the SRS.

Beta/Gamma Dosimeter Adjustments and Uncertainties. A method to consistently account for laboratory, radiological, and environment uncertainties in dosimeter readings should be developed and appropriately applied to recorded dosimeter results.

Neutron Dosimetry. Technically defensible and claimant-favorable uncertainty factors associated with application of neutron-to-photon ratios should be developed. A claimant-favorable alternative is to use the 95th percentile neutron-to-photon ratio as a point estimate for all claimants, regardless of the compensability of the claim. Further investigation into the relative effectiveness of TLNDs and their uncertainty should be completed and appropriate uncertainties applied.

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

Internal Dosimetry Assumptions. The solubility assumptions associated with organ dose derived from urine need to be discussed further. The assumption of oro-nasal breathing should be used in a manner similar to solubility, giving the claimant the benefit of the doubt with respect to mouth versus nasal breathing. Further justification for exclusion of ingestion from internal dose should be included in the TBD, especially in light of its inclusion in other TBDs.

Special Exposure Conditions. Consideration should be given to the contributions to internal and external dose from radionuclides produced in special campaigns and exposure of workers, especially those involved in tritium production and decontamination and decommissioning to special tritium compounds. The comprehensiveness and consistency in the early monitoring

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program deserves further investigation to account for potential missed dose. NIOSH should continue to pursue issues associated with construction worker exposure.

Incidents and High-Risk Jobs. NIOSH should evaluate incidents and nonroutine high-risk jobs that occurred at the SRS, and determine whether the approaches in the TBD bound these situations. This information should be provided to dose reconstructors for their consideration in the dose reconstruction process.

Data Completeness: Additional neutron exposure records in the site's record collection should be reviewed and their relative importance to dose reconstruction determined. The data from the multiple dosimetry program should be evaluated to determine whether this data is beneficial to dose reconstruction and to identify situations where nonuniform exposure was an issue.

Quality Assurance: The direction provided in the TBD should be understandable to the dose reconstructor and consistent throughout the document. Direction on application of DOE complex-wide TIBs is needed.

In light of the issues raised in this report, NIOSH should update other profiles where applicable.

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2.0 SCOPE AND INTRODUCTION

The Savannah River Site occupies 200,646 acres or about 300 square miles between Aiken, South Carolina, and Augusta, Georgia, on the coastal plain bordering the Savannah River. It was selected in 1950 to produce nuclear weapons materials, principally plutonium and tritium. Construction at SRS began in February 1951 and production operations began in December 1953. From the site's inception, E.I. DuPont de Nemours and Company (DuPont) was responsible for construction and remained its prime contactor through March 31, 1989. On April 1, 1989, Westinghouse Savannah River Company (WSRC) took management of the site. Five heavy-water pressurized reactors, two large plutonium and uranium chemical separations plants, a heavy-water production plant, nuclear fuel and target fabrication facilities, tritium processing facilities, several test reactors, and research and development laboratories were among the major facilities located at the site. After the end of the Cold War in the late 1980s, the SRS nuclear weapons mission was curtailed to tritium processing. Currently, SRS's mission also includes environmental restoration, decontamination and decommissioning of nuclear facilities, waste management, plutonium storage, and fissile material disposition activities.

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

This report provides a review of ORAUT-TKBS-0003, *Technical Basis Document for the Savannah River Site To Be Used for EEOICPA Dose Reconstructions* (Scalsky 2003), and its supporting technical information bulletins (TIBs), ORAUT-OTIB-0001, *Technical Information Bulletin: Maximum Internal Dose Estimates for Savannah River Site (SRS) Claims* (Brackett 2003), ORAUT-OTIB-0003, *Technical Information Bulletin: Savannah River Site Tritium Dose Assignment* (Duncan 2003), OCAS-TIB-006, *Interpretation of External Dosimetry Records at the Savannah River Site* (Neton 2004), and OCAS-TIB-007, *Neutron Exposures at the Savannah River Site* (Neton 2003). SC&A, in support of the Advisory Board, has critically evaluated the SRS site profile in order to:

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

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SC&A's review of the site profile and its supporting TIBs focuses on the quality and completeness of the data that characterized the facility and its operations and the methods prescribed by NIOSH for its use of these data in dose reconstruction. The review was conducted in accordance with the objectives stated in *Standard Operating Procedure for Performing Site Profiles Reviews* (SC&A 2004).

NIOSH has revised the SRS TBD since the initiation of this review process. This review reflects information currently available in Revision 2 of the SRS site profile. In a June 24, 2004, conference call with SC&A, NIOSH indicated that they were evaluating exposures to outside workers, presumably Construction Workers (see Attachment 3). Section 6.0 of the TBD, *Trades Workers*, has been reserved to specifically address Construction Workers. In an e-mail from NIOSH dated January 10, 2005, NIOSH also indicated that further updates to the SRS TBD were planned (Hinnefield 2005). The content of these changes has not been provided to SC&A. Following receipt of the remaining records requested, preparation of the construction worker dose chapter, and completion of planned updates to the SRS TBD, SC&A will revise this report if so requested by the Advisory Board.

The site profile review has been hindered to a degree by the limited and/or delayed access to SRS and NIOSH information, including site technical reports, audit reports, and critical data. SC&A conducted onsite and offsite interviews of current and previous workers from August 23-26, 2004. During this visit, SC&A copied a number of primary source records relating to SRS operations and the radiological protection program. Prior to their release, the records required review by onsite staff to evaluate their appropriateness for release to SC&A. A follow-up request based on interviews was prepared on September 9, 2004, and submitted to NIOSH for referral to the Savannah River Operations Office (SROO). Neither the records copied in August nor those requested in September were received as of November 2004. SC&A submitted a second request to NIOSH for access of these records. At the Advisory Board meeting in St. Louis, Missouri, on February 7-9, 2005, NIOSH provided SC&A with several of the requested documents, which were very helpful in permitting us to perform a more comprehensive review of the site profile. A portion of the documents copied in August 2004 were received March 2, 2005, with the remaining documents requiring additional review at the Savannah River Site. Attachment 1 provides a list of the outstanding records requested from NIOSH and the SROO. As a result of the delays in providing records, SC&A has not been able to perform a comprehensive review of records recently received and integrate this information into the current review. Although we have not received all the records requested, we believe that we have reached a point in our investigations where it is possible to provide useful review findings and commentary on the Savannah River site profile. During the review of this report by the Advisory Board, discussions can be held and judgments can be made regarding the need for further investigations and records acquisition.

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:

- 1.0 *Executive Summary*, including Summary of Strengths, Findings, and Observations, and Opportunities for Improvement

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- 2.0 *Scope and Introduction*
- 3.0 *Assessment Criteria and Method*
- 4.0 *Site Profile Strengths*
- 5.0 *Vertical Issues*
- 6.0 *Overall Adequacy of SRS Site Profile*

Based on the issues raised in each of these sections, SC&A prepared a list of issues, which are provided in the Executive Summary. Issues are designated as findings if we believe that they represent deficiencies in the TBD that need to be corrected, and which have the potential to have a substantial impact on at least some dose reconstructions. Issues are designated as observations if they simply raise questions, which, if addressed, would further improve the TBD and may perhaps reveal deficiencies that will need to be addressed in future revisions of the TBD.

Many of the issues surfaced in the report correspond to more than one of the major objectives (i.e., Strengths, Completeness of Data, Technical Accuracy, Consistency Among Site Profiles, and Regulatory Compliance.) Section 6.0 provides a list of the issues in summary form, whether the issue constituted a finding or observation, and to which objective the particular issue applies. In future site profile reviews, we may want to consider organizing the reviews according to the major chapters of the site profiles (i.e., Facilities and Processes, Occupational Medical Dose, Occupational Environmental Dose, Occupational Internal Dose, Occupational External Dose, and Trades Workers) and address each of the five review objectives as they apply to each chapter of the site profile. We believe that this approach to organizing the report will avoid unnecessary redundancy, will result in a more logical organization of the report, and result in a more readable document.

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3.0 ASSESSMENT CRITERIA AND METHOD

S. Cohen and Associates is charged with evaluating the approach set forth in the site profiles, which is used in the individual dose reconstruction process. These documents are reviewed for their completeness, technical accuracy, adequacy of data, consistency with other site profiles, and compliance with the stated objectives, as defined in *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004). This review is specific to the Savannah River Site (SRS) site profile and supporting technical information bulletins (TIBs); however, items identified in this report may be applied to other facilities, especially facilities with similar source terms and exposure conditions.

3.1 Objectives

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

3.1.1 Objective 1: Completeness of Data Sources

SC&A reviewed the site profile with respect to Objective 1, which requires SC&A to identify principal sources of data and information that are applicable to the development of the site profile. The two elements examined under this objective include (1) determining if the site profile made use of available data considered relevant and significant to the dose reconstruction, and (2) investigating whether other relevant/significant sources are available but were not used in the development of the site profile. For example, if data are available in site technical reports or other available site documents for particular processes, and if the TBD has not taken into consideration these data where it should have, this would constitute a completeness-of-data issue. The ORAU Site Profile Document database, as well as the referenced sources in the TBD, were evaluated to determine the relevance of the data collected by NIOSH to the development of the site profile. Additionally, SC&A evaluated records publicly available relating to the 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 which 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/ORAU has analyzed available data, but the technical approach used by NIOSH in the analysis of these data was found by SC&A to be scientifically unsound or not necessarily claimant favorable, this would constitute a technical accuracy issue.

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3.1.3 Objective 3: Adequacy of Data

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

3.1.4 Objective 4: Consistency Among Site Profiles

SC&A reviewed the site profile with respect to Objective 4, which requires SC&A to identify common elements within site profiles completed or reviewed to date, as appropriate. In order to accomplish this objective, the SRS TBD was compared to the Hanford TBDs. The Hanford site profile is appropriate for comparison, as the sites had similar missions. This assessment was conducted to identify areas of inconsistencies and determine the potential significance of any inconsistencies with regard to the dose reconstruction process.

3.1.5 Objective 5: Regulatory Compliance

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

In order to place the above objectives into the proper context as they pertain to the site profile, it is important to briefly review key elements of the dose reconstruction process, as specified in 42 CFR Part 82. Federal regulations specify that a dose reconstruction can be broadly placed into one of three discrete categories. These three categories differ greatly in terms of their dependence on and the completeness of available dose data, as well as on the accuracy/uncertainty of data.

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

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

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

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

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

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

In order to achieve the five objectives described above, SC&A reviewed each of the major sections of the site profile, their supplemental attachments, and TIBs, giving due consideration to the three categories of dose reconstructions that the site profile is intended to support. The Savannah River site profile is divided into six major categories of background information and guidance for use by dose reconstructors. The following briefly describes each major section of the site profile and our approach to reviewing each section.

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The first section, 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 our review, we were cognizant of the fact that the Statute or 40 CFR 42, which implements the statute, does not require the site profile. Site profiles were developed by NIOSH as a resource to 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.

The introduction is an extremely important part of the site profile because it provides a description of the facilities, processes, and historical information that serves as the underpinning for subsequent sections of the site profile. Specifically, the introduction, along with Attachment A, describes 30 facilities and processes and their associated source terms that are relevant to dose reconstruction. Our review of this section specifically addresses whether all the potentially important site activities and processes are described, and whether the characterization of the source terms seemed sufficient to support dose reconstruction.

Section 2 of the site profile provides a set of procedures for reconstructing the medical exposures experienced by workers as a requirement for employment at the Savannah River Site. SC&A reviewed this section for technical adequacy and consistency with other NIOSH procedures and other site profiles.

Section 3 of the site profile provides background information and guidance to dose reconstructors for reconstructing the doses to unmonitored workers outside of the facilities at the site, and who may have been exposed to routine and episodic airborne emissions from the facility. We 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.

Section 4 of the site profile presents background information and guidance for use by dose reconstructors for deriving occupational internal doses to workers. This section was reviewed with respect to background information and guidance regarding the types, mixes, and chemical forms of the radionuclides that may have been inhaled or ingested by the workers, the recommended assumptions for use in reconstructing internal doses based on whole-body counts and bioassay data, the methods recommended for use in the reconstruction of missed internal dose, and the methods recommended for characterizing uncertainty in the reconstructed internal doses.

Section 5 of the site profile presents background information and guidance for use by dose reconstructors for deriving occupational external doses to workers. This section was reviewed

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with respect to background information and guidance regarding the different types of external exposures (e.g., gamma, beta, and neutron) and the energy distribution of the external radiation that may have been experienced by the workers, recommendations regarding how to convert external dosimetry data to organ-specific doses, the methods recommended for use in the reconstruction of missed external doses, and the methods recommended for characterizing uncertainty in the reconstructed external doses.

Section 6 of the site profile, titled “Trades Workers,” is held in reserve and therefore is minimally addressed in this review to remind NIOSH of its importance.

In accordance with SC&A’s site profile review procedures, SC&A performed an initial review of the site profile and its supporting documentation and TIBs. SC&A then submitted questions to NIOSH with regard to assumptions and methodologies used in the site profile. These questions, along with written responses from NIOSH/ORAU regarding these questions, are provided in Attachment 2. A conference call was then conducted between NIOSH and the SC&A team allowing NIOSH to provide clarifications and explain the approaches employed in the site profile. A summary of the conference call is provided in Attachment 3.

Site expert interviews were conducted to assist the team in obtaining a comprehensive understanding of the radiation protection program, site operations, and environmental contamination. Attachments 4 and 5 provide a summary of the site expert interviews conducted by the SC&A team during a visit to Aiken, South Carolina, on August 23–26, 2004. Site experts were given an opportunity to review the interview summary for accuracy of interpretation of their input. This is an important safeguard against missing key issues or misinterpreting some vital piece of information. Although most site experts provided comments, not all site experts responded to SC&A’s request for review.

An extensive comparison was done between the methodologies used in the SRS and Hanford TBDs to determine medical occupational, environmental, internal, and external dose. This comparison focuses on the methodologies and assumptions associated with dose determination and the values used to obtain a probability of causation. A detailed analysis is provided in Attachment 6.

After compiling site expert interviews, documentation, and NIOSH input, issues raised were carefully evaluated. Information provided in the conference call by NIOSH was evaluated against the preliminary findings and observations to finalize the vertical issues¹ addressed in the audit report. To date, there were two levels of review for this report. First, the SC&A team members reviewed the report internally. Second, SC&A appointed an outside consultant, Mike Thorne, who did not participate in the preparation of this document, as an internal reviewer to go over all aspects of this report.

The outcome of this two-step review process resulted in the preparation of this report, which is referred to as a “working draft.” This working draft has been submitted to selected members of

¹ The term “vertical issues” refers to specific issues identified during our review, which were identified as requiring more in-depth analysis due to their potential to have a significant impact on dose reconstruction.

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the Advisory Board and NIOSH for factual accuracy review, and to alert NIOSH and the Advisory Board to our initial findings and observations. NIOSH and the Advisory Board will have an opportunity to review this working draft and provide comments to SC&A. Our plans are to then hold a meeting with NIOSH and Advisory Board representatives, where the working draft is discussed and the discussions recorded. Following this meeting, our plans are to revise this report and then deliver it to the Advisory Board and NIOSH. The report will, at that time, be published on the NIOSH web site and discussed at the next Advisory Board meeting. This last step in the review cycle of the TBD and its supporting TIBs will complete SC&A's involvement in the review process, unless the Advisory Board requests SC&A to participate in additional discussions regarding the closeout of issues, or if NIOSH issues a revision to the TBD or additional TIBs, and the Advisory Board requests SC&A to participate in the review of these documents.

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

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4.0 SITE PROFILE STRENGTHS

In developing a technical basis document (TBD), the assumptions used must be fair, consistent, and scientifically robust, and uncertainties and inadequacies in source data must be explicitly addressed. The development of the TBD must also consider efficiency in the process of analysis of individual exposure histories, such that claims can be processed in a timely manner. With this perspective in mind, there were a number of strengths identified in the Savannah River Site (SRS) TBD. These strengths are described in the following sections.

4.1 Completeness of Data Sources

- (1) In an effort to be comprehensive in addressing the range of facilities and processes at the SRS, NIOSH effectively compiled facility-specific information from *Facility Descriptions* (LaBone 1996), and the SRS internal and external dosimetry technical basis manuals. Facilities were divided into 30 categories, and a concerted effort was made to characterize the types and relative importance of the various radionuclides that may have contributed to internal and external exposures at the various facilities and associated processes over the life of the facility. We consider this to be one of the greatest strengths of the report.
- (2) In developing the site profile, NIOSH drew upon information contained in 274 reports cited in the reference section. These include the annual environmental reports beginning in 1964 that present the annual releases from the facility, health physics annual regional monitoring reports beginning in 1959, and numerous authoritative historical documents describing the internal and external dosimetry methods employed at the facility. In the case of the medical x-ray exposures, the TBD makes use of procurement records to determine whether and when photofluorography was used at the site. NIOSH/ORAU met with construction workers in September 2004 to identify special concerns of this group of workers. This interaction with workers could provide valuable insight into site processes and programs, and also increases public confidence in the dose reconstruction process. In addition, the multiple and substantial revisions to the site profile, along with the issuance of several TIBs, reflect an ongoing effort by NIOSH to continually improve the background information and guidance provided to the dose reconstructors.
- (3) In compiling the atmospheric source terms for deriving outdoor occupational exposures to unmonitored workers, NIOSH made a concerted effort to compile the source term data needed to reconstruct the doses to these categories of workers. Notwithstanding this effort, there are opportunities for improvement in the methods used to reconstruct the doses to these categories of workers.
- (4) For the purpose of compiling data needed to reconstruct internal doses based on historical operations, NIOSH compiled an enormous amount of data describing the radionuclides and operations at the various facilities and their associated processes. To almost a fault, NIOSH provides guidance to dose reconstructors on how to navigate through the complex mix of radionuclides required to reconstruct historical internal exposures to workers. Notwithstanding this achievement, there are opportunities for improvement in

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the data sets and instructions to the dose reconstructors with respect to reconstructing internal exposures.

4.2 Technical Accuracy/Claimant Favorability

- (1) NIOSH made a concerted effort to determine the minimum detectable levels (MDLs) that were associated with the various types of dosimeters used over the life of the facility, for the different types of operations, and for the different types of external exposures. Though we have some commentary on the external dosimetry program, especially during the early years, we believe NIOSH's description of the MDLs reflects an excellent attempt at dealing with missed external doses for most workers. In addition, the guidance provided by NIOSH for assigning missed dose based on MDLs generally gives the benefit of the doubt to the claimant.
- (2) The TBD recommended adjustments to respiration rate based on the level of exertion (i.e., heavy or light work) experienced by workers. We consider this a refinement of the models that helps to assure that reconstructed doses are not underestimated.
- (3) The TBD allows for adjustment of environmental dose based on actual number of work hours.
- (4) $H_p(10)$ MDL values are consistent with the scientific literature and, in combination with their exchange frequency, provide a technically sound basis for estimating missed photon doses.
- (5) The TBD recommends the use of the most claimant-favorable chemical form of inhaled radionuclides for the organ of interest in order to give the benefit of the doubt to the claimants.
- (6) NIOSH's use of the hypothetical intake described in ORAUT-OTIB-0001 likely overestimates the dose to non-radiological workers and minimally exposed workers.
- (7) X-ray and photofluorography have been investigated thoroughly to determine medical x-ray techniques used at SRS. Explicit consideration of photofluoroscopic examinations is especially important because of their potential for relatively large exposures, as compared to conventional x-rays.
- (8) The TBD's use of personnel monitoring data and air sample data to determine dose is consistent with the requirements outlined in 42 CFR 82.
 - Where urinalysis is available, this information is used to calculate internal dose.
 - Where beta/gamma dosimeter data is available, this information is used to determine the shallow and deep dose.
 - Where TLND data is available, this information is used to assign neutron dose.

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- (9) NIOSH published a number of TIBs that provide further direction to the dose reconstructor. These documents were beneficial in understanding the application of the TBD to the dose reconstruction process and were also reviewed.

4.3 Adequacy of Data

The TBD benefited from having access to the database that was compiled as part of the following SRS programs:

- (1) The SRS benefited from the knowledge of other predecessor DOE facilities as they established their radiation protection program. SRS was the first DOE site to perform a pre-operation environmental monitoring program, which later became an integral part of environmental monitoring programs. In addition, SRS implemented the initial radiation work permit program or equivalent at the beginning of operations, and was the first to implement administrative control limits (Attachment 4).
- (2) The radiation monitoring program was initiated in 1951 with the use of the Oak Ridge National Laboratory film badges and neutron track emulsion, Type A (NTA) badge. Within 2 years, SRS was processing beta/gamma film badges and NTA film (Taylor et al. 1995). The site established a policy for monitoring all workers entering defined Regulated Zones or Radiation Danger Zones. Dosimeters have been the responsibility of a central organization since the inception of the program. This included dosimeters issued to visitors, subcontractors, construction workers, and DOE personnel (Attachment 4).
- (3) SRS established dosimetry calibration programs consistent with the technology and processes of the time. They have historically been involved in a continued effort to improve in vivo and in vitro bioassay processes as new technology became available. The site participated in the Department of Energy Laboratory Accreditation Program (DOELAP) starting in the mid-1980s. The SRS thermoluminescent dosimeter (TLD) was DOELAP-accredited in 1989, indicating it had met the standard requirement for dosimeter calibration. The site has been cognizant of special exposure issues, such as exposure to low-energy photons in plutonium finishing and storage areas, and has made adjustments to their program as needed to account for these issues. The SRS implemented a multiple badging program with the initiation of work at the site, recognizing that there were tasks with nonuniform exposure fields (Attachment 4).
- (4) Whole-body and chest counting were initiated at SRS in 1960 and 1970, respectively. The first bioassay programs for plutonium, uranium, and tritium were initiated in 1954. Bioassay for other radionuclides, such as fission products, trivalent actinides, and neptunium, were also developed. As radiochemical techniques improved, the site adopted these techniques to improve their capabilities for detection of specific radionuclides and to improve the detection limit capabilities (Taylor et al. 1995).

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- (5) Internal doses have continued to be reevaluated as dosimetry models improved, additional bioassay data were collected, and new individuals were identified as having assimilations.
- (6) SRS has actively monitored airborne and liquid effluent releases from the site since the beginning of operations. Also, they have been actively involved in monitoring soil, vegetation, tributaries, and wildlife (DuPont 1978).

4.4 Consistency Among the Site Profiles

Although the SRS and Hanford had similar missions, there are some differences in the facility processes, design of facilities, and radiological practices. In some cases, these differences require site-specific assumptions in dose determinations. For example, due to the design of the REDOX facility at Hanford, there were substantial particulate releases of ruthenium that had to be considered. In the case of SRS, the facilities were built later and this was not an issue. NIOSH/ORAU made a concerted effort to recognize and address these differences in the TBDs. With respect to the Interactive RadioEpidemiologic Program (IREP) input parameters, the SRS and Hanford TBDs were consistent in many cases, although there is room for improvement in some areas. This consistency was especially apparent with the medical occupational exposure sections.

4.5 Regulatory Compliance

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

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

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

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5.0 VERTICAL ISSUES

SC&A has developed a list of key issues regarding the SRS site profile. These issues relate to each of the five objectives defined in *SC&A Standard Operating Procedure for Performing Site Profile Reviews* (SC&A 2004). Some issues were related to a particular objective, while others covered several objectives. A matrix relating to the particular objectives and the relative severity of each issue is provided in Section 6.0 of this report. Many of the issues raised below are applicable to other Department of Energy (DOE) and Atomic Weapons Employer (AWE) sites, and should be considered in the preparation and revision of other site profiles.

5.1 Issue 1: High-Five Approach (also referred to as the Hypothetical Intake)

The SRS TBD recommends the use of a maximizing approach for likely non-compensable claims with non-metabolic or digestive system cancers (ORAU 2004, p. 85). ORAUT-OTIB-0001, *Maximum Internal Dose Estimates for Savannah River Site Claims*, describes a “maximizing approach” for estimating internal doses for unmonitored workers for organs that do not concentrate the radionuclides in question – that is for digestive tract organs and non-metabolic 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 non-compensable cases (Bracket 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 likely 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 trades 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.

Our review of the document ORAU-OTIB-0001 has identified a number of issues related to the application of the high-five method.

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5.1.1 Regulatory Compliance

The dose reconstruction process must comply with the requirements of 42 CFR 82, *Methods for Conduction Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program Act of 2000*. As a method of effectively implementing these requirements, NIOSH has written technical guidance documents on external and internal dosimetry. 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 the guidance outlined in 42 CFR 82.

The method outlined in ORAUT-OTIB-0001 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 in the characteristics above, the intakes used in the high-five approach are calculated by SRS using ICRP 30. The organ dose is then calculated using ICRP 66 and 68 modeling and default values. Title 42, Part 82.18 (b) of the *Code of Federal Regulations* directs NIOSH to:

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

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Furthermore, in the question and answer section preceding the actual rule on dose reconstruction, 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 utilize intake information calculated using ICRP 30 methodology. Since the issuance of ICRP 30, the ICRP has developed a new lung model, which is outlined in ICRP 66, and revised dose coefficients in ICRP 68. ICRP 30, therefore, is not the most current metabolic model and is not consistent with the direction provided by 42 CFR §82.18 (b).

5.1.2 Adequacy of Data

The “high-five approach” utilizes as its basis the Savannah River Site Internal Dosimetry Registry (IDR). 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 was to ensure appropriate follow-up bioassay of individuals, and to ensure that workers with significant intakes are informed about the United States Transuranium and Uranium Registries (WSRC 2001). From 1951–1983, the site implemented the *Report of Committee II on Permissible Dose for Internal Radiation (ICRP 2)* methodology for intake determination. There was an action level concentration 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. These individuals would then be included in the SRS IDR. In about 1984, the site implemented the ICRP 30 methodology for dose calculation for radionuclides other than tritium. At that time, the criteria for inclusion in the IDR was changed to those individuals who received 100 mrem during the first year following intake (DPSOP 1987). With the release of the *Department of Energy Radiological Control Manual* (DOE 1994), the criteria was changed to 100 mrem CEDE. Most recently, the criteria for inclusion in the SRS IDR was established at 10 mrem CEDE (see Attachment 4). 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 were included in the IDR. There are approximately 1,100 individuals included in the registry.

As mentioned above, the criteria for being included in the IDR has changed over time. While the IDR contains many of the intakes that occurred at SRS, the completeness of the registry was not evaluated in the TBD or in ORAUT-OTIB-0001. SRS internal dosimetry indicated that those individuals with bioassay samples above the decision level prior to January 1, 1989, who were not involved in a recorded incident, may not be listed in the registry (see Attachment 4). For example, NIOSH/ORAU utilized a single Nb-95 intake from the registry, which occurred in 1983. The *Progress Report, December 1960, Works Technical Department (DPSP 1960)*, indicates other intakes of Nb-95 occurred prior to 1983.

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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-Pr¹⁴⁴ was detected in 66% of the personnel, Ru-Rh¹⁰⁶ in 29% and Zr-Nb⁹⁵ in 22%. The maximum body burdens found in these individuals were 49 nanocuries of Ce-Pr¹⁴⁴, 36 nanocuries of Ru-Rh¹⁰⁶, 7.2 nanocuries of Zr-Nb⁹⁵ and 24 nanocuries of Cs-Ba¹³⁷.

Furthermore, monthly Works Technical Department reports (DPSP series) submitted to senior contractor and Atomic Energy Commission managers reported the following radiation exposure problems associated with F-Area A-line:

- **January 1961** -- *“Replacement of spent silica gel in A-line columns S-8-1 and S-8-2 resulted in unusually high airborne fission product concentrations inside the column cells. During removal of the material from the columns, air activity increased to 520X10-10 ucFP/cc (173XRCG); during addition of new silica gel, air activity was observed, fresh air masks or air-supplied plastic suits were worn for further work. Body exposure rates over the open columns ranged to 200 mr/hr.”* (DPSP 1961)
- **February 1962** – *“Damage to bottoms of two denitrator vessels [in the F-Area A-line] necessitated replacement of the damaged portions. Body exposure rates ranged to 100 mr/hr during the repairs.”* (DPSP 1962a)
- **May 1962** – *“An unusual number of incidents [in the F-Area A-line] including three fires on top of the denitrator pots, one fire in a dumpster waste pan, two denitrator pot fumeouts, two denitrator pot blowouts and a broken denitrator pot shaft increased the need for greater health physics vigilance. The importance of wearing proper respiratory protection for work in the denitrator room to eliminate assimilations became increasingly evident.”* (DPSP 1962b)
- **February 1963** – *“A failed portion of the c-3-6 denitrator pot bottom was cut out and a patch was welded in. Body exposure rates ranged to 125 mrad/hr. Radiation intensities to 1 rad/hr at 2” were encountered from the failed section.”* (DPSP 1963)
- **March 1967** – *“Failed steam coils in hydrate evaporator c-2-1 were replaced [in the F-area A-Line]. Radiation from the old coils was 75 mrad/hr at 2” with transferable beta-gamma contamination of 1000 c/m at 1”. (DPSP 1967)*

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

These protection measures, notwithstanding, records indicate that 99 workers received internal doses of uranium over the history of the plant, which are well documented in site incident reports.

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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 indicate that there were 155 positive bioassays for uranium between 1953 and 1959 alone.²

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, “Largest intakes assigned at SRS (radionuclides available in IMBA),” is from the 1950s; one is from 1962 and the rest are from the 1970s (Bracket 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 record used to compile the “high five” may be inadequate to determine the highest five intakes for the listed radionuclides. There is not even one entry from the 1950s for any of the fission products in Table 2.

Our 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. We believe these types of follow-up activities likely occurred, because we note that there were over 400 cases of post-incident chelation therapy (page 114). Based on these findings, we suggest 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.

There are several radionuclides for which no bioassay technique was available in the early years during peak production. For example, Attachment D, Table D-1 of 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. Based on experience at other similar facilities, such as Hanford, there was likely Am-241 contamination in the waste streams as well as other areas. 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.

² DPSP, 56-1-7, DPSP 56-1-8, DPSP 56-1-9, DPSP-56-1-10, DPSP-56-1-11, DPSP-56-1-12, DPSP-57-1-1, DPSP-57-1-2, DPSP-57-1-4, DPSP-57-1-5, DPSP-57-1-6, DPDP-57-1-7, DPSP-57-1-8, DPSP-57-1-9, DPSP-57-1-10, DPSP-57-1-11, DPSP-57-1-12, DPSP-58-1-6, DPSP-58-1-7, DPSP-58-1-8, DPSP-58-1-8, DPSP-58-1-9, DPSP-58-1-10, DPSP-58-1-12, DPSP-59-1-1, DPSP-59-1-2, DPSP-59-1-3, DPSP-59-1-4, DPSP-59-1-5, DPSP-59-1-6, DPSP-59-1-7, DPSP-59-1-8, DPSP-59-1-9, DPSP-59-1-10, DPSP-59-1-11, DPSP-59-1-12.

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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 with respect to period of operation and to radionuclides in the source terms with due regard to the complex history of radionuclide production. Furthermore, NIOSH should consider its impact on the IDR and on other internal dose reconstructions utilizing an individual's bioassay results.

5.1.3 Technical Accuracy

There were three technical issues associated with the high-five approach. 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 highest dose to an organ. There is a fundamental problem with the comparisons of these intake retention fractions (IRF) from ICRP 30 and ICRP 68.

The hypothetical intake outlined in ORAUT-OTIB-0001 uses the Savannah River Site 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 are then averaged. The average activity (nCi) is entered into IBMA and a dose is calculated based on the *Human Respiratory Tract Model for Radiological Protection*, ICRP 66, and *Dose Coefficients for Intakes of Radionuclides by Workers*, 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 an acute inhalation occurred on January 1 in the first year of employment (Brackett 2003).

NIOSH justified the use of intakes calculated with ICRP 30 methodologies rather than the newer ICRP methodology prescribed for the dose reconstruction effort by comparing 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 Am are not significantly overestimated as stated on page 8 of the ORAUT-OTIB-0001. 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), U (types F, M, and S), Pu (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. Ruthenium-106, types M and F, are underestimated most of the time using ICRP 30 methodology. For Am-241, type M, ICRP 30 methodology may or may not underestimate the intakes, depending on the time samples are taken after the intake. Attachment 6 presents a series of calculations that demonstrate that the approach used in this ORAUT-OTIB-0001 is not claimant favorable for many radionuclides, and that ICRP 68 models would have been more claimant favorable.

Furthermore, 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

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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.” 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” (Bracket 2003, pp. 5-8). There is a fundamental problem with the comparisons of these IRFs from ICRP 30 and ICRP 68. When intakes are used to calculate organ doses, then, in general, the choice of the most soluble type is claimant friendly 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 2005, p. 152):

- *A 24-hour urine sample is collected five days after a single inhalation intake of Pu-238. The bioassay result is 1 becquerel (Bq) of Pu-238. Using the ICRP 67 model for Pu, the calculated intakes are:*
 - *For Pu-238, type S: intake of 2.2 E6 Bq (50 y committed bone surface dose is 75 sieverts (Sv), 50y committed dose to colon is 0.053Sv, 1y committed dose to the colon is 0.006 Sv).*
 - *For Pu-238, type M: intake of 2.6 E4 Bq (50 y committed bone surface dose is 24 Sv, 50y committed dose to colon is 0.042Sv, 1y 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 ICRP 30 IRF from Table 3, page 6, of ORAUT-OTIB-0001, the same bioassay result of 1 Bq of Pu-238 in a 24-hour urine sample, taken five days after a single intake, using ICRP 30 IRF from table 3, page 6, ORAUT-OTIB-0001, corresponds to intakes of:*
 - *For class Y: intake of 3.5 E5 Bq (ICRP30) (50 y committed bone surface dose is 12 Sv, 50y committed dose to colon is 0.33Sv, 1y committed dose to the colon is 0.037 Sv).*
 - *For class W: intake of 1.9 E4 Bq (ICRP30) (50 y committed bone surface dose is 17.5 Sv, 50y committed dose to colon is 0.03Sv, 1y 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 directs the dose reconstructor to use surrogate radionuclides for radionuclides included in the high-five approach, which are not available in IMBA.

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Dose for intakes of the radionuclides in Table 2 could not be calculated using either version of the IMBA code. Therefore, the Table 2 radionuclides were associated subjectively with three of the Table 1 radionuclides with similar irradiation characteristics. These Table 1 radionuclides are referred to as surrogate radionuclides in this paper. Cs-137 was assigned as surrogate for Zn-65 and Zr-95. Sr-90 was the surrogate for Ru-106, Ce-144 and Nb-95. Cm-244 was the surrogate for Cm-242 and Cf-252.

An effective dose for each Table 2 radionuclide was calculated by multiplying its intake by the largest, inhalation, 5 um AMAD, effective dose coefficient listed in ICRP 68 for the given radionuclide. Each surrogate radionuclide intake was multiplied by its ICRP 68 inhalation, 5 um AMAD, effective dose coefficient for the absorption type noted in Table 11. The effective doses were summed for each of the three radionuclide groups. The sums were divided by the respective surrogate radionuclide effective dose to determine a dose adjustment factor for each surrogate that accounts for the associated radionuclides' assumed dose contribution. The results appear in Table 12.

Annual organ doses from the IMBA runs for the surrogate radionuclides were multiplied by the Surrogate Dose Adjustment Factor to account for the doses from the intakes of the associated Table 2 radionuclides.

The surrogate radionuclide should have the same or similar distribution in the body, biological half-life, and solubility type.

The method used for determining surrogate radionuclides is not clearly explained, including the use of Type F nuclides as surrogates to Type M and S nuclides. When surrogate radionuclides are used, they should have the same or similar distribution in the body, effective half-lives, and solubility types. The initial intake for Cs-137 was calculated using a Class D solubility versus the Class Y and mixture of Classes D and W for Zn-65 and Zr-95. Cesium-137 also has a substantially different physical half-life than either Zn-65 or Zr-95. The initial intake value Sr-90 assumed 80% Class D and 20% Class W, whereas the Nb-95 value was based on a 100% Class W intake, the Ru-106 intake value was based on a Class W and Y intake, and the Ce-144 intake value was based on a 100% Class W intake. For example, for the mixture of nuclides represented by Sr-90, the dose calculated using the method presented in Table 12 of the TBD is 45% of the actual dose of the mixture using Ru-106 Type M, and 12% of the real dose of the mixture using Ru-106 Type F.

