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ADVISORY BOARD ON RADIATION AND WORKER HEALTH

Review of ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*, Revision 1

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Review of ORAUT-OTIB-0054, ***Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*, Revision 1**	Supersedes: N/A
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ABBREVIATIONS AND ACRONYMS

Al	aluminum
ATR	Advanced Test Reactor
d	day
DCF	dose conversion factor
DOE	U.S. Department of Energy
FFTF	Fast Flux Test Facility
g	gram
GA	General Atomics
HEPA	high efficiency particulate air
IMBA	Integrated Modules for Bioassay Analysis
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
IREP	Interactive RadioEpidemiological Program
IRF	intake retention fraction
kg	kilogram
kW	kilowatt
MDA	minimum detectable activity
MDL	minimum detectable level
MTHM	metric ton of heavy metal (p. 13)
MTU	metric ton unit
MW	megawatt
MWd	megawatt-day
n/cm^2-s	neutron per square centimeter per second (p. 13)
NIF	normalized intake fraction
NIOSH	National Institute for Occupational Safety and Health
ORAUT	Oak Ridge Associated Universities Team
ORIGEN	Oak Ridge Isotope Generator
ORNL	Oak Ridge National Laboratory
OTIB	ORAUT Technical Information Bulletin
pCi	picocurie
PIC	pocket ion chamber

POC	probability of causation
PWR	Pressurized Water Reactor
RF	release fraction
SC&A	S. Cohen and Associates (SC&A, Inc.)
SRS	Savannah River Site
SS	stainless steel
Sv/Bq	Sievert per Becquerel
$T_{1/2}$	half-life
TLD	thermoluminescent dosimeter
TRIGA	Training Research and Isotope Production, General Atomics
y	year

EXECUTIVE SUMMARY

This report presents a technical review of *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses*, ORAUT-OTIB-0054, Rev. 1, June 13, 2013 (ORAUT 2013, hereafter referred to as "the OTIB" or "OTIB-0054"). SC&A had previously reviewed Rev. 0 of the OTIB (ORAUT 2007) in 2008 (SC&A 2008) and its findings were subsequently discussed in several communications between SC&A and NIOSH and at Subcommittee on Procedures Review meetings. However, since the technical basis of the Oak Ridge Associated Universities Team (ORAUT) Technical Information Bulletin (OTIB) was substantially revised in Rev. 1 of the OTIB, the Subcommittee at its July 18, 2013, meeting authorized SC&A to perform a full review of the latest revision. This review is limited to those aspects of the OTIB judged by SC&A to be potentially significant to dose reconstruction, and is not a line-by-line check of data sources and calculational results.

Frequently, air-sampling or urinalysis data on worker exposure to mixed fission and activation products associated with nuclear reactors or nuclear fuel are available only in the form of gross beta or gross gamma activity unattributed to specific radionuclides. This is particularly true for exposures during the early decades of the U.S. nuclear program. For those cases, OTIB-0054 provides guidance and a standard approach to the dose reconstructor on how to assign radionuclide-specific intakes to exposed workers, as follows:

> *Interpretation of the results of these gross analyses of air samples requires knowledge of the activity ratios of the various fission and activation products in the air breathed by the worker and the beta or gamma yield of each product as measured by the detector. Interpretation of gross urinalysis results requires knowledge of the ratios of the products in urine at some time after the start of chronic intake and whether those ratios changed due to sample processing activities* (ORAUT 2013, Section 4.0).

The OTIB, however, states that its guidance should only be used in the absence of, or to supplement, site- and personnel-specific data that may be available for a particular dose reconstruction. The OTIB contains several major parts:

- A lengthy and detailed explanation of how the representative reactors are modeled and fission and activation product activities determined, resulting in the selection of four reactors and four decay times (Section 5)
- Determination of intake fractions and identification of dosimetrically significant radionuclides, along with reduction of the size of the radionuclide dataset (Section 6)
- Determination of activity and intake fraction values to assign intakes from gross gamma or beta assay data (Section 7)
- Instructions to the dose reconstructor on the use of the OTIB (Section 8)
- Examples of how to apply the OTIB (Attachment H)

The OTIB is applicable to a broad range of reactor and fuel types, operations, and situations, with several important exclusions and caveats as well as some possible extensions. The

methodology does not apply to alpha-emitting radionuclides in irradiated fuel and does not include activation products produced outside the fuel and its cladding. It also does not apply to operations involving decay times of less than 10 days and to cases where radionuclides have been deliberately extracted and concentrated for other purposes. The OTIB notes, however, that the guidance can also be extended to those cases where in-vivo or fecal bioassay data are available, or where data are available for exposures from some, but not all, relevant radionuclides.

NIOSH intends for the OTIB to be broadly applicable by encompassing a number of different reactor types and operations that were commonly encountered in the government nuclear program. The following table summarizes the five reactor categories considered along with the actual reactors chosen to represent those categories.

Table 1. Reactor Categories and Representative Reactors

Reactor Category	Representative Reactor
Plutonium Production Reactors	Hanford N-Reactor Hanford Single-Pass Reactors
Experimental Sodium-Cooled Reactors	Fast Flux Test Facility (FFTF)
Advanced Test Reactors (high enrichment)	Advanced Test Reactor (ATR)
Research Reactors	TRIGA Reactor (Al-clad fuel) TRIGA Reactor (SS-clad fuel)
Generic Reactor	Pressurized Water Reactor (PWR)

REVIEW METHODOLOGY

SC&A reviewed the OTIB in a systematic manner, following the OTIB's organization section-by-section, including the provided examples and associated workbook, and following the guidance of SC&A's review procedure, *A Protocol for the Review of Procedures and Methods Employed by NIOSH for Dose Reconstruction* (SC&A 2004). Section 1 of this review report summarizes the OTIB sections and presents SC&A's findings and other observations, Section 2 presents a checklist taken from SC&A 2004, and Section 3 contains reference citations. In a few cases, as noted in the report, SC&A independently checked some of the assumptions and tabular values appearing in the OTIB. The checklist has sections concerned with the following issues:

- The degree to which the procedure supports a process that is expeditious and timely for dose reconstruction
- Whether the procedure provides adequate guidance to be efficient in instances where a more detailed approach to dose reconstruction would not affect the outcome
- The extent to which the procedure accounts for all potential exposures, and to ensure that resultant doses are complete and based on adequate data
- Whether the procedure provides a consistent approach to dose reconstruction, regardless of claimant's exposures by time and employment locations
- Whether the procedure is fair and gives the benefit of the doubt to the claimant
- Whether the procedure adequately accounts for the uncertainty of dose estimates

- Whether the procedure strikes a balance between technical precision and process efficiency

In addition, the report assesses whether the overall process presented in the OTIB is reasonable from a technical point of view, is claimant favorable, and provides clear guidance to the dose reconstructor attempting to use it.

SUMMARY OF FINDINGS

SC&A's detailed review of the OTIB is presented in Section 1. In the interest of clarity, SC&A reviewed the OTIB "linearly," following the OTIB's development section-by-section. The review produced 10 Findings, listed in Table 2. In addition, Section 1 also identifies suggested areas where the OTIB can be improved, as well as comments supportive of the OTIB. Further, SC&A included two "Notes" (Section 1.2) pointing out apparent typos in the OTIB.

