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

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 **under**estimate 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.*

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 PuF_4, 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 ± 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.

Dosimeter Type	Yearly Exchange Frequency (*n*)	Estimated LOD (mrem)	Max. Annual Missed Dose (mrem)
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

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(LOD), while in other instances, this dose is defined as n(LOD)/2. For example for neutron doses in Table 5.3.1-1, maximum missed doses are defined as n(LOD)/2, while values in Table 5.5.2-1 (and Table E.4.1.6 of Attachment E) define these values as n(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]. . . .

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:"

Claim Status	Job Category	Exposure Geometry	Percentage
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:

	Year of X-ray		
Dose to:	1950–1970	1971–1985	1985–1999
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 **micro**rem 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

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

- 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.

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 unauditable). 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			
Issue9: 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

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 95^{th} 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

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 resuspenstion, 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 unauditable 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

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.

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

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.

(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

- 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

- 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.

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.

(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.

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.

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

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?

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 ^{85}Kr 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

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

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?

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.

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.

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 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

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.

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?

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

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 µCi/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 µCi/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?

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?

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.

- 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 though 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.

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.

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:

Plant Area	Primary Radiological Hazards
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 (RWPs). 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

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 cm^2 alpha and 80 dpm/100 cm^2 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 cm^2 and 1,000 dpm/100 cm^2, respectively. Total contamination limits were 300 dpm/100 cm^2 and 5,000 dpm/100 cm^2 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 cm^2 total alpha as a contamination limit, rather than the traditionally used 100 dpm/100 cm^2 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 cm^2 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

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

- 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

Nick Name/Acronym	Description
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

- 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 cm^2) 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

- Mud Bucket (CD container filled with paraffin in which BF_3 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 panaromic 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:

Nick Name/Acronym	Description
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

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 PuF_4 50 cm above a paraffin drum. This was to account for some of the scatter. The PuF_4 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 PuF_4 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.

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.

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

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.

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. UO_3 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.

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.)

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 is 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.

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

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 Diethylenetriaminepentaaetate (DPTA) used. Ca-DTPA was the primary type. Generally aerosol administration was used and less frequently, intramuscular injection.

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.

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

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.

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.

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

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.

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.

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.

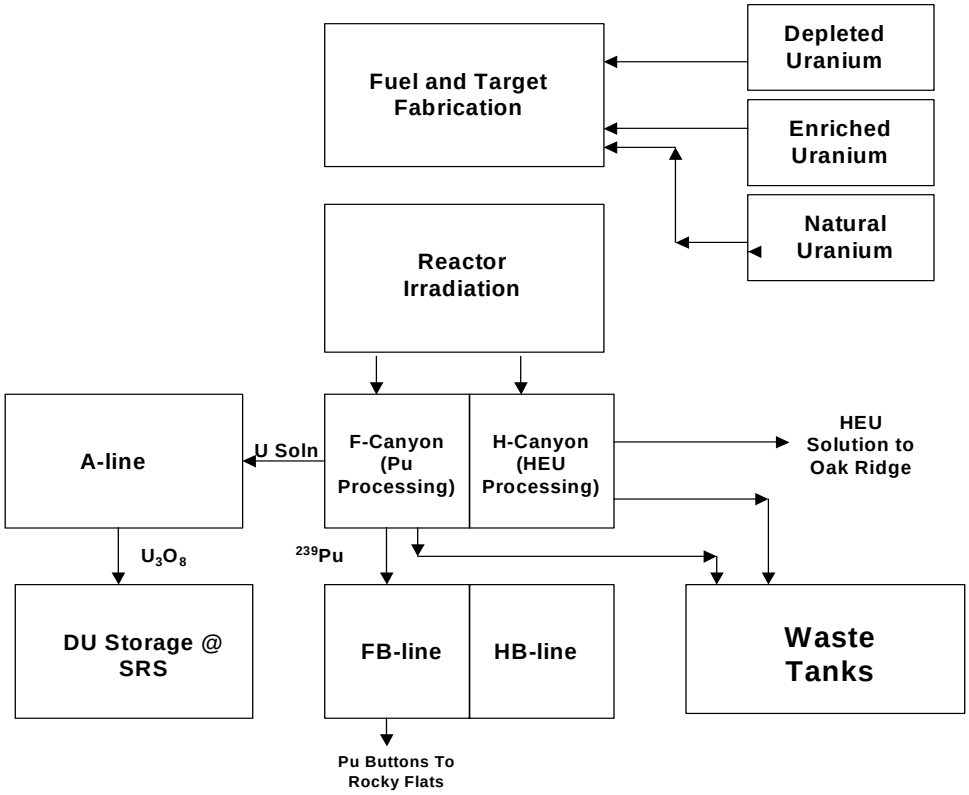

Figure 1: Outline of the Separations Process at the Savannah River Site as Described by a Site Expert

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
Posterior-Anterior View X-ray Techniques[1,2]		
<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	(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	(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).