- The 50y committed equivalent dose to the adrenals due to the sum of type F Sr-90, type M Ru-106, type M Ce-144 and type M Nb-95, for the intakes described in ORAUT-OTIB-0001, Table 12, is 7.07E-5 Sv.
- The 50y committed equivalent dose to the adrenals due to the sum of type F Sr-90, type F Ru-106, type M Ce-144 and type M Nb-95, for the intakes described in ORAUT-OTIB-0001, Table 12, is 1.41E-4 Sv.

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- Using the procedure described in the OTIB, the 50y committed equivalent dose to the adrenals due to all four nuclides is $3.19E-5$ Sv.

For example, the dose to the adrenals using the correct ICRP dosimetric and biokinetic models for each nuclide is higher than the dose using the procedure described in the document.

Curium-242, Cm-244, and Cf-252 had the same target organs and relatively consistent solubility classes; however, Cm-244 and Cf-252 deliver a higher dose to **most** organs.

A more prudent approach to the absence of radionuclides in the IMBA code is to use the dose coefficients provided in the ICRP CD-ROM and employ a linear interpolation for the radionuclides that are not explicitly given, or have these radionuclides added to the code, as this will continue to be an issue at other sites.

5.1.4 Completeness of Data

The Hypothetical Intakes were based on recorded intakes at SRS. However, the procedure does not describe the methods that were used to calculate the SRS intake values, the data upon which they were based, and if they should be used throughout the SRS. Although the SRS-derived intakes are used as a basis for an acute intake assigned by NIOSH, it is not clear whether the SRS IDR included chronic intakes as well as acute intakes. Based on the intake dates provided in ORAUT-OTIB-0001, Tables 1 and 2, the intakes appear to be the result of acute intakes only. There is no specific consideration given to chronic intakes and the relative dose consequences as compared to those determined with the hypothetical intake doses.

SC&A requested the bioassay data for each of the individuals included in the high-five intakes in order to provide a more detailed review of the high-five methodology. NIOSH indicated to SC&A that they do not have access to these data, and that it would have to be requested from the Savannah River Site. SRS was not able to provide this information in time for its consideration in this review. As a result, there is insufficient data available to reproduce the relative intakes used for the Hypothetical Intake, including an absence of bioassay data. In the absence of data, this review was also unable to determine whether the method adhered to the hierarchical process as defined in 42 CFR 82.2

5.1.5 Consistency Among Site Profiles

The approach to calculating maximum dose estimates for the SRS is different from the approach recommended in ORAUT-OTIB-0002, *Maximum Hypothetical Intake*. This approach is based on an assignment of 10% of the Maximum Permissible Body Burden (MPBB) for radionuclides associated with reactor and nonreactor facilities. The methodology is applicable from 1969 forward only. The assumption for Type F and M materials was 1 and 2 MPBBs, respectively. Derived intakes for U-234 and U-238 were multiplied by 100 to account for work in uranium facilities. A claimant-favorable solubility class was used for each radionuclide of concern to maximize organ dose. For certain radionuclides, such as uranium, the maximum plausible intakes based on a fraction of the MPBB are 5,000 nCi of U-234 and 500 nCi of U-238, which

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are much higher than the values of 105.4 nCi U-234 and 20.95 nCi U-238 recommended in ORAUT-OTIB-0001. Radionuclides included in ORAUT-OTIB-0002 that are not included in the high-five approach include Mn-54, Co-58, Fe-59, Tc-99, Y-91, Ru-103, I-131, Ce-141, Pm-147, Eu-154, Eu-155, and Th-230. Furthermore, there are a number of radionuclides not included in the hypothetical intake that have been characterized at SRS, including Se-79, Te-127m, I-129, Cs-134, Pr-144, Sm-151, Eu-152, Pu-240, Cm-243, Am-243, U-235, Th-228, Th-232, Sb-125, Ba-140, La-140, Sr-89, U-236, Ag-110, Sn-123, Te-127, Te-129, and Pu-242 (WSRC 2001; WSRC 1994; DOE 1990).

The Hypothetical Intake methodology used in the SRS TBD is also inconsistent with that described in the Hanford TBD. The basis for assignment of missed dose in the Hanford TBD is year and monitoring data-specific. For those with external monitoring and no internal monitoring, internal dose is calculated from air concentration data at 10% of the respiratory protection-required value or at a fraction of limiting air concentrations for a select exposure period (i.e., 40 hours per week for early years, 4 hours per week from 1953-present). For monitored workers, the maximum intake is determined using the minimum detectable activity (MDA) of the appropriate bioassay technique as the value for the last sample. In the case of the Hanford maximizing approach, the beta, alpha, and photon dose components from the various radionuclides were included in IREP. In SRS's case, only beta and alpha components were included.

In summary, the use of ORAUT-OTIB-0001 as a reference document for dose reconstruction is considered inappropriate. Intakes should be recalculated using the 42 CFR 82-recommended methodologies based on bioassay data rather than an intermediate product of the SRS Internal Dosimetry group, which is derived from early ICRP models.

5.2 Issue 2: Occupational Environmental Doses

The methods used to reconstruct ambient environmental doses to unmonitored workers from airborne emissions employ an atmospheric dispersion model that may be inappropriate and not claimant favorable for certain conditions. Furthermore, comparison with other site profiles indicates that there is not a consistent methodology for DOE facilities for determining ambient environmental dose.

5.2.1 Technical Accuracy

The fundamental approach employed in Chapter 3 of the site profile for deriving occupational environmental doses uses (1) a sector-averaged gaussian plume model, (2) source terms and radionuclide list originally estimated for offsite dose estimation, and (3) a resuspension factor of 10^{-9} per meter for estimating air dust loading due to radionuclides in the soil.

Chapter 3 of the site profile makes extensive use of *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) for deriving occupational environmental doses. In so doing, NIOSH has adopted the sector average gaussian plume model

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(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 Risk Assessment Corporation (RAC) in support of the Center for Disease Control (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, it is questionable whether this approach is appropriate for reconstructing onsite doses to workers for a number of reasons, which are discussed in the following paragraphs.

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, there are certain fundamental issues associated with this approach that 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). 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). Using this approach, the atmospheric dispersion factors could be more than an order of magnitude greater than those derived using average annual meteorological conditions.

Although the TBD acknowledges that the estimated annual intakes are based on average annual atmospheric dispersion factors, and that the actual instantaneous meteorology could vary from these averages by several orders of magnitude on any given day, this issue is dismissed as unimportant in the following statement on page 60 of the TBD.

However, the intake values given in Attachment C should represent a reasonable upper bound of the actual intakes that could have occurred.

If large short-term releases occurred during stable conditions, such as during low wind speeds 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. This issue is acknowledged and explicitly addressed in the Hanford TBD, where the RATCHET atmospheric dispersion code was used, instead of the standard Gaussian model employed in the SRS site profile. It is suggested that the TBD revisit this issue and confirm that

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doses from episodic releases were not, in fact, significantly underestimated because of the use of conventional Gaussian models.

The following provides examples of outdoor exposure scenarios that could not be properly reconstructed using the Gaussian model employed in Chapter 3 of the TBD. Further, the source term developed by RAC for offsite dose calculations would not capture air contamination due to these kinds of onsite activities:

- Open pan burning of Pu and other TRU contaminated solvents until 1970. Resuspension and evaporation of contaminated liquids from seepage basins.
- External doses from spills, and hot spots and internal resuspension doses, such as from the Tank Farms.

We are also concerned with the default resuspension factor employed to derive the doses to workers from the inhalation of resuspended radionuclide contaminants that were deposited on the ground from airborne emissions. In Chapter 3, NIOSH employed a resuspension factor of 10^{-9} per meter.

The methods available for deriving inhalation exposure from resuspended radionuclides include the dust loading approach and the resuspension-factor approach. The dust-loading approach is used for those scenarios where information is available on the radionuclide concentration in surface soil dust (e.g., 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 $1/\text{m}$) 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.

Dust-Loading Approach

A review of this subject is provided in a report prepared for the NRC by Battelle Pacific Northwest Labs (Sutter 1982). Table 5.1 summarizes some of the relevant information contained in that report.

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Table 5.1 Some Airborne Particulate Mass Concentrations
(reproduced from Table 2.1-2 of Sutter 1982)

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, p62
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 (stead 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

Yu et al. (1993) also presents a review of the literature on dust loadings. Table 5.2 summarizes those studies.

Table 5.2 Summary of Dust Loading Studies Cites by Yu et al. (1993) (g/m³)

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/m³ outdoors, with peak dust loadings of up to 80 mg/m³ (Rando et al. 2001; Heederik et al. 1994). For workers in the concrete industry (blasting,

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drilling, grinding), the 8-hour time-weighted average dust loading of respirable particles was measured at between 0.26 to 14 mg/m³ (Linch 2002).

In light of this review, NIOSH should evaluate the dust-loading approach, using an average work year dust loading on the order of perhaps 1 mg/m³. When using this approach, the average soil concentration should be determined over a large work area on the order of several acres.

Resuspension-Factor Approach

The resuspension of radioactive material from surfaces can be modeled by the use of an equilibrium resuspension factor (in units of length⁻¹). The resuspension factor (RF) is simply a ratio of the air concentration of radioactive material above a surface (pCi/m³) to the concentration on the surface (pCi/m²).

Measured RFs vary over very wide ranges. Kennedy and Streng (1992) reported RFs from approximately 1E-11 to 1E-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, an RF 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.

Beyeler (1999) presents a discussion of the factors that affect the RF. The experimental data and recommendations summarized in Table 5.3 are felt to be the most appropriate available information for indoor resuspension factors. Their 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.

The range of resuspension factors cited in Table 5.3 is 2 E-8 m⁻¹ to 4 E-3 m⁻¹. The reported data are generally from experiments that examined 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.

Table 5.3 Representative Reported Indoor Resuspension Data and Recommended Values

Reference	Resuspension factor or range	Comments
Barnes (1959)	4E-5 m ⁻¹ (confined space) 2E-6 m ⁻¹ (open air)	Reported for “dusty operations”; 10 ⁻⁵ m ⁻¹ recommended for most laboratory work.
Stewart (1964)	1E-6 m ⁻¹ (quiescent conditions) 1E-5–1E-4 m ⁻¹ (“operational” conditions)	Notes that excessively high particulate resuspension values indoors are likely to indicate some degree of inefficiency in the ventilation system.

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Table 5.3 Representative Reported Indoor Resuspension Data and Recommended Values (continued)

Reference	Resuspension factor or range	Comments
Brunskill (1964)	2E-4–4E-3 m ⁻¹	Measured in small rooms with various types of personnel movement, including introduction of loose contamination on coveralls. Lower recommended values were measured for a large area of “loose” contamination on concrete; “much smaller” values were found for linoleum floor.
Jones and Pond (1964)	2E-8–5E-5 m ⁻¹ 5E-5 m ⁻¹ (recommended for worst practical conditions)	Estimated that 10%–20% of total airborne radioactivity was respirable. Suggested that recommended value could be an order of magnitude lower for average conditions.
Dunster (1964)	2E-6–4E-5 m ⁻¹ 2E-6 m ⁻¹ (recommended safe value for long-term use)	Highest values from digging through dusty building rubble and in an enclosed and unventilated space.
Spangler and Willis (1964)	4E-5 m ⁻¹ (derived)	This value is calculated using equation for equilibrium airborne concentration in a small room from a surface concentration and recommended values appropriate for calculating 40-hr maximum permissible concentration (MPC) levels.
Healy (1971)	2.1 E-7–1.0 E-3 m ⁻¹ (derived)	This value is calculated using the equation for airborne concentration, assuming ventilation rate for a reasonably tight 28 m ² ×2.4 m room.
Gibson and Wrixom (1979)	2E-6–4E-5 m ⁻¹	The lower value was used in original calculation of derived working limits (DWL) for active area surfaces and might be inappropriate for widespread contamination on dusty surfaces. The higher value was obtained from measurements in a confined space and is suggested for general use.
IAEA (1970)	2E-6–3E-3 m ⁻¹ 5E-5 m ⁻¹ (recommended)	Recommended value is suggested as appropriate for general conditions of contamination on surfaces. Because of confounding factors, this effectively reduces the recommended value by a factor of 2.5 for use in calculating DWL values.
Kennedy et al. (1981)	2.5E-5 m ⁻¹ (derived)	This value is calculated using the equation for airborne concentration, assuming ventilation rate of an open transport truck and resuspension rate for a 28 m ² room.
Kennedy and Strenge (1992)	1E-6 m ⁻¹ (recommended)	Based on a review of resuspension literature. Recommended as a reasonably conservative default value to be applied to total surface concentration.
IAEA (1992)	1E-6 m ⁻¹ (recommended)	This value is recommended for use in assessing the doses associated with the handling of tools and equipment, and employs a transfer factor of 0.01 to account for the fraction of the residual surface radioactivity that is available for resuspension.
Chen (1993)	1 E-6 m ⁻¹	No justification given (based on use in Kennedy and Strenge 1992)
Draft NUREG-1720 (2002)	Lognormal distribution with mean of 3.7 E-7 m ⁻¹ and 90 th percentile of 9.6 E-7 m ⁻¹	NRC staff analyzed literature and recent field data considering realistic assumptions about decommissioned facilities and building occupancy for the D and D code. RF values best represent cleaned and aged surfaces.

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Several reports by Sehmel (1977 and 1980) revealed that there is enormous uncertainty in outdoor RFs, as there is for indoor RFs. In an investigation of RFs at the Hanford reservation, Sehmel (1977) found outdoor RFs ranging from 10^{-11} to 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 man's activities. He explains that there are many reasons for this variability, many of which have to do with 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.

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

As a final point, equations 3-2 and 3-3 on pages 52 and 53 of the TBD present the equations used by NIOSH to derive the atmospheric dispersion factors (i.e., X/Q values expressed in units of sec/m^3) 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 down wind from the release point is derived as follows:

$$Y = 1.0146X - 1.8808$$

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

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

$$Y = 1.0146(1000) - 1.8808 = 1013$$

Since X/Q values are typically a small fraction of 1 (e.g., on the order of 0.001), it appears that there is a typographical error in the equation. 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.

5.2.2 Completeness of Data

NIOSH/ORAU made a concerted effort to obtain relevant reports, technical documents, and other data relating to the SRS. This is especially evident with reports relating to environmental levels of airborne radionuclides. However, 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 the RAC (CDC 2001). The atmospheric source terms reported in these reports appear to be limited to monitored releases and are reported in terms of total annual releases by year for the purpose of deriving historical offsite doses. We are concerned that this strategy may not be entirely applicable to reconstructing the doses to onsite workers for a number of reasons.

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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 apply to exposures occurring close to the source term, especially to ground-level and/or episodic releases. For example, Chapter 4 of the CDC (2001) report makes reference 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/ORAU has not made comprehensive use of information available relating to environmental releases at the SRS. The SRS published a series of reports discussing radionuclides in the SRS environment. Included in these reports is a summary of releases from SRS facilities, including atmospheric and liquid releases, transport mechanisms, and concentration on and in the vicinity of SRS. These reports are a compilation of information from monitoring reports, and they summarize important references. These environmental reports include information on releases of activation products, americium, cesium, curium, fission products, neptunium, noble gases, plutonium, radiocarbon, radioiodine, strontium, technetium, tritium, and uranium. A number of radionuclides that are known to have been released from SRS facilities were not mentioned in the assessment of environmental dose. 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, Tc, Th-232, Xe-131m, Xe-133, Xe-135, Y-91, Zn-65, and Zr-95 (Carlton et al. 1995; Jannik 1997; WSRC 1996a and 1997b; Carlton et al. 1992a, 1996, 1993a, 1992b, and 1993b; Kantelo et al. 1993). Many of these radionuclides are mentioned in Attachment A of the SRS TBD; however, they are not included in environmental dose reconstruction. The methodology used to determine which radionuclides and source pathways are important to onsite dose assessment is not discussed in the TBD. 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. Also, since SRS has communicated to the public through technical reports that these radionuclides were released to the environment, it is prudent to acknowledge this in the TBD, even if dose is not assigned.

5.2.3 Consistency Among Site Profiles

The methodology used to determine environmental dose is not consistent between the SRS and Hanford TBDs. Atmospheric dispersion in the SRS TBD was modeled by developing atmospheric dispersion (X/Q) values. These data were converted to X/Q tables for elevated and ground releases. The RATCHET model (a Puff advection model) and an Excel spreadsheet were used to calculate intakes from airborne radionuclides in the Hanford TBD. Although it is not clearly defined in the SRS TBD, the environmental dose appears to include estimates from inhalation of radionuclides in air, direct external exposure to plumes, and exposure from

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resuspension of soil. In the case of Hanford, environmental dose includes inhalation of radionuclides from the air, direct external exposure from plumes, and physical contact with particulate radionuclides on skin. The Hanford TBD included evaluation of episodic releases, such as the release of ruthenium particles from the Reduction-Oxidation (REDOX) facility. It is not evident from the SRS TBD whether episodic releases were included in the calculations. Also, in calculating the submersion dose, the SRS TBD used Federal Guidance Report 12, *External Exposure to Radionuclides in Air, Water, and Soil* (EPA 1993), whereas the Hanford TBD used Federal Guidance Report 13, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides* (EPA 1999). NIOSH should utilize a consistent model for the calculation of environmental dose. The components of environmental dose should be clearly defined and each component discussed in the TBD.

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

5.3 Issue 3: Recycled Uranium

The site profile does not address most issues associated with processing recycled uranium (RU) at SRS. 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. Moreover, because of the extensive use of RU, estimated at about 250,000 metric tons, exposure issues and concerns regarding RU should be addressed on a DOE-wide basis.

To further bolster the assertion that recycled uranium posed unimportant risks, the 2000 report states: "These protection measures, notwithstanding, records indicate that 99 workers received internal doses of uranium over the history of the plant, which are well documented in site incident reports."³ 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 indicate that there were 205 individuals with positive bioassays for uranium between 1953 and 1960 alone.⁴

³ DOE, ESH-PEQ-2000-00059, p.2

⁴ DPSP, 56-1-7,p.506, DPSP 56-1-8, p.511, DPSP 56-1-9 p. 513, DPSP-56-1-10, p.517,DPSP-56-1-11,p. 417,DPSP-56-1-12, DPSP-57-1-1,p. 410, DPSP-57-1-2,p. 411,DPSP-57-1-4,p. 416, DPSP-57-1-5,p. 418, DPSP-57-1-6,p. 422, DPSP-57-1-7, p. 419,DPSP-57-1-8, p. 421, DPSP-57-1-9, p.416, DPSP-57-1-10, p. 420,DPSP-57-1-11, p. 421, DPSP-57-1-12, p. 418, DPSP-58-1-6, p. 420,DPSP-58-1-7, p. 416, DPSP-58-1-8, p. 416, DPSP-58-1-8, p.416, DPSP-58-1-9, p. 418, DPSP-58-1-10, p. 420, DPSP-58-1-12, p. 417, DPSP-59-1-1, p. 417, DPSP-59-1-2, p. 423, DPSP-59-1-3, p. 423, DPSP-59-1-4, p. 424, DPSP-59-1-5, p. 421, DPSP-59-1-6, p. 418, DPSP-59-1-7, p. 422, DPSP-59-1-10, p. 422, DPSP-59-1-11, p. 419, DPSP-59-1-12, P. 423, DPSP-60-1-1, p. 421,DPSP-60-1-3, p. 426,

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5.3.1 Technical Accuracy

The SRS processed significant quantities of recycled uranium, both from its own reprocessing plants and via other plants (DOE 1985; ERDA 1976; DuPont 1960; McCarty 2000; DOE 2001). It is estimated that from 1959 to 1999, some 31,355 metric tons of uranium were shipped from SRS to other 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 2001). During this same time period, it is estimated that SRS received 54,544 metric tons of uranium from other sites, such as FMPC, the DOE's gaseous diffusion plants, and the Y-12 Plant (McCarty 2001).

From 1961 to 1999, SRS processed approximately one-third of an estimated total of 250,000 metric tons recycled uranium in the DOE complex. SRS processed uranium metals, oxides, and solutions of various assays, including depleted uranium, natural uranium, low-enriched and highly enriched uranium. Enriched uranium was also extracted from domestic and foreign research reactor spent fuel. Also, from 1964 to 1969, thorium was recycled to produce U-233 (McCarty 2001). During the peak period of the Cold War, SRS generated 2,000 to 3,000 drums of RU trioxide a year. During this same period of production and processing of RU, approximately 300 workers were handling these materials annually at SRS (McCarty 2000).

Recycled uranium is so called because it 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 never before irradiated 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). The discussion of radionuclides in RU is limited to the glossary and includes only uranium isotopes (Scalsky 2004, p. 134).

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 recycled uranium. 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* states (DOE 1985):

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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 2004). However, this table does not equate with radionuclides of concern recommended by a special task force on RU convened in 1985 by the DOE, which 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, a large part of the reason that the uranium enrichment plants at Oak Ridge, Portsmouth, and Paducah were granted Special Exposure Cohort status in EEOICPA was due to the presence of transuranic trace contamination. Such trace contamination has been shown (in the following excerpted table) to have the potential of significant radiation doses, if the concentrations are high enough (DOE 2000, p. 77).

*ESTIMATED BONE SURFACE DOSES FROM RECYCLED URANIUM TO WORKERS
AT THE PADUCAH GASEOUS DIFFUSION PLANT
(Committed Effective Dose Equivalent – CDE)*

<i>Average Air Concentrations</i>	<i>Maximum Air Concentrations</i>
<i>48.06 -- 188 rems</i>	<i>599.24 -- 2,238 rems</i>

The 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....

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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.

Cruse and LaBone (2000) evaluated the dose fractions from impurities, such as plutonium in RU, for that processed and handled in SRS facilities. The source term was derived from a detailed assessment of the radionuclide mix in the 221H 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 CEDE. The analysis utilized maximizing internal dose assumptions. Dose contributions from impurities in RU was summarized by Cruse and LaBone (2000) 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 conservatism 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. Cruse 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 Cruse and LaBone (2000), McCarty 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 is sparse at best (McCarty 2000).

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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 sparse at best.

A preliminary review of historical records indicates that concentrations in uranium are tenuous at best. A DOE task force on recycled uranium reported in 1985 that, 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 proper review and concurrence that would have been required under formalized specifications. For instance, in 1976, the maximum alpha activity specification from all transuranic elements of 1,500 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 [to Y-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 percent). 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 UO₃ from the denitrator may have had the greatest contact with respirable RU particles.
- Building 321-M, where casting and machining of recycled uranium was performed. In addition, building exhaust HEPA filter change-out activities may have also created high potential for high airborne concentrations of RU.

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In summary, transuranic and fission product contaminants in RU need to be specifically evaluated for their significance, since there was no bioassay monitoring for them in uranium areas. In view of this significant uncertainty, a thorough 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.

5.3.2 Consistency Among Site Profiles

The Hanford TBD includes a discussion on RU and the impurities associated with that at Hanford (Bihl 2004, p. 24). The exclusion of the issue from the SRS TBD produces an inconsistency between sites. A consistent criterion for inclusion of impurities in organ dose should be developed and applied for DOE and AWE facilities. A formalized complex-wide policy for impurities in RU was not in affect until the later years of processing. As a result, careful consideration should be given to limits established by individual sites, and their adherence to these limits during receiving and shipping of RU.

In summary, the TBD does not address the activity fractions or the dose contribution from all pertinent impurities in recycled uranium. This dose should be included in the dose assessment for workers accessing uranium recovery facilities or handling RU. An analysis of workplace air concentrations where recycled uranium was being handled, as was done at the Paducah site (DOE 1999), may be helpful to reconstruct doses. Also, organ doses relative to impurities should be investigated further to ensure the claimant receives the benefit of the doubt with respect to organ doses.

5.4 Issue 4: External Beta/Gamma Dose Adjustments and Uncertainty Factors

The *Executive Summary* in ORAUT-TKBS-0003 informs the dose reconstructor that:

*Technical Basis Documents and Site Profile Documents are general **working** documents that provide guidance concerning the preparation of dose reconstructions at particular sites or categories of sites. [Emphasis added.]*

ORAUT-TKBS-0003, Section 1.2 *Scope* further states that:

This document also presents the technical basis of methods used to prepare the SRS worker dose records for input to the NIOSH Interactive RadioEpidemiological Program (IREP) and the Integrated Modules for Bioassay Analysis (IMBA) computer codes used to evaluate worker dose. Because information on measurement uncertainties is an integral component of the NIOSH approach, this document describes how the uncertainty for SRS exposure and dose records is evaluated.

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*The main body of text in this document provides the description of the facilities and processes, historical information related to worker internal and external exposures, and environmental data for use when actual monitoring data are unavailable. **The attachments represent the critical data and tables required by the dose reconstructors for performing the individual claimant dose reconstructions. These attachments should suffice as a stand-alone document for dose reconstruction. Additional details, if necessary, could be found in the main body of the text.** [Emphasis added.]*

The intent of Section 5.0 of the TBD is to provide sufficient historical and technical data that would serve to identify limitations/uncertainties of past dosimetry practices and dosimeter designs used to assess external exposures to photons and neutrons, and explain the technical basis for the need to:

- Amend select dosimeter recorded external photon doses
- Substitute select dosimeter recorded neutron doses
- Quantify the lower limit of detection for specific dosimeters
- Account for missed dose

The methodology used by the TBD to assign beta and photon external exposure does not account for all uncertainty associated with dosimeter measurements. The following are the categories under which the TBD needs to be more specific and complete:

- Calibration of dosimeters at 0E are often not representative of incident angles encountered in the field and 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 keV.
- The TBDs 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%.
- There is a lack of guidance pertaining to interpretation of shallow dose.
- Dosimeter adjustment factors for SRS are inconsistent with DOE complex-wide technical information bulletins.

5.4.1 Dosimeter Calibrations

Dosimeters used to monitor personnel for external radiation for various time periods between 1952 and the present are identified and described in Sections 5.3.1 through 5.3.5 of the TBD. This discussion includes Table 5.3.1-1, which summarizes user dates, dosimeter exchange frequencies, laboratory MDLs, and maximum annual missed dose based on $n(\text{LOD})$.

Also presented are data in behalf of two independent studies that assessed performance characteristics of early film dosimeters as well as TLDs. The first study, conducted by the International Agency for Research on Cancer (IARC), evaluated the on-phantom dosimeter response to various photon energies and exposure geometries. For comparison, all recorded dosimeter responses were standardized to the true $H_p(10)$ dose. A summary of these data is reproduced herein as Table 5.4.

Table 5.4 IARC Testing Results of United States Beta/Photon Dosimeters

Geometry	Phantom	118 keV		208 keV		662 keV	
		Mean ^a	SD/Mean	Mean ^a	SD/Mean	Mean ^a	SD/Mean
US-2 (Two-element film dosimeter)							
A-P	Slab	3.0	2.1	1.3	1	1.0	0.8
A-P	Anthropomorphic	3.0	4.2	1.2	1.9	1.0	1.8
Rotational	Anthropomorphic	2.2	2	1.4	3	1.2	3.2
Isotropic	Anthropomorphic	1.5	4.4	1.1	1.6	1.0	2.7
US-8 (Multi-element film dosimeter)							
A-P	Slab	1.0	1.5	1.0	0.8	0.8	1.7
A-P	Anthropomorphic	0.8	9.5	0.9	6	0.8	1.8
Rotational	Anthropomorphic	1.2	1.9	1.2	17	1.1	1.8
Isotropic	Anthropomorphic	1.0	3	1.2	9	1.0	2.3
US-22 (Multi-element thermoluminescent dosimeter)							
A-P	Slab	0.9	4.4	0.9	3.9	0.9	3.5
A-P	Anthropomorphic	0.8	3.1	0.9	2.1	0.9	3.9
Rotational	Anthropomorphic	1.1	3.1	1.2	1.5	1.0	4.1
Isotropic	Anthropomorphic	0.9	0.3	1.0	2.5	0.9	1.6

^a Ratio of recorded dose to $H_p(10)$

Data from a second study cited in the TBD are those of a 1990 study by Wilson et al. (1990), which also assessed film dosimeters and TLDs to five different photon energies under Anterior-Posterior (AP) and rotational exposure geometries. As reproduced herein as Table 5.5, the dosimeter responses in this study were also standardized to the $H_p(10)$ doses, but no data are presented that quantify the uncertainty of individual dosimeter responses within a given group.

Table 5.5 Testing Results for Hanford Two-Element and Multi-Element Film Dosimeters for Energy and Angular Response

Beam (energy, keV)	Anterior-posterior (AP) exposure			Rotational exposure		
	Film dosimeters		TLD, 1972- present	Film dosimeters		TLD, 1972-93
	Two-element 1944-56	Multi-element 1957-71		Two-element 1944-56	Multi-element 1957-71	
16 ^b	0.1	0.98				
59 ^b	0.5	1.1				
M150(70)	0.7	0.70	0.95	1.31	1.31	1.77
H150(120)	1.6	0.64	0.87	3.00	1.20	1.64
¹³⁷ Cs(662)	1.0	1.0	1.0	1.46	1.46	1.46

^a Divide recorded dose by table value to estimate $H_p(10)$.

^b Based on Wilson et al. (1990).

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On the basis of these data, the TBD concluded the following:

- In the IARC study, the two-element dosimeter design significantly overestimated $H_P(10)$ for all irradiation geometries and increasingly for the lower energy photon energy at 118 keV.
- Study data presented by Wilson et al. (1990) are similar to those of the IARC study.
- In Section 5.3.4.1, the TBD states:

*Based on the collective information, SRS dosimeters are expected to reasonably measure the $H_P(10)$ dose under **all SRS workplace radiation fields**. . . . and laboratory irradiations of the two-element dosimeter have shown an **over-response** of the actual $H_P(10)$ dose by about a factor of 2 to photons greater than 100 keV. A **claimant-favorable** approach is proposed to **ignore** this over-response because of the complexity of workplace photon energies and exposure geometries that tend to result in an **under-estimate** of the $H_P(10)$ dose. . . . As such, a reasonable estimate of deep dose, compared to $H_P(10)$, is expected for SRS beta/photon workplace radiation. [Emphasis added.]*

In summary, the ORAU Team concluded that the dosimeters' **over-response** at low photon energies was offset by **under-responses** caused by calibration methods, angular response, environmental factors, etc., and that the recorded dose for all types of dosimeters employed was, in fact, a reasonable estimate of the 1,000 mg/cm² deep dose with only the following two minor corrections, as explained in Section 5.3.3.1:

SRS dosimeters were originally calibrated using primarily uranium and ²²⁶Ra using in-air (i.e., no phantom) exposures to selected levels. K-fluorescent x-rays were used to develop dosimeter response characteristics for the lower energy photon fields in plutonium facilities. This practice is similar to other AEC sites. Taylor et al. (1995) describes adjustments to SRS recorded dose to estimate $H_P(10)$ based on SRS preparations for DOELAP performance testing in the mid-1980s. At that time, it was concluded that:

- *Prior to 1 January 1986 the recorded dose of record (i.e., photon) dose should be multiplied by a factor of 1.119 (11.9%).*
- *Prior to 1 January 1987 [i.e., for the year 1986] recorded dose of record (i.e., photon) should be multiplied by a factor of 1.039 (3.9%).*

*These changes in the recorded dose should be made to arrive at an assured **claimant-favorable** treatment. Common sources of laboratory **bias** are shown in Table 5.3.3.1-1 for personnel beta/photon dosimeter calibration based on comparison of the recorded dose with $H_P(10)$. [Emphasis added.]*

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As referenced above, in order to account for dosimeter **uncertainty**, the dose reconstructor is directed to Table 5.3.3.1-1, which cites the following values:

<u>Dosimeter Type</u>	<u>Years When Used</u>	<u>Uncertainty</u>
Two-element film	1957 – 1959	±25%
Multi-element film	1960 – 1969	±20%
TLD (SRS-TLND and Current Panasonic TLD)	1957 – present	±10%

A generic uncertainty value applicable to all dosimeters is also provided in Section 5.7.2, which identifies a standard deviation of 30%.

Performance characteristics of personnel dosimeters (that include two-element and multi-element film dosimeters and three different TLDs) varied significantly, as shown in Tables 5.4 and 5.5. Table 5.4, however, suggests that the over-response of dosimeters is largely confined to the two-element film dosimeter, and moreover was limited to photon energies below 200 keV. The two-element film dosimeter's use was restricted to the time period between March 3, 1952 and November 8, 1959. Inspection of Table 5.4 shows that the multi-element film dosimeter and subsequent TLDs generally yielded responses that were equivalent to the $H_p(10)$, and in some cases significantly **lower** (e.g., the multi-element film and TLDs yielded values of 0.8 and 0.9 for all energies under AP geometry).

Although NIOSH/ORAU acknowledged the fact that multiple factors may contribute to an under-response of a film dosimeter or TLD, it was concluded that the resultant **under-response** is offset by the energy-dependent **over-response**. (SC&A, however, pointed out the fact that the over-response was principally limited to the two-element film and only at low-photon energies.) NIOSH/ORAU, therefore, erroneously concluded that no adjustments (other than for on-phantom calibration) needed to be made to recorded dosimeter doses.

While SC&A endorses the need for simplifying dose reconstruction whenever possible, such simplification must, however, favor the claimant. There is reason to believe that dosimeter doses after 1959 may have been underestimated by as much as 40%, as explained below.

In a 1994 study, Fix et al. (1994) assessed the angular response of dosimeters for various discrete angles of exposures and subsequently “evenly weighted” these responses to simulate a rotational exposure geometry. These data are reproduced below in Tables 5.6 and 5.7. These empirical data suggest a potential average under-response of 25% to 40% among the three types of dosimeters employed at SRS.

Table 5.6 Measured Angular Response of Hanford Dosimeters to Three Radiation Sources Using an Anthropomorphic Phantom Expressed as a Bias Relative to the Normal Position (i.e., 0°)

(Source: Fix et al. 1994)

Angle of Exposure	Hanford Multi-Element Film Dosimeter			Hanford Multi-Element Thermoluminescent Dosimeter		
	M150	H150	¹³⁷ Cs	M150	H150	¹³⁷ Cs
180°	0.27	0.42	0.35	0.11	0.16	0.36
-135°	0.27	0.55	0.23	0.11	0.14	0.33
-90°	(a)	(a)	0.88	0.87	0.95	1.01
-45°	1.01	0.88	0.96	0.98	1.00	0.98
0°	1.00	1.00	1.00	1.00	1.00	1.00
+45°	0.94	0.67	0.98	0.97	0.98	1.01
+90°	(a)	(a)	0.98	0.66	0.66	0.98
+135°	0.30	0.36	0.45	0.12	0.15	0.37

(a) For these angles (=90°), the film response would have been noted as an obviously abnormal result because the metallic filter image would not have been observed on the film.

Table 5.7 Estimate Bias Resulting from On-Phantom Angular Response of Hanford Dosimeters for Evenly Weighted Contribution from Angles Presented in Table 3^(a)

(Source: Fix et al. 1994)

Source	Hanford Two-Element Film Dosimeter	Hanford Multi-Element Film Dosimeter	Hanford Thermoluminescent Dosimeter
M150 (70 keV)	0.64 ^(b)	0.64 ^(c)	0.60
H150 (120 keV)	0.71 ^(b)	0.65 ^(c)	0.63
¹³⁷ Cs (662 keV)	0.68 ^(b)	0.73	0.76

(a) Divide recorded dose by table value to estimate deep dose.

(b) Values estimated from 0° and 180° irradiations of multi-element film dosimeters.

(c) Data for 90° exposure angles were not used.

Empirical measurements of angular sensitivity of film to photons have also been reported by Hine and Brownell (1956) and are reproduced below in Table 5.8.

Table 5.8 Relative Film Badge Sensitivity in Free Air for Gamma-Rays Incident at Various Angles

(Source: Hine and Brownell 1956)

Angle of Incidence	0.11 MeV	0.20 MeV	1.2 MeV
0E (perpendicular incidence)	1.00	1.00	1.00
22.5E	0.87	0.92	0.97
45E	0.46	0.73	0.91
67.5E	0.33	0.45	0.92
90E	0.16	0.41	0.94

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In summary, the above-cited data consistently identify a significant **under-response** based on the angle of incidence in behalf of the SRS multi-element film and TLDs, which did **not** suffer from the photon-energy dependent **over-response**.