Table 2. Review Findings

Finding No. and Report Section	Finding Summary
1 (Sect. 1.1)	SC&A is not able to evaluate the appropriateness of the input parameters used for the ORIGEN2 runs since they are not specified or references cited in the OTIB.
2 (Sect. 1.1)	The OTIB does not provide sufficient information to allow evaluation of its downselect from the initial seven to the final four representative reactors chosen.
3 (Sect. 1.1)	While Rev. 0 of the OTIB (Section 5.2) provides extensive discussions of the ORIGEN2 runs for each reactor, Rev. 1 does not for the ORIGEN-S runs. For each of the nine representative reactor cases, the OTIB (Table 5-2) specifies the specific power, irradiation time, and burnup, and includes a basis (e.g., "maximum burnup at nominal power" for ATR 1), but does not say how the values were selected or cite any reference; Rev. 0 made extensive use, for example, of the DOE report, *Source Term Estimates for DOE Spent Nuclear Fuels*, DOE/SNF/REP-078, Rev. 0, March 2003 (DOE 2003). SC&A cannot fully evaluate the appropriateness of the values chosen for each case without such information.
4 (Sect. 1.1)	SC&A notes that Table 5-1 of the OTIB lists both aluminum and stainless steel-clad TRIGA reactors as belonging to the initial set of seven reactors. However, Table 5-2, which lists the four reactors chosen as references, as well as the accompanying text, do not indicate which cladding was assumed for the TRIGA reactor. The OTIB also does not indicate what fuel enrichment was chosen, give a source for the specific power or the chosen burnups, or provide justification for its assumptions.
5 (Sect. 1.2)	In selecting release fractions for exposures to airborne radionuclides associated with reactor operations, the OTIB starts with the fuel inventory. However, it might have been more appropriate to use the mix of radionuclides in the gas gap or primary coolant as the starting point for assigning the isotopic composition in urine samples. Also, if a worker was involved in handling waste streams, such as ion exchange resins or HEPA or charcoal filters, using the isotopic mix in fuel as the starting point might not be appropriate and might lead to non-claimant-favorable results. These issues should be addressed in the OTIB.

Table 2. Review Findings

Finding No. and Report Section	Finding Summary
6 (Sect. 1.2)	The use of effective dose conversion factors (i.e., DCFs that relate to effective whole-body dose) is appropriate for screening purposes if the objective of the OTIB was to reconstruct whole-body doses, but not necessarily claimant-favorable for organ doses. For example, a radionuclide that does not contribute significantly to the whole-body dose could be an important contributor to an organ dose and might be eliminated. For certain reactor scenarios, some radionuclides that were not present in Table E contribute to the intake in a significant way and can deliver important contributions to organ doses. The OTIB would benefit from some discussion of this matter.
7 (Sect. 1.2)	Intakes and organ doses should be calculated using the same set of radionuclides as used to derive the contributions to the total beta excretion rate results.
8 (Sect. 1.2)	The OTIB explains that it was recognized that some of the methods used to determine the beta/gamma concentration of fission and activation products in urine would miss certain radionuclides, such as radioiodines. The OTIB claims that this is not a problem because the isotopic assignments of intake are based on the predicted relative concentrations of 17 radionuclides in air. Hence, these radionuclides are not missed when deriving doses. This seems reasonable, except if a large fraction of the activity is lost during the analysis of the urine samples. This, of course, would result in an underestimate of the actual gross beta/gamma composition of the urine, which would underestimate the radionuclide intake. The OTIB states that this issue was taken into consideration, and it was found to be important at the Savannah River Site (SRS). For this reason, a separate protocol is used in the SRS site profile. It is not apparent, however, how the dose reconstructor deals with situations where the airborne mix of radionuclides that might be associated with reactor operations or maintenance bears little resemblance to the mix of radionuclides in the fuel.
9 (Sect. 1.5)	The current OTIB workbook (Workbook 1.01) needs to be revised to match the current version of OTIB-0054 (Rev. 1), and then re-evaluated.
10 (Sect. 1.6)	Our primary concern with this OTIB is that although NIOSH developed a protocol that simplifies and likely overestimates the radionuclide intakes for individual workers, the protocol seems to be somewhat arbitrary when applied to a particular worker. In the process of developing the protocol, indicator radionuclides are used to derive intake values of the dosimetrically significant radionuclides **that do not necessarily relate to the real intakes and excretion rates for any given worker**. NIOSH does not show the degree of realism or conservatism built into the dose reconstruction for a given worker due to the scenario that was assumed by all other assumptions taken to derive the values in the tables presented in Section 7. The methods described in the OTIB will provide intakes and doses not necessarily correlated with the real ones. The differences between the intakes provided through the use of the document and the real ones are unknown and depend heavily on the scenario (periods of fuel irradiation and decay), the reactor type, and detection methods. It could be argued that as long as the protocol is scientifically valid and claimant favorable for a given worker, the approach is consistent with the letter and intent of the rule. However, we can envision a situation where two workers are assigned a dose using the OTIB, where in one case, the protocol is extremely claimant favorable and the worker is compensated, and in the other case, the protocol is less claimant favorable and the claim is denied. The OTIB would benefit from a discussion of this particular concern.

1.0 SC&A EVALUATION OF ORAUT-OTIB-0054, REVISION 1

OTIB-0054 is quite complex and consists of a number of sections, first describing the process of developing the tables used by a dose reconstructor, then providing guidance to the dose reconstructor on how to estimate doses for a variety of different situations. Hence, we found it beneficial to explain our understanding of the OTIB as part of the process of making comments on it. For convenience and clarity, we generally followed the OTIB document section framework, but grouped our discussion into three primary sections on reactor modeling, determination of intake fractions, and application instructions to the dose reconstructor. In addition to reviewing the body of the OTIB, SC&A also reviewed the examples provided in an appendix and the associated worksheet provided on the NIOSH computer system.

1.1 REACTOR MODELING

The goal of the modeling portion of the OTIB is to reduce a large amount of representative reactor fuel isotopic data into a form easily usable by a dose reconstructor looking at actual claimant cases. The starting point for this OTIB is the radionuclide mix in spent fuel for a number of different reactor types, each with different operating conditions (specific power, irradiation time, and burnup), calculated at several different decay times following removal from the reactor. It is assumed that an actual claimant case condition will fit somewhere within the parameter space defined by the OTIB. Section 5 of the OTIB begins by choosing seven actual reactors to represent five different general categories. Table 3 summarizes the reactors and cases:

Table 3. Reactor Categories and Representative Reactors

Reactor Category	Representative Reactor
Plutonium Production Reactors	Hanford N-Reactor Hanford Single-Pass Reactors
Experimental Sodium-Cooled Reactors	Fast Flux Test Facility (FFTF)
Advanced Test Reactors (high enrichment)	Advanced Test Reactor (ATR)
Research Reactors	TRIGA Reactor (Al-clad fuel) TRIGA Reactor (SS-clad fuel)
Generic Reactor	Pressurized Water Reactor (PWR)

Source: ORAUT-OTIB-0054, Rev. 1, Table 5-1

The OTIB uses the ORIGEN2 Version 2.1 isotope generation and depletion code (Croff 1980) to calculate isotopic inventories for the 7 representative reactors of Table 3 in 11 runs (4 reactors had 2 runs each, using 2 different sets of cross section and fission product yield libraries), decayed for 10 days, 40 days, 60 days, 90 days, 180 days, 1 year, and 3 years after irradiation had ceased. The code contains nuclear data libraries for the FFTF, ATR, and PWR, but not for the two Hanford production reactors and the two TRIGA reactors. For each of those cases lacking an ORIGEN2 library, ORAUT ran the code twice with different available cross section and fission product yield libraries chosen to attempt to bound the irradiation results.

