Paper

INTERNAL DOSE RECONSTRUCTION UNDER PART B OF THE ENERGY EMPLOYEES COMPENSATION ACT

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Abstract—**The reconstruction of internal doses under Part B of the Energy Employees Occupational Illness Compensation Program Act differs in multiple ways from that used in a typical operational setting. There are, for example, no limits at or above which doses must be assessed; all doses, including unmonitored or potentially undetected doses, must be reconstructed. In addition, the primary dose of concern is that delivered to the organ in which the cancer originated, and only the dose delivered to that organ prior to the time the cancer was diagnosed is relevant. Additional challenges are presented in the requirement to partition dose by radiation type and energy rather than by radionuclide, the need to include any potential dose that could have been received but was unmonitored or undetected, the inability to collect follow-up samples, and, in many cases, a general lack of information regarding the employee's work history, such as specific duties or location within a site. To overcome these challenges, the NIOSH dose reconstruction program has adopted a set of default values that include assumptions that are favorable to the claimant when there is more than one plausible choice. Due to the large number of claims that must be reconstructed, efforts are continuously underway to expedite the rate at which they can be processed. This is being achieved by taking advantage of situations in which it can be documented that more detailed evaluations would not change the outcome of the adjudication of the claim.**

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INTRODUCTION

The nature of the National Institute for Occupational Safety and Health (NIOSH) Radiation Dose Reconstruction Program requires an understanding of the history of occupational internal dosimetry because the data collected for these programs are the bases on which the internal dose is reconstructed. The primary purposes of such programs have always been to: (1) verify that exposure control measures (e.g., engineering controls, air monitoring, workplace sampling) are adequate and functioning properly; (2) demonstrate regulatory compliance; and (3) evaluate the intake to an individual should one occur. Because regulations have changed over the years and technology has advanced, methods for monitoring individuals and reporting the results have also changed.

In contrast, the objective of the reconstruction of internal doses for claims filed under the Energy Employees Occupational Illness Compensation Program Act (EEOICPA) is to estimate the dose that contributes to the risk of cancer resulting from the intake of radionuclides. The basic tenets of an internal dose calculation under EEOICPA are consistent with those of the International Commission on Radiological Protection (ICRP); the underlying concepts, models, and equations are the same. The departure comes in the application of these methods because the required quantities differ from those used in radiation protection practice. The purpose of internal dose reconstruction under EEOICPA is to provide input to the probability of causation (PC) calculation. The PC is calculated by applying risk coefficients to the radiation dose received by a particular organ using the Web-based NIOSH Interactive RadioEpidemiological Program (NIOSH-IREP), which is described in more detail elsewhere in this issue (Kocher et al. 2008).

CONTRASTS IN INTERNAL DOSE CALCULATION FOR REGULATORY COMPLIANCE AND DOSE RECONSTRUCTION

At a more basic level, the primary differences between doses calculated for purposes of dose reconstruction as contrasted to those for regulatory compliance can be summarized as follows:

- Title 42 CFR Part 82 (U.S. DHHS 2002) requires that the current ICRP models be applied in the calculation of internal doses. This includes use of the ICRP Publication 66 (1994a) respiratory tract model and the biokinetic

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models for radionuclides contained in ICRP Publications 67, 68, 69, and 71 (1993, 1994b, 1995a and b);

- Occupational programs are designed to comply with regulatory limits; the regulations have always specified or implied a dose or dose rate above which monitoring was required. Conversely, this implies that monitoring is not required below some level. In dose reconstruction, all dose is considered to contribute to the risk of cancer induction. This includes dose that could have gone undetected by the monitoring method or dose that was unmonitored;
- The dose of interest is that to the organ in which the cancer was diagnosed, and only the dose that is delivered from the start of exposure through the date of cancer diagnosis, assessed on a calendar year basis. As such, the effective and committed doses (i.e., those doses received in the 50 years following intake), quantities that are currently reported as part of occupational protection programs, are of no use. The same is true for the earlier practices of reporting radionuclide intakes as a fraction of secondary radiation limits, such as the Maximum Permissible Body Burden, Maximum Permissible Organ Burden, or the Maximum Permissible Concentration, rather than as a dose. The first two are indicators of the instantaneous dose rate and cannot be converted to dose without knowing all assumptions used in their determination; and
- The dose must further be divided into the contributions from the individual radiation components rather than assigning total doses from radionuclides.