Thus, the TBD's stated conclusion that: "... A claimant-favorable approach is proposed to ignore this **over-response** because of the complexity of workplace photon energies and exposure geometries that tend to result in an **underestimate** of the $H_p(10)$..." is factually incorrect and clearly not claimant favorable for recorded dosimeter doses post-1959, which employed multi-element film dosimeters and TLDs.

A serious limitation associated with personnel dosimeters (film and TLD) involves the under-response caused by the dosimeter's angular sensitivity. When personnel dosimeters are calibrated, the incident radiation is normal (i.e., 0°) to the plane of the dosimeter and yields a dose response (i.e., calibration factor) that is optimal. At incident angles that deviated from 0° the response of the dosimeter is greatly diminished, which leads to an **underestimate** of the true exposure that is being measured.

5.4.2 Dosimeter Correction Factors

In the mid-1980s, SRS implemented changes in 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. This correction factor was based on a study conducted by Taylor (1995).

Based on photon energies that characterize Ra-226 (and daughters) and Cs-137, it is uncertain whether the assumed correction factor is appropriate for photon energies of ambient radiation fields that characterize SRS as explained below.

Radiological Uncertainty: Backscatter. Backscatter may significantly influence the dose-response of a dosimeter and reflects the calibration protocol. Fix et al. (1994) states that "... In 1984, the dosimeter calibration procedure was changed to "on-phantom" as opposed to "in-air" to better simulate the dose to workers."

This implies that prior to 1986, dosimeters were calibrated in free air, and after 1986, calibration of personnel dosimeters was performed on-phantom. For these two calibration conditions, differences in recorded dose are profoundly affected by the photon energy.

For illustration, suppose that a dosimeter is placed at the point P on the surface of the phantom and the amount of radiation is measured in a given length of time (see Figure 5.1). Then suppose that the phantom is removed, leaving the dosimeter P at exactly the same point in space, and the exposure is run for an equal length of time. It will be found that the dose recorded by the dosimeter at P will be considerably less in the second case. This is because part of the radiation observed in the first case is radiation that is scattered back from the phantom to the point P . The backscatter factor is defined as follows:

$$\text{Backscatter factor} = \left(\frac{D_s}{D_a} \right)$$

and, the percentage backscatter as:

$$\text{Percentage backscatter} = \left(\frac{D_s - D_a}{D_a} \right)$$

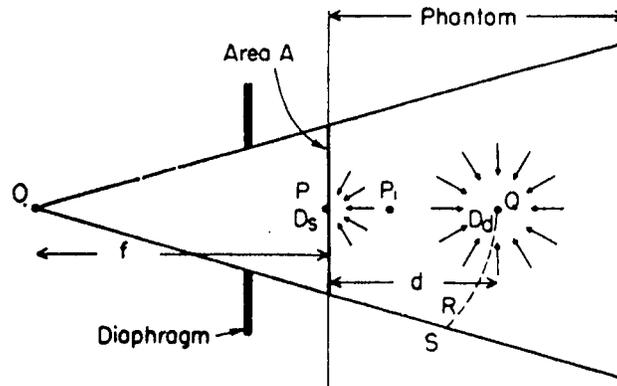


Figure 5.1 Diagram to Illustrate the Meaning of Surface Backscatter and Percentage Depth Dose

Here, D_a stands for the dose measured by the dosimeter in air, and D_s is the corresponding dose with the scattering material (i.e., phantom) in place.

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. Backscatter factors related to radiation quality and field size are summarized in Figure 5.2. The data indicate that for photons with HVL between 0.6 mm Cu and 1.0 mm Cu (or ~60 keV-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 DCFs with photon energies between 30 keV and 250 keV, which is commonly assumed for SRS workers.

From data presented in Figure 5.2, it is likely that the cited on-phantom correction factor of 1.119 may be too low for photon energies between 30 and 250 keV.

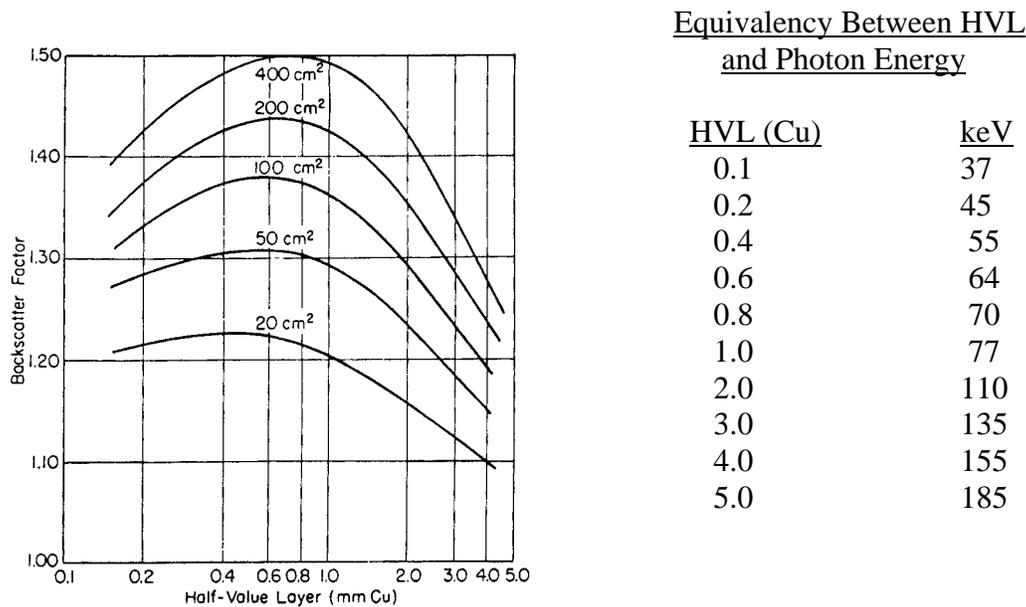


Figure 5.2 Variation of the Backscatter Factor with the Quality of the Radiation for a Number of Field Sizes
(Source: Hine and Brownell 1956)

5.4.3 Dosimeter Uncertainty

For recorded dose, NIOSH/ORAU stated that “. . . measured doses are treated as a normal distribution with a standard deviation of 30% . . .”

A standard deviation of about 30% appears to be a reasonable value that is, however, limited to what is commonly attributed to “laboratory uncertainty.” For film dosimeters, laboratory uncertainty includes all the uncertainties introduced in calibration protocols, chemical processing of films, reading their optical densities, comparing these densities with the densities of unexposed and calibration films, and in interpreting the measured densities in terms of exposure.

Thus, under highly **controlled laboratory** radiation exposure conditions of dosimeters, film development and processing uncertainties of 20% to 30% are commonly noted (NRC 1989). However, under field exposure conditions, two other sources of uncertainty may significantly increase the total uncertainty. These include **radiological uncertainties** and **environmental uncertainties**.

Radiological uncertainties are contributed by variation in the photon energy spectrum (that may differ from the calibration source), the body-wearing position of dosimeter (i.e. angular sensitivity), and radiation backscatter (as discussed above). **Environmental** uncertainties relate to the field conditions to which a film dosimeter/TLD may be exposed during the wear-period. Environmental factors affecting dosimeters include moisture, light, high temperatures, chemical

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exposures, pressure, etc. The relative contributions of **laboratory**, **radiological**, and **environmental** uncertainties are thoroughly described and quantified in behalf of film dosimeters that were used during the atmospheric nuclear testing program, which spanned from 1945 to 1962 (NRC1989). When combined, the average total uncertainty was generally found to be about 40% for dosimeter exposures that ranged between 0.2 and 2 rem.

In summary, the TBD's generic assigned standard deviation value of 30% is likely to be low for film dosimeters used prior to 1971. Early film dosimeters are likely to have a **workplace** standard deviation of at least 40%.

(It should also be noted that Table 5.3.2.1-1 of the TBD, which is reproduced here as Table 5.4, should be corrected as follows: As a measure of uncertainty, the headings defined as "SD/Mean" are, in fact, expressed as **percentage** (%) values. In the original Thierry-Chef et al. 2002 study, Table 5.6 identifies these columns as "SD/Mean (%)." Thus, the TBD shows uncertainties that are two orders of magnitude too high.)

5.4.4 Missed Photon Doses

A minor deficiency of this TBD and/or Section 5.0 is the absence of guidance pertaining to the interpretation of open-window dose, shallow dose, 7 mg/cm² dose, and/or skin dose. Although these terms are defined in the glossary and mentioned in the Executive Summary (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, *Interpretation of Dosimetry Data for Assignment of Shallow Dose* (Merwin 2005).

5.4.5 Consistency Among Site Profiles

The recommended adjustments to recorded photon dose in the TBD are inconsistent with those recommended in ORAUT-OTIB-0010, *Technical Information Bulletin: A Standard Complex-Wide Correction Factor for Overestimating External Doses Measured with Film Badge Dosimeters*, and ORAUT-OTIB-0008, *Technical Information Bulletin for a Standard Complex-Wide Conversion/Correction Factor for Overestimating External Doses Measured with Thermoluminescent Dosimeters*. The time periods for the adjustments are also inconsistent with those recommended in Taylor et al. 1995. The transition year for beta/gamma adjustment factors is 1986 rather than 1987.

The directions provided in the SRS TBD for adjustment of deep photon dose are as follows:

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Table 5.9 Adjustments to Reported SRS Deep Photon Doses

Time Period	Dosimeter	Facility	Step	Adjustments to reported dose
Prior to 1987	All beta/photon dosimeters	All facilities	A	Multiply reported TLD-Deep photon dose by a factor of 1.119 to estimate Hp(10).
For the year 1987	TLD beta/photon dosimeter	All facilities	B	Multiply reported TLD-Deep photon dose by a factor of 1.039 to estimate Hp(10).
Post 1987	TLD beta/photon dosimeter	All facilities		No adjustments made.

ORAUT-OTIB-0010 recommends a standard correction factor of 2.0 for film badges from 1970 forward. This would apply only to the first quarter of 1970 for SRS. ORAUT-OTIB-0008 also recommends a standard correction factor of 2.0 for TLD badges. This would apply to recorded deep dose from the second quarter of 1970 to the present. The SRS adjustment factors applied to recorded dose are substantially lower than those recommended in the DOE complex-wide documents.

In summary, the TBD has underestimated the true exposure being measured by the dosimeter. 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. The general standard deviation value is too low for film dosimeters prior to 1971. Furthermore, the SRS dosimeter adjustment factors are lower than those recommended in DOE complex-wide TIBs.

5.5 Issue 5: Neutron Dosimetry

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 (Pu-Al alloys) in the 300 Area. Each of these facilities not only differs 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 SRS time period, when neutron exposure was assessed by NTA film, as explained below.

Neutron dosimeters used to monitor individual workers at SRS involved three different designs. The first involved the neutron track emulsion, Type A film (NTA) 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 that the worker was exposed, 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 MeV, as reported by others), as well as other factors contributing to the dosimeter's uncertainty,

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NIOSH/ORAUT concluded that NTA monitoring data used from 1953 through the end of 1970 was 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 a 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 the need to match 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 makes use of 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., $n + \text{Li-6} \rightarrow \alpha + \text{H-3}$). 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/ORAU 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. (Note: One location where the use of NTA film dosimeters was considered useable is the Fuel Fabrication Area (321 M Area).) NIOSH/ORAU 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. Monitoring data for these dosimeters are regarded by NIOSH/ORAU as "reasonably accurate" and are, therefore, considered useable for dose reconstruction, but not without "adjustment." Since 1971, neutron doses recorded by TLDs were based on neutron quality factors in 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 ICRP 60 weighting factors. In compliance with 42 CFR 82, NIOSH/ORAU 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 in behalf of **post-1971 recorded** neutron doses.

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 that the TLND provides significant improvements over NTA film.

5.5.1 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 to not use of 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 5.10 below summarizes surrogate post-1971 data that is to be used for neutron dose reconstruction prior to 1971.

Table 5.10 Neutron-to-Photon Ratio Values Used as Surrogate Data for NTA Film Dosimeters

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

Inspection of Table 5.10 shows that not only are there large differences in neutron-to-photon ratios among the four general areas, but there exist even larger differences 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 three independent uncertainties of the **post**-1971 TLND neutron dosimeter, (2) the uncertainty of the **post**-1971 TLD photon dosimeter, and (3) the variability of the neutron-to-photon ratios among locations within a given area, such as the FB-Line.

In addition to these three uncertainties are two more uncertainties that contribute to the actual pre-1971 neutron dose. The fourth uncertainty is the **pre**-1971 **photon** dose (which must be multiplied with the **post**-1971 neutron-to-photon ratio); and the fifth 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, etc., during the assessed post-1971 era were, in fact, identical/comparable to those that existed between 1953 and 1970. Table 5.11 summarizes the five separate uncertainties that collectively define the overall uncertainty of pre-1971 neutron doses that are derived by the photon-to-neutron ratio method.

Table 5.11 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	$\pm 50\%$ to $\pm 75\%$ ^(a)
- multi-element film	$\pm 40\%$ to $\pm 60\%$ ^(a)
2) post-1971 TLND dose	$\pm 50\%$ to $\pm 75\%$ ^(a)
3) post-1971 photon dose	$\pm 20\%$ to $\pm 30\%$ ^(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 non-quantifiable uncertainties. The aggregate of these uncertainties preclude the use of guidance, as given in Section E.4.1.6 of Attachment E of the TBD, which states:

*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 to 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. Table 5.10 reveals the magnitude of the effect of using the 95th percentile adjustment factors as opposed to the geometric mean (i.e., a factor of about 4-fold increase in the neutron doses).

5.5.2 Performance Characteristics of the TLND

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

*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.]

The above referenced Figure 5.3.4.2-2 is reproduced below as Figure 5.3. Neither statement (as emphasized above) is supported by data shown in Figure 5.3:

- 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 Pu), 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.

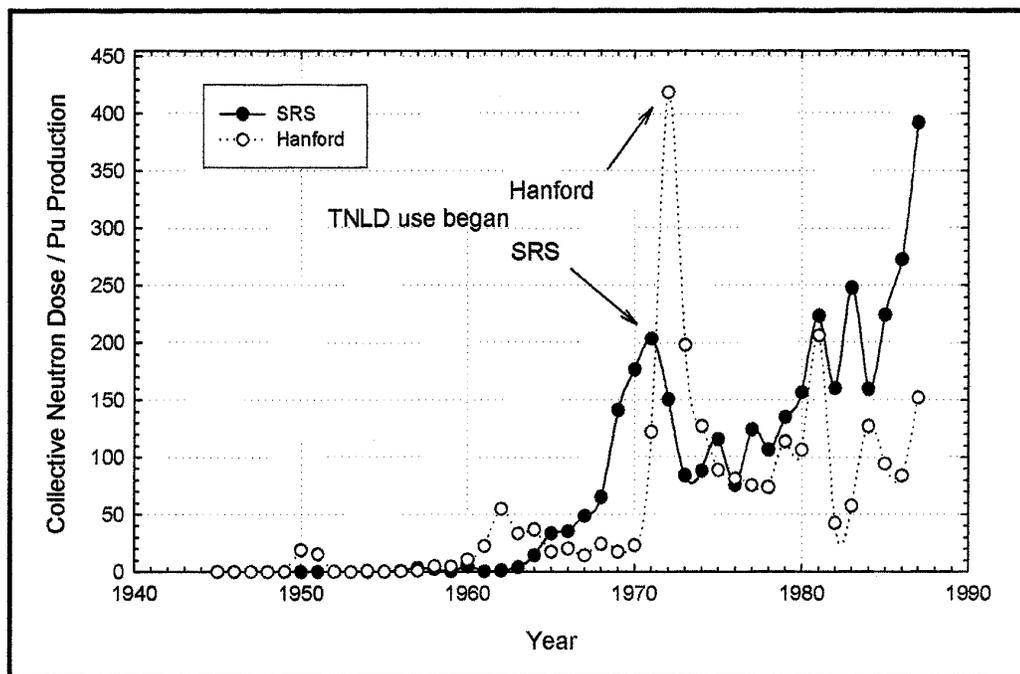


Figure 5.3 Trends in SRS and Hanford Collective Neutron Dose Normalized to Plutonium Production

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The uncertainty of the TLND's response is discussed in Section 5.3.5 of the TBD and includes the following statements:

*As reported in the PNNL report, measurements with the TEPC, multisphere system and ³HE 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. [Emphasis added.]*

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 $\pm 50\%$ 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.") [Emphasis added.].

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 recommends the use of a 95th percentile value for the TLND neutron dose of record.

In summary, the approach to assigning neutron dose is not technically defensible or claimant-favorable.

5.6 Issue 6: Tank Farm Workers

The F- and H-area Tank Farm characterization in the TBD is inadequate for dose reconstruction guidance in several respects. Moreover, no references are provided for the Tank Farm discussion in the TBD, and there is no analysis indicating how the conclusions were arrived at. NIOSH may be planning to address some of these areas under the reserved trades-workers section. However, given that Tank Farm workers entering radiological control areas appear to have been subject to monitoring requirements, regardless of individual worker designation, the concerns discussed in this section are broader and would apply to a larger set of employees. The

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following are the categories under which the TBD guidance needs to be more specific and complete:

- 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.
- 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.

5.6.1 Radionuclide Lists

The TBD (Scalsky 2004, p. 31) gives the radionuclides 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 .

The list in Table A-14 of the TBD also 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. This is confusing. No references are provided for the lists. The TBD also does not contain any analysis as to how these lists were prepared and how NIOSH made the determination that these 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.

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, Table 1.

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Similarly, the Tank Farm data bank contains records of internal Ru-106 exposure. Further, 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. Rhodium-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.

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. The internal radionuclide list is incomplete in other ways; for instance, Tc-99 is missing from it. Finally, several radionuclides that were produced, processed, or used as target material were not included in the Tank Farm radionuclide list. They include Th-232, Np-237, Pu-242, and U-233. Since no analysis is presented in the TBD in regard to 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, their use is discussed in the TBD, but they are not included in the Tank Farm radionuclide list for reasons that are not explained. If they have been evaluated, the analysis should be presented. If not, they should be evaluated. NIOSH/ORAU also needs to be aware of the differences in the constituents of the tanks based on the processes that fed them.

5.6.2 Early Tank Farms 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. (as quoted in Makhijani, Alvarez, Blackwelder 1986, p. 20.)

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 instance, there is no entry in the data bank explicitly showing the amount of worker exposure prior to 1960, where there are many after that date. The changes in the frequency of entries per year in the Tank Farm data bank are 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, the 1960s, and at least part of the 1970s. The following table, reproduced from Makhijani, Alvarez, and Blackwelder 1986, p. 30, shows the increasing frequency of Tank Farm data bank entries:⁵

⁵ 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. It was requested from NOISH as part of the SRS document request, but has not been received. A similar data bank for the F- and H-area canyons also has not been received.

Table 5.12 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.
1960-65	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 containing 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 SHI index is also incomplete, as acknowledged by the site, it raises the question of how complete the record of incidents might be in individual worker dose records, at least for Tank Farm workers. This is a crucial issue, since NIOSH dose reconstruction procedure for SRS relies heavily on the DOE dose records being essentially complete and looks to the 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. 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 R per hour, tens of R per hour, and sometimes even hundreds of R per hour (Makhijani, Alvarez, and Blackwelder 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 in regard to missing incidents. SC&A does not have access to the master incident list and hence is unable to evaluate this problem. NIOSH has not been using this list in its SRS dose reconstructions.

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5.6.3 External Exposure Geometry Issues Related To Tank Farms Workers

There were a large variety of exposure geometries experienced by employees working in the F- and H-area Tank Farms. The question of the location of the badge in relation to geometry of exposure is especially important in the Tank Farm area due to the highly non-uniform 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 done on piping, in junction boxes, and other Tank Farm equipment, as well as during clean-up 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 in regard to external dose due to 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 or not multiple badges were used
- What entries were made into the records when multiple badges were used

Although SRS had an established multiple badging program, it is unclear whether multiple badging was used during Tank Farms work. NIOSH/ORAU should investigate the specific exposure conditions of the Tank Farm workers, including an evaluation of the incident exposure versus the badge location.

5.6.4 Radiological Zone Designation

In addition to these issues, there is the question of 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-Tank Farms were designated as radiation zones, but that the entire F- and H- Tank Farm area was not so designated (see Attachment 4). Given that incidents may have been missed due to lack of recording, the potential for 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 issue and other instances like it 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.

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

The above discussion indicates that NIOSH has not evaluated site data, including the crucial Tank Farm data bank and the master list of incidents, in the course of preparing the TBD and of

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revising it. Hence, NIOSH's data evaluation is incomplete in 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 that this document is of primary importance in assessing radiological conditions in that part of SRS, and in determining 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. 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 radionuclides lists that were chosen for the F- and H-area Tank Farms.

It is unclear whether there is adequate data to reconstruct any but the minimum tank doses for Tank Farm workers because of the various issues, data gaps, and uncertainties discussed above. The situation in regard to early workers is especially unclear. A clear judgment on this question will be difficult or impossible without a careful evaluation of the available literature, and without 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.

5.7 Issue 7: Internal Dose Assumptions

Solubility, oro-nasal breathing, and ingestion should be carefully considered in regard to internal dose reconstruction.

5.7.1 Solubility Assumptions

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

5.7.2 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 Downtwon Site St. Louis, Missouri*. That finding is also applicable to workers at the SRS. 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 was a mouth breather or not should be addressed and a determination made whether NIOSH should continue to strictly follow ICRP models, which do not address oro-nasal breathing, or

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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).

5.7.3 Ingestion

NIOSH/ORAU has assumed that inhalation is the only pathway for internal exposure at SRS. During our discussion with them, an 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 (is important at AWEs), 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. Further, in order to take ingestion into account using bioassay data, the inhalation component has to 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. As another example, the problem would be solvable if fecal analysis and urinalysis data were both available for estimating the intake in question. The TBD must then specify a procedure for solving for the inhalation and ingestion intakes. The TBD does not provide any procedure for doing this. Moreover, the CATI does not ask about food intake (SC&A 2005, Chapter 5). Moreover, some food consumption may have been associated with unauthorized practices. At Fernald, for instance, Plant 5 workers ate their lunch on the K-65 silos on nice days. The SRS TBD does not discuss the corresponding issues at SRS that might affect the assumption of negligible internal dose. The environmental component of the TBD may also be a factor in this regard, given that issues like plutonium-contaminated solvent burning are not discussed.

As a result of this gap, this could be an issue at all DOE sites. Also note, this is not just an issue for the early period. This will surely not show up on any worker records.

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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, of course). If one assumes inhalation = 0 in interpreting the bioassay data, then one can get 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.

In summary, while inhalation is likely to be the predominant intake mode the vast majority of the time, an overall analysis in the TBD is needed 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. In the context of the need to evaluate this issue, SC&A notes that NIOSH has evaluated some issues where doses are very low. For instance, NIOSH has estimated intakes as low as 2.42E-04 Bq per year for Pu-238 in Table C-18 (SRS TBD, p. 180).⁶ This is appropriate when there are questions, since it settles the issue if the analysis is sound. The same would apply to ingestion, even if only to show that NIOSH's assumption is analytically based.

NIOSH/ORAU has stated above that ingestion is not usually considered at major DOE sites. For worst-case internal dose assumptions at Hanford, ingestion doses have been assumed and included as a part of the total internal dose. Ingestion dose should not be selectively applied at one facility and ignored at another.

5.8 Issue 8: Special Tritium Compounds (STCs)

The technical information bulletin for assignment of tritium dose disregards the potential dose from special tritium compounds. ORAUT-OTIB-0001 states (p. 11) the following:

Organically bound tritium (OBT) historically has been ignored for occupational assessment and SRS assumes that there are no significant quantities of stable metal tritides (SMT).

ORAUT-OTIB-0003 provides a basis for determining missed dose and the default missed dose values. Tritiated water is the only form of tritium considered in missed dose calculations, which are applied to monitored workers as well as unmonitored workers. While the assumption of tritium as tritiated water is generally claimant favorable, it is not so in specialized situations. Specifically, the TBD does not consider organically bound tritium and stable metal tritides as important.

OBT and SMTs are present at Savannah River Site, especially in relation to tritium production. Most tritium handled in the process areas was in the form of tritium gas (HT or T₂) or tritium oxide. However, tritium handling operations can form other compounds, such as organically bound tritium and stable metal tritides. For instance, carbon sources in the tritium processing area can contribute to the production of organic tritium forms by hydrogenation or exchange

⁶ SC&A has noted elsewhere in this review that the environmental intake estimates and methods used for making them are questionable and may be underestimated by substantial margins. The point in this context is that estimates of intakes that NIOSH considered to be low were made and published.

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reactions. The processes where this might have occurred include the following (Milham and Boni 1976):

- Graphite crucibles used in Li-Al 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

Although organic tritium compounds have been identified in the exhaust from the tritium-processing areas, the form of OBT is not certain (Milham and Boni 1976). In addition, 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 uptake of molecular tritium (HT) by living pine needles. 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.

The effective dose per 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 that for HTO. Thus the effective dose calculated for each Bq excreted in urine is 4 times higher, considering it is due to OBT instead of HTO.

Processes at the Savannah River Site have also produced stable metal tritides. 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.

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SMTs, often existing as small particles, are encountered in some facilities used to process significant quantities of tritium. These particles are solid substances containing tritium that do 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.

Because SMTs are relatively insoluble, and the retention of this type of tritium is longer than 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 lungs. Some of the tritium may leach out in the lung fluids and then be incorporated into the body water. These particles may also produce organically bound tritium from contact with lung tissue, which would further complicate the metabolic process (DOE 2004).

Furthermore, Special Tritium Compounds (STCs) 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). 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 these individuals, tritides will build up for a few days followed by a traditional exponential decay. The *ICRP Database of Dose Coefficients: Workers and Members of the Public* 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 metal tritides should be equal to the generic types F, M, and S, if the ICRP recommendations are followed.

Although under most circumstances the concentrations of STCs would be a small fraction of the exposures to tritiated water, this may not be the case for those workers involved in tritium production or decontamination and decommissioning of tritium facilities. The relative impact of these radionuclides compared to the default missed dose assignment should be investigated to ensure the missed dose bounds potential dose from STCs. In addition, dose reconstructors should be made aware of the characteristics of STC excretion in urine to enable them to identify intakes of STCs as compared to tritiated water. 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).

5.9 Issue 9: Internal Dose from Transplutonium and Non-Military Radionuclide Production

While the main products produced at the Savannah River Site were plutonium and tritium, a variety of other isotopes were produced during the transplutonium program and for non-military 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

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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. As a high neutron flux was required to produce transplutonium isotopes, the site established a 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 Savannah River included the thorium campaigns for the production of U-233, 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.

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 (see p. 65, Rev. 2) and not a material produced in its own right. The use of Pu-242 as a radiobioassay tracer beginning in 1981 (Scalsky 2004, p. 65) 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.

In summary, the impact of internal and external exposure to radionuclides from special campaigns should be analyzed and included in the TBD.

5.10 Issue 10: Incidents and High-Risk Jobs

Incidents and high-risk jobs are not listed in the TBD or referenced to alert dose reconstructors of unique exposure conditions.

5.10.1 Incidents

NIOSH/ORAU appears to have decided that incidents are not significant to the dose evaluation process where routine monitoring occurred, and therefore, NIOSH/ORAU does not incorporate them into the TBD or other supporting documents. Furthermore, the TBD does not discuss the various site sources available for incident identification. However, including references to where these reports can be found is prudent, so that dose reconstructors can access this information if there is reason to believe that a given claimant may have been involved in an incident or accident.

This issue was discussed with NIOSH, as follows:

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SC&A Inquiry: *Is the rationale that an incident database is not needed as part of the site profile based on the premise that DOE records contain the data needed in the worker dose records?*

NIOSH Response: *In many cases, the workers will state there was no incident, but the records do, in fact, identify incidents. It is possible for a claimant to say that there was an incident but no incident is identified in the records. There are not many cases when a worker says there was an incident and it was not in the dose record. 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.*

Since the high-five approach is limited only to internal dose for non-metabolic cancers, the approach is unlikely to capture doses resulting in external exposure from incidents, spills, and over-exposure conditions. In addition, 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 workers with little or no apparent internal dose.

Exposure conditions that may present themselves during an incident or occurrence have not been addressed in the TBD. Per the telephone conversation held between SC&A and NIOSH (see Attachment 2), the DOE exposure file and the CATI provide the mechanisms for identifying incidents, and many incident reports are included in the SRS personnel radiation exposure file. However, there are some instances where an incident will not be placed in the individual exposure file. The CATI interview is used as a secondary source of information on incidents or occurrences. This creates issues for family member claimants, since they are far less likely to be aware of incidents, and in some cases do not even know the definition of an incident or occurrence (SC&A 2005). As a result, there is an uneven playing field 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.

During site expert interviews (see Attachments 4 and 5), several methods for documenting of incidents and occurrences were identified. Many, but not all, incident investigations are included in the Personnel Radiation Exposure File. Other sources of incident information were identified, including Special Hazards Investigations (SHI) reports, the SRS incident database, and the Tank Farms data bank. These reports provide brief descriptions of the incidents and were maintained separate from the individual radiation exposure records. SC&A recently (February 2005) obtained a copy of the SHIs, the associated log, and a procedure related to their generation. NIOSH has not included these reports in their case reviews. 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 Farms data bank for this review. SRS also developed a database containing minor and major incidents through 1999. Included in this database are some of the classified incidents. Westinghouse Safety Management Solutions (WSMS) currently has possession of this database. Based on site expert interviews, it appears that this incident database is not readily available to NIOSH.

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There should be a redundant system for identifying incidents, because the dose reconstructor cannot always rely on incident records being included in the individual's dosimetry file, and other incident's sources may not be easily searchable. In addition, an evaluation of the Special Hazards Bulletins and the Tank Farms data bank indicates that incident databases may not always be complete.

DPSOP-40, SHB 2, *Investigating Radiation and Contamination Incidents* (DPSOP 1981), contains the standard operating procedure for special hazards investigations. SC&A's copy of this DPSOP is dated March 1981. Among the incidents required to be investigated under this procedure are the following:

- *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.*

By these criteria, the log of Special Hazards Investigations (DuPont 1990) is incomplete, as illustrated by Table 5.13, which contains entries from the Tank Farm data bank. These entries correspond to one or more of the criteria for an SHI quoted above. Yet in most cases, there was no entry in the SHI index corresponding to the date and area in which the incident listed in the Tank Farm data bank took place.⁷

Table 5.13 Comparison of F-Area and H-Area Tank Farm Data Bank Entries with SHI Log
(See Note 1)

Date and Area of data bank entry	Tank Farm data bank entry summary	In SHI log?
Feb. 1968, F-area	Failed tank 1B evaporator pump. "Body exposure ranging to 30R/hr at 18 inches" during replacement. Asphalt also contaminated to 5 rads/hr at 5 inches. "Total estimated exposure was 0.8 R."	No
July 1971, F-area	Total of 3.4 R worker exposure "while lightening packing glands." "Exposure resulted from high radiation levels in feed pump enclosure."	No
11-20-1972, F-area	Exposure during removal of a valve. "Hands, face and personal items contaminated to 2,000 c/m beta-gamma. Bioassay – 13 nCi, Cs-137/1.5 L. Body count = 84 nCi Ru-106, 368 nCi Cs-137."	No

⁷ SC&A used the Tank Farm data bank entries as compiled in Table 1, Part II, of Makhijani et al. 1986, and which contains only a subset of all data bank entries. The data bank used in Makhijani et al. 1986 only goes up to 1982.

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Table 5.13 Comparison of F-Area and H-Area Tank Farm Data Bank Entries with SHI Log (continued)
(See Note 1)

Date and Area of data bank entry	Tank Farm data bank entry summary	In SHI log?
08-21-1975 F-area	“Contaminated soil encountered [sic] during excavation around Riser 6 Tank 3. 350 R/hr @ 1 inch from steam supply line to the jet in Tank 3. Probably the result of suckback and leak. Soil contained about 50 Ci 137-Cs.”	No
Feb. 1979, F-area	3 workers exposed to CTS loop. No badges. Exposure estimated 65 mR. Cause accidental removal of a fence.	No
03-14-1979, F-area	“Construction worker got 8,000 c/m beta-gamma on gloves. Worked in 241-F regulated area without health physics coverage.	No
02-15-61, H-area	Raising of thermocouple from Tank 16 annulus plug caused contamination of 2 workers “up to 6,000 c/m” and equipment was contaminated “15 r/hr on a riser plug.” “Approved procedures” not followed.	No
Sept. 1966, H-area	“Radiation Exposure, See SHI 243.”	Yes
Nov. 1968, H-area	“Film badge of sep. dept. supv. indicated skin exposure of 14,590 mrads during Oct. exceeding AEC manual quarterly standard of 10 rem....See special hazards investigation 266.” Note SHI 266 describes this as in 221-H, not 241-H	Yes
02-28-74, H-area	1 pint contaminated liquid sprayed from a leak. “. . .grating of catwalk around evaporator cell was contaminated to 8 rads/hr...2 maintenance mechanics were contaminated by falling droplets. Nasal contamination up to 1,345 d/m. Body contamination 300 mr/hr at 2" from arm, 1 st mech. bioassay = 12 nCi; Cs-137/1.5L Chest count = 262 nCi....2 nd mech. Bioassay = 64 nCi, Cs-137/1.5L Rec'd (2% MPBB).” Note: NBS 1969 MPBB = 30 µCi. Therefore 2% MPBB = 600 nCi.	Yes
02-01-77, H-area	Tank 29, liquid spill during repairs. Exposure rates 150 rads/100R/hr. at 5 cms. “Personal shoes” contaminated.	No
May 1977, H-area	“High personnel exposures to T&T workers on hot job.”	Maybe

Source: Makhijani et al. 1986, Part II, Table 1, “Worker Exposures.”

Note 1: Entries for this table were summaries of data bank entries, except for the parts that are in quotes, which were taken verbatim. Original data bank no longer available for verification.

This short list, selected from the Tank Farm data bank, contains three estimated internal exposure entries (one in the F-area in 1972 and two in the same incident in the H area in 1974) that are larger than the lowest two values listed in ORAUT-OTIB-0001, Table 1, for the “high-five” Cs-137 intakes in ORAUT-OTIB-0001. The average for the NIOSH “high-five” Cs-137 intake is about 361 nanocuries, while the average that takes into account the highest five from among the above data points and Table 1 of ORAUT-OTIB-0001 is about 475 nanocuries, or about 31% higher. Moreover, it is not clear that these values would represent the five highest. Given the

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gaps in the data bank, it is unlikely. Furthermore, SC&A has not had access to the data bank relating to the F- and H-canyons that could indicate exposures to fission products or other radionuclides higher than those listed in Table 1 of ORAUT-OTIB-0001. Finally, it is unclear how the standard operating procedure for special hazards investigations has changed over time. The TBD contains no discussion of this issue.

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). A redundant system for incident identification is necessary for an effective evaluation of incidents and accidents.

High-Risk Exposure

For cases where the site profile does not fit a particular individual, there is a need to provide guidance within the technical basis document on how to address the following special exposure conditions.

- Construction, subcontract, and decontamination and decommissioning workers
- Workers involved in U-233 and thorium recovery and processing
- Workers involved in production of certain transuranic radionuclides (i.e., Pu-242, Am-243, and Cm-244)
- Off-normal or unauthorized practices and exposures (e.g., eating fish from Par Pond)
- Open-burning of spent tributyl phosphate at the burial grounds
- Authorized procedures that may have caused significant unrecognized or unreported external and internal exposures, such as opening high-level waste tank risers for visual checks of the tanks

The TBD does not provide a list of special exposure conditions that require an individualized dose reconstruction. For consistency among dose reconstructions, the TBD should alert the dose reconstructor to conditions when a deviation from the standard dose reconstruction methodology is needed.

In summary, incidents and high-risk exposures may present situations where application of methodologies in the site profile is inappropriate. Dose reconstructors should be alerted to these situations. Based on records storage practices, redundant systems are necessary to develop a complete list of incidents.

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5.11 Issue 11: Early Worker Radiological Monitoring Completeness

The TBD does not address the consistency of the SRS internal and external monitoring program for different operations and through time. This is especially important in the case of early workers who may not have had routine monitoring commensurate with their exposure.