The OTIB describes how the reactors were modeled in ORIGEN2. Material compositions were given based on a single fuel element or per metric ton of fuel and burnup, and "nominal values for burnup and specific power were selected as appropriate given the composition basis."

> **Finding 1:** SC&A is not able to evaluate the appropriateness of the input parameters used for the ORIGEN2 runs, since they are not specified or references cited in the OTIB.

The OTIB notes that, "No decay (shutdown) intervals or other power history considerations were included. Activity ratios for short-lived radionuclides, are, therefore, generally overstated." SC&A concurs with this statement.

ORIGEN2 is a widely-used, industry-standard code, developed by Oak Ridge National Laboratory (ORNL), to calculate the isotopic inventory after buildup and decay of isotopes in a nuclear reactor under conditions specified by the user. The code uses a matrix exponential method to solve coupled linear, first-order differential equations with constant coefficients. The code contains data libraries, including cross sections, fission product yields, decay data, and decay photon data. SC&A believes that the use of ORIGEN2 is appropriate and consistent with common practice in the nuclear community.

The 11 ORIGEN2 runs produced activity data for 879 fission product nuclides and 688 activation product nuclides, with a large overlap of nuclides between the 2 categories, and including some stable nuclides in addition to the radioactive ones. The activities in each dataset were normalized to that of Cs-137 at 10 days of decay (as stated in Note 1 of the OTIB, that choice of decay time minimizes loss of short-lived radionuclides).

The OTIB states, "Based on comparison of the fission and activation product relative activity data for the 11 ORIGEN2 runs, 4 reactors were selected as having isotopic inventories and ratios that would be representative of the variety of reactors and fuel combinations under consideration: the ATR, the FFTF, the N-Reactor, and the TRIGA reactor."

> **Finding 2:** The OTIB does not provide sufficient information to allow evaluation of its downselect from the initial seven to the final four representative reactors chosen.

ORAUT then switched from the ORIGEN2 to the ORIGEN-S code for further fission and activation product inventory calculations for the four representative reactors remaining. ORIGEN-S is a more modern version of ORIGEN2 and is part of the SCALE code system (ORNL 2011) for nuclear safety analysis and design developed and maintained by ORNL. Other modules in the SCALE system were used to create case-specific cross section libraries and preprocess data for ORIGEN-S. Multiple sets of irradiation parameters were defined for each of the four representative reactors, resulting in nine ORIGEN cases; parameters (specific power, irradiation time and burnup) for the nine resulting cases are summarized in Table 5-2 of the OTIB. Bases for the nine cases are also listed in the table, illustrating how the selections attempt to span the potential ranges of the parameters for each reactor.

Finding 3: While Rev. 0 of the OTIB (Section 5.2) provides extensive discussions of the ORIGEN2 runs for each reactor, Rev. 1 does not for the ORIGEN-S runs. For each of the nine representative reactor cases, the OTIB (Table 5-2) specifies the specific power, irradiation time, and burnup, and includes a basis (e.g., "maximum burnup at nominal power" for ATR 1), but does not say how the values were selected or cite any reference; Rev. 0 made extensive use, for example, of the DOE report, *Source Term Estimates for DOE Spent Nuclear Fuels*, DOE/SNF/REP-078, Rev. 0, March 2003 (DOE 2003). SC&A cannot fully evaluate the appropriateness of the values chosen for each case without such information.

Notwithstanding the preceding finding, SC&A attempted to verify the parameters used in the ORIGEN-S runs as best it could using the information provided in the OTIB as well as in several outside sources. Table 5-2 of the OTIB is reproduced here as Table 4 for convenience.

Table 4. ORIGEN-S Irradiation Parameters for the Nine Representative Reactor Cases

Case	Parameters	Basis
ATR 1	Specific power = 2,379.1 MW/MTU Irradiation time = 132.27 days Burnup = 314,684 MWd/MTU	Maximum burnup at nominal power
ATR 2	Specific power = 8,651.2 MW/MTU Irradiation time =36.4 days Burnup = 314,904 MWd/MTU	Maximum burnup at maximum assembly power
ATR 3	Specific power = 2,379.1 MW/MTU Irradiation time = 56 days Burnup = 133,230 MWd/MTU	Nominal burnup at nominal power
FFTF 1	Specific power = 163.8 MW/MTHM Irradiation time = 929.4 days Burnup = 152,230 MWd/MTHM	Maximum burnup at nominal power
FFTF 2	Specific power = 163.8 MW/MTHM Irradiation time = 488.3 days Burnup = 79,984 MWd/MTHM	Nominal burnup at nominal power
N-Reactor 1	Specific power = 10.4 MW/MTU Irradiation time = 114.2 days Burnup = 1,188 MWd/MTU	Production of weapons-grade plutonium (nominal 6% Pu-240 content) at nominal power
N-Reactor 2	Specific power = 10.4 MW/MTU Irradiation time = 285.6 days Burnup = 2,970 MWd/MTU	Production of fuel-grade plutonium (nominal 12% Pu-240 content) at nominal power
TRIGA 1	Specific power = 15.57 MW/MTU [(a)] Irradiation time = 730.1 days Burnup = 11,368 MWd/MTU	Maximum burnup at nominal power
TRIGA 2	Specific power = 15.57 MW/MTU Irradiation time = 115.2 days Burnup = 1,994 MWd/MTU	Nominal burnup at nominal power

(a) OTIB Table 5-2 typo misstates "MWd" rather than the correct "MW"
Source: ORAUT-OTIB-0054, Rev. 1, Table 5-2

SC&A made a quick check by multiplying the specific power by the irradiation time for each case, which should then produce the last entry, the burnup. The "arithmetic" is essentially correct except for the TRIGA 2 case, where (15.57 MW/MTU) × (115.2 days) =

1,794 MWd/MTU. The table lists 1,994 MWd/MTU, however, which might just be a typo since it is off in only one digit; this is not seen as an important difference in any event.

The four representative reactors will now be examined individually.