IDENTIFICATION OF THE ORGAN OF INTEREST

The first step to be taken before calculating the dose is to determine the organ for which the dose is to be reconstructed. Internal dose may result from radioactive material that is taken into the body through inhalation, ingestion, or absorption. The material distributes throughout various organs and tissues, the organs being dependent on the element and the form of the material. The distribution of material is specified in the ICRP (1993, 1994b, 1995a and b) biokinetic models. The decay of the radionuclide results in the irradiation of organs and tissues throughout the body, yielding a dose from a source internal to the body. The organs in which the material deposits are termed *source tissues* or *source regions* by the ICRP (1979, 1994b), while the organs in which the radiation is absorbed are known as the *target tissues*. Table 1 lists the source and target organs used in the calculations.

There are two basic steps in the ICRP dosimetric scheme:

Table 1. Source and target organs used in the calculation of internal doses using IMBA.[a]

Source organs	Target organs
Adrenals	Adrenals
Bladder (contents)	Urinary bladder (wall)
Brain	Brain
Breast	Breast
Gall bladder (contents)	Gall bladder (wall)
Heart content	Heart (wall)
Heart wall	Kidney
Kidneys	Liver
Liver	Muscle
Muscle	Ovaries
Ovaries	Pancreas
Pancreas	Testes
Testes	Thyroid
Thyroid	Red bone marrow
Red bone marrow	Bone surfaces
Bone	Stomach (wall)
Cortical volume	Small intestine (wall)
Cortical surface	Upper large intestine (wall)
Trabecular volume	Lower large intestine (wall)
Trabecular surface	Skin
Stomach (contents)	Spleen
Small intestine (contents)	Thymus
Upper large intestine (contents)	Uterus
Lower large intestine (contents)	ET[b]
Stomach wall	Lung
Skin	Colon
Spleen	ET1[b]
Whole body	ET2[b]
ET1[b]	LNET[c]
ET2[b]	BBsec
ET[b] sequestered	BBbas
ET bound	bbsec
LNET[c]	AI
BB1 (bronchial fast)	LNTH[e]
BB2 (bronchial slow)	Esophagus
BB sequestered	
BB bound	
bb1 (bronchial fast)	
bb2 (bronchial slow)	
bb sequestered	
bb bound	
AI[d]	
AI bound	
LNTH[e]	

[a] James AC. User manual for IMBA Expert OCAS/ORAU-Edition (Version 4.0). Richland, WA: ACJ & Associates, Inc.; 2005.
[b] ET = extrathoracic.
[c] LNET = extrathoracic lymph node.
[d] AI = alveolar-interstitial region.
[e] LNTH = thoracic lymph node.

1. Calculate the number of transformations of the radionuclides of interest in the appropriate *source* tissues; and
2. Calculate the dose delivered to the *target* tissue of interest from the transformations in all source tissues.

The number of transformations, $U_{S,j}$, of radionuclide, j, in source tissue, S, that occur over the time interval from time t_1 to time t_2 is given by

$$U_{S,j} = \int_{t_1}^{t_2} q_{S,j}(t)dt, \quad (1)$$

where $q_{S,j}(t)$ is the quantity of radionuclide, *j,* in source tissue, *S,* as a function of the time *t* post intake. In the standard ICRP schema, t_1 is zero (i.e., the time intake began) and t_2 is 50 y post intake (i.e., 50 y committed dose). As noted above, for PC determinations, the organ dose in each calendar year from the time of intake to the time the cancer was diagnosed is the required quantity. The transformations, *U*, can be computed using eqn (1) in which t_1 and t_2 represent 1 January and 31 December, respectively, in the same year.

For reconstruction purposes, H_T (the annual organ dose to target tissue *T* from radionuclide *j*) can be computed using eqn (2), which is derived from ICRP (1989) dosimetric models:

$$H_T = \sum_S \sum_j U_{S,j} SEE(T \leftarrow S)_j, \quad (2)$$

where $SEE(T \leftarrow S)$, Specific Effective Energy, is the energy imparted per unit mass of a target tissue as a consequence of the emission of a specified radiation, *j,* from a transformation occurring in source tissue (*S*). This is commonly expressed in units of Sv $(Bq\ s)^{-1}$.

When the cancer originates in one of the target tissues identified by the ICRP, this analysis is straightforward; however, there are more than a few situations where the organ of interest is not an ICRP target tissue. For example, while prostate cancer is among the most common forms of cancer, the prostate is not identified as a target organ. To deal with this situation, surrogate organs are determined through an application of the current ICRP biokinetic models for individual elements. A limited number of organs are specifically named as source tissues in each model, while others are modeled indirectly through a soft tissue compartment that accounts for a small amount of the intake being incorporated into "all other soft tissue." For example, the strontium biokinetic model recommended in ICRP Publication 67 (1993) specifies parametric values for the bladder, cortical bone volume and surface, trabecular bone volume and surface, and the upper and lower large intestines; all other tissues are included in the soft tissue compartment. For the purpose of the dose reconstruction program, the organs and tissues included in this soft tissue compartment are referred to as "nonmetabolic organs" because it is necessary to refer to specific organs and tissues rather than a grouping of them. When the energy emitted by a radionuclide is in the form of alpha or beta radiation, the soft tissue compartment is the source of dose unto itself (this is also the case for each organ named in the biokinetic model), that is, the absorbed fractions for particulate radiations are assumed to be 1.0. As a result, all nonmetabolic organs receive equal doses.