5.11.1 Consistency in Field Implementation of the Monitoring Requirements

As described in Attachment 4, the Radiological Control Organization was not centralized for the site. 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 determined the requirements for routine and special bioassay and dosimetry with 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 (RDZ's).*
- *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, bioassay and dosimeter requirements also had to be approved by Operating and Health Physics Departments (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 non-routine 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. The requirements for external monitoring were the same as defined above (DPSOP 1971, 1973, 1974, and 1976). By the late 1980s, the dosimeter and bioassay requirements were clearly outlined in DPSOP-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

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1 mrem/hour (DPSOP-1989b). Routine and special bioassay requirements were also outlined for both in vitro and in vivo counting (DPSOP 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 in the early years.

In February 1999, SRS underwent an independent assessment of their 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.

Given that these types of procedural non-conformances were identified in 1999, it would seem that these types of non-conformances may have also occurred in the early years. It is not apparent in the TBD how the potential for such non-conformances should be handled by dose reconstructors. The TBD would benefit from a discussion of this potential issue.

The TBD indicates that neutron exposure should be explicitly addressed if employees worked in specific areas of the site (i.e., 736A, 773A, reactors, 221F FB-line, 221H HB-line, 235F, 772F, and 321M). When the work area is unknown or not clear, the dose reconstructor is provided with the following guidance (Neton 2003):

General indications of potential neutron exposures

1. *If an energy employee was monitored for neutron exposure in 1971 or later, and they did not change jobs or work area, the energy employee should be considered to have been exposed to neutrons prior to 1971. The monitoring for neutrons increased dramatically after the implementation of the TLND in 1971, thus contemporary monitoring is a good indicator of potential for neutron exposure.*
2. *External dosimetry records indicate the 17 keV calibration curve was used for interpretation of the shallow dose. This is an indication of exposure to plutonium and therefore neutrons. This indicator could be for work in the 100, 200, or 300 areas.*
3. *Neutron exposure indicated in external dosimetry records between 1958-1962 (codes 32 and 33). This neutron dose might or might not have been separated in the HPAREH summary sheet. In addition, the dosimetry cards prior to 1958 also contained an area for fast neutrons (NF) and slow neutrons (NS) to*

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be recorded. A close investigation of the dosimetry records should be conducted to evaluate the potential for neutron exposure.

There are a number of issues with respect to this approach for assigning missed neutron dose. First, there are a number of claimants who were early site workers and were not monitored for neutron exposure before 1971. It is not apparent whether SRS was consistent in applying the 17 keV calibration curve to plutonium workers outside the B-line and plutonium-alloy operations. The completeness of the routine film badge logbooks is unknown, especially with respect to monitoring results less than the Minimum Detectable Level (MDL) for NTA film. These logbooks should be compared with neutron processing records and neutron summary reports to determine the completeness of this information. In addition, as discussed later, NIOSH/ORAU has not retrieved all available personnel neutron exposure data. Since the assignment of missed neutron dose is partially dependent on the existence of neutron monitoring over the period of employment, the knowledge of whether individuals were monitored for neutron exposure is critical.

Although early procedures clearly defined when beta/gamma dosimeters were to be worn, the requirements for the use of neutron dosimeters and taking bioassay samples were left up to the field support organization. Without a single organization determining neutron dosimeter and bioassay requirements, there may have been inconsistencies in actual practices. This also includes the request for special interpretations of film badges. Based on the potential for inconsistencies in determining monitoring requirements, the completeness of monitoring at SRS should be evaluated.

5.11.2 Dose Assessment for Early Workers

NIOSH is aware that early worker dose reconstruction poses difficult questions, as illustrated by the following exchange taken from Attachment 3:

***SC&A Inquiry:** How is NIOSH dealing with the many issues relating to early workers?*

***NIOSH Response:** A lot of the early worker dose reconstruction has been postponed. However, if an early worker case that can be compensated is identified, the dose reconstruction is performed. These cases are difficult. One denial of an early worker that comes to mind was a cafeteria worker at Oak Ridge. He worked for 6 months and developed prostate cancer at age 70. A maximum dose assessment was done, and it resulted in a denial. Co-worker data will be used to see who were monitored versus who were unmonitored. This is the basic approach intended for early workers.*

***SC&A Inquiry:** Will this result in a TIB?*

***NIOSH Response:** This will likely be a revision to the technical basis document itself. Early Y-12 folks are in the process and the evaluations are close to being completed. There are a large number of claimants that were not monitored.*

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SC&A has not performed a comprehensive look at early worker related issues. However, the above statement indicates that NIOSH is confronted with serious technical issues that need to be resolved by a further TBD revision or issuance of a Technical Information Bulletin. Such a comprehensive look was not done in Revision 2 issued in October 2004.

The discussion of early workers, to the extent it is present, as for instance with pre-1971 neutron doses, is lacking in depth. There are also issues with early data that are not discussed in the TBD. Further, even the date that defines the term “early worker” needs to be understood in the specific context of the issue, as is clear in the following discussion pertaining to early worker dosimetry.

Before proceeding with this section, it is important to understand the difference between missed and unmonitored doses as used in this report. The term “missed dose” is used for workers who were issued dosimeters, but the records report “zero” because the exposures were below the limits of detection. Missed dose could also apply to workers who were monitored, but the records were lost or the worker claims he was monitored, but no records exist. In addition, records could actually report a value that is below the limits of detection, or reported as <MDL. These situations are referred to as missed dose. The term “unmonitored exposures” applies to workers who were not issued dosimeters. This could occur under circumstances where there was good reason not to monitor a given worker based on his job responsibilities, or it could also apply to a worker who should have been monitored, but wasn’t.

Unfortunately the TBD itself, in Section 5.5 (page 111 of 242), defines missed dose in a way that is not very helpful:

Missed photon dose for SRS workers would occur where: 1) there is no recorded dose because workers were not monitored or the dose is unavailable and 2) a zero dose is recorded for the respective SRS dosimeter systems or any dosimeter response less than the minimum detectable level.

This is not a very useful definition because methods required to deal with a monitored worker that has a zero reading because it is below the limits of detection should be different than that for a monitored worker whose records are missing, an unmonitored worker who was unmonitored for good reason, and an unmonitored worker who should have been monitored. The ambiguity in the definition of missed and unmonitored doses is further exacerbated by ambiguous definitions provided in OCAS-IG-001 on pages 5-7 and page 16.

Given the cited definitions, the following identifies certain issues related to unmonitored workers who may have experienced neutron doses during the early years.

The TBD appears to apply MDLs for unmonitored dose (Scalsky 2004, p. 111). This approach is appropriate for missed dose (i.e., the worker was monitored but the results came back as zero or below the MDL). However, this is not necessarily claimant favorable for unmonitored workers without an actual history of why a particular worker was not monitored (making the use of co-worker data necessary for dose reconstruction). Assignment of the administrative control limit may be more appropriate in some cases. The interpretation of zero entries needs more work and

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has more uncertainties than are discussed in the TBD. Appendix E, page 238, mentions co-workers briefly and says that co-worker “situations do require careful examination.” However, the TBD contains no discussion of how that careful examination should be done. Use of co-worker data as a means to fill in non-existent exposure data for claimants is especially important for early neutron exposures (i.e., prior to 1971). This issue is acknowledged in the TBD in Appendix E, Section E-4.1.6, page 238. We suggest that a statistical model be employed for using co-worker data for neutron dose assignment, with consideration being given to 95th percentile values to ensure that the benefit of the doubt is given to the claimant if there is reason to believe that the claimant was exposed but not monitored or his records are missing.

Table 5.3.4.2.1.2-1 of the SRS TBD shows the trends in SRS and Hanford collective neutron dose normalized to plutonium production. The table clearly shows there was minimal collective neutron dose prior to 1964 (Scalsky 2004). Review of the monthly reports from the Works Technical Department for December 1956, 1957, 1958, 1959, and 1960 indicates 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, the dosimeter exchange schedule was weekly. The number of badges processed likely includes multiple badges for an individual. From this information, it appears that minimal numbers of individuals at the Savannah River Plant 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.

Furthermore, monthly reports from the Works Technical Department contain information indicating that the tritium-monitoring program may not have been comprehensive. The number of tritium urinalysis samples was tracked by area in these reports. 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-0003 (Duncan 2003).

In summary, 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.

5.12 Issue 12: Availability of Additional Source Documents and Data

Based on searches performed for records pertaining to dose reconstruction, SC&A has identified sources of records that are not currently provided to NIOSH/ORAU by the site, but which may be important to dose reconstruction.

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5.12.1 Individual Neutron Exposure Data

In addition to the data sources identified in the site profile, History Associated Incorporated (HAI) prepared site-specific guides to epidemiologic and health-related records at a number of sites including the SRS. A list of records available at the Savannah River Site is available through the DOE Office of Epidemiologic Studies website. Included in this inventory of records are the following:

- Neutron Pencil Results (1952)
- Neutron Exposure Reports (1971)
- NTA Film Badge Inventory and Calibration Sheets, (1950-62, 1964-69)
- Neutron Dosimetry Data Logs (1972)
- Dosimetry Logs (1954-1978)
- 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)

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 source of additional data may be relevant to OCAS-TIB-007, which provides guidance on how to determine when there was a potential for neutron exposure prior to 1971.

5.12.2 Multiple Dosimetry

There are a number of conditions that can result in partial body exposures, or portions of the body being exposed unevenly. The SRS recognized this early in 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 more accurately capture doses to various parts of the body at risk of greater exposure than would be indicated by a single dosimeter worn at the pocket level. The TBD has not mentioned the multiple badging programs and how it 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/ORAU. 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, ICRP 26 weighting factors are used. The whole-body dose is the only number which is documented in the individual’s dosimetry file.

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All dosimeters worn were processed and results were recorded by area of the body (see Attachment 4). NIOSH/ORAU should evaluate the results of multiple dosimeter processing by body position to determine whether partial-body exposures are an issue to specific organs, and to evaluate whether the doses provided by the multiple dosimetry are more claimant favorable. As this information is not routinely included in individual files, a special request will be required.

In summary, the additional neutron data is important in the evaluation of neutron dose, especially since the dose reconstructor is instructed to choose the higher of the missed neutron dose or the recorded neutron dose. Multiple dosimetry results may provide valuable information on nonuniform exposures and may be of assistance in determining organ dose for some workers.

5.13 Issue 13: Quality Assurance

SC&A has inserted a section on quality assurance as a separate part of regulatory compliance to deal with quality assurance issues that are indirectly related to compliance. This section considers the clarity of the dose reconstruction direction provided in the TBD and the consistency within the TBD and between the TBD and relevant TIBs.

ORAUT-PLAN-0001, *NIOSH Dose Reconstruction Project Quality Assurance Program Plan (QAPP)*, outlines the quality assurance elements to be implemented in all aspects of the dose reconstruction process. The QAPP states the following (p. 15):

Dose reconstructions shall be performed in accordance with the requirements of 42 CFR 82, NIOSH Technical Guides, written work direction and procedures received from NIOSH and as described in approved Project Management documents.

Well-developed technical documents are necessary to insure 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.

5.13.1 Ambiguous Dose Reconstruction Direction

Our review has identified that, in several sections, the writing style is not clear and unambiguous, especially in Chapters 4 and 5 covering internal and external dose assignment, respectively. The document is not complete and it is difficult to understand. 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.

The SRS TBD discusses the historical in vivo and in vitro bioassay program and air sampling data in Section 4.0, *Occupational Internal Dose*. Within this section, dose assignment methodology is described for urinalysis data, whole-body counting, and chest counting. Intake assumptions are provided for trivalent actinides, plutonium, uranium, neptunium, some fission

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products, some activation products, and tritium. Further direction for the use of information is provided in Attachment D, *Occupational Internal Dose*. Attachment A, *Description of Facilities and Processes*, provides isotopic ratios that are referred to in Section 4.0. The hypothetical intake assigned to energy employees who have a non-metabolic or digestive system cancer is described in ORAUT-OTIB-0001. The technical basis for assignment of missed tritium dose is explained in ORAUT-OTIB-0001 and in ORAUT-OTIB-0003. In addition, there is no reference in the TBD to DOE complex-wide TIBs, including ORAUT-OTIB-0010 (Stewart 2004), ORAUT-OTIB-0008 (Fix and Stewart 2003), ORAUT-OTIB-0007, *Technical Information Bulletin: Occupational dose from Elevated Ambient Levels of External Radiation* (Strom 2003), and ORAUT-OTIB-0002 (Rollins 2004). It is difficult to determine when the various techniques described in these TIBs and in the TBD should be used by the dose reconstructor. It is also unclear whether these complex-wide TIBs are applicable to SRS.

A major factor that limits the readability and, therefore, comprehensibility of Section 5.0 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.

Another factor with the potential for contributing to confusion is the failure to separate and **highlight** conclusions from a variety of data that only serve to support a given conclusion. For example, Table 5.3.2.1-1 and Table 5.3.2.1-2 (in Section 5.3.2.1 of the TBD) correspond to two independent sets of study data that evaluated the two-element film, multi-element film, and TLDs that were used at SRS (see Tables 1 and 2 therein). Footnotes to each table suggest/imply that the dose reconstructor may use the values contained in the tables to adjust a claimant's recorded dose to arrive at a corrected $H_p(10)$ dose.

The actual use of these tables, as suggested by the footnotes, would lead to a significant error, since several pages later, Section 5.3.4.1 informs the reader in a very ambiguously stated manner that “. . . A claimant-favorable approach is proposed to ignore this **over**-response [as cited in Tables 5.4 and 5.5] because of the complexity of workplace photon energies and exposure geometries that tend to result in an **underestimate** of the $H_p(10)$ dose.” [Emphasis added.]

This begs the questions: (1) Why the two studies were acknowledged in the first place? and (2) Why they were dismissed as inconsequential in spite of select dosimeter limitations that were observed at various energies and exposure geometries?

A similar and equally ambiguous discussion involves the use and limitations of the NTA film neutron dosimeter. In Section 5.3.2.2, it is only suggested that “. . . NTA personnel neutron dosimeters cannot be effectively used.” A second suggestion about its limited value appears in Section 5.3.4.2.1.2, which states the following:

It is evident in this figure [Figure 5.3.4.2-2] that the collective neutron dose was under-recorded prior to implementation on January 1971 of the SRS TLND . . . The extent of the under-estimate is difficult to estimate.

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Without firmly informing the reader about the need for a surrogate approach that would substitute for NTA film dosimetry, there follows an extensive discussion in Section 5.3.4.2.1.2 about neutron-to-photon ratios at four major SRS locations.

For the Fuel Fabrication Area (321 M) discussed in Section 5.3.4.2.4.2, the reader is, nevertheless, informed of the following:

Since the Pu-Al alloy spectra is a similar energy to PuF₄, approximately 90% of the spectra is greater than that 500 keV threshold for NTA film. As a result the NTA film measurements are expected to be reasonably accurate to within parameters discussed in the External Dose Reconstruction Implementation Guideline . . .

At this point, a first-time reader would have ample reason to question the dubious use and role of NTA film data and the purpose of the neutron-to-photon discussion.

To add to the confusion, Section 5.3.5 discusses SRS workplace uncertainty and provides Table 5.3.5-1, which cites uncertainties for three photon dosimeters and **two** neutron dosimeters (note that the commercial Panasonic neutron dosimeter is **not** included). In behalf of the NTA film dosimeter, the workplace uncertainty is defined as $\pm 100\%$ along with the statement: “. . . need to use another method.” The text provides **no** recommendation pertaining to what “other method(s)” the dose reconstructor has the option to consider.

Lastly, the ambiguous role of NTA film in neutron dose reconstruction is supported in Section 5.5.2 that discusses **missed neutron dose**. Table 5.5.2-1 provides the following data for NTA film and assignments for missed neutron dose.

<u>Dosimeter Type</u>	<u>Yearly Exchange Frequency (n)</u>	<u>Estimated LOD (mrem)</u>	<u>Max. Annual Missed Dose (mrem)</u>
ORNL-NTA (1951–8/3/1953)	52	~50	2,600
SRS NTA (8/3/1953–7/13/1960)	52	~40	2,080
SRS NTA (7/14/1960–12/31/1970)	26	~40	1,040

It is only in Section 5.5.2.1 that the reader encounters a firm statement that reads as follows:

Due to the uncertainty in whether an energy employee’s NTA badge would respond to the workplace neutron spectra, using a [neutron-to-photon] ratio to the measured photon dose is recommended as a claimant-favorable option to reconstruction an individual worker neutron dose.

In summary, Sections 4.0 and 5.0 of the TBD are poorly structured, ambiguously worded to the point of being evasive, and confusing in providing procedural guidance. If used as the sole

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guidance document, multiple readings are likely required to decipher the intent of these sections and to gain the necessary confidence for its use as a guidance document.

5.13.2 Inconsistencies in the TBD

A couple of inconsistencies or errors were noted in the TBD. Section 4.0 summarizes the radionuclides of concern listed in the SRS *Internal Dosimetry Technical Basis Manual* (WSRC 1990). Attachment A further elaborates on this list by providing radionuclides of concern by facility. Several of the radionuclides of concern are not included in either the hypothetical intake or in the standard method for assignment of internal dose. These radionuclides include Am-242, Ce-143, Co-58, Cr-51, Eu-152, Fe-59, I-133, Np-239, Sr-92, and Tc-99. The TBD does not explain the rationale for excluding these radionuclides as potential sources of internal exposure.

On page 25 of the TBD, the statement is made that U-235 and U-236 are principal contributors to internal dose. This is not correct. For enriched U, most of the dose is from U-234, not U-235. For low-enriched U, U-238 is next in importance. Uranium-235 never contributes more than 5% of the radioactivity of uranium at any enrichment. This error simply may be a typo or an actual technical error.

Throughout Section 5.0, Attachment E, and other NIOSH/Guidance documents, numerous tables provide data on “maximum Annual Missed Dose” for photons, electrons and neutrons. In some instances, maximum annual missed dose is defined as $n(\text{LOD})$, while in other instances, this dose is defined as $n(\text{LOD})/2$. For example for neutron doses in Table 5.3.1-1, maximum missed doses are defined as $n(\text{LOD})/2$, while values in Table 5.5.2-1 (and Table E.4.1.6 of Attachment E) define these values as $n(\text{LOD})$.

5.13.3 Meaningless Precision

Throughout the TBD and inclusive of Section 5, numerous instances exist where dose estimates, dose adjustments, or dose refinements are introduced that either have no intrinsic value to dose reconstruction or imply a level of precision that is unwarranted. The following examples serve to illustrate this issue.

Example #1. Following a lengthy discussion about large uncertainties of photon dosimeters that involve film processing, energy dependence, exposure geometry, etc., Table 5.4.1-1 provides the following two photon dose adjustments, which are defined to the **third** decimal point.

- prior to 1986 multiply reported deep dose by a “factor 1.119”
- for the year 1986, multiply the reported deep dose by a factor 1.039

Example #2. For a facility as complex as SRS, source terms and their corresponding radiation fields are highly variable in time and space. This is acknowledged in Section 5.6.1, which states: “. . . The selection of a worker orientation is important to the calculation of the organ dose [i.e., selection of the organ-specific DCF value]. . . .

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Unfortunately there is no definitive process to determine the exposure geometry for each worker.”

Having acknowledged this difficulty, NIOSH/ORAU, nevertheless, provide the default geometries that are based on compensability and further differentiate “worker” from “supervisor:”

<u>Claim Status</u>	<u>Job Category</u>	<u>Exposure Geometry</u>	<u>Percentage</u>
Likely non-compensable	All	AP	100%
Compensable–workers	All	AP	50%
		ROT	50%
Compensable–supervisors	All	AP	50%
		ISO	50%

In the absence of a technical basis for these assumptions and the trivial differences between rotational versus isotropic DCF values, this level of detail (that is also time consuming to the dose reconstructor) is difficult to justify in context of so many other and more significant uncertainties that affect dose reconstruction.

Example #3. Perhaps the most improper attempts to introduce levels of “precision” involve occupational medical doses in Section 3.0 of the TBD. For example, Table B-2 cites the following period-specific doses for the testes in behalf of a PA chest x-ray:

<u>Dose to:</u>	<u>Year of X-ray</u>		
	<u>1950–1970</u>	<u>1971–1985</u>	<u>1985–1999</u>
Testes	0.983 mrem	0.00044 mrem	0.00033 mrem

Given the variabilities of kVp, current, distance, collimation, filtration, and the worker’s physical dimensions, organ doses of less than 1 **microrem** lack credibility and relevance (to dose reconstruction) by several orders of magnitude.

In summary, quality assurance is an important part of maintaining a consistent and defensible dose reconstruction program. NIOSH/ORAU 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 should be corrected or explained. The level of precision within the TBD should reflect the precision of the original data from which values are derived.

5.14 Issue 14: Subcontractors and Construction Workers

NIOSH has reserved Section 6 in the SRS TBD to specifically address trades workers. They have also been in the process of evaluating issues related to construction workers, and are currently working with the Center to Protect Worker Rights (CPWR) to further evaluate this situation. SC&A mentions this issue here to remind NIOSH of its importance at both SRS and

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other DOE sites. NIOSH/ORAU should also consider including the decontamination and decommissioning workers in this special evaluation.

Subcontractors and construction workers pose special dose reconstruction issues because they are typically mobile, wear temporary dosimetry for each visit, and are difficult to track down for follow-up bioassay.

The site profile should consider including more data and analysis to address dose reconstruction for unmonitored radiation workers, including the following:

- Off-normal practices and exposures
- Workers who performed construction work in non-radiological areas, but who also worked inside or outside radiological work areas without being monitored
- Group dose estimates from one or a few badges and validity for individual exposures, with complex issues relating to geometry of badge for the individuals who were monitored and the geometry of the individuals monitored relative to non-monitored individuals
- Site data describing who was and wasn't monitored for various periods

Each DOE site relied on construction workers to build the facilities currently in use. DuPont, who initially operated the SRS, had a separate division known as DuPont Construction. As time progressed and the site was completed, the scope of work for construction workers changed. The site maintained trade support to a limited extent onsite; however, specialized trades were obtained from the local union hall. These trades were typically brought in under subcontractors to provide specialized maintenance support or to provide staff augmentation. There were thousands of construction trades employees brought in for either temporary or regular employment. The period of time and the number of times an individual was assigned to support work at the SRS depended on the type of work to be completed. Subcontractor and construction trades continue to be used today, especially to support decontamination and decommissioning projects, as well as building new facilities to support changes in site missions.

There are several issues associated with subcontractors and/or construction trade workers that should be addressed, including the following:

- Monitoring periods are based on the length of the contract and the tasks performed rather than an established frequency, leading to more uncertainty in recorded and missed dose
- Exposure scenarios tend to be of short duration with high potential for exposure
- This particular workforce tends to be utilized in several areas of the site, as opposed to being assigned to a single area

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- Due to their specialty as maintenance support, construction trades tend to perform more hands-on work
- The monitoring program for subcontractors and construction trade workers may be incomplete, especially in the early years of operation
- Construction worker employment and dosimetry records are stored separately from production worker records
- Subcontractors and construction trade employees are not always available for follow-up monitoring when positive bioassays are observed
- Coworker data may not be applicable to these individuals

NIOSH should carefully evaluate the unique conditions associated with subcontractor and/or construction trade workers, which make clear distinctions among construction workers, maintenance workers, trades workers, employees who worked outdoors, and perhaps other categories of outdoor workers and subcontractors.

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

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

6.1 Satisfying the Five Objectives

The SC&A review procedures, as approved by the Advisory Board, require that each site profile be evaluated against five measures of adequacy: completeness of data sources, technical accuracy, adequacy of data, site profile consistency, and regulatory compliance. **The SC&A review of the Savannah River Site profile finds that the profile generally satisfies these objectives, although shortcomings and potential issues exist of varying significance that need to be addressed.** Many of the issues involve lack of sufficient conservatism in key assumptions or estimation approaches, incomplete site data or incomplete analysis of that data, or incomplete reflection of operational or dosimetry history. Key issues are summarized below and in Table 6.1, which provides a matrix representation of the identified issues sorted according to the SC&A findings and observations. Detailed evaluation of these issues are provided elsewhere in this report.

6.1.1 Objective 1: Completeness of Data Sources

The breadth of data sources used as a basis for the SRS site profile is evident in the 274 reports cited as references, including a number of authoritative historical documents dating back to start of operations in the early 1950’s. As noted in Section 4.0, NIOSH effectively compiled facility specific information and proceeded to characterize, by each of 30 facility categories, the types and relative importance of various radionuclides that may have contributed to internal and external exposures.

However, data gaps were evident for a number of key aspects of the TBD. While substantial atmospheric source term data were derived to estimate outdoor occupational exposures to unmonitored workers, the treatment of episodic and ground level releases were not given as rigorous an evaluation and remain in question. For the Hypothetical Intakes (“high-five approach”), used as a maximizing approach for likely non-compensable claims (Issue 1), it was not possible to independently validate whether the approach considered chronic intakes (as well as acute intakes) because NIOSH was unable to provide access for SC&A to individual bioassay data (rendering this question unauditably). Likewise, for Tank Farm workers,

Table 6.1 Issue Matrix for the Savannah River Site Technical Basis Document

Description	Issue Classification	Obj. 1: Completeness of Data	Obj. 2: Technical Accuracy	Obj. 3: Adequacy of Data	Obj. 4: Site Profile Consistency	Obj. 5: Regulatory Compliance
Issue 1: High-Five Approach	Finding	X	X	X	X	X
Issue 2: Environmental Modeling	Finding	X	X		X	
Issue 3: Recycled Uranium	Finding		X		X	
Issue 4: External Beta/Gamma Dose Adjustments and Uncertainty	Finding		X		X	
Issue 5: Neutron Dosimetry	Finding		X			
Issue 6: Tank Farms Workers	Finding	X	X	X		
Issue 7: Internal Dosimetry	Finding		X		X	
Issue 8: Special Tritium Compounds	Observation		X			
Issue 9: Internal Dose: Transplutonium and Non-military Rad Production	Observation		X			
Issue 10: Incidents and High-Risk Jobs	Observation	X	X			
Issue 11: Early Worker Radiological Monitoring Completeness	Observation	X	X	X		
Issue 12: Availability of Additional Sources and Data	Observation	X				
Issue 13: Quality Assurance	Observation					QA
Issue 14: Subcontractors/Construction Workers	Observation	X	X			

NIOSH's site data evaluation appears to be incomplete with regard to exposure conditions and uncertainty estimates because primary data sources were not reviewed. For early SRS workers, the site profile lacks a comprehensive evaluation of the early monitoring program with respect to

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its consistent application, adherence to procedures, and recordkeeping, all of which hold significant implications for reconstructing doses for unmonitored workers during the early years. Similar gaps in data availability were noted for individual neutron 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, recycled uranium, and environmental releases.

6.1.2 Objective 2: Technical Accuracy/Claimant Favorability

The SC&A review found more issues related to satisfying this objective than any other, most related to the technical basis (or lack thereof) for assumptions and analytic approaches used in the TBD. 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 non-claimant favorability. Internal dose did not include all radionuclides to which individuals were exposed, including trace radionuclides in recycled uranium and radionuclides from special campaigns. For modeling of airborne radionuclide releases, one potentially significant issue is the non-conservatism of the standard Gaussian model used in the TBD where it pertains to “non-standardized” 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 1×10^{-9} per meter may not be claimant favorable by 3 to 4 orders of magnitude. For external dose, the indicated adjustment factors and uncertainty analysis do not take into account the total uncertainty associated with the beta/gamma and neutron dosimetry program, which likely results in underestimation of external dose. For internal dose, the assumption that inhalation is the only pathway for internal exposure at SRS is questionable, given evidence that work practices and large particle sizes may have had a role in making ingestion a contributing pathway. Also, another overarching issue is the use of the geometric mean when using surrogate data. NIOSH should consider using the 95th percentile values when using surrogate data to support dose reconstruction for workers that were not monitored.

6.1.3 Objective 3: Adequacy of Data

The TBD generally benefited from the internal and external monitoring program and improvements in dosimetry and radiobioassay technologies. As that operational history began in the early 1950's, the SRS radiation-monitoring programs benefited from experience elsewhere in the then AEC complex, with established technologies such as ORNL film badges and NTA film being employed from the onset. NIOSH has reflected the available databases effectively in its site profile, albeit SC&A identified several key issues.

Questions regarding data adequacy, where they arise in the SC&A evaluation, have largely focused on three aspects of the TBD: (1) the adequacy of the SRS Internal Dosimetry Registry (IDR) as a basis for the “high-five approach,” when evaluated against incident and operational history, and individual bioassay data; (2) the questionable adequacy of dose data for the F- and H-area Tank Farm workers given the lack of primary records review, including key operations

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and incident documentation for early workers; and (3) the adequacy and comprehensiveness of available dosimetry data for the early monitoring program where unmonitored workers are more likely to have had “missed dose” due to inconsistent monitoring in the field (due largely to decentralized health physics management at that time).

6.1.4 Objective 4: Consistency Among Site Profiles

As noted previously in Section 4.0, while SRS and Hanford had similar missions, marked distinctions existed and continue to exist in facility design, operations, operational history and radiological practice. NIOSH has appreciated this distinction and tailored its TBD assumptions and analytic approaches to the unique histories and conditions at the two sites, while mirroring those assumptions and approaches where justified. However, the SRS profile or TBD was the first one generated by NIOSH and suffers from shortcomings that were remedied in subsequent site profiles, such as Hanford, leading to some of the apparent inconsistencies.

Attachment 6 provides, in tabular form, an evaluation and comparison of the default assumptions for each element of exposure (i.e., occupational medical dose, internal dose, external dose, and environmental dose). The lapses in consistency noted by SC&A include inconsistent methodologies and assumptions regarding external, internal, and environmental dose for almost identical monitoring and exposure conditions at SRS and Hanford; the lack of treatment in the SRS site profile of issues (e.g., recycled uranium) that were addressed in the Hanford TBDs; and inconsistencies between the assumptions and approaches indicated in the SRS profile versus that prescribed by ORAU DOE complex-wide procedures (e.g., ORAU-OTIB-001, ORAU-OTIB-008 and ORAU-OTIB-0010). One example is the use of inconsistent dosimeter adjustment factors. Of particular concern is the lack of consistency in the components included in internal (e.g., inhalation and ingestion) and environmental dose (e.g., soil resuspension, large particle depositions on skin, airborne emissions, liquid effluents) between the two site profiles. These are all detailed in this report in the SC&A findings and observations under “site profile consistency.”

6.1.5 Objective 5: Regulatory Compliance

NIOSH has effectively complied with the hierarchy of data required under 42 CFR 82 and its implementation guides with one notable exception. SC&A was unable to ascertain whether the relative intakes used for the “high-five” Hypothetical Intake were in conformance with the requirements of 42 CFR 82.2, rendering this question unauditably at this time.

6.2 Usability of Site Profile for Intended Purpose

SC&A has identified seven criteria that reflect the intent of the EEOICPA, the Final Rule, and the regulatory requirements of 42 CFR 82 for dose reconstruction. Due to the fact that the purpose of a site profile is to support the dose reconstruction process, it is critical that the site profile assumptions, analytic approaches, and procedural directions be clear, accurate, complete, and auditable (i.e., sufficiently documented). The following seven criteria are directly applicable

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and relevant to SC&A's evaluation of the SRS site profile from the standpoint of ascertaining whether it can adequately do so (with key words/phrases underlined):

- Objective #1: Determine the degree to which procedures support a process that is expeditious and timely for dose reconstruction.
- Objective #2: Determine whether procedures provide adequate guidance to be efficient in select instances where a more detailed approach to dose reconstruction would not affect the outcome.
- Objective #3: Assess the extent to which procedures account for all potential exposures and ensure that resultant doses are complete and based on adequate data.
- Objective #4: Assess procedures for providing a consistent approach to dose reconstruction regardless of claimants' exposures by time and employment locations.
- Objective #5: Evaluate procedures with regard to fairness and the extent to which the claimant is given the benefit of doubt when there are unknowns and uncertainties concerning radiation exposures.
- Objective #6: Evaluate procedures for their approach to quantifying the uncertainty distribution of annual dose estimates that is consistent with and supports a DOL probability of causation estimate at the upper 99% confidence level.
- Objective #7: Assess the scientific and technical quality of methods and guidance contained in procedures to ensure that they reflect the proper balance between current/consensus scientific methods and dose reconstruction efficiency.

In several sections, this review has identified that the writing style is not always clear and unambiguous, especially Chapters 4 and 5. The document is not complete, and it is difficult to understand. For example, with reference to ORAUT-0TIB-0001:

- Data for Tables 1 and 2 are not referenced and it is not possible to verify the data or to relate it to specific jobs.
- There is insufficient information to reproduce the relative intakes on Tables 3 and 5.
- It is difficult to accept the reasons to use the ICRP 30 calculated intakes as the worst case estimates, when it underestimates the ICRP 68 calculated intakes for several radionuclides.
- The calculation of annual organ doses from hypothetical intakes for Table 2 nuclides is not clear.
- The use of surrogate radionuclides is not well explained, including the use of type F nuclides as surrogate to types M and S. The choice of surrogate nuclides is very subjective as are the comparisons on Tables 3-10.

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The TBD does not mention the process SRS used to derive intakes in the IDR, including the assumptions used by SRS to calculate the intakes other than solubility. This information is important in establishing the claimant favorability of the approach. In addition, the procedure for the assignment of tritium dose should be complemented by the information contained in the document ORAUT-OTIB-0003.

The procedures described are difficult to understand and reproduce. There is no clear indication regarding which data were used to derive the hypothetical intakes. Our review could not determine if the procedure adheres to the hierarchical process as defined in 42 CFR 82.2. From ORAUT-OTIB-0001, Table 3-10, one can speculate that urine bioassay was used for assigning intakes for Pu, Am, U, Np, Cm, and Sr, and in vivo methods were used for Co and Cs. There is no mention of the method that was used to calculate intakes from Ce, Cf, Nb, Ru, Zn, and Zr.

The document is not consistent with other procedures that are part of the hierarchy of procedures employed by NIOSH for dose reconstruction, including:

- The standard procedures in 42CFR 82 and OCAS-IG-002 recommend the use ICRP new models for interpreting bioassay results. However, in this document, intakes were calculated using ICRP 30 models.
- The approach to calculate maximum dose estimates for SRS is different from the approach recommended for other DOE similar facilities (ORAUT-OTIB-0002) and not always claimant favorable. For certain radionuclides, such as uranium, the maximum plausible intakes, based on a fraction of the maximum permissible body burden, are 5000 nCi of U-234 and 500 nCi of U-238, which are much higher than the values recommended in ORAUT-OTIB-0001; i.e., 105.4 nCi of U-234 and 20.95 nCi of U-238.

The document addresses employees with no detectable activity in their monitoring results and employees who were not included in the bioassay program. The document does not explicitly refer to unmonitored periods of exposure and to missing dosimetry data. For tritium, missing dosimetry data is addressed in the document ORAUT-OTIB-0003.

We also reviewed the degree to which this procedure is claimant friendly in instances of missing data, in instances of unknown parameters effecting dose estimates, and in instances where the claimant was not monitored. The procedure described in the document is not clear, and does not apply the same claimant-favorable approach for all radionuclides for the following reasons:

- The use of ICRP 30 models, instead of the current ICRP models, to calculate the hypothetical intakes are not claimant favorable for most radionuclides.
- The procedure recommended for the use of surrogate radionuclides does not appear to be claimant favorable.

For the purpose of compiling data needed to reconstruct internal doses based on historical operations, NIOSH compiled an enormous amount of data describing the radionuclides and operations at the various facilities and their associated processes. To almost a fault, NIOSH provides guidance to dose reconstructors on how to navigate through the complex mix of

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radionuclides required to reconstruct historical internal exposures to workers. Notwithstanding this achievement, there are opportunities for improvement in the data sets and instructions to the dose reconstructors with respect to reconstructing internal exposures.