Advanced Test Reactor (ATR)

The ATR, which is a pressurized, light-water moderated, beryllium-reflected research reactor, has operated at Idaho National Laboratory (INL) for over 45 years (it is still in operation). The reactor uses highly enriched uranium fuel (93.15% nominal enrichment) arranged in curved fuel element plates, and is run at very high burnup levels to permit accelerated testing of materials for radiation damage. An INL website states the following:

> *The Advanced Test Reactor (ATR) is located at the ATR Complex on the Idaho National Laboratory site and has been operating continuously since 1967. The primary mission of this versatile facility was initially to serve the U.S. Navy in the development and refinement of nuclear propulsion systems, however, in recent years the ATR has been used for a wider variety of government and privately sponsored research. DOE's designation of the ATR as a National Scientific User Facility enables DOE to facilitate research that would not have been feasible for many researchers and allows for fuller utilization of the facility.*
>
> *The ATR design exploits a unique serpentine core configuration that offers a large number of test positions. It has large test volumes in high-flux areas, and is designed to permit simulation of long neutron radiation exposures in a short period of time. The maximum thermal power rating is 250 MW, with a maximum unperturbed thermal neutron flux of 1.0 × 10^{15} n/cm^2-s.* [INL 2013]

SC&A consulted *Source Term Estimates for DOE Spent Nuclear Fuels* (DOE 2003) for technical information about the ATR. The document was prepared by the Department of Energy (DOE) to support Yucca Mountain safety analysis studies by characterizing source terms for a wide range of spent nuclear fuel held by DOE at its various facilities. DOE 2003 explicitly estimates the ATR spent fuel source terms as Template 12. The DOE report's upper-end burnup of 314,683 MWd/MTHM (megawatt-day per metric ton of heavy metal) corresponds to the burnup used in the OTIB. The OTIB modeled two power level and irradiation time cases at maximum burnup: Case ATR 1 simulates a nominal power level with a long irradiation time; and Case ATR 2 simulates a maximum power level with a short irradiation time. The third case, ATR 3, simulates a nominal power level with an intermediate irradiation time producing a lower fuel burnup.

DOE 2003 (p. A-104) lists 10 MW as the maximum power level per fuel element and the total uranium (U-235, U-238, U-234, U-236) loading in a fuel element as 1166.89 g. The latter is equivalent to 0.001167 MTU. The specific power can then by found by dividing the power level by the fuel loading:

> 10 MW/0.001167 MTU = 8,569 MW/MTU.

The OTIB ATR 2 case (maximum burnup at maximum power) of 8,651 MW/MTU is reasonably close to this. However, it is not evident what assumptions the OTIB used to derive the ATR 1 case (maximum burnup at nominal power) and the ATR 3 case (nominal burnup at nominal power) values.

Fast Flux Test Facility (FFTF)

The FFTF was a 400 MW_{th}, liquid sodium-cooled, FFTF, located at the DOE Hanford site that operated from April 1982 to April 1992 (DOE 2007). DOE 2003 states that the fuel was mixed oxide, with plutonium (86 wt% Pu-239) amounting to 29 wt% of the heavy metal (uranium plus plutonium); the driver assemblies were clad with 316 SS; and the assemblies were subjected to high burnups.

The OTIB considers two cases: FFTF 1, with a maximum burnup of 152,230 MWd/MTHM at nominal power; and FFTF 2, with a nominal burnup of 79,984 MWd/MTHM at nominal power. DOE 2003 (p. A-6) notes that the highest burnup was 152,230 MWd/MTHM, with most in the range of 70,000–90,000. Hence, the OTIB choice of burnups appears to have adequately bracketed the FFTF operating conditions.

DOE 2003 (Table 3 and p. A-6) gives the average power for a standard fuel assembly as 5.4 MW. Dividing the average power by the specific power of 163.8 MW/MTHM assumed by the OTIB, yields the heavy metal fuel loading in a fuel assembly of 0.03297 MTHM/assembly (or, 32,970 g heavy metal/assembly). By comparison, DOE 2003 (Table 1) lists the total heavy metal composition (Pu-239, Pu-240, Pu-241, Pu-242, Am-241, U-235, and U-238) of a FFTF fuel assembly as 32,918 g, which supports the OTIB value used in its reactor modeling.

Hanford N-Reactor

The features of the Hanford N-Reactor, which operated from 1963 to 1987, are summarized in DOE 2003 (Template 7, p. A-45), which, as noted in that document, were largely taken from Bergsman 1994; the latter characterizes and inventories the spent fuel stored at the Hanford site:

> *The … N-Reactor is a graphite-moderated,* [slightly enriched] *pressurized water-cooled reactor … initially designed for plutonium production for national defense* [and, later] *modified to produce steam to be used by the Washington Public Power Supply System to generate electricity. N-Reactor was the only dual-purpose reactor in the United States. The core of the N-Reactor was a 1800-ton graphite block, 33 feet (10 meters) high by 33 feet (10 meters) wide by 39 feet (12 meters) long* (DOE 2003, p. A-46).

Bergsman notes that the N-Reactor primarily used two different types of fuel elements, the Mark IV and the Mark IA, both clad with Zircaloy-2. The Mark IV elements have a pre-operations enrichment of 0.947% U-235 in both of its concentric tubes, while the Mark IA elements have a pre-operations enrichment of 1.25% U-235 in the outer tube and 0.947% in the inner tube. The Mark IV elements, though, contain a greater uranium weight than the Mark IA elements—50.0 lb (22.7 kg) vs. 35.9 lb (16.3 kg) (Bergsman 1994, p. 3). DOE 2003 based its

radioactive inventory determination on the Mark IV elements. The total fuel loading is 11.6 MTU (DOE 2003, p. A-53).

The OTIB selects a representative burnup of 1,188 MWd/MTHM for the first case, N-Reactor 1, and a greater value, 2,970 MWd/MTHM, for the second case, N-Reactor 2. The former value is given in Schwarz 1997 for an outer Mark IV fuel element, and it results in a discharge Pu-240 content of 6%. As far as SC&A could check, the OTIB appears to have taken its data for the N-Reactor from authoritative sources.

TRIGA

The low-power, water-cooled, pool-type UZrH-fueled reactors are manufactured by General Atomics (GA) primarily for research and educational purposes and are considered inherently safe due to their very large prompt negative temperature coefficient of reactivity that quickly drops the reactor subcritical in response to any power excursions (which raise the water coolant temperature). This feature also allows the TRIGA to be pulsed to a high power level, followed by a quick, automatic self-shutdown. GA has supplied over 65 reactors throughout the world over the past 45 years, ranging in (thermal) power from 20 kW to 16 MW and containing 60–100 fuel elements clad either with aluminum or stainless steel (General Atomics). The models progressed from Mark I through Mark III, with a number of specially modified units also produced. The early models used highly enriched uranium (70%), but the later models greatly reduced the enrichment (to 20%) in response to U.S. government proliferation concerns, and the earlier models were converted to the lower fuel enrichment as well. Pertinent information for constructing an ORIGEN model can be found in the INL report, Sterbentz (1997), and the Westinghouse Hanford report, Schmittroth and Lessor (1996). An aluminum-clad fuel element contains 180 g of uranium, and a stainless steel-clad fuel element contains 195 g of uranium (Sterbentz 1997).

> **Finding 4:** SC&A notes that Table 5-1 of the OTIB lists both aluminum and stainless steel-clad TRIGA reactors as belonging to the initial set of seven reactors. However, Table 5-2, which lists the four reactors chosen as references, as well as the accompanying text, do not indicate which cladding was assumed for the TRIGA reactor. The OTIB also does not indicate what fuel enrichment was chosen, give a source for the specific power or the chosen burnups, or provide justification for its assumptions.