For radionuclides where photon emissions are a major source of the emitted energy, the nonmetabolic organ doses can vary in relation to their distance from the source tissues in which the radionuclides deposited, depending on the intensity of the radiation. Therefore, when the cancer has originated in an organ that is not included in the ICRP target tissues, the dose assigned to that organ is the highest estimated for any nonmetabolic organ. This concept is referred to as the "highest nonmetabolic organ" and is used to assign dose to organs such as the prostate. Two examples of how this determination is made are shown in Table 2. In the case of ^{239}Pu (type M), an alpha-emitting radionuclide with only low energy gamma rays, the dose coefficients for all of the nonmetabolic organs are equal, so the dose to an organ not specified by the model is also assigned using this coefficient. ^{131}I decays by beta emission with associated gamma rays of higher energies, so there are differences between the dose coefficients for the nonmetabolic organs. The thymus and esophagus, organs in proximity to the thyroid (the primary organ of deposition), receive higher doses than those at a greater distance from the radiation source. In this case, the nonmetabolic organ closest to the organ of cancer origination would be the most appropriate surrogate organ.

Identification of the target organs for cancers of the mouth is another example of this challenge. The model in ICRP Publication 66 (1994a) includes the mouth in the

Table 2. Dose coefficients for two nuclides—see text for explanation of their use in identifying appropriate surrogates for nonmetabolic organs.

Organ[b]	Equivalent dose coefficient $(Sv\ Bq^{-1})$[a]	
	^{239}Pu (type M)[c]	^{131}I (type F)[c]
Adrenals	1.78×10^{-6}	2.52×10^{-11}
Urinary bladder	*1.78×10^{-6}*	*3.66×10^{-10}*
Brain	1.78×10^{-6}	8.12×10^{-11}
Breast	1.78×10^{-6}	3.16×10^{-11}
Gall bladder	1.78×10^{-6}	2.53×10^{-11}
Heart wall	1.78×10^{-6}	3.99×10^{-11}
Kidneys	*4.16×10^{-6}*	2.36×10^{-11}
Liver	*2.12×10^{-4}*	2.51×10^{-11}
Muscle	1.78×10^{-6}	7.06×10^{-11}
Pancreas	1.78×10^{-6}	2.80×10^{-11}
Thyroid	1.78×10^{-6}	*2.09×10^{-7}*
Red bone marrow	*4.81×10^{-5}*	5.62×10^{-11}
Bone surfaces	*1.01×10^{-3}*	6.95×10^{-11}
Skin	1.78×10^{-6}	3.75×10^{-11}
Spleen	1.78×10^{-6}	2.54×10^{-11}
Thymus	1.78×10^{-6}	8.52×10^{-11}
Uterus	1.78×10^{-6}	3.35×10^{-11}
Esophagus	1.78×10^{-6}	8.52×10^{-11}
Gonads	*1.35×10^{-5}*	3.00×10^{-11}

[a] ICRP Publication 68 (1994b).
[b] Includes all systemic organs in the ICRP biokinetic models.
[c] Italicized values indicate a modeled organ for the given radionuclide.

extrathoracic (ET2) region, but the dose estimate for that region is not based on any tissues in the mouth; the estimate is based on the tissues of the pharynx. The pharynx is posterior to both the mouth and the nasal cavity, so activity passing through it is not likely to be representative of the mouth for inhalation intakes. While the decision was made not to write models specific to the dose reconstruction program, it was necessary in this case to interpret the dose that the ICRP lung model was attributing to the mouth rather than the pharynx. This resulted in the mouth being considered part of the soft tissue compartment rather than a part of the ET2 region, thereby requiring application of the highest nonmetabolic organ concept. This interpretation later was substantiated by the recently revised ICRP human alimentary tract model (ICRP 2006).

In still other cases, the cancer diagnosis is not specific to a single organ or tissue. For example, there is a cancer that is coded as "malignant neoplasm of the endocrine gland, site unspecified." Endocrine glands are located throughout the body, so a review of the cancer diagnosis documentation by a medical doctor is required on a case-specific basis to determine the appropriate organ for dose calculation.