Chapter 5.0 of the TBD on external dosimetry was difficult to understand and is incomplete. The mingling of beta/photon and neutron exposure, which require different analysis techniques, was one source of confusion. Considerable amounts of data from independent studies were provided in this chapter, which were ultimately dismissed as inconsequential to dose reconstruction. In order for the dose reconstructor to effectively apply the appropriate external dose, several sections of the TBD had to be consulted. The TBD is poorly structured, ambiguously worded to the point of being evasive, and confusing in providing procedural guidance. If used as the sole guidance document, multiple readings are likely required to decipher the intent of the section and to gain necessary confidence for its use as a guidance document.

6.3 Unresolved Policy or Generic Technical Issues

A number of issues were identified that are common to both the Savannah River Site and Hanford site profiles, and in some cases, represent potential generic policy issues that transcend any individual site profile. These issues may involve the interpretation of existing standards (e.g., oro-nasal breathing), how certain critical worker populations should be profiled for historic radiation exposure (e.g., construction workers and early workers), and how exposure itself should be analyzed (e.g., treatment of incidents and statistical treatment of dose distributions). Some of these issues are acknowledged by NIOSH (e.g., construction workers, oro-nasal breathing) and are currently being addressed to ascertain whether and how all site profiles should reflect these considerations. The following represent those issues identified in the SRS site profile review that in SC&A's view represent transcendent issues that need to be considered by NIOSH as **unresolved policy or generic technical issues**.

- (1) An absence of direction on applicability of the TBD and/or TIBs to individual dose reconstructions.
- (2) Mobility of work force between different areas of the site. Site expert testimony that many workers moved from one plant to the next is a complicating factor. Establishment of an accurate worker history is crucial in such cases. This will be especially difficult for family member claimants.
- (3) Statistical technique and subsequent application of the data to individual workers.
- (4) Assessment of dose from impurities and/or daughter products in radioactive material received and processed at sites.
- (5) Assumptions on solubility, oro-nasal breathing, and ingestion.
- (6) Direction with respect to consideration of incidents and high-risk jobs in individual dose reconstructions.

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- (7) Availability of monitoring records for subcontractor and/or visitors and potential exposure while working on or visiting a facility.
- (8) Assessment of dose to construction workers and other early workers.
- (9) Unique exposure conditions for decontamination and decommissioning workers.

The relative impact of each of these items on dose reconstruction is site specific and requires independent evaluation in each TBD.

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ATTACHMENT 1

OUTSTANDING RECORDS REQUESTED OF NIOSH AND THE SAVANNAH RIVER SITE TO SUPPORT SRS SITE PROFILE REVIEW

SC&A conducted onsite and offsite interviews of current and previous workers from August 23–26, 2004. As a follow-up to these interviews, a request for documentation was made on September 9, 2004 and submitted to NIOSH for referral to the Savannah River Operations Office (SROO). After not receiving requested information, SC&A submitted a second request in November 2004. Several of the documents requested from SRS were provided to SC&A at the Advisory Board meeting in St. Louis, Missouri, on February 7–9, 2005. An additional shipment of records was received from the Savannah River Site on March 2, 2005. Several of the documents copied during the visit to SRS and subsequently requested following the visit have not been received as of the date of issuance of these reports. A list of outstanding records, which may be useful in SC&A review and in any future modifications made to the SRS site profile by NIOSH/ORAU is as follows.

- Written Responses to SRS Medical Interview
- Inter-Office Memo - Medical Radiation Doses
- Inter-Office Memo - EED X-Ray Facility, 723-A (Room 200)
- Memo - Use of Diethylenetriaminepentaacetic Acid (DTPA) by DOE and it's Predecessor Agencies
- Inter-Office Memo - DTPA Chelate Follow-up
- Follow-up To Cuprimine Evaluation (DTPA Administration)
- C.N. Wright's personal files available on microfilm, including information on studies from the Personnel Meters group, instrument studies, calibration, and x-rays
- SRS Worker Dose and Contamination History
- Radiation Safety Functional Appraisal of the SRS External & Internal Dosimetry Programs
- Memo - Uniformity of Radiation Fields - 736A Calibration Wells
- Radiological Physics - Radiation & Radiation Dose Measurement - C.N. Wright
- Letter - Panasonic 809 Dosimeters
- Algorithm for Panasonic 802 and 809 Dosimeters for the Rocky Flats Dosimetry Badge
- Notes On Current Neutron Dosimetry Status
- Letter - Luminescent Dosimetry for Personnel Monitoring
- Neutron Film Monitoring Techniques

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- Response To The Independent Assessment of the SRS Internal Dosimetry Program External Dose Analysis
- Radiobioassay and in vivo count data utilized to determine the high-five intakes
- Technical Position, Compliance with 10 CFR 835 Appendix D for Pu-242 Solutions in HB-Line
- Type B Accident Investigation Board Rpt. of the Plutonium Intake, Between - Aug. 4, 1996 - Feb. 10, 1997 By a Crane Operator at the SRS F-Canyon
- Letter to Mr. Hekman: Type B Investigation into the F-Area Unusual Occurrence
- Letter to Mr. Schwallie: Type B Incident Investigation Into The F-Area Unusual – Occurrence
- Attachment 1 - Inter-Office Memo - Type B Investigation of Potential Overexposure of F-Canyon Operator
- Gollob Analytical Service - Radioassay Analysis
- Inter-Office Memo - Radon Dose Measurements
- Letter - RC & HP Response to Verbal Inquiries
- Inter-Office Memo - Dose From Radon
- Inter-Office Memo - Radon
- Special Tritium Compounds In Rust and Dust At SRS
- Radiological Control Program for Special Tritium Compounds
- Memo - Comments on the Draft DOE Handbook on Radiological Control Programs for Special Tritium Compounds
- Tritium Panel Draft Compilation Paper
- Inter-Office Memo - SRS "Planned" Environmental Stack Radioactivity Releases
- Memo - Film Badge Survey of SRP Background
- Letter - Radiation Safety Considerations for Work In Lower Pen Branch (U) (Rev. 1)
- Letter - Radiation Safety Considerations for Forest Service Personnel
- Environmental GAMMA Dose Measurements In Cities and Towns Near SRP
- Memo - Descriptions of Historical Record HP Instruments
- Progress In Health Physics Instruments
- Inter-Office Memo - Calibration of Ionization Chambers Used at SRS for Measuring Tritium in Air
- Memo - Argon 41 Radiation Measurement Problem
- Memo - Kurz Alpha CAM Comments

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- General Specifications - Uranium Solidification Facility Alpha Constant Air Monitors
- Memo - Alpha CAM Evaluation (RTA-199-HP), Victoreen Model 758
- Memo - Alpha CAM Evaluation (RTA-199-HP), Eberline Model Alpha – 6
- Memo - Evaluation of Alpha Constant Air Monitors
- Memo - Impactor Air Samples, B-Line
- A Continuous Monitor for Prompt Detection of Airborne Plutonium - June 1964
- Memo - Proposed SRS Personal Air Sampling Program
- Inter-Office Memo - QA Record Requirements
- Annular Kinetic Impactor - Works Technical Department Data Record
- Memo - DP-188, A Continuous Monitor for Airborne Plutonium
- A Continuous Monitor for Prompt Detection of Airborne Plutonium - May 1963
- Operational Experience With Kanne Chambers - Works Technical Dept. Data Record.

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ATTACHMENT 2

KEY QUESTIONS FOR NIOSH/ORAU REGARDING SITE PROFILE DOCUMENTS

SC&A submitted questions relating to the Savannah River Site Profile on April 30, 2004. NIOSH provided responses in writing during the May 11-12, 2004 meeting in Cincinnati. The questions and responses from this exchange are provided below.

General

1. Are there representative examples of how site profile data are being used in dose reconstruction? More specifically, any that relate to the Savannah River Site profile? Do there exist specific NIOSH policies or procedures governing how site profiles are to be used in this manner and how they are communicated? What quality assurance process is in place for NIOSH to assure that ORAU's profile development process is adequate?

Response: (JJF) The SRS Site profile contains attachments that summarize approaches to assess environmental, medical, internal and external sources of worker dose using the respective DOE individual worker records regarding external dosimetry records, internal dosimetry records, diagnostic (i.e., occupational x-ray) dose records, incident investigation reports, and other methods of analysis, the relevant ORAU guidance and the dose reconstruction review process will illustrate the extent of the issues considered in dose reconstruction.

2. Understanding that ORAU did not rely on "raw" data and documents to prepare the Site Profiles, did ORAU perform a screening check of databases and of the secondary documents in order to ensure that the secondary documents were a sound basis for the Site Profiles? Specifically, did ORAU review the kinds of documents and data bases mentioned in the SC&A document request made to the NIOSH Advisory Board on April 21, 2004?

Response: (JJF) Assuming that "raw" data represent a calibrated laboratory evaluation that can be related to radiation exposure, it is not entirely accurate that ORAU did not use raw data to prepare the site profiles. For example, the results of individual processed dosimeters and bioassay results were used in the development of probability distributions of worker dose and distribution parameters. The DOE reported dose results for each worker (i.e., claim), as adjusted for identified technological limitations or missed results, are a primary input to the Interactive RadioEpidemiological Program (IREP). The basic types of information used to develop the site profiles were generally associated with: 1) site or facility type-specific workplace radiation type, energy and exposure geometry field characteristics, 2) administrative practices such as policies concerning selection of monitored workers, handling low dose results, performing occupational related medical x-ray examinations, etc., 3) technological characteristics of the monitoring methods with regards to radiation type and spectral characteristics and 4) calibration practices that relate laboratory to workplace conditions.

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(BAN) As a further explanation for the Hanford environmental exposures, there were two major sources of information – the Hanford meteorological records for the last 25 years and the Hanford Environmental Dose Reconstruction (HEDR) project records of site operations and effluents [and those derived from them, like the Radiation Assessment Corporation (RAC) source term report]. The Hanford meteorology records are very slightly-processed raw information. The HEDR (and RAC) source term information has been extensively peer reviewed, and some published, and is of generally acknowledged very high quality. The RAC report content compares well with the HEDR results.

(DEB) Documentation of 50 years of operation at major, complex DOE sites at the level of detail desired for the technical basis document is hard to come by. Authors use whatever sources of information they can acquire. In the case of bioassay at SRS, much of the information came from the existing SRS Internal Dosimetry Technical Basis Document and from A History of Personnel Radiation Protection Dosimetry at the Savannah River Site. In the case of mixed fission product urinalysis in the early years, it was assumed that the SRS procedure was similar to a Hanford procedure at the time. Bioassay reporting levels were evident by simply looking at the recorded values for many workers over the years.

3. ORAU is using worker dosimetry and bioassay data as if they were basically sound, with some gaps to be filled in. It is not clear, from the standpoint of internal policy and implementation, how dose reconstructors are addressing some obvious questions that arise with this approach, such as the following:
 - a. Is ORAU using this assumption across the board for all DOE sites, or is it examining some dose records of individual workers at each site and making a site-by-site determination?

Response: (JJF) The general focus is to examine site specific radiation fields, administrative practices, technology and calibration parameters that may impact systematic bias as relevant in site-wide trends in collective dose. Bias adjustments to dose (as described in the site-specific profiles) are used to arrive at claimant-favorable estimates. Probability analyses are performed to arrive at reasonable, claimant-favorable uncertainty estimates.

For example, the evaluation of recorded external dose is based on selecting the radiation quantities used in the current DOE Laboratory Accredited Program (DOELAP) performance tested dosimetry systems and working back through time historically to examine potential impacts of changes in dosimetry systems, organizations performing dosimetry, etc. Bias adjustments are made for identified technological limitations (such as radiation type or energy response), missed dose, etc., and the uncertainty estimated from the overall dosimetry information. The estimated radiation type, dose fraction, worker dose and uncertainty are primary considerations involved in dose reconstruction and calculation of organ dose.

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These steps are an integral component of the dose reconstruction process described in OCAS-IG-001. Supporting analyses are presented in the site profiles to address subsets of workers, such as those with recorded neutron dose, skin dose, etc., to identify limitations to be considered in dose reconstruction.

(DEB) As much as possible, bioassay MDAs, decision levels, and reporting levels are site-specific. A random sampling of claims records are reviewed to try to spot issues that the DRs might encounter that are not covered generically in the TBD. DRs also call the TBD author when they encounter something in the record not covered in the TBD. Limitations in the bioassay data, such as not being adequate to detect all radionuclides in mixtures, are pointed out in the TBDs when known. Biases in bioassay data are pointed out when known, e.g., instructions to ignore a set of chest counts for Pantex workers because of documented bias.

- b. Has ORAU examined whether the assumption of basically sound dose records (with gaps) is valid for all periods for which dose calculations are being made?

Response: (JJF) A primary objective of the site profiles is to assess the historical integrity of the available DOE dose records. Certainly, it is recognized that current technology, radiation protection guides, and workplace conditions have substantially changed compared to historical conditions, and that the historical situation is generally of most relevance to NIOSH dose reconstruction for individual workers. The site profiles describe monitoring policies, record keeping practices, monitoring technology, detection limits, etc., and the available worker exposure records.

- c. How is ORAU taking into account studies and worker testimony regarding records that may be deeply flawed for various reasons (such as putting badges in back-pockets, when a dose threshold is reached)?

Response: (JJF) A part of the claimant interview process concerns the identification of a worker's supervisors and co-workers to allow confirmation of various aspects of exposure monitoring, such as in this example of a potential practice that is undoubtedly in violation of site monitoring policies and worker training. For workplace conditions with significant exposure potential there is typically the presence of dedicated radiation monitoring (i.e., radiation technologists) staff with the objective to assess workplace conditions, monitor exposure, ensure radiation safety practices are followed and control the dose received by workers. If the situation as described actually occurred, the dose reconstruction process is to provide a claimant-favorable estimate of the actual dose received. There are other pertinent details such as the type and location of the cancer, when the exposure occurred, the extremes of the potential dose based on the work performed, etc.

(DEB) Concerning internal dose, the TBDs provide instructions on how to account for missed bioassay or unmonitored workers. Task 5 has also generated procedures for handling these situations.

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- d. How are problems of the geometry of exposure being dealt with, especially for periods for which only one badge was worn? What other obstacles, uncertainties or similar challenges have been identified with respect to ascertaining doses in circumstances where data is incomplete?

Response: (JJF) The respective parameters of radiation types, spectral energies, and exposure geometries all represent potential problems in dose reconstruction. This is considered in assessing radiation response characteristics of the respective dosimetry systems. These problems were well-recognized historically and there are some recognized historical limitations such as for neutron radiation. The use of a single whole body badge is typical in most all situations, with the caveat that pocket ionization chambers were commonly used even for sites operations beginning in the 1940s. Records of portable radiation protection instrumentation, although certainly the primary workplace method of exposure measurement, are generally not available and cannot easily be used in dose reconstruction. There have been laboratory studies of dosimeter response characteristic to selected photon radiation sources at various irradiation angles, and for dosimeters mounted on anthropomorphic phantoms rotated in the radiation field.

4. The SRS and other DOE site profiles have sections on “occupational environmental dose” that refer to “workers outside of the facilities.” From the standpoint of worker categorization:

- a. How are “outside workers” defined?

Response: (JJF) The potential for onsite (but outdoors) ambient external dose or intakes of airborne nuclides from environmental releases is one exposure pathway considered in the site profiles.

(BAN) From the standpoint of preparing the exposure/dose rate information, it is immaterial. For the Hanford TBD the calculated “dose rate fields” and “concentration fields” are applicable to any person outdoors.

- b. Are outside non-radiation workers distinguished from radiation workers and, if so, how?

Response: (JJF) Site policies for categorizing radiation or non-radiation workers, site policies for monitoring, etc., is a consideration in the preparation of the site profile. An objective of the site profile is to assess the potential significance of the different exposure pathways, and, as necessary, to assign a claimant-favorable dose if no dose was measured or to adjust, as necessary, measured doses.

- c. Does this refer to outside radiation workers, non-radiation workers, or both?

Response: (JJF) The ORAU approach has been to focus on the potential exposure pathways and to assess dose to workers (monitored or not) as appropriate. Certainly, if

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there is essentially no likelihood of a significant ambient environment dose to a monitored worker then this is useful information. Likewise, there have been analyses of the potential for a systematic negative bias in recorded worker external dose because control dosimeters were located at onsite control points and the results of these dosimeters used to "background" compensate the worker dosimeter measured dose. Fundamentally, the objective has been to identify and evaluate all significant sources of potential bias.

5. The SRS and other DOE site profiles (Hanford, Rocky Flats) use source terms intended for off-site populations.
 - a. Does this produce the right set of radionuclides for on-site workers? Specifically, are the screening calculations used by RAC to determine the radionuclide list for detailed study also valid for on-site radionuclide screening?

Response: (BAN) Based on analyses of the HEDR scoping studies examining many additional radionuclides and potential exposure pathways, the included radionuclides account for well over 99% of the dose.

- b. How is the issue of the composition of recycled uranium being handled for various facilities and periods of operation?
- c. Does this approach yield the right pathways for on-site exposure?
- d. Do you think this captures the bulk of the dose on-site received by outside workers? And what is the basis for this? For instance, how does it capture open pan burning of Pu-contaminated solvents at SRS, which occurred in about the first two decades? How does it capture the shine from spills?

Response: (BAN) The process used will capture the bulk of the dose; however, it does not include shine from spills or intentional disposals (such as from disposals into the Hanford BC Trenches in the later 1950s). There are far too many small (and large) contamination incidents to be able to catalog and evaluate all of them.

- e. Is Gaussian dispersion modeling the right way to approach doses for outside on-site from source terms emanating from stacks?

Response: (BAN) Gaussian models work fine for the Hanford analyses; the distances between sites are large. The RATCHET model developed for the Hanford releases was used. It is a Lagrangian-trajectory, Gaussian-puff model. The grid was only 1 km, so it is better than most straight-line models in resolution.

- f. How are episodic releases to be treated for on-site exposures? What kind of modeling and source terms would be needed?
- g. Did you review the RAC source terms for accuracy and completeness?

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Response: (BAN) The RAC atmospheric source terms have been reviewed at Hanford and found to be acceptable. These comparisons have been discussed directly with the RAC author. The source terms are intentionally conservative, and overestimate the releases.

6. NIOSH and ORAU stated during the PACE local meeting in Richland that experience with reconstruction of doses for monitored workers will allow better reconstruction for unmonitored workers. What is the procedure that NIOSH and ORAU expect to follow to allow for the extrapolation from monitored to unmonitored workers? Specifically, what is the assumed relationship between the radiological conditions experienced by monitored workers inside facilities and those experienced by outside unmonitored workers who did radiological work?

Response: (JJF) The objective of the ORAU site profile is to understand site policies for selecting workers to monitor, the monitoring practices, technology, etc. The completeness of dose records for those selected to be monitored is an important consideration. Certainly if the policies, practices and technology pertinent to monitored workers are understood, then by implication upper bounds on potential dose to unmonitored workers can be estimated.

Savannah River Site Profile

1. Has ORAU gone back to basic dosimetry, environmental monitoring, and bioassay data, etc., to validate the adequacy and completeness of SRS worker profiles?

Response: (JJF) Basic data in the context of final individual dosimeter and bioassay results for individual samples have been used in the analyses. The dose reconstruction process considers whether a dose result is available for each routine exposure period. For example, if personnel dosimeter results are expected on a two-week exchange, a missed dose is assigned for any missing or "0" recorded dose using the minimum detection level.

Response: (BAN) For Hanford, dispersion patterns and concentrations were modeled using monitored ⁸⁵Kr results in the 1980's.

(DEB) DRs use bioassay data not site internal dose data to recalculate internal dose.

2. Has ORAU developed an overall list of basic assumptions regarding potential dose received for exposures to specific radionuclides in each major fabrication, separation, extraction, waste handling, etc. process performed in each building or location at SRS over the time span of SRS operations? Are such summaries available?

(DEB) To the extent possible, radionuclide source terms developed for specific facilities. Assumptions about radionuclides potentially present but not directly monitored in bioassay were incorporated into the TBDs so DRs would account for the unmonitored radionuclides, e.g., Pu-241 and Am-241 in plutonium mixtures; Y-91, Cs-137, Ru-106

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and others in early fission product urinalyses. In making these assumptions the guiding principle was “plausible but claimant-favorable.”

3. What source data were used to perform the validation process described in question 1 above and are these data sources digitized for review or were hardcopy records available for review? Can such data sources be made readily available to SC&A?
4. Are there records available for and have they been reviewed by ORAU for each major SRS accident, off-normal event, unusual exposure spike, contamination spill, major radionuclide release etc., and were assumptions adjusted and/or will assumptions and doses assigned be adjusted based on such reviews?
5. 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?

Response: (DEB) Gene Rollins will have to answer the first question for SRS. Hanford airborne releases are well documented. Internal doses are determined from bioassay data with adjustments for unmonitored radionuclides. “Potentials for inhalation” are accounted for by estimating missed dose or accounting for unmonitored periods. Ingestion is not usually considered at major DOE sites (is important at AWEs), 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. Some bioassay techniques are well documented, other not. The TBD author uses whatever information he/she can get. Bioassay uncertainties are generally not well understood or documented until recent times, say 1980s or 1990s. However, uncertainty in internal dose is usually most heavily influenced by things other than the bioassay analysis uncertainty; such as, intake date and intake mode (acute or chronic), inhalation absorption class, particle sizes, whether the worker followed urine collection instructions, and day-to-day variability in excretion patterns.

6. Has ORAU developed a complete picture of operating parameters, procedures or protocols that might increase or decrease worker dose potential and are these well documented and utilized in determining worker or group dose?

Response: (JJF) SRS has prepared an historical analysis “A History of Personnel Radiation Dosimetry at the Savannah River Site” (WSRC-RP-95-234) which is similar in scope to the Hanford reports of historical dose. These reports describe the scope of the SRS program to monitor and measure worker dose since the beginning of operations. SRS has also been the subject of epidemiologic studies of workers exposure to radiation and these records are maintained by ORAU.

7. Has the change in dosimetric techniques with each technique’s inherent uncertainty been adequately taken into account in evaluation of worker’s recorded dose or in computing the worker’s missed dose?

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Response: (DEB) Changes in bioassay techniques, MDAs, and reporting levels are documented as best can be determined. Bioassay uncertainties are generally not documented (at SRS) but don't have much impact on the dose assessment. At Hanford, measurement-specific uncertainties have been reported for each result since the mid 1980s.

8. Have general overviews been developed of measured levels within each facility and for each type operation and have these been correlated with recorded dosimetry values for these operations/areas to see if they are in reasonable agreement?

Response: Question is not clear – what measured levels?

9. Has ORAU spent time trying to tie together various collections of measured data and its correlation with recorded dosimetry data (i.e., variations in average levels by year) and which groups are likely to be at the upper end of the ranges of exposure?

Response: Assumptions about groups that higher potential for exposure and groups with lower potential have been used at some sites, mostly AWEs and semi-AWEs like the Iowa Army Ammunition sites. This approach is especially important when bioassay data are lacking. Trying to tie exposure rates to production throughout at the sites like SRS or Hanford probably wouldn't prove much because of continually improving radiation protection practices.

10. Reactor and packaging areas are often associated with significant releases over the years. In order to review the potential for higher exposures, has ORAU taken a closer look at high pressure operations in packaging areas and accidental airborne releases via equipment and operator failures?

No answer was given to this question.

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ATTACHMENT 3

CONFERENCE CALL WITH NIOSH AND SC&A

General Discussion

The purpose of the conference call with the National Institute for Occupational Safety and Health (NIOSH) was twofold. First, the S. Cohen and Associates (SC&A, Inc.) team wanted to clarify the NIOSH/Oak Ridge Associated Universities (ORAU) position on items within the Savannah River Site (SRS) Technical Basis Document (TBD). Secondly, the SC&A team wanted to inquire about information not currently addressed in the technical basis document. The SC&A team wants to gain an understanding of the procedures for development of the technical basis documents. Is this process formalized? What is the rationale behind decisions made with respect to the TBD and what was included?

During the introductory comments it was suggested that the beginning of each site profile contain a "scoping statement" that will help the reader understand what worker exposure categories the site profile addresses and what worker exposure categories it does not address, and why. The scoping statement should also explain what supplements to the profile are planned or in preparation that will address issues that may be important, but are currently not addressed in the profile. Jim agreed that this would be helpful.

NIOSH encouraged the SC&A team to understand the intent of the current versions of the technical basis documents. The original version of the TBD was developed to address a majority of the cases, but not all the intricate issues which may arise. The TBDs are meant to be living documents which are periodically revised as new information or methods become available. The current goal is to evaluate only cases that between the TBD and the energy employee records have enough data to make a definitive decision. The TBDs are used in conjunction with energy employee records and not as a sole source for determining dose. In the event that there is a gap in the TBD, NIOSH can develop a Technical Information Bulletin (TIB) which provides additional guidance.

SC&A inquired as to the status of the request for source documentation used to develop the documentation. Jim Neton indicated that he was in the process of investigating getting authorized online access to the NIOSH database for approved SC&A team members. This was an issue he had to bring up with the Centers for Disease Control and Prevention. Access may require completion of training with respect to information technology policies and the Privacy Act. NIOSH will follow up with SC&A on the progress with respect to providing source data.

John indicated that the SC&A team had identified some major areas of concern including unauthorized practices, unmonitored workers with the potential for exposure, and the construction worker issue. The latter is currently under investigation by NIOSH, which is preparing a construction worker profile guide for dose reconstruction for this group of workers. Additional items were addressed by specific members of the team. The issues discussed are outlined below.

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Accidents and Incidents

SC&A Inquiry: The identification and integration of accident and incident data into the dose reconstruction were a concern of SC&A. How were accidents and incidents identified for a particular individual? How were these accidents and/or incidents incorporated into the dose reconstruction?

NIOSH Response: The evidence of accidents and incidents is available in the Personnel Radiation Exposure Record. In addition, they may be identified in the Computer Assisted Telephone Interviews (CATI) process. If the claimant indicates that there was an incident with no information in the Personnel Radiation Exposure Record, NIOSH will make additional records requests of the site. Incidents are considered in dose reconstruction.

SC&A Inquiry: Is the rationale that an incident database is not needed as part of the site profile based on the premise that DOE records contain the data needed in the worker dose records?

NIOSH Response: In many cases, the workers will state there was no incident, but the records do, in fact, identify incidents. It is possible for a claimant to say that there was an incident but no incident is identified in the records. There are not many cases when a worker says there was an incident and it was not in the dose record. 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.

SC&A Inquiry: How are incidents involving external exposure dealt with?

NIOSH Response: Most workers who were involved in external exposure incidents had film badges. Dose reconstructions for unmonitored workers haven't been done yet.

Comparison of In Vivo and In Vitro Bioassay Data

SC&A Inquiry: Is urine bioassay verified with in vivo counts when the information is available for the same worker? This was found to be very helpful at Fernald and assisted in addressing solubility issues.

NIOSH Response: A validation of this type is not done because the most claimant-favorable assumptions regarding solubility class are assumed.

Decontamination and Decommissioning Workers

SC&A Inquiry: With respect to the "Construction Workers" chapter under development for SRS, does this chapter address Decontamination and Decommissioning (D&D) workers?

NIOSH Response: Define D&D workers. (Clarification by SC&A: Workers involved in decommissioning facilities.) Construction workers like those who replaced heat

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exchangers were considered. The "Construction Workers" chapter concentrated on subcontractors that may not have been monitored by the site. It was felt that current D&D workers are monitored.

Early Workers and Outside Workers

SC&A Inquiry: How is NIOSH dealing with the many issues relating to early workers?

NIOSH Response: A lot of the early worker dose reconstruction has been postponed. However, if an early worker case that can be compensated is identified, the dose reconstruction is performed. These cases are difficult. One denial of an early worker that comes to mind was a cafeteria worker at Oak Ridge. He worked for 6 months and developed prostate cancer at age 70. A maximum dose assessment was done, and it resulted in a denial. Co-worker data will be used to see who were monitored versus who were unmonitored. This is the basic approach intended for early workers.

SC&A Inquiry: Will this result in a TIB?

NIOSH Response: This will likely be a revision to the technical basis document itself. Early Y-12 folks are in the process and the evaluations are close to being completed. There are a large number of claimants that were not monitored.

External Exposure Geometry and Angular Dependence

SC&A Inquiry: There are a number of external exposure geometry issues. How are partial body exposures which may result in a higher dose to the organ of interest than recorded on the film badge dealt with in dose reconstruction?

NIOSH Response: A technical information bulletin has been drafted addressing the partial body exposures from glove box work.

SC&A Inquiry: Hans referred NIOSH to pp. 97 and 98 of the SRS TBD. Both tables make reference to angular response. Angular dependence is very dependent on photon energy and can be great. The site profile acknowledges this angular dependency, but does not make a reference to how this is taken into account. The correction factors used are 1.119 prior to 1986 and 1.039 for 1986. This would not accommodate angular dependence. It appears that angular dependence is not accounted for in dose reconstruction.

NIOSH Response: Angular dependence can affect the film badge dose. An A-P exposure orientation is assumed as the calibrations were performed in this orientation. Other orientations, including rotational, showed a lower response depending on the energy of the incident radiation. If badges were calibrated using a rotational geometry, a rotational correction factor would need to be applied. For many organs, the DCF is a factor of two lower for rotational geometry.

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SC&A Inquiry: What about the factors on page 227? Do you further refine the doses to account for the adjustment for angular dependence?

NIOSH Response: The particular corrections listed here are accounting for many different factors. These are not explicitly rotational factors. Angular dependence is accounted for in using the A-P geometry primarily. The 1.119 has nothing to do with a rotational geometry adjustment.

SC&A Inquiry: At this stage angular dependence is accounted for with the use of A-P geometry?

NIOSH Response: A-P geometry is used and therefore we do not use an angular dependence correction factor. If you wanted to use an angular response and apply a correction factor, this would also be okay. It would give more or less the same result. Low photon exposure is most likely in the plutonium facilities where individuals perform glove box work. This exposure would be A-P.

SC&A Inquiry: Reproducibility is dependent on development time of film. Are there data for this? Were process procedures used that controlled time and temperature of the bath?

SC&A Response: Kathy indicated that she had access to film processing procedures for the 1950's and 1960's and would provide them. These procedures addressed film processing.

NIOSH Response: Early dosimetry was provided by Oak Ridge National Laboratory until 1953.

SC&A Inquiry: On page 97, it indicates that the original badge calibrations were performed in-air (no phantom). Did previous calibrations include radiation backscatter?

NIOSH Response: Film badges would have read higher because of backscatter. It is energy dependent. There is not much backscatter from low energy photons.

Hazard Ranking

SC&A Inquiry: There are certain classes of workers that have unique exposure conditions or more significant potential for exposure. Is NIOSH performing hazard ranking with relation to groups of workers?

NIOSH Response: This has not been done this yet; however, groups have been identified with special exposure circumstances.

SC&A Inquiry: How is the priority established for deferrals when there are many workers who have been waiting for a long time?

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NIOSH Response: There is no formal queue. Informally, it is based upon magnitude of claims.

Medical X-ray Exposure

SC&A Inquiry: A review of historical literature yielded significant information on photofluorography. Many studies were performed during the period when photofluorography was used. Historical documents indicate that there is an enhanced dose to the bone surfaces and bone marrow due to transition from the soft tissue to the bone surfaces (as much as 5 times higher). In the TBD the air dose and bone dose are 1 and 0.0605, respectively. Bone dose should be higher because of higher energy deposition in bone. This becomes an issue with respect to bone surfaces and bone marrow dose. The actual dose to marrow stem cells is equivalent to bone surface dose. Has this historical research been taken into account in determining the dose to the bone surfaces and trabecular bone marrow?

NIOSH Response: The SRS TBD does not reflect that; however, other TBDs do. We are preparing a program evaluation report for SRS claims that have been completed. We are aware of the photofluorography issue.

The source document used for medical x-ray exposure was ICRP 74. An average of 30 to 250 keV photons is used. This is one of the reasons for the change to ICRP 34 after SRS. So we are re-doing the SRS TBD. No bone cancers to date have been non-compensable. Since this is the case, it is uncertain whether it is worth looking into this.

SC&A Comment: The issue of soft tissue to bone interface is a crucial one. More energy is delivered at 6 to 9 cm depth than to the skin. This would include marrow cavity stem cell dose. If bone cancers are compensable, it may not make much difference here; however, it would also affect leukemias. Hans indicated he would provide NIOSH with his write-up with respect to medical x-ray dose and angular dependence of dosimeters.

Reactor and Packaging Areas

SC&A Inquiry: NIOSH did not provide a response to Question 10 on the list of SRS Questions previously submitted.

Reactor and packaging areas are often associated with significant releases over the years. In order to review the potential for higher exposures, has ORAU taken a closer look at high pressure operations in packaging areas and accidental airborne releases via equipment and operator failures?

NIOSH Response: This issue hasn't been investigated yet.

Recycled Uranium

SC&A Comment: The TBD did not address the assignment of dose from transuranics contaminants in recycled uranium. The workers were not monitored for these

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transuranics in the bioassay program. The number of affected individuals may have been significant.

NIOSH Response: TRU contamination in recycled uranium has not been covered. The “High-Five” approach would compensate for the lack of TRU bioassay data for non-metabolic cancers. No consideration to date has been given to the specific situation of TRU contamination in recycled uranium and the lack of bioassay. We are aware of this issue; however, we do not know the status of the work completed by ORAU.

SC&A Inquiry: What are the decision criteria for excluding the recycled uranium issue from the TBD?

NIOSH Response: We are currently focusing on monitored workers. We have been looking at workers who were monitored for Pu and other TRU. There is a possibility internal dose from TRU in recycled uranium was missed for those considered to be uranium workers.

Tritium Exposure

SC&A Inquiry: Workers with the potential for exposure to tritium were not always monitored. An example of an incident in April 1959 in the 100 Area at SRS that resulted in an estimated 37 rem dose was given. Is the air monitoring data a better indication of tritium exposure in this type of situation compared to the assignment of missed dose or the use of bioassay?

NIOSH Response: The general approach is to process monitored workers and assign generous missed dose. SRS had a strong monitoring policy and an intake of any significance would likely not be missed. There may be incidents that were missed based on no monitoring data.

SC&A Inquiry: Table 13 in the Internal Dosimetry Technical basis document and Figure 1 in the Tritium Information Bulletin described two slightly different approaches to assigning missed dose to workers with urine concentrations less than the reporting level (5 $\mu\text{Ci/L}$). Which approach is used to assign missed dose?

NIOSH Response: Both approaches are used. 355 mrem is the annual dose corresponding to excretion of 5 $\mu\text{Ci/L}$ every day throughout the year. This dose is assigned for individuals who were not assigned a dose. If bioassay data were available, the method was refined to reflect the process described in Figure 1. When the detection levels dropped, the process was further refined to reflect the new detection level.

Unmonitored Workers with the Potential for Exposure

SC&A Inquiry: Has NIOSH/ORAU performed any comparison studies between air monitoring and bioassay data?

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NIOSH Response: NIOSH does not have access to air concentration data. A comparison of this data would not be an accurate comparison as people wore personnel protective equipment which was intended to prevent uptake. Individuals in encapsulated suits would have low bioassay but air concentration would be high. It is difficult to line up individuals with air samples. Dose reconstructions on individuals who are not monitored but had the potential for tritium exposure at this time are not being completed at this time.

SC&A Inquiry: What approach will be applied to individuals who were not monitored for tritium but had the potential for exposure? In particular, will air sample or coworker data be used?

NIOSH Response: NIOSH is not to the point of reconstructing doses based on co-worker data yet. Not all assumptions have to be claimant-friendly. This is only appropriate for situations where there are two approaches to a particular issue. The preference is to use co-worker bioassay data rather than air monitoring data to assigned doses to unmonitored workers with the potential for exposure.

SC&A Inquiry: An example of a clerk located outside the radiological area adjacent to a source locker was given as an additional example of an individual who may have been exposed to a radiological hazard, but not have been monitored. Also, ventilation failures and potential exposure outside the controlled area were discussed. How will doses for these individuals be reconstructed?

NIOSH Response: These are very individualized situations, and there doesn't need to be a systematic approach to these situations. An alternative approach is to use co-worker data from monitored workers. NIOSH has not done this yet.

Unauthorized Practices

SC&A Inquiry: Is NIOSH considering unauthorized practices in dose reconstruction? For example, fish fries were held at Par Pond and workers routinely sat on uranium ingots at Fernald. The concern with respect to unauthorized practices was that unique exposure scenarios were present which would not necessarily be captured in routine monitoring programs. For example, workers who sat on ingots would likely have high gonad, bladder and prostate doses from external exposure which would not be reflected on the film badge.