In the next step, the OTIB reduces decay times under consideration from the initial seven to four representative ones: 10 days, 40 days, 180 days, and 1 year. OTIB Table 5-3, reproduced here as Table 5, provides guidance to the dose reconstructor for choosing an appropriate decay time for specified worker activities.

Table 5. Fission and Activation Product Decay Times for General Steps in the Fuel Cycle

Activity	Decay Time
Reactor operations, spent fuel storage, fuel examination; activity from fuel leakage/failure or segmentation	10 days
Dissolution of fuels from early production reactors (e.g., Hanford production reactors 1940s–1950s)	40 days
Fuel dissolution; general or later years	180 days
Waste management	1 year

Source: ORAUT-OTIB-0054, Rev. 1, Table 5-3

Hence, after a number of steps, the OTIB has transformed the original set of cases to four different types of reactors, each with different operating conditions, and four different decay times. These are the reference reactors that will be used by dose reconstructors as the starting point for reconstructing the doses to a given claimant with a given type of cancer.

1.2 INTAKE FRACTIONS

For convenience and clarity in the following discussion, SC&A informally labels different portions of the OTIB as "Step 1," "Step 2," etc.

Step 1

As discussed above, the starting point for this OTIB (described in OTIB Section 5) is the derivation of the radionuclide mixture in spent fuel for four different representative reactors, each with different operating conditions (specific power, irradiation time, and burnup), and four different decay times (10 days, 40 days, 180 days, and 1 year). The dose reconstructors will use these reference reactors as a basis for reconstructing the doses to a given claimant with a given type of cancer.

Given the isotopic mixtures of Section 5 of the OTIB, Section 6 explains how the list was reduced through a multi-step process. First, radionuclides that could not contribute significantly to the internal dose, such as very short-lived radionuclides and noble gases, were removed from the list. Then, the activities of the remaining radionuclides were multiplied by the following assumed release fractions (RFs): 1.0 for tritium; 0.5 for volatiles, such as radioiodines; and 0.01 for semi-volatiles, like Cs-137. For particulates and solids, the OTIB uses an RF of 0.01, which is cited in the report as being conservative, since solids are likely to have release fractions of 0.001 or less. The OTIB cites DOE 1997, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, as the basis for these release fractions. Appendix B of the OTIB presents the assumed RFs for the radionuclides under consideration. The outcome of this step is an expression of the activity of each radionuclide that is likely to be airborne. This process is performed for each of the nine categories of fuel and four different decay periods.

SC&A believes that these RFs, although crude, are reasonable and will likely result in a claimant-favorable reconstruction of the isotopic mix of radionuclides in urine, assuming the worker is involved in fuel dissolution processes. However, as discussed below, it is not apparent

that using the core inventory as the starting point for deriving doses is always scientifically appropriate and claimant-favorable.

> **Note 1:** There appears to be a typo in the RF for iodines in Attachment B of the OTIB. Appendix B lists an RF of 0.05, but we believe 0.5 is the value that the OTIB intends to use.

> **Finding 5:** In selecting release fractions for exposures to airborne radionuclides associated with reactor operations, the OTIB starts with the fuel inventory. However, it might have been more appropriate to use the mix of radionuclides in the gas gap or primary coolant as the starting point for assigning the isotopic composition in urine samples. Also, if a worker was involved in handling waste streams, such as ion exchange resins or HEPA or charcoal filters, using the isotopic mix in fuel as the starting point might not be appropriate and might lead to non-claimant-favorable results. These issues should be addressed in the OTIB.

Step 2

It is our understanding that the mix of airborne radionuclides obtained in Step 1 was then expressed in terms of fractions of the total activity, where the sum of the assigned fractions for each set of airborne activities was set equal to 1.0. Section 6.2 of the OTIB provides a description of the methods used to further reduce the large number of radionuclides present in irradiated fuel to a manageable number (without compromising the dose reconstruction) by a two-step process. The first cut was to multiply the release fraction of each radionuclide by the 50-year effective dose commitment coefficient (e.g., Sv/Bq inhaled) taking into consideration the different solubility types of each radionuclide. We assume that, for each reactor type and decay time, this step in the process results in the product of a release fraction with a DCF, which in turn is expressed in terms of a metric referred to as a dosimetrically significant source term. Each of these metrics was again converted to a fraction, and is presented in Attachment D. A total of 36 isotopes are included in the tables of this attachment. This list was further reduced by eliminating those radionuclides that contributed less than 0.01 to what is referred to as the normalized intake fractions (NIFs). Attachment E presents the revised list of NIFs, which now has been reduced to 17 radionuclides.

> **Finding 6:** The use of effective dose conversion factors (i.e., DCFs that relate to effective whole-body dose) are appropriate for screening purposes if the objective of the OTIB were to reconstruct whole-body doses, but not necessarily claimant favorable for organ doses. For example, a radionuclide that does not contribute significantly to the whole-body dose could be an important contributor to an organ dose and might be eliminated. For certain reactor scenarios, some radionuclides that were not present in Table E contribute to the intake in a significant way and can deliver important contributions to organ doses. The OTIB would benefit from some discussion of this matter.

Step 3

It is our understanding that, for each set of normalized intake fractions using Cs-137 and Sr-90 as the normalizing (indicator) radionuclides, the relative concentrations of each radionuclide in urine assuming a chronic intake for 2 years, followed by a 24-hour urine sample, was calculated and given in Attachment G. However, these are theoretical radionuclide concentrations. In reality, processing of urine samples requires some time, where some ingrowth and decay might occur, and a certain amount of some radionuclides might be lost in the preparation of the sample for counting.

The OTIB makes adjustments to the relative radionuclide concentrations in urine to account for this ingrowth and decay. The NIFs from Table D-1 are multiplied by the corresponding intake retention fractions (IRFs) of Table F-1 (obtained from the Integrated Modules for Bioassay Analysis (IMBA) program for the least possible solubility type for all radionuclides, except for Sr-90 for which Type F was used), and then adjusted according to the physical decay and ingrowth during the delay time between sample collection and counting time. The maximum delay time considered was 30 days. The half-lives ($T_{1/2}$) for decay corrections are given in Table F-2.

> **Note 2:** The $T_{1/2}$ for Cs-136 in Table F-3 is listed as 13.16 y instead of the correct 13.16 d.

Tables G-1, G-2, and G-3 give the relative radionuclide concentrations in urine at the time of gross beta or gross gamma counting. Tables G-4 and G-5 list the contributions from Table D-1 radionuclides to gross beta and gross gamma activity, respectively, for air concentration or surface concentration.

Step 4

For each reactor scenario considered, the contribution of Sr-90 to the total beta activity of a urine sample collected after 2 years of chronic inhalation intakes is calculated using the values in Tables G-1 and G-2 and then divided by the total beta contributions of all the other radionuclides of that scenario to yield a fractional value. In a similar fashion, the relative gamma activity of Cs-137 in the urine sample collected after 2 years of chronic inhalation intakes is calculated using the values of Table G-3.