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	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). After 1970, the assignment of missed dose is based on the limit of detection provided in Table E-10 (Scalsky 2004, pp. 241-242). It appears that an ICRP 60 correction factor is applied to missed dose; however, this is unclear in the TBD (Scalsky 2004, p. 110).	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).
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	For the maximizing approach, a value of one is used (TBD, p. 61). For the best-fit analysis, the dose conversion factors in the external dosimetry guide for the relevant exposure geometry. OCAS-IG-001 Appendix A (NIOSH 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)	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.

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

Description of Assumption	SRS	Hanford
Exposure geometry	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)	Default exposure: Likely non-compensable workers - 100% AP Compensable worker – 50% AP, 50% ROT Compensable supervisor – 50% AP, 50% ISO. (Fix 2004, p. 77)
Photon Adjustment Factors (Recorded Dose)	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). 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.	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).
IREP Dose Distribution Type for recorded photon dose	Constant. The adjustment factor encompasses the uncertainty so no additional uncertainty factors are included. [1]	Constant.[1]
Recorded Neutron Dose Adjustment Factor (Prior to 1971 – SRS; Prior to 1972 Hanford)	NTA film is considered inadequate for use in dose reconstruction due to the energy dependence. The missed neutron dose approach is applied for this period of time. 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).	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).
Recorded Neutron Dose Adjustment Factor (7/78-12/83)	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).	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).

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.

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	(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose (Scalsky 2004, p. 84; Bracket 2003, p. 2). (2) For those individuals with external monitoring but no or limited internal monitoring, an annual missed tritium dose and environmental dose from uranium, plutonium, and ^{131}I are assigned as internal dose. It is also reasonable to pick a fission or activation product that produces the largest dose to the organ of interest (Scalsky 2004, p. 84; Bracket 2003, p. 8). (3) Highest five intakes for various nuclides are applied to those individuals with non-metabolic or digestive system cancers (Bracket 2003, p. 2).	(1) Individuals with no external or internal monitoring data were assigned an environmental internal dose, (Bihl 2004, p. 48) (2) For those individuals with external monitoring but no or limited internal monitoring, the approach was year dependent. For 1947 through 1952, daily intakes at 10% of the respiratory protection required value for 40 hours/week were assumed. Iodine was assumed to be at 0.1 times the vapor index. For 1953 through 1988, daily intakes were based on an exposure to airborne concentrations at 10% of the limiting air concentration for four hours per week, (Bihl 2004, p. 49). (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). (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).

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 ***except*** PFP, 209E, 120, the 300 Area Fuel Fabrication Facilities, 108-B, and U-Plant. Tritium intakes are assigned for the 108-B Building, the 300 Area Test Reactors, and in some situations where work locations are unknown or variable. The particular beta/gamma radionuclide and its solubility class are determined based on the organ of concern. For some facilities and periods of time it is specified (Bihl 2004, pp. 51-52).
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., Table5.2.5-2, 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).

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 *Estimation of Ingestion Intakes* (NIOSH 2004). (Bihl 2004, p. 8)
Surrogate Radionuclide in IMBA for $^{65}Zn/^{95}Zr$	^{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 $^{106}Ru/^{144}Ce/^{95}Nb$	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 $^{242}Cm/^{252}Cf$	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]

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	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.	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).
Other Comments	None.	Informs the dose reconstructor of limited use radionuclides such as ^{14}C, ^{232}Th, radon, ^{90}Y, ^{227}Th, ^{227}Ac, and ^{32}P (Bihl 2004, p. 32)

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

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	*Radioactive Releases from the Savannah River Plant 1954-1989* (Cummins 1991), *Savannah River Site Dose Reconstruction Project Phase II: Source Term Calculation and Ingestion Pathway Data Retrieval, Evaluation of Materials Released from the Savannah River Site* (CDC 2001), SRS meteorology data, SRS environmental reports for 1993-2001.	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)

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, *Cancer Risk Coefficients for Environmental Exposure to Radionuclides*, 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/m3 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 50th-percentile and a geometric standard deviation. A 95th percentile for the source term is estimated as 25% greater than the 50th 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.

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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class Y ICRP 30 Urine IRF	Intakes 30/60	IRF 30/68
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

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 ICRP 68 Urine IRF	Class Y ICRP 30 Urine IRF	Intake 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class D ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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.

4. 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 ICRP 68 Urine IRF	Type S ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class D ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 WB IRF	Class W ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Tot. Body	Type D ICRP 30 Tot. Body	Intakes 30/68	IRF 30/68
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 ICRP 68 WB IRF	Class W ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 Urine IRF	Class W ICRP 30 Urine IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 WB IRF	Class W ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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 ICRP 68 WB IRF	Class D ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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

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 ICRP 68 WB IRF	Class W 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 ICRP 68 WB IRF	Class Y 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 ICRP 68 WB IRF	Class Y ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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 form 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 ICRP 68 WB IRF	Class D ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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

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 ICRP 68 WB IRF	Class W ICRP 30 WB IRF	Intakes 30/68	IRF 30/68
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