NIOSH Response: NIOSH is not at the stage in the dose reconstruction requiring integration of these situations into the dose reconstruction process. Current investigations are underway regarding consumption of contaminated produce at various facilities. Other situations will be investigated in the future.

SC&A Inquiry: How are workers who were exposed in unusual ways evaluated? Were these situations reasonable for inclusion in the technical basis documents? How do these issues get put into the pipeline for investigation?

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NIOSH Response: These issues are dealt with as they are brought to the attention of NIOSH. Actions are taken on these issues in the form of a technical information bulletin or a revision to the technical basis document.

Visitor Cards

SC&A Inquiry: Are visitor badge doses integrated into dose reconstruction of workers at SRS? Temporary visitor badges were issued periodically to workers who were already on routine monitoring programs and the concern was that some of their monitored dose would be reflected on their temporary badge.

NIOSH Response: Visitor badge data are included in dose calculations if the individual radiation exposure data provided by the site includes this information. Missed doses are assigned for zero doses. NIOSH is assuming that the site is providing a complete set of exposure records.

Request for Specific Documents

A request was made by SC&A team members for the following documents.

- 1) "A History of Personnel Radiation Dosimetry at the Savannah River Site."
- 2) Film processing procedures used at SRS.

NIOSH indicated that the first document was available at the OSTI website. Procedures for film processing are available through Kathy Robertson-DeMers for a portion of the 1950's and 1960's.

Highlights of NIOSH responses

- A worker profile is under development for outside workers.
- TRU contamination in recycled uranium is not taken into account.
- For unmonitored tritium workers, co-worker data will be used.
- SRS personnel radiation exposure files and CATI interviews are used to identify accidents and incidents.
- D&D workers are not included in the "Construction Workers" chapter as NIOSH assumes that they are usually monitored.
- Unapproved practices are not currently considered in dose reconstruction.
- For unmonitored early workers co-worker data will be used to assign dose.
- There has been no validation of bioassay data with air monitoring data or in vivo counts.

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- The A-P exposure geometry is assumed. No correction for rotational exposure is necessary. No correction factor for angular dependence is applied.
- Technical basis documents are written to address a majority of the claims for a particular site. They are living documents.

Path Forward

NIOSH agreed that many of the issues raised by SC&A deserved further investigation, and that NIOSH was looking into or will look into selected issues. SC&A suggests that NIOSH identify in writing or through communication with the Advisory Board, those issues which they intend to investigate further. This will help the overall process and assist in deciding which issues will be pursued in Phase II of the Savannah River Site review.

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ATTACHMENT 4

SRS FACILITY SITE EXPERT INTERVIEW SUMMARY — FORMER AND CURRENT RADIATION PROTECTION STAFF

Organization

DuPont was responsible for construction of the Savannah River Site and operated the facility through March 31, 1989. Westinghouse Savannah River Company took over SRS operations on April 1, 1989. Currently, SRS has a single contract with several companies involved in the work including Westinghouse Savannah River Company and Bechtel Savannah River Inc (BSRI).

C.M. "Pat" Patterson was responsible for setting up the original radiological control program at the Savannah River Plant. Prior to his arrival at SRP he had worked with K.Z. Morgan at ORNL and Herb Parker at Hanford. DuPont requested that "Pat" come to SRP to lead the radiation protection program. "Pat" in turn asked experienced individuals from other facilities to join his team. "Pat" was the original Radiation Protection Manager. From 1951 to 1991, there were only two Radiation Protection Managers. The individuals in the health physics program were generally consistent from the plant inception through the late 1970s. There was very little new hiring during this period of time. Initially, industrial hygiene and environmental monitoring were a part of the organization. The radiological control organization was fairly stable during the first decades of operation.

Initially, the radiological control organization consisted of an Area Central Safety Committee for each major area of the site (i.e., separations, reactor/heavy water, raw materials, and SRL), and a Plant Central Safety Committee. The area committees met periodically to discuss radiation protection issues. The managers of these areas would then in turn provide reports to the Plant Central Safety Committee. With the arrival of Westinghouse, the health physics organization was centralized.

Currently, Radiological Protection Services consists of Health Physics Services (dosimetry and radiological monitoring); Regulatory and Radiological Technology, Training, Procedures, and Programs; and matrixed organizations that are composed of field health physics support. In January 2003, the health physics organization decentralized and went to a matrixed type organization. The matrixed organizations include all major projects onsite including Decontamination and Decommissioning. Subcontractor involvement in radiation work has been minimal at the site until the mid-1990s. Subcontractors either use in-house procedures for performing radiological control work or they implement project-specific procedures, which are approved by the Westinghouse Radiological Control Manager. The contractual process is used to ensure subcontractors comply with the health and safety procedures and policies of the site. There is also tight control on sources and radiation-generating devices brought onsite. Outside organizations are required to use the radiological control services provided by Westinghouse, such as instruments and radiological field support, or have alternative instruments and staff approved by a WSRC subject matter expert. In terms of dosimetry, subcontractors are required to use site dosimeters and bioassay programs.

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Field Procedures and Policies

DPSOP-40, *Radiation and Contamination Control*, provided general information on radiation and contamination control, including permissible limits, personnel monitoring, radioactive material transportation, and waste disposal. This procedure was similar to the current Radiological Control manuals. DPSOP-40-1, *SRP Radiation and Contamination Control*, and other facility procedures provided specific radiological control requirements for particular facilities. These documents constituted the standard operating procedures for the field.

From the field perspective, the radiological hazards of concern by area during early operations were:

<u>Plant Area</u>	<u>Primary Radiological Hazards</u>
Reactors	Tritium, direct exposure from fission and activation products
Separations	Fission and activation products, transuranics
Raw Materials	Natural uranium (also beryllium)
Heavy Water (Rework Area)	Tritium

The guidelines provided for determining bioassay and dosimetry requirements were documented in procedures (presumably DPSOP-40-1).

Special Work Permits (SWPs) were initiated with the startup of production. Initially they were used in the reactor area, and then the use was expanded to all facilities. SRS continued to use SWPs until the mid-1960s. These were used for "hot jobs" where limits had to be set. The inspector and operations supervisor or laboratory supervisor approved them. Individuals had to acknowledge they were working under a specific SWP. Surveys and air sampling were performed during these jobs. In 1992, the site implemented the use of Radiation Work Permits (RWP). There were standing RWPs (repetitive work) and job-specific RWPs. The standard RWPs required monthly sign-in on a log sheet. The job-specific RWPs required a sign-in on each entry. In the absence of RWP/SWPs, the requirements were documented in DPSOP-40-1 or facility-specific procedures.

Procedures initially provided some guidance on preparing work permits, but field personnel based requirements on judgments. Training provided to personnel on RWP/SWP preparation. As the Radiological Control program matured, procedures provided more guidance on work permit preparation. The site currently has an RWP procedure that covers how the requirements are determined. These requirements are based on actual or potential radiological conditions. For larger projects, a special job plan or work package is developed which includes the RWP and other documents related to the job. Of the areas, the requirements have historically been the most restrictive in the Separations areas followed by SRL. The reactor and raw materials permit requirements were less restrictive.

Work permits included requirements for Personnel Protective Equipment (PPE), dosimetry, supplemental dosimetry, timekeeping, exit requirements, health physics coverage, and other

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special instructions. Early PPE used at SRS included coveralls, gloves, booties, hoods, shoe covers, and respiratory protection (i.e., assault masks, respirators, bubble suits). The goal was to prevent or minimize personnel contamination. Initially, assault masks with cartridges were used to prevent uptakes of radionuclides other than tritium. The highest use was in the 200 areas. Masks were sent to the laundry after each use.

SRS participated in the development of the bubble suit to protect individuals from coming in contact with tritiated water and for use in high airborne areas. Prior to the use of bubble suits, knee-high rubber boots with straps and slickers were used. There were several iterations of bubble suits prior to the development of the finished product. The modern bubble suit was developed by SRS in the 1970-1972 time frame. There were situations where timekeeping was used in lieu of bubble suits for tritium intake control. This was always the case. Lead-lined gloves were used in the separations facility glove lines. Lead aprons were also available.

The RWP could be linked with the field monitoring data (i.e., survey and air sampling data), especially in the case where lapels were used. During the time period when RWP/SWPs were not used, this would be difficult. The field records are stored in hardcopy by date and Health Physics Office, so retrieval would be difficult and records review cumbersome. The purposes for RWP are to notify individuals of the hazards associated with a task and for the purposes of trending.

Radiological Posting and Contamination Control

At the inception of the site, there were no common postings throughout the complex. Each site established its own posting criteria. ORNL developed the first postings with the tri-foil on it. Initially, the posting and contamination limits were set at 10 dpm/100 cm² alpha and 80 dpm/100 cm² beta/gamma for a Regulated Zone. Radiation Danger Zones were similar to what is currently referred to as a High Radiation Area or Very High Radiation Area. With the implementation of DOE Order 5480.11, *Radiation Protection for Occupational Workers* (Issued December 12, 1988), definitions of radiological areas and posting limits changed. Regulated Areas and Radiation Danger Zone went away. The order required the establishment of Contamination Areas, High Contamination Areas, Radiation Areas, High Radiation Areas, Very High Radiation Areas, and Airborne Radioactivity Areas. The removable alpha and beta/gamma contamination limits were 20 dpm/100 cm² and 1,000 dpm/100 cm², respectively. Total contamination limits were 300 dpm/100 cm² and 5,000 dpm/100 cm² for alpha and beta/gamma, respectively. There were slight changes made with the implementation of the *DOE Radiological Control Manual*. Note that SRS has used 500 dpm/100 cm² total alpha as a contamination limit, rather than the traditionally used 100 dpm/100 cm² total alpha limit documented in DOE Order 5400.5 and its supplementing documents. This is primarily due to the fact that total alpha contamination cannot be detected with current instrumentation at a level of 100 dpm/100 cm² with a scanning survey.

Radiological Training

Prior to the implementation of the DOE Radiological Control manual, a formalized Radiological Worker training program did not exist in the DOE. Requirements for subcontractor training

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would have been developed within the last 20 years. Early training included incident training for Health Physics staff, criticality safety and fissionable material handling training for individuals working with fissile material, and Hydrogen Sulfide (HS) training for individuals working in Heavy Water production. Current Radiation Worker training includes direction on how to wear dosimetry.

Real-time Dose Tracking

Personnel were required to track their day-to-day dose on dose tracking records. This included documenting their Pocket Dosimeter readings. Neutron dose was also estimated where neutrons were an exposure issue. The neutron/gamma ratio was known for the different areas onsite. Personnel dose tracking records (e.g., time keeping records) used these ratios to document an estimated neutron exposure. The neutron/gamma ratios are documented in technical reports. For example, the neutron to gamma ratio for FB-line and HB-line is 3:1. If readings on the dose tracking records were greater than 25% different from the film badge results and the film badge results were greater than 100 mrem, a Radiation Exposure Data Investigation (REDI) was completed. DuPont Savannah River was the first DOE contractor to implement an administrative control limit (ACL). The initial ACL was set at 3 Rem.

Routine work did not occur in areas where more than the allowable dose could be received. If nonroutine work had to be performed, special procedures were considered, so that higher exposure levels were limiting in time of exposure. There are few high-level exposures and if they accidentally occurred, they will be documented in the individual radiation exposure file.

Air Sampling

The purpose of the air-sampling program is to verify engineering controls, document radiological conditions, detect changes in radiological conditions, determine appropriate postings, and provide input to the internal dosimetry program. This helps minimize internal exposure to the workers by providing early indicators. There are several types of air sampling including ambient air monitoring (high occupancy area), containment verification monitoring (close to potential release point), and job coverage. A job coverage air sample is used for a specific radiological job evolution.

Historically, radiological engineers were involved in the development and implementation of the air-sampling program. Air-sampling equipment used in the past or currently used at SRS includes the following.

- Annular Kinetic Impactor
- Retrospective Air Sampler (RASr)
- Portable Tritium Bubbler
- SRS Alpha LVAM (developed by SRS)
- SRS Beta/Gamma LVAM
- Eberline AMS-3/4 Beta-Gamma Continuous Air Monitor (CAM)
- Eberline Alpha 6 CAM
- Canberra Sentry Alpha CAM

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- SRS Kanne Chamber (developed by SRS)
- Aptec-NRC TAM-100D (NRC bought out by Canberra)
- Scintrex Model 209/309
- Mobile SRS alpha HVAM
- Fixed SRS alpha HVAM.

The RASr uses glass fiber filters. The Annular Kinetic Impactor and the HVAMs use greased planchets as a sample media. Teflon PTFE filters are used with the Canberra Alpha CAMs to allow for real time Alpha Energy Analysis. Water and Ethylene Glycol are used in the Portable Tritium Bubbler. Each of these units is addressed in detail in *The Savannah River Site Workplace Air Monitoring Technical Basis Manual* (WSRC 2001b). Lapel/breathing zone air samplers were implemented early in 2003. There are procedures in place to direct their usage. Rotameters and vacuum gauges are factory-calibrated and verified prior to initial use and on an annual basis. All SRS LVAMs and Portable HVAMs have been retired.

Air-flow studies have been completed for all indoor facilities. Patterns vary depending on the facility. They range from a wind tunnel atmosphere to stagnant air. The canyons and sampling aisles have good air flow. In the FB-line it depends on the particular room, but it can be stagnant or have a low turnover. The HB-line ventilation is better as it was built after the FB-line.

In general, the technology shortfalls associated with air monitoring instrumentation are limitations on the use of impactors. The efficiency of the impactors falls off with small particle size.

Air Sampling Acronyms

<u>Nick Name/Acronym</u>	<u>Description</u>
HVAM	High Volume Air Monitor (~ 40 cfm)
LVAM	Low Volume Air Monitor
Alpha LVAM	Alpha Low Volume Air Monitor (only used for a short time)
RAS	Retrospective Air Sampler

Radiological Instrumentation

SRS has a display of historical instruments used at the SRS from the 1950's forward. Included with some of the instruments are information cards indicating what they are and how they were used. Instruments used at SRS (past or present) include the following.

- NE Electra Plus (present)
- Eberline Model AC-3 Alpha Scintillation Probe (present)
- Eberline RO-2 Ionization Chamber (present)
- Eberline ASP-1 NRD Rem Ball (present)
- Eberline RO-2S

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- Eberline RO-7
- Eberline RO-20
- Eberline Geiger Counter E-120/E-120G
- 110 Pancake Probe
- Eberline RO-2S-1 Beta/Gamma Survey Meter
- SRL Neutron Survey Meter (designed by W.F. Splichal)
- Fast Neutron Survey Meter (Oak Ridge design)
- Jordan Electronic, Inc. CD-710 Survey Meter
- Victoreen Instrument CD-720 Survey Meter
- Eberline PRM-4B Portable Count Rate Meter
- Victoreen 471 Rate Meter
- SRP Scintillation Counters (DP-110)
- Eberline PAC-1SA Portable Survey Meter used with an alpha scintillation probe
- The Nucleus Model K2 Scaler (Oak Ridge design)
- Overhoff and Associates, Inc. SP-4000
- Sampson Alpha Survey Meter (Raychronix- Phased out in 1970)
- SRL Alpha Survey Meter with Flash light probe (>100 cm²) or the Chicken Wing Probe.
- Victoreen THYAC Model 489 Beta/Gamma Survey Meter
- SRL TRYAC Beta/Gamma Survey Meter
- AEC Juno Survey Meter
- Technical Associates Model 7 Alpha/Beta/Gamma Survey Instrument
- Espey Manufacturing Company, AEC Model SIC-17C High Range Instrument (Yellow Juno)
- Applied Physics Electrometer (Tritium Sampler)
- Victoreen "R" Meter (Source Calibration)
- Landsverk Charger/Reader with Electrometer Ion Chambers
- Victoreen 496 Survey Meter
- Landsverk Pocket Ionization Chamber (1950s and 1960s)
- RAYCHRONIC/NUCOR Cutie Pie (CP) Dose Rate Meter
- Cutie Pie Mark V (Oak Ridge Instrument)
- Sampson Alpha Survey Meter

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- Mud Bucket (CD container filled with paraffin in which BF₃ proportional counter was inserted.)
- Raychronix “Hurst” Neutron Survey Instrument.

The Victoreen THYAC was the main Beta/Gamma Survey Meter in the early years of operation. The SRL Alpha Survey Meter was the main alpha survey meter until it was replaced with the AC-3. Cutie Pies were the primary dose rate instruments until they were replaced with the RO-2.

The original calibration facility was housed in Building 736A. The calibration facility has moved to Building 735-2B. SRS uses a number of calibration sources to calibrate instruments and dosimeters. These include the panoramic dosimeter irradiator, beta beam irradiator, gamma beam irradiator, low scatter irradiator, x-ray beam irradiator, and americium irradiator. The low scatter irradiator is used for neutron calibrations and can be used moderated or bare. These sources are traceable through National Institute for Standards and Technology. In the past they were traceable to the National Bureau of Standards. Sources are calibrated using X-Radian and Capintec Ionization Chambers with electrometers. The Raychronix/Nucor CP with electrometers has also been used. The portable instrument technical basis document has more information on source calibrations.

Instrument Acronyms:

<u>Nick Name/Acronym</u>	<u>Description</u>
PDI	Panaramic dosimeter irradiator
BBI	Beta beam irradiator
GBI	Gamma beam irradiator
LSI	Low scatter irradiator
XBI	X-ray beam irradiator
AmI	Americium irradiator
NI	Neutron irradiator

External Dosimetry

The monitoring technologies have changed over time; however, the philosophy has not changed. There have been no adjustments to the dose of record based on changes in technology. Correction factors were developed in the dosimetry history document for the purpose of comparison. The correction factors included considerations for change in calibration source (i.e., Ra-236 to Cs-137), contribution from backscatter, and changes in μ_x for photons. These factors would result in an upward adjustment of earlier data by 11.9%. As a part of Department of Energy Laboratory Accreditation Program implementation, the factors described above had to be implemented. For more information refer to page 105 of the dosimetry history report. The definition of deep dose from a 2 cm depth to a 1 cm depth occurred with the implementation of the Panasonic dosimeter.

The site is DOELAP-accredited for all predominant radionuclides at the site. The site is not accredited for low-energy photons due to the absence of this category in the DOELAP program. Calibration sources are NIST traceable. There have been no changes in the dosimeters since

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1995. Although all dosimeters have small technology shortfalls, the SRS dosimeter meets the world standard for dosimeters.

Initially, TLDs were stored in badge racks. Since September 1992, employees have been allowed to take badges home. Temporary badges have been issued to workers who forgot their dosimeter in the past. Since September 11, 2002, the dosimeter has been married to the security badge. If an individual forgets their security badge, they are required to go home to retrieve it.

Individuals in plutonium areas were exposed to the low-energy photons of transuranics. The film badges did not adequately respond to these photons. As a result, the site interpreted badges with the x-ray calibration curve. This was more representative of the low-energy photons encountered in the field. This special interpretation of film badges for the HB and FB-line began in approximately 1958. There were also sections of the M-Area where low-energy photons were an issue, and thus the special interpretation was used. The special film badge interpretation was limited to these areas. In areas with low-energy photons, the gonad and lens of eye dose were expected to be greater than that of the whole body.

Neutron personnel monitoring criteria for workers at the initial startup of reactors are not clearly known. Historically, neutron monitoring has occurred when the general area dose rate was greater than or equal to one mrem per hour. Note that early neutron survey instruments were used for measurement of fast neutrons. The site relied on area monitoring as an indicator of when personnel monitoring was necessary. There was likely intermittent neutron monitoring during this period, and dosimeters were turned in at the end of the cycle.

The original neutron source used to calibrate NTA film was a semi-moderated PuBe source. In 1965, this source was replaced with a plutonium fluoride source. The badges were irradiated with the PuF₄ 50 cm above a paraffin drum. This was to account for some of the scatter. The PuF₄ source improved the accuracy of the dosimeter, as it was more representative of the neutron energies encountered in the field. In general, the PuBe had too much moderation and the PuF₄ had too little moderation.

The NTA film used by the site underestimated neutron dose due to lack of response at <500 keV energy and partial response at 500 keV-1,000 keV. There is a steep curve representing the NTA film response between 500 keV and 1,000 keV. The TLND was sensitive to all neutron energy ranges. One method for correcting the underresponse in the NTA film is to determine the energy underresponse by comparing the TLND and NTA film. Factors influencing the outcome of the total dose, such as rate of production, would have to be considered. The error bars would be large on this type of estimation. For example, at the Pu facilities, the underestimate for NTA film was about one-third based on this type of comparison.

PNNL completed a neutron characterization survey to determine energy spectra at various areas on the SRS. Most of the facilities were in operation at the time. A slow neutron component was identified in facilities. SRS paid a great deal of attention to the neutron-to-photon ratio as it was used for daily dose tracking. In general, applying a neutron-to-photon ratio to recorded deep dose for 1981 data and forward would overestimate the dose, as compared to the badge reading.

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Multiple dosimetry systems have been used at SRS when nonuniform exposure is expected. Early health physics staff indicates that multiple dosimeters were used from the inception of operation in one form or another. There has been a change in the way multiple badge results are recorded in HPAREH or its equivalent. Prior to 1992, 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, a subcontractor was working onsite in the 1990s. He defeated the interlock system on his radiography unit and inadvertently put his head in the beam. The dose to the back of the head was calculated to be 11 Rem. This dose was higher than that measured by his chest dosimeter. His whole-body dose was assigned as 11 Rem. In about 1992, the methodology for assigning whole-body dose from multiple badges changed. Each organ was assigned a weighting factor. The whole-body dose was calculated as the sum of the weighting factor times the appropriate dosimeter value. There were a total of eleven dosimeter points possible in a multiple dosimeter pack. A new chest dosimeter was worn to act as a reference point, and the routine dosimeter was left in the badge rack for this period of time.

Although each dosimeter was processed, multiple dosimeter results were not routinely included in the Personnel Radiation Exposure Record. The only dose from multiple badging that occurs in the individual records was the whole-body dose. The results from multiple dosimeters are stored separately from individual dosimetry records, except the calculated whole-body dose.

The Savannah River Site has had an area dosimetry program. This program was used to verify postings and reevaluate radiological boundaries.

Subcontractors

The Savannah River Site dosimetry department has always been centralized. As a result the records for construction workers, subcontractors, visitors, and employees have been maintained by the same group. The basis for monitoring was and continues to be based on the facility, rather than the individual. Entry into a radiological area requires the use of a dosimeter regardless of the individual entering.

In the last 10 years of operation, there has been an increase in subcontractors working onsite. In general, the site provides health physics support to the subcontractors. In some cases they bring their own health physics services. In these cases, the radiological control procedures used by the subcontractor must be reviewed and approved by the site radiological control organization. In the case where subcontractors or visitors bring their own dosimetry, SRS still assigns them a site dosimeter. Monitoring requirements for visitors and subcontractors is at times more rigorous than for the site employee.

Radiobioassay

Per DOE policy, bioassay is used to determine internal dose. Although air sampling data is reviewed in the internal dose assessment process, it is not usually used in the calculation. As a result, limited work on comparison between air concentration data and bioassay data has occurred. The site started a DAC-hour tracking program in 2003, which can be used to assign or monitor internal dose.

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There are a number of reasons why individuals are put on routine bioassay sampling. If there is a potential for generating airborne radioactivity or if respiratory protection is required, individuals are put on a bioassay program. Prior to 1990, the field radiological control groups determined the need for special bioassay in the event of an incident or occurrence. The field eventually lost control of this function due to inconsistencies in their approach and errors. From 1990 forward, the field has been required to contact the Internal Dosimetry group when an incident has occurred. Internal Dosimetry in turn determines the need for special bioassay. They also set the guidelines for determination of routine bioassay requirements.

Monitoring has most recently been determined based on the potential to receive 100 mrem in a year. The monitoring requirements are tied to the area/facility rather than the individual. This is the way it has been since startup. SRWPs and RWPs are used to communicate these requirements. This means that anyone (i.e., operations, construction, subcontractors, etc.) who enters the area must comply with the dosimetry requirements outlined for that area/facility.

Currently in the bioassay program, one radionuclide is typically not used as an indicator for the presence of other radionuclides. Although many radionuclides occur together (e.g., Am-241 and Pu-239, or Sr/Y-90 and Cs-137), it is possible to find these radionuclides separate from one another. The current practice is to characterize field samples and use this as an indicator of the radionuclides present. With the routine monitoring program, there are no assumptions made with respect to ratios. Prior to 1990, the field was involved in choosing which bioassay samples were taken. As a result, it is unknown whether one radionuclide was used as a surrogate for another radionuclide.

There has always been monitoring for tritium exposure, although the detection level of tritium has changed over time. The tritium monitoring program has been inclusive of the reactor workers. The highest cumulative tritium dose at the site is from the tritium facilities.

Stable metal tritides may be formed during specific operations in the tritium processing facilities. These are not prevalent throughout the site, however. Monitoring is available to detect tritium in the form of special tritium compounds (STCs). With respect to evaluating STCs, an enhanced tritium-monitoring program has been implemented in the last ten years. The prior control of tritium exposure occurs in the field. Air sampling and limiting stay time are two techniques employed.

In the early 1980s, the site began monitoring releases of OBTs and metal tritides to the environment. OBTs were included in the evaluation performed by the Center for Disease Control. Other STCs were not addressed. A second independent reviewer from the Agency for Toxic Substance and Disease Registry looked at the historical potential for STCs.

The uncertainty in in vivo analysis was primarily based on the detectors being used. The most pronounced improvement in the in vivo program occurred when the phoswich detectors were replaced with germanium detectors. The improvement in calibration processes also improved the accuracy of the counting systems over time. The introduction of the Livermore phantom did not have as substantial an effect on improvement of in vivo counting technique. Also, it is

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important to note that the decision level/minimum detectable concentration was calculated differently through time. The paper listed below addresses some of these issues.

Dean, P.N., Langham, W.H., and Ide, H.M., *External Measurement of Plutonium Lung Burden*. 15th Annual Bioassay and Analytical Chemistry Conference.

There is always some potential that individuals not on a bioassay program are exposed to an internal hazard. The frequency of this is unknown. Termination in vivo counts for all employees would detect more obvious intakes.

For the most part, the site has been successful in obtaining bioassay samples from longer-term contractors and site employees. There has been some difficulty with getting bioassay samples from short-term or mobile construction and subcontract workers. By the time the site realizes they are delinquent, these individuals are gone.

As a result of the limited information available electronically, it is not possible to determine how often the bioassay results exceeded the detection limits through the years. The data is also not available to do a comparison of which process likely resulted in the greatest number of intakes.

Technology shortfalls in the bioassay program are related to detecting insoluble plutonium in the absence of americium. This is discussed in the *Savannah River Site Internal Dosimetry Technical Basis Document*.

The similarity between the bioassay program at Hanford and SRS has not been evaluated. Early radiobioassay techniques were adopted from Oak Ridge National Laboratory. It is uncertain how SRS accounted for intakes of radionuclides prior to the implementation of a bioassay technique at the site. This was not an issue after 1970 as the bioassay program was well developed.

Internal Dosimetry

Savannah River developed what is referred to as the Savannah River Site Registry. Historically, this was a database of confirmed assimilations. Individuals were classified as having a confirmed assimilation if they had two positive bioassay samples. In 1984, the criteria for inclusion in the SRS IDR changed. Individuals with 100 mrem Annual Effective Dose Equivalent or greater were included in the registry. As of 1993 when the DOE implemented CEDE, individuals with 100 mrem CEDE were included in the SRS IDR. Currently, individuals with known incidents that likely resulted in a committed dose of 10 mrem or more are included.

Doses for individuals in the SRS Registry were calculated using the ICRP 30 methodology in the 1990s. In the 1986-1987 time frames, anytime an individual terminated from the site, his/her bioassay data was evaluated and an internal dose assigned as applicable. This procedure was discontinued for individuals other than those meeting a 5-6 Rem CEDE threshold. All internal doses after January 1, 1989 are required to be assessed per regulation. Individuals with bioassay above the decision level prior to January 1, 1989 and having no corresponding incident may not have a dose calculated via the ICRP 30 methodology.

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When performing an internal dose assessment, the Internal Dosimetry group varies the assumptions made with respect to solubility class, date of intake, etc. Guidance on the assumptions is documented in the internal dosimetry technical basis document. Historically, assumptions are not an issue as uptakes have been reassessed per the ICRP 30 guidance. Although particle size studies may have been completed by the field, these data are not used in the internal dose calculation.

There is a potential of exposure to special chemical forms of radionuclides such as highly insoluble plutonium oxide and tritides. The FB-line and associated waste streams may contain extremely insoluble forms of plutonium oxide. One case evaluated based on an event in 1999, indicated an intake of extremely insoluble plutonium. For this evaluation the new ICRP 60 models were used. Tritides are found onsite; however, they are typically contained.

SRS does not assign missed dose based on decision levels related to *in vivo* and *in vitro* analysis.

Recycled Uranium

Initially, the recycled uranium program at SRS did not involve the fabrication of fuel rods. The uranium was processed through the separations facility to form UNH. The resulting uranium mixture was then either stored onsite or shipped to the Y-12 Plant in Oak Ridge. UO₃ powder was sent to the Gaseous Diffusion Plants and mixed with virgin uranium.

SRS characterized the source terms involving recycled uranium shortly after the gaseous diffusion plants were identified as having a plutonium source term. The internal monitoring program concentrated on monitoring individuals for those radionuclides which compose 90% of the dose delivered, or for those that serve as an indicator for other radionuclides. In the case of recycled uranium, the impurities often did not meet this criterion. Part of the processing of recycled uranium involved monitoring the radionuclide makeup of the product. The waste stream from the uranium facilities was also monitored. An internal evaluation of dose from impurities in recycled uranium was also conducted. A methodology was developed by operations to keep the Pu/U ratio to a level such that the dose contribution from Pu constituted <10% of the dose. In addition, air sampling was used in areas handling recycled uranium to monitor for airborne contamination.

The first recycled uranium onsite was for the purposes of R&D. This material arrived in the late 1950s or early 1960s. Eventually the site was involved in production processing of recycled uranium. After the cladding had been dissolved from the fuel, the uranium was sent to the A-lines of the separations facilities. Initially the resulting material was stored. Much of this material was removed from the site in the process of decommissioning. In the 1960s, the site started recycling material for use in fuel manufacturing. The UNH retrieved from the A-line process was returned to the Y-12 Plant to be processed into UO₃ powder. The powder was then processed through the gaseous diffusion plants along with fresh material.

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Radiological Records

The interaction between the Radiological Records group at the Savannah River Site and NIOSH or its contractor has been limited to providing personal dosimetry information. The data provided includes information from the quarterly logbook data, microfiche monthly (cycle) dose records, the individual Personnel Radiation Exposure Files, microfilm roll copies of individual Personnel Radiation Exposure Files from early years, the Health Protection Annual Radiation Exposure History Database (HPAREH), the archived Health Protection Radiation Exposure Database (HPRED) monthly (cycle) dose records for some years, HPRED, and visitor or temporary badge cards. This constitutes the total SRS personnel exposure record. SRS has a database which tracks the requests from NIOSH and when the material is provided to NIOSH or its contractors. The site has a database called EDWS that is used to store the retrieved dosimetry data for each claim in .pdf format. Follow-up requests for information have primarily been limited to getting better copies of information and requesting specific information from staff.

Not all tritium and neutron logbooks have been located. Records from 1951-1957 have the beta/gamma and neutron dose reported on the same record. From 1958 through the first quarter of 1963, tritium and neutron doses can be determined by a code. After the first quarter of 1963 thru 1972 there is no neutron-specific data available. From 1973 to the late 1980s microfiche copies of neutron-specific logbooks are available. For the period from second quarter 1963 through fourth quarter 1972, there is no way to distinguish what portion of the open window and shielded dose is neutron dose. Semi-annual tritium data is available for the second quarter of 1963, but the records are missing for the second half of 1963 and for 1964 and 1965. Tritium dose is available in semi-annual tritium reports from 1966 until 1979 and quarterly from 1980 through the first quarter of 1989. Tritium bioassay results are also available on bioassay cards.

There has been a possibility that an individual could be on a routine dosimeter program and be assigned a temporary dosimeter. When a permanent employee was issued a temporary badge, the dose from the visitor card was incorporated into the total dose assigned to that individual. Historically, visitor card information can be used as a source of data for employees who were issued temporary badges.

Long-term subcontractors were assigned a routine badge. Subcontractor/construction force doses were recorded in the logbooks using Payroll Numbers other than 1 and 2. This also included DOE staff. Short-term subcontractors were assigned temporary badges. Temporary badge results for the early years are stored on 3" x 5" index cards or what is commonly known as the visitor cards prior to 1979. Since 1979/1980, the records have been available in HPAREH or the comparable system.

Currently, the PRORAD software is used to provide access control and dosimetry record storage. Field radiological records are not maintained by the Radiological Record group. The different field offices have the responsibility for maintaining field surveillance records (e.g., air samples, survey reports, Radiological Work Permits, timekeeping records, etc.)

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Bioassay sample results were originally documented in logbooks. This information was transferred to individual bioassay cards which were placed in the Personnel Radiation Exposure File. With the current system of analysis, the data is computerized.

There are a number of sources of incident information. A majority of the incident records are available in the Personnel Radiation Exposure Files. If an incident report was written, it was supposed to make it into the record. A write-up was placed in each involved individual's file. There is a suspicion that not all the incident records made it to the dosimetry files. Facility personnel responding to incidents were not included in incident report write-ups.

A REDI was also issued when there was a lost or damaged dosimeter or the dosimeter or bioassay was not returned. This may or may have not been filed in the Personnel Radiation Exposure files. The REDI has been used at least since 1978. Prior to the REDI, Missing Exposure Investigations were used. When reconstructing the dose, the site made use of time and motion studies, radiological survey data, air-sampling data, and coworker data.

There are a number of methods for documenting incidents, occurrences, or abnormal events. Events are captured in the field logbook and in radiation survey reports. This would include minor situations (e.g., spread of contamination) where personnel exposure was not an issue. Starting in 1992, the field issued Radiation Deficiency Reports. There are also Problem Identification Reports. In the case of incidents, there has been some sort of form completed since the inception of the site radiological control program. During the DuPont era, there were also Special Hazard Bulletins that were generated by an investigating board. These were separate from the Special Hazard Investigation reports.

Incidents were documented as Special Hazards Investigations (SHIs) from the beginning of production through 1989 when DuPont left. Since the SHIs are not easily searchable, these are not included in the information sent to NIOSH or its contractor unless copies were already in the individual Personnel Radiation Exposure Files. SRS has a database referred to as the SRS Incident Database. It contains minor and major incidents through 1999 including all the SHIs. Some of the information in this database is classified. Westinghouse Safety Management Solutions (WSMS) currently owns this database. As a result, the database is not readily available for SRS to provide to NIOSH or its contractor.

There were more process upsets in the early years than in the later years in terms of environmental release and occupational dose. It is important to note that reporting criteria for incidents has become more prescriptive over time. (e.g., An incident which would be considered minor in the early days would be reportable by today's standards.)

Environmental Exposure

Tritium is fairly dispersible in the environment and can be found in soil, groundwater and vegetation. Tritium is detectable in the onsite environment and is the largest contributor to offsite doses (i.e., < 1 mrem per year). There were higher levels of tritium in the environment during the production years.

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Fission products were released to the environment during some periods of operations. The largest release from the site involved I-131. There was not a special mechanism in place in the separations facilities to confine iodine gas. This was usually not an issue as the iodine would decay prior to processing it in the separations area. Also detectable in the environment is Cs-137. The Cs-137 in the environment is partly from site releases and partly due to the atomic bomb testing.

Uranium is primarily found in the M-area and in waste streams. The release of actinides is localized around the chemical separations plants. Environmental doses from these releases are primarily limited to onsite personnel. Radiological releases have resulted in some soil and liquid effluent contamination. There was a high demand for product in the early 1960s prior to the Test Ban Treaty. As a result, the holding time was reduced to 90 days. The shorter holding time meant the iodine in the fuel was at a higher concentration when it went to the separations process. The largest iodine releases at the site occurred during this period of time.