Table 7-1a defines the contribution of Sr-90 to gross beta urinalysis for minimally processed samples, Table 7-1b defines the contribution of Cs-137 to gross gamma urinalysis for minimally processed samples, and Table 7-2 shows the contribution of Sr-90 to the gross beta urinalysis following major chemical processing. The activity fractions in Tables 7-1a and 7-2 are used as appropriate to determine Sr-90 contribution to gross beta analysis results in all urine samples taken from a worker. In the same way, Table 7-1b is used to calculate the Cs-137 activity contribution to the gross gamma result in all urine samples taken from a worker.

SC&A Calculation: SC&A performed a sample calculation to enhance its understanding of the procedure and to check the use of the tables to determine the beta activity fraction. The problem assumes a worker is exposed chronically for 2 years through inhalation to Sr-90 from case

ATR 1 fuel with 10-day decay [ATR 1 (10d)]. A 24-hour urine sample is taken on the last day of exposure and is minimally processed before it is counted 30 days after sample collection. Note that for a delay of 30 days between sample collection and counting, the adjustment factor for Sr-90 is 1.0.

The excretion of Sr-90 in urine at the end of those 2 years can be expressed as:

$$u(\text{Sr-90, 2y}) = 0.231 \times I(\text{Sr-90}) \quad (1)$$

where,

0.231 is the Intake Retention Function (IRF) for Sr-90 from Table F-1, and

I(Sr-90) is the Sr-90 intake.

For the ATR 1 (10d) case:

$$I(\text{Sr-90}) = 4.01 \times 10^{-4} \times I(\text{total}) \quad (2)$$

where,

4.01×10^{-4} is the Sr-90 Normalized Intake Fraction (NIF) from Table D-1, and

I(total) is the total daily beta activity intake for a worker exposed in the ATR 1 (10d) scenario

Then, substituting Eq. (2) in to Eq. (1),

$$u(\text{Sr-90, 2y}) = 0.231 \times 4.01 \times 10^{-4} \times I(\text{total}) = 9.24 \times 10^{-5} \times I(\text{total}) \quad (3)$$

Note that the value of 9.24×10^{-5} calculated in Eq. (3) corresponds to the "contribution to gross beta activity for an unprocessed urinalysis" value for Sr-90 in the ATR 1 (10 d) scenario appearing in Table G-1.

For each radionuclide in Table G-1, the corresponding urine excretion of a particular radionuclide after 2 years chronic exposure is:

u(radionuclide, 2y) = I(total) × value for that radionuclide in the first column of Table G-1 (i.e., the contribution to gross beta activity for that radionuclide in the ATR 1 (10d) scenario). (4)

The total beta activity in urine after 2-year chronic intake is equal to the sum of the contributions of all radionuclides in Table G-1, which is 4.32×10^{-3} for the ATR 1 (10d) scenario (adding values in first column of Table G-1). Then,

$$u(\text{total beta, 2y}) = 4.32 \times 10^{-3} \times I(\text{total}) \quad (5)$$

Solving for I(total) in Eq. (5) gives

$$I(\text{total}) = u(\text{total beta, 2y})/(4.32 \times 10^{-3}) \quad (6)$$

Substituting the expression for I(total) from Eq. (6) into Eq. (3) gives

$$u(\text{Sr-90, 2y}) = 9.24 \times 10^{-5} \times u(\text{total beta, 2y})/(4.32 \times 10^{-3}) \quad (7)$$

Doing the divsion in Eq. (7) yields

$$u(\text{Sr-90, 2y}) = 2.14 \times 10^{-2} \times u(\text{total beta, 2y}) \quad (8)$$

If a worker has a set of gross beta readings in urine, all taken after 2 years of chronic exposures, the contribution from the indicator Sr-90 to the gross beta activity in the urine samples is calculated using the activity fractions in Table 7-1a or Table 7-2 (depending on the degree of sample processing) to determine the indicator radionuclide activity. IMBA is then applied to the set of Sr-90 activities in urine, u(Sr-90, 2y), and the best fit intake of Sr-90 is calculated. Intakes for the other radionuclides are then assigned using the appropriate values of Tables 7-3a through 7-3i.

Tables 7-3a through 7-3i (for the different reactor scenarios) are based on the reduced list of dosimetrically important radionuclides, listed in Table E-1. The OTIB claims that it is claimant favorable to assign doses to the radionuclides in Table E-1 instead of to the larger set of radionuclides in Table D-1. Even if true, it is not correct to use Table D-1 intakes to derive the contributions from each nuclide to total urine excretion and then back-calculate the intake rates using a different table, i.e., Table E-1, with a reduced number of nuclides.

> **Finding 7:** Intakes and organ doses should be calculated using the same set of radionuclides as used to derive the contributions to the total beta excretion rate results.

The OTIB does not explain why chronic intakes shorter than 2 years were not considered. For example, all the example problems in Attachment H assume a worker whose urine samples were taken at a minimum of 2 years after the first day of employment. SC&A tested the assumption of 90 days chronic exposure instead of 2 years to see if there is a significant change. For the scenario ATR 1 (10d) and 90 days chronic exposure, the fraction of Sr-90 in minimally processed samples is equal to 0.0254 instead of 0.0214. Taking into consideration all the uncertainties and approximations throughout the procedure, SC&A concludes that there is no relevant difference and the assumption of a 2-year chronic intake scenario is acceptable. SC&A also determined that the Cs-137 gross gamma activity fraction in urine for minimally processed samples for 90 days continuous exposure is lower than that for 2 years' exposure. Thus, the scenario of 2-years continuous exposure assumed by NIOSH is claimant favorable for gammas as well.

Step 5

Each set of chronic isotopic intakes determined as outlined above is then used as input to IMBA to derive doses to the organ of concern. The intake that results in the highest organ dose is used as input to the Interactive RadioEpidemiological Program (IREP) to derive probability of causation (POC).

Finding 8: The OTIB explains that it was recognized that some of the methods used to determine the beta/gamma concentration of fission and activation products in urine would miss certain radionuclides, such as radioiodines. The OTIB claims that this is not a problem because the isotopic assignments of intake are based on the predicted relative concentrations of 17 radionuclides in air. Hence, these radionuclides are not missed when deriving doses. This seems reasonable, except if a large fraction of the activity is lost during the analysis of the urine samples. This, of course, would result in an underestimate of the actual gross beta/gamma composition of the urine, which would underestimate the radionuclide intake. The OTIB states that this issue was taken into consideration, and it was found to be important at the SRS. For this reason, a separate protocol is used in the SRS site profile. It is not apparent, however, how the dose reconstructor deals with situations where the airborne mix of radionuclides that might be associated with reactor operations or maintenance bears little resemblance to the mix of radionuclides in the fuel.