The TBD, Revision 2 (pp. 58 of 232) states the following:

Soil sampling and analysis were not routinely performed at the Savannah River Site during the period of greatest atmospheric releases (from 1955 through the late 1960's).

This is an incorrect statement. The first environmental samples were completed as a pre-environmental assessment. This included all types of samples. Sampling has continued throughout the operation of the site.

Medical Exposure

DuPont always maintained a substantial occupational medical program, as they wanted to keep their employees healthy and on the job. Historically, medical exams were used for surveillance and to determine qualification for jobs. DuPont had established corporate guidelines for medical exams. The exams were also a part of the benefits provided by the employee. As with the current exams there were pre-employment, periodic and exit exams. In the early years of operation these exams were offered on an annual basis.

Medical exams historically included the following items.

- CBC
- Urinalysis
- Blood chemistry
- Chest x-ray (pre-employment required/follow-up optional)
- Drug screening (new hires)
- Height, weight, blood pressure and pulse
- Pulmonary function test
- Tonometry
- EKG
- Hearing test

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Medical exams are currently performed in accordance with *Contractor Occupational Medical Program*, DOE Order 5480.8a and applicable Federal regulations and other standards (ANSI) such as asbestos, hearing conservation, lead, beryllium, and respiratory protection. Each employee receives a pre-employment exam to determine fitness for duty. The frequency and elements of the exam are based on regulatory requirements. The employee is asked to complete an exit survey when leaving the site. The answers on this survey determine the evaluations performed in the exit exam. Overall, the level of exams has decreased over the past 16 years. In the last 4-5 years, the medical division has been limiting exams to the minimum required elements and frequencies.

Current pre-employment exams involve almost all of the tests above. Follow-up exams are performed per requirements. A biennial (sometimes annual) Pulmonary Function Test is performed on those individuals qualified to use respiratory protection. The medical staff tries to take advantage of the opportunity when employees are scheduled for medical exams, as scheduling can be difficult.

There were as many as thirteen medical facilities on the Savannah River Plant at one time. This included a medical facility in the 200H area. Three medical facilities now exist on the site. The Medical Department was originally part of the Human Resources Department when the plant started.

The type of x-ray equipment used at the Savannah River Site has changed over time. Fixed and portable units have been used at the site. Photofluorography has not been used at the site per the medical staff. Normal 14" x 17" film is used for chest x-rays. The x-rays are shot in the PA orientation. Lateral shots were not done unless the doctor saw something on the PA x-ray. Film wastage is estimated at about 1/10 of a percent or less. Historically, one film was shot at the beginning of each morning to determine a densitometer reading. The site maintained the x-ray films, which are currently stored in a records repository.

No registration of x-ray equipment is required with the South Carolina DHEC. Although SRS does not fall under the jurisdiction of the state of South Carolina, they do implement the requirements for x-ray unit inspections and maintenance. A qualified vendor has routinely provided maintenance of x-ray equipment. The site currently has a subcontract in place with a firm from Charleston, South Carolina to perform annual x-ray inspections. An inspection report is provided to the site following completion of the inspection. Prior to implementation of this contract, it is believed health physics performed surveys. There has historically been area dosimetry posted on the outside wall of the x-ray room.

There have been 407 individuals administered chelation therapy over time at the Savannah River Site. This number does not account for the multiple administrations that occurred for some individuals. With the administration of chelation therapy, blood tests were taken to monitor patients. The use of chelation was strongly encouraged because of its proven decorporation effect. The doctor chose the type of Diethylenetriaminepentaetate (DTPA) used. Ca-DTPA was the primary type. Generally aerosol administration was used and less frequently, intramuscular injection.

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Currently, the criterion for administering DTPA is based on receiving a committed dose of 2 Rem or more. When an incident occurs, the field notifies Internal Dosimetry. Based on the field indicators, a preliminary dose estimate is determined. If the dose is 2 Rem or more, a physician is notified for the consideration of chelation therapy. The physician ultimately makes the decision to offer chelation therapy. The worker is required to sign an informed consent form if they agree to chelation therapy.

Records relating to the administration of chelating agents and the results of blood tests are maintained in the medical file. Bioassay results are maintained in the Personnel Radiation Exposure File. Chelation records are also forwarded to Oak Ridge Associated Universities in Oak Ridge, who maintains the records for all DOE-related chelations.

There have been no lung lavage procedures performed at SRS. Wound excision or treatment was done in conjunction with the health physics staff. Individuals with tritium uptakes were encouraged to drink a lot of fluids. There have been no treatments at SRS for acute radiation sickness to the knowledge of the current medical staff.

Medical files at SRS are very detailed. These files include the following information:

- Medical exam results
- Injury and illness reports
- Correspondence to and from their personal physician
- Death certificates
- Other medical related records.

Non-Radiological Worker Exposure

Each area usually has a building that houses administrative staff. In the reactor areas this building is located outside the area fence. In the separations areas the administrative building is within the fence for that area. This area also includes storage facilities and process laboratory facilities. There are no administrative employees housed inside the canyon or reactor buildings. Some buildings in the 300 Area where radioactive material is stored or handled have offices in the front of the buildings.

SRNL houses both administrative offices and laboratory facilities. The administrative offices are located near the front of the building. Surveys indicate that the dose rates in this area are very low. The back of the building houses the laboratory facilities where radioactive material is handled and stored.

Unauthorized Practices and Group Monitoring

Radiological Control was not aware of any unauthorized practices in the field related to dosimetry. Workers in general followed the radiological control rules. They did like to complain though. The construction force relied heavily on health physics and trusted them. There were some individuals who hunted and fished onsite regardless of the rules not to do so.

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There are a lot of anecdotal stories for former workers on group monitoring; however, this was not an acceptable practice at any time in SRS history. In the 1990s, SRS instituted a policy for escorts and visitors to wear a dosimeter, and if any positive dose was assigned from it, it was assigned to each member of the group. Note that this only applied to visitors (as currently defined by DOE), not workers visiting from off-site to perform work at SRS.

Also, in areas where neutron dosimetry was required historically, workers routinely assigned to the area wore their neutron dosimeter at all times. A worker visiting from another facility was only required to don a neutron dosimeter if they entered a facility where the neutron dose rate was 1 mrem/hour. Thus, in such facilities you could have regular workers wearing both beta/gamma dosimeters and neutron dosimeters and a visiting worker only wearing a beta/gamma dosimeter.

Relationship with the State

The site interacts with both the South Carolina Department of Health and Environmental Control (SCDHEC) and the Georgia Department of Natural Resource (GDNR). The interaction involves primarily environmental issues and emergency response activities. There is a designated individual at the site who provides interaction with the state agencies. When working in Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) and/or Resource Conservation and Recovery Act (RCRA) space, there has to be state agreement on closure standards.

Concerns

With the initial development of the TBD, minimal site expert input was solicited from the SRS Internal Dosimetry, External Dosimetry, and Radiological Records groups. This may be due to the potential conflict of interest issues. Key radiological control staff have not read the SRS TBD.

NIOSH has used a matrixed approach to assigning hypothetical intakes to workers that were not on a monitoring program. The internal monitoring program at SRS used air sampling and field indicators, as well as bioassay, to detect potential intakes. The field monitoring results were used as a method for triggering personnel monitoring. It is very unlikely that an intake resulting in a dose of 5 rem CEDE or more of transuranics would be missed without some indication of a problem in the air sampling or other field data. Based on the technology used for air sampling and personnel monitoring, it is possible that an intake resulting in a dose of 1 Rem CEDE of transuranics was missed for some individuals. The application of an exposure matrix for unmonitored individuals is scientifically flawed and results in credibility issues for the site.

HPAREH has over the period of time included fields which may only be applicable to specific years of operation. For example, the AEDE field was populated during the period DOE required reporting of individual dose in terms of AEDE. With the switch to CEDE, no additional data was included in this field. When using the HPAREH file, NIOSH should be cognizant of this, and not assume all fields are complete for all periods of time. In terms of internal dose, the most appropriate value to be used is the most current calculation for that individual. Note that internal

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dose is recalculated as new bioassay becomes available and as new ICRP models are mandated by DOE.

Final Comments

Among the most challenging radiological conditions at SRS, is the FB-line in 221F Building. This is an aging facility that has had a number of upset conditions in the past. The chemical separations areas account for over 25% of the collective dose at the site. In addition, a majority of the Price Anderson Amendment Act violations and B level occurrences have occurred in these facilities. There is a high level of glove failures in this area, thus respiratory protection is required despite the engineering controls in place. Another challenging radiological control situation is first time evolutions for Decontamination and Decommissioning work. There is considerable uncertainty in what radiological conditions will be encountered during this type of work.

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ATTACHMENT 5

SRS FACILITY SITE EXPERT INTERVIEW SUMMARY — PRODUCTION AND CONSTRUCTION WORKER STAFF

Movement from the Site

Historically, staff turnover has not been as big of an issue as it has been in the last 15-20 years. As an example, we will discuss the health physics department. Professional staff was hired from all over the country. DuPont Operations did their primary hiring in the 1953-1955 time frame. There was very little turnover in this department through the 1950s, 1960s and 1970s. In 1966 SRS hired its first two new professional Health Physicists. Late in the 1970s and 1980s the original staff started to retire and was replaced with new staff. Today, not as many individuals stay with one company their entire career as in the past.

Movement on the Site

In general, the operations personnel remained within the same area of the plant. Supervisors were moved around to different facilities as they were needed. Most administrative support personnel were housed in Building 704. A few of these individuals were located in the production areas.

DuPont had three divisions at the SRS: Construction, SRP Operations, and SRL. Initial staff was hired as early as 1951. There was movement among these three divisions. With respect to the nonexempt staff, individuals were initially hired as General Service Operators (GSOs). DuPont had an extensive internal training program. After working with this group for a year or so, they had the option to move into specialty work such as operations, laboratory work, health physics, etc. DuPont developed its own in-house training program. Additional training was required for GSOs that become reactor operators, laboratory technicians, or health physics technicians. GSOs, in general, were not radiation workers. The GSO concept was similar to the hiring hall concept.

Full-time workers were typically assigned to a specific facility. Those without seniority were often assigned to shift work as the plant operated around the clock. Once an individual graduated from shift work they were offered an option to move wherever they wanted to within the limits of their job functions. For example, an individual would be hired on as a GSO and work at that job for a couple of years. After receiving further training, for example, as a Chemical Separations Operator, he would be put on a rotational shift. After 10-years or so, he had the option to move to a more favorable job. Often, these individuals moved to a less radiologically hazardous job, which could have involved a different work unit onsite.

Normal progression for longer-term employees was from nonexempt to exempt. After about 15 years, nonexempt staff would be promoted to supervision. As a supervisor they, in general, receive less radiation exposure.

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There were some exceptions to this generally stationary work force. Health physics technicians were deliberately rotated so that they had experience at a variety of facilities. In the last 20-years there have been multipurpose technicians who can be loaned between facilities (e.g., F-canyon technicians can be loaned to H-canyon).

Operations

The CMX and TNX facilities were built to serve as prototypes for full-scale operations in 1951. The CMX and TNX facilities produced a large number of products. There were three or four test reactors/critical assemblies onsite. These small reactors were used for materials testing. These test reactors were similar to a research reactor. The neutron exposure hazard in these areas would be expected to be low; however, it would have resulted in a higher portion of the total whole-body dose. The first radioactive material to arrive onsite was likely used at the 777M Test Reactor.

Reactor targets and fuel (i.e., uranium rods, lithium targets, and other target material) were fabricated in the Raw Materials (300) Area. Fuel was irradiated in the reactors. The irradiation time depended on the material desired (i.e., Pu-238, Pu-239, H-3, Cf-252, Cm). Reactors consisted of the zero or ground level (top of the reactor), the Minus-20 level (heat exchanger area), and the Minus-40 level (water coolant pump area). The Pin Room was under the bottom of the reactor. When the reactor was shut down, the fuel was removed from the reactor and put into the disassembly basin. The fuel was pulled out of the reactor with a crane in the Crane Area. As the fuel was pulled out of the reactor, it was sprayed with water. The crane then put the fuel in the channel and it was moved to the disassembly area for storage. During this process, the fuel was out of the water for a few minutes. The charge machine would reload the reactor. The irradiated fuel was allowed to decay in the disassembly basin for a preset period of time (e.g., typically 90-180 days for Pu-239).

Railroad cars would back into the disassembly area. A shielded cask was loaded with decayed irradiated fuel in the disassembly basin and placed on railroad cars. The material was then transported down to the 200 Areas for processing. The railroad car would back into the appropriate separations canyon, depending on the type of material being abstracted from the fuel. The fuel was loaded into the dissolver to remove the aluminum jacket on the rod. The resulting uranium and plutonium advanced down the A- and B-lines, respectively. Liquid plutonium was converted into metal buttons. Further information on the Separations Process can be found in the Bebbington document. The Separations facilities were approximately 800 feet long and divided into 18 sections. Refer to Figure A5-1 for a schematic of the separations process.

SRP produced Pu-238 for the deep space program. In this process an Np target was put in the reactor and irradiated. After removal and decay, the target was transferred to the separations facility. The target was liquefied, fission products are extracted, and further chemistry was done to separate the Pu-238. The purified plutonium underwent a finishing process. There were special challenges associated with Pu-238. Plutonium-238 is more difficult to detect due to the absence of Pu-241 in the mixture. In other processes, the Pu-241 decays to Am-241, which is easily detectable. The dose conversion factor for Pu-238 is the same as for Pu-239. Personnel

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monitoring for Np started with the establishment of the production of Heat Sources for the deep space program. This operation did not occur at the site in the 1950s.

The site had a number of laboratory facilities. The Process Control Laboratories (Buildings 772F and 772-1F) were responsible for analyzing samples pulled from the separations process. SRNL was involved with a number of Research and Development activities. Exposure conditions were highly dependent on the particular activities an individual was involved in. SRNL work involved a large number of radionuclides including fission products, activation products, Cf-252, and actinides. SRP was the chief supplier of Cf-252 sources prior to the commercialization of this process.

The SRS Tank Farms consist of primarily double-shelled tanks, with a few single-shelled tanks. As a result of the high radiation in those tanks, radiolytic decomposition of water resulted in the formation of hydrogen gas. There have been ventilation system problems with the tanks from off gassing. Due to the nature of the waste stream, the waste must be cooled. To minimize waste volumes, evaporation of waste is performed to remove excess water in the waste. This water is necessary to ensure the waste drains to the waste tanks, but once there, much of it becomes excess. The waste separates in different media including salts and sludge. Stress corrosion has been an issue with the tanks resulting in some tank leaks. There are thousands of transfers of waste each year. Leaks and spills are most likely during these transfers. Contamination at the Tank Farms is predominantly beta/gamma; however alpha contamination is present. It is cleaned up as detected.

The burial grounds are located between the 200F and 200H areas. Plutonium trenches are separate from fission/activation product trenches. In the early years of operations, there were spent solvents from the 200F and 200H areas burned in the middle of the burial grounds.

There was a variety of fuel processed through the separations facility. Spent fuel was received from the Y-12 Plant, research reactors, Idaho National Engineering and Environmental Laboratory, Hanford, the Navy and other DOE sites. The diagram below shows the general process flow for separations activities.

Radiological Hazards

During the DuPont years at SRS, the contractor was very safety conscious. Individuals were required to follow safety and radiological control rules. If they chose not to do this, disciplinary action was taken. Generally, the most hazardous areas of the plant involved separations and Cf-252 production. The original production of Pu-238 was challenging due to the out-of-date facilities used for initial separation and purification of this material. The highest personnel exposures onsite likely occurred on the FB-line.

Workers were required to acknowledge the Special Work Permits for jobs where they were used. At some facilities, timekeeping was used. This was done to limit personnel exposure and track daily dose. Bioassay was based on the worker's task assignment.

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When the reactors originally started up, the field personnel identified leaks in the reactor shielding, which allowed xenon gas to escape into occupied areas of the reactor. The holes were plugged to prevent further occurrences. There was observed airborne activity from noble gases in the reactor areas from time to time. It is uncertain whether impact of submersion dose from these gases was assessed for particular employees. Reactor workers did wear dosimeters and were subject to whole-body counts. The whole-body count would have been effective in detecting intakes of some particulate and absorbed gas fission products. In general, reactor health physics felt the reactor shields were intact.

The neutron doses in the production reactors were quite low. The reactors had shielding and access to more hazardous areas of the reactor was controlled. There was no access to the Crane Area of the reactor when the reactor was operating. Entries into the Crane Area to perform maintenance were made when the reactor was down.

The earlier periods of operation at the F and H Canyons required more hands-on work. Prior to the implementation of ALARA, doses were generally higher. In the mid-1960s, separations, maintenance, reactor, and health physics personnel received an average of 2.5 Rem per year. Since the inception of operations, SRS has had an Administrative Control Level with respect to cumulative annual dose. Initially the ACL was set at 3 Rem per year (external and tritium).

Construction Workers

When DuPont constructed the site, they had a division referred to as DuPont Construction. Following the takeover of the site by Westinghouse, this construction division went away. Many construction workers were hired from the union halls. Trades workers included iron workers, asbestos installers, bricklayers, pipefitters, laborers, and other maintenance and craft workers. Some of the construction workers had regular jobs at the site, while others were temporary employees. These individuals worked at multiple facilities on the site. They were involved with evasive work including maintenance, repair, and demolition, as well as construction of new facilities.

There were a number of radiological issues associated with the subset of workers at the Savannah River Site and other facilities. The jobs performed by construction workers were often short-term, high-risk jobs. Temporary workers, who numbered in the thousands, wore badges only during their time onsite, and had to obtain a new badge when they returned to the site for additional work. Also, it was difficult to get follow-up bioassay on these workers as they did not necessarily stay in the immediate geographical area. Construction work history and radiation exposure records were stored separately from those of operational personnel, especially in the early years.

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Additional References

Refer to the SRS histories and ERDA-1737 (SRP Environmental Impact Statement) for a list of valuable references on SRS operations. The various Safety Analysis Reports, Technical Standards, and Operating Standards will also provide information. These references will provide additional names of individuals involved in operations.

Another source of information is the local Citizen's Advisory Board. The SRS CAB was formed about 12 years ago in order to provide suggestions to DOE. It is currently composed of four committees, which concentrate on waste management, facility disposition and site remediation, nuclear materials, and strategic and legacy management. Their meeting minutes are available through the SRS web site under CAB.

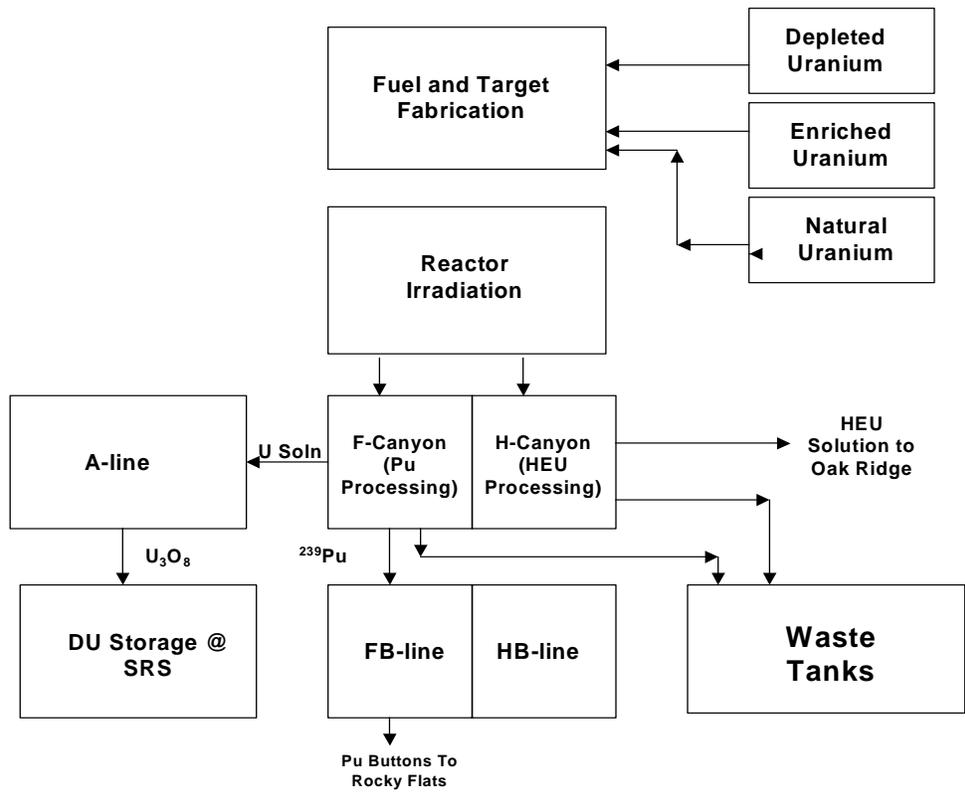


Figure 1: Outline of the Separations Process at the Savannah River Site as Described by a Site Expert

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ATTACHMENT 6

CONSISTENCY BETWEEN SAVANNAH RIVER SITE AND HANFORD SITE PROFILES

Table A.6.1 Occupational Medical Exposure Default Assumption Comparison for the Savannah River Site and Hanford

Description of Assumption	SRS	Hanford
Frequency of chest x-rays (Default)	One annual x-ray procedure for each year or partial year.	Posterior-Anterior View: Before 1946 – 1/1982: Pre-employment, annual, and termination 1/82-1/83: Pre-employment, annual, and termination for over 50 years; Biennially for 40-49 years; Every third year for 39 years or younger. 1/83-3/90: Biennially for over 50 years; Every third year for 40-49; and Every five years for 39 years and younger. 3/90 – present: Every five years Lateral chest x-rays also given periodically prior to 4/1997.
Organ Dose Conversion Factors	Obtained from ICRP 34 (1982)	Obtained from ICRP 34 (1982)
IREP Radiation Rate	Acute	Acute
IREP Radiation Type	Photons, 30-250 keV	Photons, 30-250 keV
IREP Dose Distribution Type	Constant	Constant
Total uncertainty	30% (x-ray dose multiplied by 1.3 and entered as a constant)	30% (x-ray dose multiplied by 1.3 and entered as a constant)
Conversion Factor from PA to Lateral	2.5	2.5
Chest Thickness	PA View: 26 cm Lateral View: 34 cm	PA View: N/A Lateral View: N/A
Substitute dose conversion factors for thyroid, eye/brain, ovaries and analogues, testes, and uterus	Substitute view and organ DCFs applied to minimally collimated beams prior to 1970. (Scalsky 2004, p. 50)	Use DCFs for lung for all other organs in thoracic cavity; for organs in abdomen, use DCFs for the ovary (Scalsky 2003, p. 10)
Analogue organ for Thymus	Lung	Lung
Analogue organ for Esophagus	Lung	Lung
Analogue organ for Stomach	Lung	Lung
Analogue organ for Bone Surface	Lung	Lung
Analogue organ for Liver, gall bladder, spleen	Lung	Ovary
Analogue organ for Remainder Organs	Lung	Ovary
Analogue organ for Urinary/bladder and colon/rectum	Ovary	Ovary
Analogue organ for Eye/brain	Thyroid	Thyroid

**Table A.6.1 Occupational Medical Exposure Default Assumption Comparison for the Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
<i>Posterior-Anterior View X-ray Techniques^{1,2}</i>		
<1946	Site not in operation.	kVp: Unknown mAs: Unknown SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 120 mR
2/1946 – 12/ 1950	kVp: 80 mAs: 30 SSD: 152 cm SID: 183 cm Filtration: 1.5 mm Al ESE: 108 mR	kVp: 80 mAs: 25 SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 79 mR
1/1951 - 4/19/59	kVp: 80 mAs: 30 SSD: 152 cm SID: 183 cm Filtration: ESE: 108 mR	kVp: 80 mAs: 10 SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 79 mR
4/1959 – 12/1970	kVp: 80 mAs: 30 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 108 mR	kVp: 80 mAs: 10 SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 40 mR
1/1971 – 1/1983	kVp: 110-120 mAs: 10 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 44 mR	kVp: 80 mAs: 10 SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 40 mR
1/1983 – 7/1985	kVp: 110-120 mAs: 10 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 44 mR	kVp: 100 mAs: 10 SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al ESE: 35 mR
8/1985 – 3/1990	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 100 mAs: 10 SSD: 72" (183 cm) SID: 183 cm Filtration: 2.5 mm Al; 4.0 mm Al for CONX Type 12 ESE: 35 mR

Table A.6.1 Occupational Medical Exposure Default Assumption Comparison for the Savannah River Site and Hanford (continued)

Description of Assumption	SRS	Hanford
3/1990 – 4/1997	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 6.7 SSD: 72 " (183 cm) SID: 183 cm Filtration: 4.0 mm Al ESE: 21 mR
4/1997 – 2/1998	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 10 SSD: 183 cm SID: 183 cm Filtration: 4.0 mm Al ESE: 17 mR
2/1998 – 5/1999	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 5 SSD: 183 cm SID: 183 cm Filtration: 4.0 mm Al ESE: 11 mR
5/1999 – present	kVp: 120 mAs: 7.5 SSD: 152 cm SID: 183 cm Filtration: 3.5 mm Al ESE: 33 mR	kVp: 110 mAs: 5 SSD: 183 cm SID: 183 cm Filtration: 4.0 mm Al ESE: 11 mR
Photofluorography		
Technique Factors	kVp: 100 mAs: 60 SID: 102 cm Filtration: 2.5 mm Al ESE: Applies from 1951-1957	kVp: 80 to 100 kVp mAs: not specified SID: 102 cm Filtration: 2.5 mm Al ESE: 1.53 R Applies 1945 to 1962

¹ Refer to Scalsky 2004, pages 41-47 for SRS x-ray technique discussion.

² Refer to Scalsky 2003, page 18 for Hanford x-ray technique summary.

³ N/A = not applicable; PA = posterior-anterior; LAT = lateral; kVp = kilovolt potential; mAs = milliampere-second; SSD = source-to-skin distance; SID = source-to-image distance; ESE = entrance skin exposure

Table A.6.2 External Exposure Default Assumption Comparison for the Savannah River Site and Hanford

Description of Assumption	SRS	Hanford
Missed Photon Dose Application	Applies to workers with no recorded dose because they weren't monitored or their results are unavailable; and workers who have a zero recorded dose (Scalsky 2004, p. 111).	Applies to workers with no recorded dose because they weren't monitored or their results are unavailable; and workers who have a zero recorded dose, (Fix 2004, p. 75).
Missed Photon Dose Methodology	<ol style="list-style-type: none"> (1) For a claimant-favorable maximum potential missed dose, use the limit of detection (LOD) multiplied by the number of zero doses (Scalsky 2004, pp. 111 and 238) (2) Divide the limit of detection (LOD) 2, and multiply by the number of zeros and not monitored periods; (Scalsky 2004, p. 242), or (3) Missed doses are added to measured doses and treated as a constant. 	Divide the MDL by 2, and multiply by the number of zeros and not monitored periods (Fix 2004, p. 75). Table 6E.6 (Fix 2004), provides potential maximum photon dose by year.
IREP Dose Distribution Type for missed photon dose	<ol style="list-style-type: none"> (1) When using the Limit of Detection (LOD)/2 methodology, a lognormal distribution with a geometric standard deviation of 1.52 in Parameter 2 of the IREP input is used (Scalsky 2004, p. 116). (2) When simply adding the missed and measured dose, a constant is used. 	Lognormal distribution with a geometric standard deviation of 1.52. ¹ The assessment at Hanford was based on the assumption that uncertainties from individual sources followed independent lognormal distributions. For each uncertainty source, a factor is assigned reflecting bias (B) and a 95% uncertainty factor (K); the uncertainty factor was determined so that the interval obtained by dividing and multiplying by this factor would include 95% of all observations (Fix 2004, p. 27).
Missed Neutron Dose Application	Assign a missed neutron dose if there is neutron monitoring between 1958 and 1962, if there is neutron monitoring in 1971 or later, or there is indication of use of the 17 keV calibration curve for interpretation of beta/gamma film. Also applies to those who worked with Cf or Cm, maintenance workers, those involved in the PuAl target campaign, and those on routine plutonium bioassay. If the recorded neutron dose is greater than the calculated dose, the calculated dose is used (Neton 2003).	Assign a missed neutron dose if the individual worked in a facility with a potential for neutron exposure, The vast majority of neutron dose to Hanford workers was received at the 200 West Area Plutonium Finishing Plant (PFP) facilities (p. 74.) There is potential for significant missed dose in the 300 Area plutonium laboratory (308, 309, 324), the 100 Area reactor facilities (i.e., reactors, (B, D, F, H, DR, C, KW, KE), the 300 Area accelerator (3754B), the calibrations facilities (3745, 318) and the Fast Flux Test Reactor (p. 73). (Fix 2004).

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**Table A.6.2 External Exposure Default Assumption Comparison for the Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Missed Neutron Dose Methodology	<p>A neutron-to-photon ratio is applied to missed and recorded photon dose for nonmonitored worker and workers with inadequate neutron monitoring (i.e., prior to 1971). The upper 95% value is used for the maximizing technique. The geometric mean value is used for the best-fit technique (Scalsky 2004, pp. 240-241).</p> <p>After 1970, the assignment of missed dose is based on the limit of detection provided in Table E-10 (Scalsky 2004, pp. 241-242).</p> <p>It appears that an ICRP 60 correction factor is applied to missed dose; however, this is unclear in the TBD (Scalsky 2004, p. 110).</p>	<p>A neutron-to-photon ratio is applied to missed and recorded photon dose for nonmonitored worker and workers with inadequate neutron monitoring. The upper 95% value is used for the maximizing technique. The mean value is used for the best-fit technique (Fix 2004, pp. 75-77).</p>
IREP Dose Distribution Type for missed neutron dose	Lognormal distribution with a geometric standard deviation of 1.52. ¹	Lognormal distribution with a geometric standard deviation of 1.52. ¹
IREP Exposure Rate	Acute for beta and photon Chronic for neutron (Scalsky 2004, pp. 87 and 235, respectively).	Acute for beta and photon Chronic for neutron (Fix 2004, pp. 8, 59, and 69, respectively)
IREP Radiation Type (default)	Photon, 30-250 keV Electron, > 15 keV, Neutron, 0.1-2 MeV (Scalsky 2004, pp. 49, 236, and 237, respectively)	Photon, 30-250 keV Electron, > 15 keV Neutron, 0.1-2 MeV (Fix 2004, p. 29)
Organ dose conversion factor	<p>For the maximizing approach, a value of one is used (TBD, p. 61).</p> <p>For the best-fit analysis, the dose conversion factors in the external dosimetry guide for the relevant exposure geometry. OCAS-IG-001 Appendix A (NIOSH 2002) contains a detailed discussion of the conversion of measured dose to organ dose equivalent, and Appendix B contains the appropriate dose conversion factors (DCFs) for each organ, radiation type, and energy range based on the type of monitoring performed. (Scalsky 2004, p. 242)</p>	<p>The dose conversion factors for each, organ, radiation type, and energy ranged from OCAS-IG-001 are used. If the exposure geometry cannot be determined, default values are found in Table 6E-9 (Fix 2004, p. 77). No separate value is provided for the maximizing approach.</p>

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**Table A.6.2 External Exposure Default Assumption Comparison for the Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Exposure geometry	<p>Default exposure: Likely non-compensable workers - 100% AP Compensable worker – 50% AP, 50% ROT Compensable supervisor – 50% AP, 50% ISO. Dose reconstructor has the option to choose the most appropriate exposure geometry for the individual. (Scalsky 2004, p. 242)</p>	<p>Default exposure: Likely non-compensable workers - 100% AP Compensable worker – 50% AP, 50% ROT Compensable supervisor – 50% AP, 50% ISO. (Fix 2004, p. 77)</p>
Photon Adjustment Factors (Recorded Dose)	<p>Multiply by 1.119 for years prior to 1987. Multiply by 1.039 for 1987. No adjustment is needed post-1987 (Scalsky 2004, p. 238).</p> <p>Note: Taylor et al. (1995) indicates that the 1.119 adjustment factor should be applied through 1985 and the 1.039 adjustment factor should be applied for 1986. No correction is required for 1987 and after.</p>	<p>No adjustment for the multi-element dosimeter, TLD, or gamma dose. For 200 Area plutonium workers prior to 1957, the 20% of the open window dose is added to the penetrating dose (Fix 2004, p. 73).</p>
IREP Dose Distribution Type for recorded photon dose	<p>Constant. The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included.¹</p>	<p>Constant.¹</p>
Recorded Neutron Dose Adjustment Factor (Prior to 1971 – SRS; Prior to 1972 Hanford)	<p>NTA film is considered inadequate for use in dose reconstruction due to the energy dependence. The missed neutron dose approach is applied for this period of time. If the measured dose from the NTA is greater than the calculated dose, this value is used and the ICRP 60 conversion factor is applied (Scalsky 2004, p. 238).</p>	<p>NTA film is considered inadequate for use in dose reconstruction due to the energy dependence. The missed neutron dose approach is applied for this period of time (Fix 2004, p. 48).</p>
Recorded Neutron Dose Adjustment Factor (7/78-12/83)	<p>In order to calculate the dose input for the IREP, Table E-1, the recorded neutron dose must be separated into neutron energy groups as shown in Table E-3 and subsequently converted to ICRP 60 (1990) methodology (Scalsky 2004, 235-238).</p>	<p>When using the four-chip HMPD during the period of its use from July 1978 through December 31, 1983 in Hanford 200 and 300 Area plutonium facilities only, multiply the recorded neutron dose by 1.35. At all other times, divide the dose into the facility specific neutron energy bins, and multiply by the ICRP 60 conversion factor (Fix 2004, p. 74).</p>

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**Table A.6.2 External Exposure Default Assumption Comparison for the Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Recorded Neutron Dose Adjustment Factor (1/72-6/78, 1/84 – present)	In order to calculate the dose input for the IREP, Table E-1, the recorded neutron dose must be separated into neutron energy groups as shown in Table E-3 and subsequently converted to ICRP 60 (1990) methodology (Scalsky 2004, 235-238).	Divide the recorded neutron dose into the facility specific neutron energy bins, and multiply by the ICRP 60 conversion factor (Fix 2004, pg 74).
IREP Dose Distribution Type for recorded neutron dose	Constant. The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included. ¹	Constant ¹
Shallow Dose Adjustment Factors	Shallow dose adjustments factors are not addressed in the TBD or SRS TIBs.	Shallow dose adjustments factors are not addressed in the TBD.
Low-energy photons (< 30 keV)	1954-1981 Subtract the reported deep dose from the shallow dose for plutonium workers. 1982-present. Plutonium workers are those individuals that worked in 321M, 221H – B line, 221F – B line, 772F, 235F, 773A, 736A, and other plutonium storage areas (Neton 2004). (For testicular, breast, or skin cancer)	The stated Hanford practice to include 1/5 of the shallow dose based on a 16-keV calibration to the deep dose for Hanford plutonium facilities workers could resolve this source of potential under-response around 17 keV (Fix 2004, pg 26). For 200 Area workers prior to 1957, the 20% of the open window dose is added to the penetrating dose, (p. 14).
IREP Dose Distribution Type for recorded shallow dose	Shallow dose is addressed from a technical perspective in the TBD, but no direction is provided to the dose reconstructor (Scalsky 2004, p. 97).	Not included in the TBD.
IREP Radiation Type for recorded dose	Specific to the particular facility for beta, photon, and neutron dose. For example, in the reactor area 100% of the beta doses is assumed to be >15 keV, 50% of the photon dose is >250 keV, and 50% of the photon dose is 30-250 keV (Scalsky 2004, p. 98).	Specific to the particular facility for beta, photon, and neutron dose. For example, in the reactor area 100% of the beta doses is assumed to be >15 keV, 75% of the photon dose is >250 keV, and 25% of the photon dose is 30-250 keV (Fix 2004, p. 29).

¹ These parameters were obtained from review of several dose reconstruction IREP input sheets.