1.3 INSTRUCTIONS TO DOSE RECONSTRUCTOR

Section 8 of the OTIB summarizes what the document contains and provides instructions (beginning in the third paragraph) to the dose reconstructor on how to apply the methodology contained in the OTIB. In short, activity fraction data (Tables 7-1a/b, 7-2, or 7-4a/b) are applied to determine the indicator radionuclide activities for a given gross beta or gamma measurement. Decay times are selected from consideration of which worker category (Table 5-3) is most appropriate for the case. After the appropriate number of indicator radionuclide activity levels are found, IMBA and activity fraction ratios are used to determine radionuclide intakes. Finally, indicator radionuclide doses are entered into IREP to determine the POC for the cancer under consideration.

The OTIB cautions that site- or claimant-specific information should be used if available:

> *This document provides activity and intake fraction values for use in interpreting gross beta or gross gamma activity measurements related to reactor source terms. The values were derived using representative parameters and models; site- and claim-specific information, if available, should take precedence over the guidance in this document. This includes site-specific models and guidance in Site Profiles* (ORAUT 2013).

The OTIB provides significant guidance in cases where there is no specific information available:

> *If a specific decay time or times cannot be established, the dose evaluation will need to consider all four possibilities. Similarly, in most instances, all nine of the representative reactor cases will need to be considered. If the dose reconstructor does not know if significant separations chemistry was performed prior to a gross beta urinalysis, it is favorable to the claimant to assume significant chemistry was performed… If both gross beta and gross gamma results are available for a given*

> *assay then the more sensitive measure [i.e., that with the lowest minimum detectable activity (MDA)] should be used. If the dose reconstructor does not know if a given assay result was a gross beta or a gross gamma count (and both are realistic choices for the site under consideration) then both should be evaluated to determine which yields the larger assigned dose* (ORAUT 2013).

SC&A believes that following the above guidance will result in claimant-favorable outcomes. Overall, SC&A believes that, except for a few possible issues, such as those raised regarding the starting point for the calculation (i.e., the isotopic inventories of radionuclides in fuel), the protocol described in OTIB-0054, Rev. 1, seems reasonable (if our understanding of the protocol is correct).

1.4 EXAMPLE PROBLEMS

Because of the complexity of the protocol, the OTIB provides three example calculations in Attachment H to guide the dose reconstructor in applying the OTIB. The following presents our review of two of the example calculations in order to ensure that our understanding of the protocol is accurate.

1.4.1 Example 1: Gross Beta Bioassay Data (with Major Chemical Processing) from Waste Management Worker

In this example, the worker was involved in waste management activities and had a series of eight gross beta urine sample analyses from 1964 to 1966 that ranged from less than the minimum detectable level (MDL) of 20 pCi/day to 57 pCi/day. The introduction to the example explains that:

> *…in dose reconstructions intakes and doses to the organ of concern have to be calculated for all the reactor cases to determine the one that maximizes the organ. This is necessary to justify use of the constant (upper bound) distribution. However, the user can choose to determine the intakes for one reactor case only and use the OTIB-0054 tool* [to] *calculate the intakes (and organ doses) for the other reactor cases without having to use IMBA more than once.*

In accordance with Table 5-3 of the OTIB, 1-year old fuel is assumed for the waste management function. Also, it was determined that the mix applicable to ATR 1 is appropriate, and, since the bioassay data are gross beta, Sr-90 Type F is used as the indicator radionuclide. Table 7-2 (major chemical processing of the urine) indicates that the activity fraction of Sr-90 for 1-year old ATR 1 fuel is 0.471 of the gross beta results. Based on these assumptions, and using IMBA, the chronic Sr-90 intake rate that yields the derived urine excretion rates reported in Table H-1 of the OTIB is 76 pCi/d. This value is used as the normalization factor to derive the intake fractions of the other radionuclides (see Table H-1 of the OTIB).

This exercise was performed for each of the other eight reactor types (by scaling from the ATR 1 results) to derive the maximum dose to the organ of concern. This approach seems to be claimant favorable.

1.4.2 Example 2: Gross Beta Bioassay Data (with Minor Chemical Processing) from Worker Involved in Spent Fuel Storage Operations

In this example, the worker was involved in spent fuel storage operations. The bioassay data are the same as used in Example 1, but a 10-day decay time is assumed from the listings in Table 5-3. The gross beta fraction for Sr-90 in urine is 0.0214 for 10-day old fuel for the ATR 1, as listed in Table 7-1a (minor chemical processing of the urine). We understand this to mean that the chronic intake of Type F Sr-90 for 2 years results in a relative concentration of Sr-90 in urine of 0.0214. Hence, 0.0214 of the gross beta activity excreted in urine in Table H-3 is Sr-90, or about 0.0214 times gross beta activity observed in urine or about 0.6 pCi/d. Table H-4 presents the chronic intake rates for all radionuclides by normalizing against Sr-90 (i.e., assuming the relative intake of Sr-90 intake is one and the other radionuclides have an intake that is proportional to that of Sr-90, as listed in Table 7-3a of the OTIB).

The OTIB states that, using IMBA, the intake rate of Sr-90 that results in a urinary excretion rate of 0.6 pCi/day is 3.51 pCi/d. As a check, SC&A calculated an intake rate of 3.52 pCi/d using the Sr-90 excretion rates, which is in agreement with the OTIB value.

1.5 WORKSHEET

As part of reviewing the OTIB, SC&A also evaluated the associate workbook (Workbook 1.01), which is used by the dose reconstructors to automate the dose reconstruction process. It should be noted that while Rev. 1 of the OTIB was released on June 13, 2013, Version 1.01 of the Workbook was released on October 9, 2007 (after Rev. 0 and before Rev. 0, PC-1 of the OTIB).

SC&A analyzed the three examples provided in the OTIB to verify the methodology and compatibility of the examples with the text of OTIB-0054. SC&A then ran the intake calculations for these three examples using the OTIB-0054 Workbook 1.01; a sample input screen is shown in Figure 1.

[This figure has been **redacted** as it contains information protected by the Privacy Act.]

Figure 1. Example of OTIB-0054 Workbook 1.01 Data Input Screen

After SC&A ran the intake calculations for these three examples using the workbook, it was found that some of the radionuclides (Eu-155, Fe-55, etc.) and intake values provided by the workbook were different than those listed in the tables in the examples, i.e., Table H-2, H-4, and H-5. The resulting values provided by the workbook analyses did not match those listed in the corresponding tables for the three examples, nor did the ratios of the radionuclides match.

For Example 1, the Ru-106/Sr-90 value derived from Table H-2 is 0.917, but the workbook returns a corresponding ratio of 3.80. Additionally, the workbook does not list Co-60, but the OTIB table did, and the OTIB table does not list Fe-55 while the workbook does. Tables 6 and 7 help to illustrate this issue.