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Table A.6.3 Comparison of Default Assumptions for Internal Exposure at Savannah River Site and Hanford

Description of Assumption	SRS	Hanford
Particles Size (default)	5 micron (Scalsky 2004, Section 4.0, Attachment D)	5 micron (Bihl 2004, p. D-10)
Intake Type (default)	Chronic (Scalsky 2004, Section 4.0, Attachment D)	Chronic (Bihl 2004, p. 7-9)
Default Excretion Volume	1.4 liters/day (Volumes less than 1.4 liters/day are corrected by normalizing the actual volume to 1.4 liters/day. Samples recorded as activity per 1.5 liters are not corrected.) (Scalsky 2004, p. 70)	Uses a urinary excretions value of 0.2 ug/d for elemental analyses, 0.15 dpm/d for ²³⁴ U and ²³⁸ U and essentially anything detected for ²³⁵ U (Bihl 2004, p., 27)
Solubility Class	For the maximizing approach, the most claimant-favorable solubility type for the organ of interest is used. For the best-fit approach the most appropriate solubility type can be used (Scalsky 2004, p. 85).	For the maximizing approach, the most claimant-favorable solubility type for the organ of interest is used. For the best-fit approach the most appropriate solubility type can be used. Inhalation class and lung absorption type for uranium is found in Bihl 2004, Table 5.2.5-3, p. 24).
Intake Date for Hypothetical Intake (excluding tritium)	Acute inhalation on January 1 in the first year of employment (Scalsky 2004, p. 85; Bracket 2003, p. 3).	First day of employment or the first day of operation of the facility where the worker was assigned. For separation plants, chronic intakes would apply from either the first day of work for the worker or the start-up of the plant, December 1944 for T Plant and April 1945 for B Plant (Bihl 2004, p. 8).
Tritium Missed Dose Application	Assigned to workers monitored for external dose, but having no bioassay. For workers not in the dosimetry or bioassay-monitoring program, the missed internal dose is based on environmental intake only. Scalsky 2004, p. 84; Duncan 2003, pp 6 and 12)	Assigned to workers who worked in 108-B, the 300 Area Test Reactors, and in some cases where work location was unknown or variable. Those who never wore a dosimeter and had no bioassay results were assigned environmental doses (Bihl 2004, pp. 21-22).
Basis for Tritium Missed Dose	Dose calculated based on the tritium reporting level for a particular time period (Scalsky 2004, p. 67; Duncan 2003, p. 6).	Tritium urinalysis was not perfected until 1961. Liquid scintillation counting for tritium likely was started in 1958 (Bihl 2004, pp. 21-22). From 1949 to 1960 the MDA was 5 uCi/L and from 1961 to 1981 the MDA was 1 uCi/L. Later in 1982 the MDA changed to 10 dpm/ml and in 1991 to 20 dpm/ml, (Bihl 2004, p. 22). Tritium intakes were accounted for as part of external dose until about 1986-87 (TBD does not explain methodology), when they were entered in the dose database as internal dose (Bihl 2004, pp. 12 & 22).

**Table A.6.3 Comparison of Default Assumptions for Internal Exposure at Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Hypothetical Intake Application	Applied to claims with non-metabolic and digestive tract cancers (Scalsky 2004, p. 85; Bracket 2003, p. 2).	Applied to individuals who wore a dosimeter but did not have any bioassay (Bihl 2004, p. 48).
Basis for missed internal dose from radionuclides other than tritium	<p>(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose (Scalsky 2004, p. 84; Bracket 2003, p. 2).</p> <p>(2) For those individuals with external monitoring but no or limited internal monitoring, an annual missed tritium dose and environmental dose from uranium, plutonium, and ¹³¹I are assigned as internal dose. It is also reasonable to pick a fission or activation product that produces the largest dose to the organ of interest (Scalsky 2004, p. 84; Bracket 2003, p. 8).</p> <p>(3) Highest five intakes for various nuclides are applied to those individuals with non-metabolic or digestive system cancers (Bracket 2003, p. 2).</p>	<p>(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose, (Bihl 2004, p. 48)</p> <p>(2) For those individuals with external monitoring but no or limited internal monitoring, the approach was year dependent. For 1947 through 1952, daily intakes at 10% of the respiratory protection required value for 40 hours/week were assumed. Iodine was assumed to be at 0.1 times the vapor index. For 1953 through 1988, daily intakes were based on an exposure to airborne concentrations at 10% of the limiting air concentration for four hours per week, (Bihl 2004, p. 49).</p> <p>(3) From 1989 through the present, a daily exposure at 5% of the limiting air concentration for 4 hours per week was assumed, (Bihl 2004, p. 50).</p> <p>(4) For monitored workers with no confirmed intake, a maximum intake is determined by using the MDA of the last sample as an upper bound (Bihl 2004, p. 47).</p>

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**Table A.6.3 Comparison of Default Assumptions for Internal Exposure at Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Radionuclides included in the Hypothetical Intake	^{241}Am / ^{241}Pu (M), ^{244}Cm (M), ^{60}Co (S), ^{137}Cs (F), ^{237}Np (M), ^{238}Pu (M), ^{239}Pu (M), ^{90}Sr (F), ^{234}U (F), and ^{238}U (F) (Bracket 2003, p. 9)	Variable by facility and organ of interest. Alpha intakes are assigned for the Plutonium Finishing Plant (PFP), the 200 Area Fuel Separations Plants, U-Plant, C-Plant, the 300 Area Fuel Fabrication Facilities, 209E, 120, 324, 325, 327, the Tank Farms and evaporator facilities (0.5 times the alpha intake), and where work location is unknown or highly variable. Alpha intakes are based primarily on ^{234}U or ^{239}Pu . Beta/gamma intakes are assigned for all facilities <i>except</i> PFP, 209E, 120, the 300 Area Fuel Fabrication Facilities, 108-B, and U-Plant. Tritium intakes are assigned for the 108-B Building, the 300 Area Test Reactors, and in some situations where work locations are unknown or variable. The particular beta/gamma radionuclide and its solubility class are determined based on the organ of concern. For some facilities and periods of time it is specified (Bihl 2004, pp. 51-52).
Default Activity Ratios Pu Mixture	Ten-year old 12% plutonium mix (Scalsky 2004, p. 66).	Not specified in the TBD.
Activity Fractions for other Mixtures	Activity fractions are facility dependent. The activity fractions are taken from the Internal Dosimetry Technical Basis Manual (WSRC 1990). The information for these ratios was obtained from safety analysis reports, personal interviews, open literature, etc.	Activity fractions are provided for uranium mixtures, Table 5.2.5-3, page 24, weapons and fuel grade plutonium, Table 5.2.1-3 page 16, and recycled uranium impurities., Table 5.2.5-2, page 24. Default mixtures based fission product urinalysis was developed by time period and organ of concern (Bihl 2004, p. 10, Attachment D).
Radionuclides of Concern for Monitored Workers	Radionuclides of concern were based on the in vivo and in vitro bioassay data of the individual (Scalsky 2004, pp. 66 & 67). Although the TBD provides activity fractions in Attachment A, it is not clear how these activity fractions are used in dose calculations.	Radionuclides of concern were based on the in vivo and in vitro bioassay data of the individual, or the minimum detectable activity for a particular radionuclide. Radionuclide assumptions varied by facility and organ of interest (Bihl 2004, p. 13).

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Table A.6.3 Comparison of Default Assumptions for Internal Exposure at Savannah River Site and Hanford (continued)

Description of Assumption	SRS	Hanford
Tritium Dose for Monitored Workers	Based on the reporting level if the tritium bioassay is less than this level, or the actual bioassay result if it is greater than the reporting level. Organically Bound Tritium and Stable Metal Tritides are not considered (Bracket 2003, p. 6).	Tritium urinalysis was not perfected until 1961. Liquid scintillation counting for tritium likely was started in 1958 (pp. 21-22). From 1949 to 1960 the MDA was 5 uCi/L and from 1961 to 1981 the MDA as 1 uCi/L. Later in 1982 the MDA changed to 10 dpm/ml and in 1991 to 20 dpm/ml (p. 22). Tritium intakes were accounted for as part of external dose until about 1986-1987 (TBD doses not explain methodology), when they were entered in the dose database as internal dose (pp. 12 & 22). (Bihl 2004, pp. 12 & 22)
Internal Dose for radionuclides other than tritium	Based on either actual bioassay values or detection levels for bioassay techniques. For non-metabolic cancers, the maximizing approach is used (Scalsky 2003, p. 85).	Based on either actual bioassay values for positive values. Based on a chronic intake over the entire exposure period with the last sample assumed to be at the MDA (Bihl 2004, p. 47).
Basis for pre-bioassay program doses	Not included in the TBD.	Air concentration tolerance or limits, (Bihl 2004, pg. 7)
Ingestion	Not included in the TBD.	Assigned during periods were air sampling was used to determine internal dose. The quantity is based on the air concentration level or on the guidance provided in <i>Estimation of Ingestion Intakes</i> (NIOSH 2004). (Bihl 2004, p. 8)
Surrogate Radionuclide in IMBA for $^{65}\text{Zn}/^{95}\text{Zr}$	^{137}Cs used as a surrogate. Surrogate Adjustment factor = 2.43. (Brackett 2003, p. 9)	Not included in the TBD.
Surrogate Radionuclide in IMBA for $^{106}\text{Ru}/^{144}\text{Ce}/^{95}\text{Nb}$	Radionuclides not available in IMBA. ^{90}Sr used as a surrogate. Surrogate Adjustment factor = 7.25 (Brackett 2003, p. 9).	Not included in the TBD.
Surrogate Radionuclide in IMBA for $^{242}\text{Cm}/^{252}\text{Cf}$	Radionuclides not available in IMBA. ^{244}Cm used as a surrogate. Surrogate Adjustment factor = 1.09 (Brackett 2003, p. 9).	Not included in the TBD.
IREP Radiation Types for Hypothetical Intake	Alpha Beta: >15 keV Tritium: < 15 keV (Bracket 2003, pp. 8 & 12)	Alpha ¹ Beta: >15 keV ¹ Photon: > 250 keV ¹ Tritium: < 15 keV ¹
IREP Dose Distribution Type	Constant (Brackett 2003, p. 12)	Constant ¹

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**Table A.6.3 Comparison of Default Assumptions for Internal Exposure at Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Internal Dose Uncertainty	<p>For the missed dose assignments, the value entered includes the uncertainty. ¹ No direction is provided to the dose reconstructor for dose assignments based on monitoring data.</p>	<p>For the missed dose assignments, the value entered includes the uncertainty. For dose assignments based on monitoring data, the following values can be applied as a standard deviation: (1) 0.3 times the MDA or reporting level, or (2) 0.5 times the MDA for chest counting. Actually report errors can be used if available (Bihl 2004, p. 46). For air concentration data, a triangular distribution with zero as the minimum, the derived values as the mode, and twice the mode as the maximum is used (Bihl 2004, p. 7).</p>
Other Comments	None.	<p>Informs the dose reconstructor of limited use radionuclides such as ¹⁴C, ²³²Th, radon, ⁹⁰Y, ²²⁷Th, ²²⁷Ac, and ³²P (Bihl 2004, p. 32)</p>

¹ These parameters were obtained from review of several Hanford dose reconstruction IREP input sheets.

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Table A.6.4 Comparison of Default Assumptions for Environmental Exposure at Savannah River Site and Hanford

Description of Assumption	SRS	Hanford
Application	Apply the annual internal and external environmental dose for each full or partial year of employment for the maximizing approach. Dose reconstructors are instructed to use only the maximum annual intakes in Table C-17 for the maximizing approach (Scalsky 2004, p. 179). For the best-fit approach, modifications can be made for partial year of employment. No environmental dose is assigned if the background is not subtracted from the workers badge (Scalsky 2004, p. 62).	Environmental doses are assigned to personnel with no bioassay and no evidence of having worn a dosimeter at the Hanford Site (Bihl 2004, p. 48).
Sources of Environmental Releases Considered	The TBD heavily references the Cummins (1991) and CDC (2001) documents, and dose not include many of the base assumptions from those reports in the TBD. It is apparent that releases from the reactors and separations areas were considered.	T-plant particles and iodine, B-Plant particles and iodine, REDOX particles and iodine, PUREX particles and iodine, Z-Plant particles, reactor noble gases, and tritium from 108B Building (Savignac 2003, p. 18).
Source Term Basis	<i>Radioactive Releases from the Savannah River Plant 1954-1989</i> (Cummins 1991), <i>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</i> (CDC 2001), SRS meteorology data, SRS environmental reports for 1993-2001.	Hanford Works environmental reports; Methods for Estimating Radiation Doses from Short-Lived Gaseous Radionuclides and Radioactive Particles Released to the Atmosphere During Early Operations at Hanford (Till et al. 2002).
Methodology	Gaussian model (Scalsky 2004, Section 3.1.1)	Puff advection (RATCHET) model (Savignac 2003, p. 14)
Type of Releases	The TBD heavily references the Cummins (1991) and CDC (2001) documents, and dose not include many of the base assumptions from those reports in the TBD.	Calculations included routine and identified non-routine releases. Estimates include inhalation of radionuclides in air, direct external radiation from plumes, and physical contact with particulate radionuclides on skin.
Ventilation Rate (m3/year)	2,400 (default); Adjustments can be made for light and heavy work (Scalsky 2004, p. 162).	2,400 (default); Based on 1.2 m ³ /hour ± 0.4 m ³ /hour (Savignac 2003, p. 16)
Exposure Time (hours/week)	40 with a 1.25 conversion factor to increase the exposure time to 50 hours/week (Scalsky 2004, p. 61).	40 (Savignac 2003 p. 24)

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Table A.6.4 Comparison of Default Assumptions for Environmental Exposure at Savannah River Site and Hanford (continued)

Description of Assumption	SRS	Hanford
Mobile Workforce	Assign the maximum dose listed for any area onsite.	Information not included in the TBD.
Facility Specific Workforce	Assign the maximum dose listed for any area onsite for the maximizing approach. Assign an area specific environmental dose based on the work location of the worker for the best-fit approach (Scalsky 2004, p. 61).	Information not included in the TBD.
Radionuclides Considered for External Dose	⁴¹ Ar, (Scalsky 2004, p. 60)	⁴¹ Ar, ¹³¹ I, ¹⁰⁶ Ru (Savignac 2003, pp. 19 and 23)
Radionuclides Considered for Submersion Dose	⁴¹ Ar, (Scalsky 2004, p. 59)	⁴¹ Ar, page 17, ¹³¹ I, ³ H Kathy – can't find evidence that these last two belong here.
Submersion DCF	Assumed values from the Federal Guidance Report 12 (EPA 1993). (Scalsky 2004, p. 60)	Federal Guidance Report No. 13, <i>Cancer Risk Coefficients for Environmental Exposure to Radionuclides</i> , 1999.
Radionuclides Considered for Internal Dose.	³ H, ¹³¹ I, ²³⁸ Pu, ²³⁹ Pu, ²⁴⁰ Pu, ²³⁴ U, ²³⁵ U, and ²³⁸ U (Scalsky 2004, p. 51)	³ H, ¹³¹ I, ^{131m} Xe, ¹⁴⁴ Ce, ¹⁴⁴ Pr, ¹³⁷ Cs, ¹³⁷ Ba, ²³⁹ Pu, ¹⁰³ Ru, ^{103m} Rh, ¹⁰⁶ Ru, ¹⁰⁶ Rh, ⁹⁰ Sr, ⁹⁰ Y, ⁹⁵ Zr, ⁹⁵ Nb (Savignac 2003, p. 8)
Soil	Density = 1,600 kg/m ³ Surface Factor = 0.08 Resuspension Factor = 1E-9/m (Scalsky 2004, p. 59)	Not included in the TBD.
Liquid Effluents	Not included in the TBD.	Not included in the TBD.
Organ Dose Conversion Factor	1.0 is used in the maximizing approach. The organ dose conversion factors in the external dosimetry guide for the relevant exposure geometry are used in the best-fit analysis (Scalsky 2004, p. 61).	Not included in the TBD.
IREP Rate	Chronic (Scalsky 2004, pg. 61)	Chronic ¹
IREP Radiation Type	Photon, 30-250 keV ⁴¹ Ar, 100% photon, > 250 keV (Scalsky 2004, pp. 60 & 61)	Photon, 30-250 keV ¹
IREP Dose Distribution Type	Constant. Doses and intake quantities provided with a 50 th -percentile and a geometric standard deviation. A 95 th percentile for the source term is estimated as 25% greater than the 50 th percentile (Scalsky 2004, p. 60).	Constant. Doses and intake quantities provided with a geometric mean and standard deviation. There is no direction on how these values should be entered into IREP.

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**Table A.6.4 Comparison of Default Assumptions for Environmental Exposure at Savannah River Site and Hanford
(continued)**

Description of Assumption	SRS	Hanford
Special Considerations for Uranium and Plutonium	The isotope yielding the maximum organ dose was assumed at 100% rather than applying a mixture (Scalsky 2004, p. 59).	Not applicable.
Other	1955 values are assigned to 1952, 1953, and 1954 (Scalsky 2004, pg 54)	The four chemical separations plants, T Plant, B Plant, REDOX Plant and the PUREX plant, along with the plutonium handling Z plant are shown in Figure 4.1.1 to be the most important release points at Hanford (Savignac 2003).

¹ These parameters were obtained from review of several Hanford dose reconstruction IREP input sheets.

ATTACHMENT 7

EVALUATION OF INTAKES DERIVED USING ICRP 30 VERSUS ICRP 68 METHODOLOGIES

The tables below compare the relative intakes for all radionuclides listed in tables 1 and 2, pages 4 and 5, ORAUT-OTIB-0001, derived using ICRP 30 Intake Retention Fractions (IRFs) and ICRP 68 Intake Retention Fractions (IRFs). Intakes were back-calculated assuming a constant bioassay monitoring result (unitary bioassay result, for example), measured at different times after intake

1. Comparison of Pu relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: Pu-238

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	2.30E-04	2.83E-04	8.13E-01	1.23E+00
5	3.90E-05	5.39E-05	7.24E-01	1.38E+00
10	1.50E-05	2.54E-05	5.91E-01	1.69E+00
50	8.50E-06	1.45E-05	5.86E-01	1.71E+00
100	6.80E-06	1.11E-05	6.13E-01	1.63E+00
180	5.60E-06	8.50E-06	6.59E-01	1.52E+00
200	5.10E-06	7.85E-06	6.50E-01	1.54E+00
300	4.20E-06	5.80E-06	7.24E-01	1.38E+00
360	3.85E-06	4.90E-06	7.86E-01	1.27E+00

The intakes from Pu, type M, are underestimated using ICRP 30 methodology.

Radionuclide: Pu-238, Pu-239, Pu-241

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type S	Class Y	Intakes 30/60	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	2.30E-06	1.55E-05	1.48E-01	6.74E+00
5	4.50E-07	2.87E-06	1.57E-01	6.38E+00
10	2.20E-07	1.29E-06	1.71E-01	5.86E+00
50	1.70E-07	7.56E-07	2.25E-01	4.45E+00
100	1.60E-07	6.97E-07	2.30E-01	4.36E+00
180	1.60E-07	7.25E-07	2.21E-01	4.53E+00
200	1.60E-07	7.34E-07	2.18E-01	4.59E+00
300	1.60E-07	7.75E-07	2.06E-01	4.84E+00
360	1.70E-07	7.80E-07	2.18E-01	4.59E+00

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The intakes from Pu, type S, are underestimated using ICRP 30 methodology.

2. Comparison of Am relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: Am-241

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	ICRP 68 Urine IRF	ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
1	1.80E-03	6.66E-03	2.70E-01	3.70E+00
5	7.20E-05	5.24E-05	1.37E+00	7.28E-01
10	4.90E-05	4.97E-05	9.86E-01	1.01E+00
50	2.00E-05	3.48E-05	5.75E-01	1.74E+00
100	1.50E-05	2.22E-05	6.76E-01	1.48E+00
180	1.10E-05	1.25E-05	8.80E-01	1.14E+00
200	1.00E-05	9.70E-06	1.03E+00	9.70E-01
300	8.00E-06	5.47E-06	1.46E+00	6.84E-01
360	7.20E-06	4.50E-06	1.60E+00	6.25E-01

For Am, type M, ICRP 30 methodology may or may not underestimate the intakes. It will depend on the time samples are taken after the intake.

Radionuclide: Am-241

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Am-241 in matrix of Type S compounds of Pu

Time (d) after intake	Type S	Class Y	Intake 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	3.01E-05	3.76E-04	8.01E-02	1.25E+01
5	1.40E-06	2.21E-06	6.33E-01	1.58E+00
10	9.90E-07	1.87E-06	5.29E-01	1.89E+00
50	5.28E-07	1.91E-06	2.76E-01	3.62E+00
100	4.59E-07	1.97E-06	2.33E-01	4.29E+00
180	4.30E-07	2.04E-06	2.11E-01	4.74E+00
200	4.27E-07	2.06E-06	2.07E-01	4.82E+00
300	4.18E-07	2.13E-06	1.96E-01	5.10E+00
360	4.14E-07	2.15E-06	1.93E-01	5.19E+00

The intakes from Am, type S, are underestimated using ICRP 30 methodology.

3. Comparison of U relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: U-234 - U-235 - U-238

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type F	Class D	Intakes 30/68	IRF 30/68
	ICRP 68	ICRP 30		
	Urine IRF	Urine IRF		
1	1.80E-01	1.87E-01	9.63E-01	1.04E+00
5	4.20E-03	1.31E-02	3.21E-01	3.12E+00
10	2.70E-03	7.26E-03	3.72E-01	2.69E+00
50	3.00E-04	6.67E-04	4.50E-01	2.22E+00
100	1.00E-04	1.11E-04	9.01E-01	1.11E+00
180	4.40E-05	4.40E-05	1.00E+00	1.00E+00
200	2.40E-05	5.15E-06	4.66E+00	2.15E-01
300	8.90E-06	1.80E-06	4.94E+00	2.02E-01
360	6.00E-06	1.70E-06	3.53E+00	2.83E-01

The intakes from U, type F, are underestimated using ICRP 30 methodology, for samples taken up to 180 days after the intake. For type F, it is very unlikely that samples are taken after 180 days exposure.

Radionuclide: U-234 - U-235 - U-238

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68	ICRP 30		
	Urine IRF	Urine IRF		
1	2.30E-02	4.13E-02	5.57E-01	1.80E+00
5	7.30E-04	2.69E-03	2.71E-01	3.68E+00
10	5.40E-04	1.75E-03	3.09E-01	3.24E+00
50	1.90E-04	4.80E-04	3.96E-01	2.53E+00
100	1.10E-04	2.43E-04	4.53E-01	2.21E+00
180	7.00E-05	7.00E-05	1.00E+00	1.00E+00
200	5.80E-05	7.49E-05	7.74E-01	1.29E+00
300	3.20E-05	2.33E-05	1.37E+00	7.28E-01
360	2.30E-05	1.00E-05	2.30E+00	4.35E-01

ICRP 30 underestimates U type M intakes for all reasonable times of collecting samples, after an intake occurred.

Radionuclide: U-234 - U-235 - U-238

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type S	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	7.00E-04	2.23E-03	3.14E-01	3.19E+00
5	2.20E-05	1.31E-04	1.68E-01	5.95E+00
10	1.60E-05	8.42E-05	1.90E-01	5.26E+00
50	5.70E-06	2.34E-05	2.44E-01	4.11E+00
100	4.10E-06	1.87E-05	2.19E-01	4.56E+00
180	3.45E-06	1.83E-05	1.89E-01	5.29E+00
200	3.20E-06	1.81E-05	1.77E-01	5.66E+00
300	2.80E-06	1.83E-05	1.53E-01	6.54E+00
360	2.68E-06	1.83E-05	1.47E-01	6.81E+00

The intakes from U, type S, are underestimated using ICRP 30 methodology.

- Comparison of Np relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: Np-237

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Type S	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	6.20E-03	3.42E-03	1.81E+00	5.52E-01
5	3.40E-04	3.02E-05	1.13E+01	8.88E-02
10	1.30E-04	2.56E-05	5.08E+00	1.97E-01
50	6.20E-05	1.78E-05	3.48E+00	2.87E-01
100	4.20E-05	1.13E-05	3.72E+00	2.69E-01
180	2.75E-05	6.50E-06	4.23E+00	2.36E-01
200	2.40E-05	4.97E-06	4.83E+00	2.07E-01
300	1.60E-05	2.82E-06	5.67E+00	1.76E-01
360	1.30E-05	2.40E-06	5.42E+00	1.85E-01

The intakes from Np, are underestimated using ICRP 68 methodology.

5. Comparison of Cm-242 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: Cm-242

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	1.80E-03	6.63E-03	2.71E-01	3.68E+00
5	7.00E-05	5.13E-05	1.36E+00	7.33E-01
10	4.70E-05	4.77E-05	9.85E-01	1.01E+00
50	1.60E-05	2.81E-05	5.69E-01	1.76E+00
100	9.80E-06	1.45E-05	6.76E-01	1.48E+00
180	6.24E-06	7.20E-06	8.67E-01	1.15E+00
200	4.40E-06	4.15E-06	1.06E+00	9.43E-01
300	2.20E-06	1.53E-06	1.44E+00	6.95E-01
360	1.50E-06	1.53E-06	9.80E-01	1.02E+00

For Cm-242, type M, ICRP 30 methodology may or may not underestimate the intakes. It will depend on the time samples are taken after the intake.

6. Comparison of Cm-244 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: Cm-244

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	1.77E-03	6.63E-03	2.67E-01	3.75E+00
5	7.17E-05	5.24E-05	1.37E+00	7.30E-01
10	4.85E-05	4.97E-05	9.75E-01	1.03E+00
50	2.02E-05	3.46E-05	5.84E-01	1.71E+00
100	1.48E-05	2.20E-05	6.74E-01	1.48E+00
180	1.08E-05	1.52E-05	7.11E-01	1.41E+00
360	6.80E-06	7.07E-06	9.62E-01	1.04E+00

For Cm-244, type M, ICRP 30 methodology most of the time underestimates the intakes.

7. Comparison of Sr-90 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine bioassay data:

Radionuclide: Sr-90

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type F	Class D	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	6.80E-02	8.57E-02	7.93E-01	1.26E+00
5	9.20E-03	2.45E-02	3.76E-01	2.66E+00
10	4.10E-03	1.04E-02	3.94E-01	2.54E+00
50	3.30E-04	1.94E-04	1.70E+00	5.88E-01
100	9.80E-05	1.26E-04	7.78E-01	1.29E+00
180	6.40E-05	8.40E-05	7.62E-01	1.31E+00
200	5.00E-05	7.42E-05	6.74E-01	1.48E+00
300	2.90E-05	5.04E-05	5.75E-01	1.74E+00
360	2.20E-05	4.02E-05	5.47E-01	1.83E+00
400	1.80E-05	3.71E-05	4.85E-01	2.06E+00

For Sr-90, type F, ICRP 30 methodology underestimates the intakes for most of the times samples are taken.

Radionuclide: Sr-90

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type S	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	8.10E-04	1.34E-03	6.04E-01	1.65E+00
5	1.30E-04	4.22E-04	3.08E-01	3.25E+00
10	6.10E-05	1.87E-04	3.26E-01	3.07E+00
50	8.70E-06	1.62E-05	5.37E-01	1.86E+00
100	4.40E-06	1.55E-05	2.84E-01	3.52E+00
180	3.40E-06	1.51E-05	2.25E-01	4.44E+00
200	3.00E-06	1.50E-05	2.00E-01	5.00E+00
300	2.40E-06	1.48E-05	1.62E-01	6.17E+00
360	2.20E-06	4.65E-06	4.73E-01	2.11E+00

For Sr-90, type S, ICRP 30 methodology underestimates the intakes.

8. Comparison of Co-60 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo bioassay data (whole-body counting):

Radionuclide: Co-60

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	4.90E-01	5.66E-01	8.66E-01	1.16E+00
5	9.10E-02	2.06E-01	4.42E-01	2.26E+00
10	7.20E-02	1.63E-01	4.42E-01	2.26E+00
50	4.40E-02	9.78E-02	4.50E-01	2.22E+00
100	3.10E-02	5.77E-02	5.37E-01	1.86E+00
180	2.30E-02	3.46E-02	6.65E-01	1.50E+00
200	1.90E-02	2.44E-02	7.79E-01	1.28E+00
300	1.30E-02	1.40E-02	9.29E-01	1.08E+00
360	1.06E-02	1.15E-02	9.22E-01	1.08E+00

For Co-60, type M, ICRP 30 methodology underestimate the intakes.

9. Comparison of Cs-137 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo bioassay data (whole-body counting):

Radionuclide: Cs-137

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type F	Type D	Intakes 30/68	IRF 30/68
	ICRP 68 Tot. Body	ICRP 30 Tot. Body		
1	6.00E-01	6.22E-01	9.65E-01	1.04E+00
5	4.30E-01	5.72E-01	7.52E-01	1.33E+00
10	4.10E-01	5.43E-01	7.55E-01	1.32E+00
50	3.20E-01	4.19E-01	7.64E-01	1.31E+00
100	2.30E-01	3.05E-01	7.54E-01	1.33E+00
200	1.20E-01	1.61E-01	7.45E-01	1.34E+00
300	6.40E-02	8.55E-02	7.49E-01	1.34E+00

The intakes from Cs-137 are underestimated using ICRP 30 methodology.

10. Comparison of Ce-144 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo bioassay data (whole-body counting):

Radionuclide: Ce-144

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	TYPE M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	4.96E-01	5.96E-01	8.32E-01	1.20E+00
5	9.30E-02	2.44E-01	3.81E-01	2.62E+00
10	7.97E-02	2.06E-01	3.87E-01	2.58E+00
50	6.19E-02	1.52E-01	4.07E-01	2.46E+00
100	5.08E-02	1.13E-01	4.50E-01	2.22E+00
200	3.78E-02	7.58E-02	4.99E-01	2.01E+00
300	2.87E-02	5.60E-02	5.13E-01	1.95E+00
400	2.20E-02	4.26E-02	5.16E-01	1.94E+00

The intakes from Ce-144 are underestimated using ICRP 30 methodology.

11. Comparison of Cf-252 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from urine data:

Radionuclide: Cf - 252

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	TYPE M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 Urine IRF	ICRP 30 Urine IRF		
1	1.30E-03	3.20E-03	4.06E-01	2.46E+00
5	1.42E-05	2.68E-05	5.30E-01	1.89E+00
10	1.32E-05	2.54E-05	5.20E-01	1.92E+00
50	8.43E-06	1.76E-05	4.79E-01	2.09E+00
100	5.75E-06	1.12E-05	5.13E-01	1.95E+00
180	3.72E-06	5.87E-06	6.34E-01	1.58E+00
360	1.86E-06	2.55E-06	7.29E-01	1.37E+00

The intakes from Cf-252, type M, are underestimated using ICRP 30 methodology.

12. Comparison of Nb-95 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo data (whole-body counting):

Radionuclide: Nb-95

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	4.90E-01	5.82E-01	8.42E-01	1.19E+00
5	8.30E-02	2.11E-01	3.93E-01	2.54E+00
10	6.10E-02	1.54E-01	3.96E-01	2.52E+00
50	1.90E-02	4.60E-02	4.13E-01	2.42E+00
100	5.50E-03	1.17E-02	4.70E-01	2.13E+00
200	5.20E-04	8.79E-04	5.92E-01	1.69E+00
300	5.10E-05	7.66E-05	6.66E-01	1.50E+00
400	5.10E-06	7.17E-06	7.11E-01	1.41E+00

The intakes from Nb-95, type M, are underestimated using ICRP 30 methodology.

13. Comparison of Ru-106 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo data (whole-body counting):

Radionuclide: Ru-106

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type F	Class D	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	5.10E-01	5.35E-01	9.53E-01	1.05E+00
5	2.10E-01	3.47E-01	6.05E-01	1.65E+00
10	1.70E-01	2.88E-01	5.91E-01	1.69E+00
50	8.30E-02	1.39E-01	5.97E-01	1.67E+00
100	5.50E-02	9.36E-02	5.87E-01	1.70E+00
180	4.00E-02	2.90E-02	1.38E+00	7.24E-01
200	3.60E-02	6.13E-02	5.87E-01	1.70E+00
300	2.70E-02	4.65E-02	5.80E-01	1.72E+00
360	2.30E-02	3.98E-02	5.78E-01	1.73E+00

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The intakes from Ru-106, type F, are underestimated using ICRP 30 methodology, most of the times.

Radionuclide: Ru-106

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M		Class W	
	ICRP 68 WB IRF	ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
1	4.90E-01	5.84E-01	8.39E-01	1.19E+00
5	9.90E-02	2.36E-01	4.20E-01	2.38E+00
10	8.00E-02	1.91E-01	4.20E-01	2.38E+00
50	4.70E-02	1.09E-01	4.32E-01	2.31E+00
100	3.10E-02	6.31E-02	4.91E-01	2.04E+00
180	2.10E-02	1.52E-02	1.38E+00	7.24E-01
200	1.70E-02	2.75E-02	6.18E-01	1.62E+00
300	1.10E-02	1.64E-02	6.70E-01	1.49E+00
360	9.00E-03	1.31E-02	6.87E-01	1.46E+00

The intakes from Ru-106, type M, are underestimated using ICRP 30 methodology, most of the times.

Radionuclide: Ru-106

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type S		Class Y	
	ICRP 68 WB IRF	ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
1	4.90E-01	5.85E-01	8.38E-01	1.19E+00
5	8.60E-02	1.98E-01	4.34E-01	2.30E+00
10	7.10E-02	1.63E-01	4.36E-01	2.30E+00
50	4.70E-02	1.39E-01	3.38E-01	2.96E+00
100	3.50E-02	1.19E-01	2.95E-01	3.39E+00
180	2.75E-02	8.69E-02	3.17E-01	3.16E+00
200	2.50E-02	8.93E-02	2.80E-01	3.57E+00
300	1.80E-02	6.83E-02	2.63E-01	3.80E+00
360	1.55E-02	5.76E-02	2.69E-01	3.72E+00

The intakes from Ru-106, type S, are underestimated using ICRP 30 methodology.

14. Comparison of Zn-65 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo data (whole-body counting):

Radionuclide: Zn-65

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type S	Class Y	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	5.39E-01	6.05E-01	8.91E-01	1.12E+00
5	2.68E-01	3.96E-01	6.77E-01	1.48E+00
10	2.48E-01	3.66E-01	6.78E-01	1.48E+00
50	1.80E-01	2.85E-01	6.32E-01	1.58E+00
100	1.37E-01	2.27E-01	6.04E-01	1.66E+00
200	8.75E-02	1.50E-01	5.83E-01	1.71E+00
300	5.67E-02	1.00E-01	5.67E-01	1.76E+00
400	3.69E-02	6.72E-02	5.49E-01	1.82E+00

The intakes from Zn-65, type S, are underestimated using ICRP 30 methodology.

15. Comparison of Zr-95 relative intakes (ICRP 30/ICRP 68), assuming intakes were calculated from in vivo data (whole-body counting):

Radionuclide: Zr-95

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type F	Class D	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	5.40E-01	5.80E-01	9.31E-01	1.07E+00
5	2.30E-01	3.76E-01	6.12E-01	1.64E+00
10	1.80E-01	3.08E-01	5.85E-01	1.71E+00
50	8.20E-02	1.40E-01	5.85E-01	1.71E+00
100	4.70E-02	8.06E-02	5.83E-01	1.71E+00
180	2.10E-02	3.36E-02	6.25E-01	1.60E+00
200	1.60E-02	2.71E-02	5.91E-01	1.69E+00
300	5.40E-03	9.09E-03	5.94E-01	1.68E+00
400	1.80E-03	3.05E-03	5.90E-01	1.70E+00

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The intakes from Zr-95, type F, are underestimated using ICRP 30 methodology.

Radionuclide: Zr-95

Intake: Inhalation

Aerosol size: 5.0 micron AMAD (68) and 1.0 micron AMAD (30)

Time (d) after intake	Type M	Class W	Intakes 30/68	IRF 30/68
	ICRP 68 WB IRF	ICRP 30 WB IRF		
1	4.90E-01	0.594	8.25E-01	1.21E+00
5	8.50E-02	2.03E-01	4.19E-01	2.38E+00
10	6.60E-02	1.69E-01	3.91E-01	2.56E+00
50	3.00E-02	7.51E-02	4.00E-01	2.50E+00
100	1.50E-02	3.24E-02	4.63E-01	2.16E+00
200	4.30E-03	7.97E-03	5.40E-01	1.85E+00
300	1.30E-03	2.41E-03	5.39E-01	1.86E+00
400	4.30E-04	7.85E-04	5.48E-01	1.83E+00