Table 6. ATR 1 Radionuclide Intake Rates Derived from Sr-90 Intake Rate of 76.0 pCi/d

Radionuclide	1-yr Ratios
Co-60	1.41E-02
Sr-89	4.02E+01
Sr-90	7.60E+01
Y-90	7.60E+01
Y-91	9.35E+01
Zr-95	1.44E+02
Nb-95	3.10E+02
Ru-103	6.63E+00
Ru-106	6.97E+01
I-131	4.81E-09
Cs-134	4.20E+01
Cs-137	7.68E+01
Ce-141	3.45E+00
Ce-144	9.58E+02
Pr-143	7.90E-05
Pm-147	1.84E+02
Eu-154	1.69E+00
Total Daily Intake	**2.08E+03**

Source: ORAUT-OTIB-0054, Rev. 1, Table H-2

Table 7. Results of OTIB-0054 Workbook 1.01 for Example 1: Radionuclide Intake Ratios Derived from Sr-90 Intake

Radionuclide	Ratio to Sr-90
Ce-141	2.10
Ce-144	7.80
Cs-134	0.31
Cs-137	1.30
Eu-155	0.08
Fe-55	0.81
Nb-95	3.00
Pm-147	3.10
Ru-106	3.80
Sr-89	0.31
Sr-90	***1.00***
Y-91	0.77
Zr-95	1.40

Investigating further, SC&A found that columns X, Y, Z, and AA of the workbook list the radionuclides and their ratios based on Sr-90 or Cs-137 for a decay period of 10 days, 40 days, 180 days, and 1 year. These values and ratios match the results the OTIB-0054 workbook provided for each example.

Considering the dates of the issuance of the OTIB-0054 Workbook 1.01 (2007) and the current OTIB-0054, R1 (2013), and the Publication Record Description on page 2 of the OTIB, it is apparent that the current version of the OTIB is a complete rework of the methodology and

reactor data used. Therefore, the current version of the OTIB and the 2007 version of the workbook do not provide the same intake values, ratio values, or radionuclide lists. However, SC&A's hand calculations (using the data tables in the OTIB) for the three examples did provide the same resulting values as stated in the examples (see Section 1.4).

Finding 9: The current OTIB workbook (Workbook 1.01) needs to be revised to match the current version of OTIB-0054 (Rev. 1), and then reevaluated.

1.6 ASSESSMENT OF OVERALL TECHNICAL BASIS AND USABILITY OF OTIB

Finding 10: Our primary concern with this OTIB is that although NIOSH developed a protocol that simplifies and likely overestimates the radionuclide intakes for individual workers, the protocol seems to be somewhat arbitrary when applied to a particular worker. In the process of developing the protocol, indicator radionuclides are used to derive intake values of the dosimetrically significant radionuclides **that do not necessarily relate to the real intakes and excretion rates for any given worker**. NIOSH does not show the degree of realism or conservatism built into the dose reconstruction for a given worker due to the scenario that was assumed and by all other assumptions taken to derive the values in the tables presented in Section 7. The methods described in the OTIB will provide intakes and doses not necessarily correlated with the real ones. The differences between the intakes provided through the use of the document and the real ones are unknown and depend heavily on the scenario (periods of fuel irradiation and decay), the reactor type, and detection methods. It could be argued that as long as the protocol is scientifically valid and claimant favorable for a given worker, the approach is consistent with the letter and intent of the rule. However, we can envision a situation where two workers are assigned a dose using the OTIB, where in one case, the protocol is extremely claimant favorable and the worker is compensated, and in the other case, the protocol is less claimant favorable and the claim is denied. The OTIB would benefit from a discussion of this particular concern.

2.0 PROCEDURE CHECKLIST

In addition to the technical review of the OTIB described elsewhere, SC&A also reviewed it in accordance with its procedure, *A Protocol for the Review of Procedures and Methods Employed by NIOSH for Dose Reconstruction* (SC&A 2004). Table 8 is taken from that procedure. Since the table has general applicability, not all of its items are germane to the review of the OTIB.

Table 8. Procedure Review Outline/Checklist

No.	Description of Objective	Rating 1-5*	Comments
1.0	**Determine the degree to which procedures support a process that is expeditious and timely for dose reconstruction**		
1.1	Is the procedure written in a style that is clear and unambiguous?	4	
1.2	Is the procedure written in a manner that presents the data in a logical sequence?	5	
1.3	Is the procedure complete in terms of required data (i.e., does not reference other sources that are needed for additional data)?	5	
1.4	Is the procedure consistent with all other procedures that are part of the hierarchy of procedures employed by NIOSH for dose reconstruction?	N/A	This review did not examine other procedures.
1.5	Is the procedure sufficiently prescriptive in order to minimize the need for subjective decisions and data interpretation?	4	
2.0	**Determine whether the procedure provides adequate guidance to be efficient in instances where a more detailed approach to dose reconstruction would not affect the outcome**		
2.1	Does the procedure provide adequate guidance for identifying a potentially high probability of causation as part of an initial dose evaluation of a claim?	N/A	Not the intention of the procedure.
2.2	Conversely, for claims with suspected cumulative low doses, does the procedure provide clear guidance in defining worst-case assumptions?	N/A	Not the intention of the procedure.
3.0	**Assess the extent to which procedures account for all potential exposures and ensure that resultant doses are complete and based on adequate data**		
3.1	Assess quality of data collected via interviews:	N/A	
3.1.1	Is scope of information sufficiently comprehensive?	N/A	
3.1.2	Is the interview process sufficiently flexible to permit unforeseen lines of inquiry?	N/A	
3.1.3	Does the interview process demonstrate objectivity, and is it free of bias?	N/A	
3.1.4	Is the interview process sensitive to the claimant?	N/A	
3.1.5	Does the interview process protect information as required under the Privacy Act?	N/A	
3.2	Adequacy and use of site-specific data pertaining to:	N/A	
3.2.1	Personal dosimeters (e.g., film, TLD, PICs)	N/A	
3.2.2	In-vivo/in-vitro bioassays	N/A	
3.2.3	Missing dosimetry data	N/A	
3.2.4	Unmonitored periods of exposure	N/A	

Table 8. Procedure Review Outline/Checklist

No.	Description of Objective	Rating 1-5*	Comments
4.0	**Assess procedure for providing a consistent approach to dose reconstruction regardless of claimant's exposures by time and employment locations**		
4.1	Does the procedure support a prescriptive approach to dose reconstruction?	5	
4.2	Does the procedure adhere to the hierarchical process as defined in 42 CFR 82.2?	4	
5.0	**Evaluate procedure with regard to fairness and giving the benefit of the doubt to the claimant**		
5.1	Is the procedure claimant favorable in instances of missing data?	4	
5.2	Is the procedure claimant favorable in instances of unknown parameters affecting dose estimates?	4	
5.3	Is the procedure claimant favorable in instances where claimant was not monitored?	4	
6.0	**Evaluate procedure for its ability to adequately account for the uncertainty of dose estimates**		
6.1	Does the procedure provide adequate guidance for selecting the types of probability distributions (i.e., normal, lognormal?)	N/A	
6.2	Does the procedure give appropriate guidance in the use of random sampling in developing a final distribution?	N/A	
7.0	**Assess procedures for striking a balance between technical precision and process efficiency**		
7.1	Does the procedure require levels of detail that can reasonably be accounted for by the dose reconstructor?	4	
7.2	Does the procedure avoid levels of detail that have only limited significance to the final dose estimate and its POC?	4	
7.3	Does the procedure employ scientifically valid protocols for reconstructing doses?	4	

* Rating system of 1 through 5 corresponds to the following: 1=No (Never), 2=Infrequently, 3=Sometimes, 4=Frequently, 5=Yes (Always), N/A indicates not applicable

3.0 REFERENCES

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