DOSE CALCULATIONS

In the context of EEOICPA, a dose calculation does not necessarily equate to the most accurate estimate of *dose* that can be constructed. Because this is a compensation program, the goal is to reach an accurate *compensation* decision. That is, approximations can be made and efficiency methods employed while still arriving at the correct determination of a compensation decision. For example, it is possible to complete some cases using an overestimate of the dose that was actually received by the worker. If, under these conditions, the dose results in a non-compensable determination, it is not necessary to further refine the dose estimate because it would not change the outcome of the decision. This allows some flexibility when dealing with unknowns. A variety of techniques have been developed, the applications of which vary depending on the specific details of the case and the data available. As a result, the dose reconstructors must exercise professional judgment in determining the most appropriate method to use.

Similarly, a *best estimate* of dose does not mean that the goal is to achieve the most accurate reconstruction of the true dose received by the worker. Rather, it is the best approximation that can be made within the constraints and requirements of the dose reconstruction program. The majority of cases do not require a best estimate because it has been found that methods to expedite the cases can frequently be employed. These methods, referred to as efficiency methods, are described in more detail later in this section. Only if an efficiency method is not consistent with the outcome (i.e., an overestimate produces a compensable decision and an underestimate produces a non-compensable decision) is a best estimate performed. Such an approach reduces the time needed to complete cases and lessens the need for detailed knowledge of all site processes and worker activities. This is especially useful for the assessment of internal doses, where the possible intake patterns are frequently unknown, particularly when all results are less than the detection limit, but determination of upper and lower bounds are possible.

Assumptions and claimant favorability

A significant difference between dose reconstruction and a typical situation in a radiological protection program is the availability of data. Dose reconstruction is performed after the fact and must rely on available information while a protection program functions in real time and can often collect additional information identified as necessary for a dose assessment. The requirement for timeliness of compensation decisions has an impact on the approaches to dose reconstruction; a balance must be struck between efficiency and accuracy while continuing to be grounded in the best available science. To meet this requirement, default assumptions are applied on both site-specific and complex-wide bases (Merwin et al. 2008a).

In the early days of the U.S. Department of Energy (DOE) complex, information regarding such things as incidents, specific job functions, and selection of individuals for monitoring was often recorded in logbooks that may not be readily accessible or may be in a format such that data extraction is difficult. Analytical techniques were in the developmental stages and could change frequently in an effort for improvement, and pertinent information, such as data needed to determine detection levels, is lacking. If a specific parameter is unknown, the value that is most favorable to the claimant (i.e., that which results in the largest dose) must be selected from the plausible values. Examples of how this is applied include:

- Early monitoring methods frequently could not distinguish between several nuclides; samples were analyzed using gross analytical techniques, such as a gross alpha, beta, or gamma analysis. Lack of information about specific work activities or location of the individual makes it difficult to determine which radionuclide(s) is (are) applicable to a given worker. Site information and the analytical procedure, when available, are used to narrow the list of possibilities. To

determine the intake for an individual, the list of possible radionuclides is assessed and the one that delivers the largest dose is assigned;

- An individual working in multiple or unknown locations within a facility that handled many forms of uranium has the potential for exposure to a range of solubility types—F, M, and S. Given sufficient data, the material type can be determined by performing a fit to the bioassay results; this is discussed in more detail below. In the vast majority of cases, however, bioassay results are sporadic and variable and do not provide adequate information. In such cases, the material type that maximizes the dose to the organ in which the cancer formed is used to determine the intake, as long as no bioassay results are in disagreement with the assumption. For example, the dose calculated from urine samples may be maximized by the assumption of a type S material, but the predicted lung content may greatly exceed that actually measured by chest counts, so this intake is ruled out; and
- Uranium has frequently been measured in mass units because of the concern for nephrotoxicity. Only radiation dose is of interest under EEOICPA, so activity is needed to calculate a dose; this requires knowledge of the enrichment of the uranium material that was handled. Some sites used many different levels of enrichment in different areas over the lifetime of the site, so location and date of potential exposure are frequently of importance to the determination of any potential intake(s). In cases where information is inadequate to determine a worker's location precisely, the enrichment yielding the largest activity is applied.

Missed dose

Missed dose, for the purpose of this program, is defined as the potential dose that could have been received by a bioassay program participant but, because of limitations in the monitoring system, was undetected. Therefore, missed dose is calculated only for individuals who were monitored. Missed dose is assigned when the bioassay results for an individual are less than the detection limit. Given no other information, the true dose could have been between zero and that which would be delivered by an intake consistent with a bioassay result at the detection level. The lack of regular bioassay data and associated intake information makes the construction of possible intake scenarios difficult, particularly when there are no positive results. For an individual performing the same job with the same type of material throughout a given time frame, it is assumed in most cases that the exposure was chronic and lasted for the duration of the possible exposure period. This is more realistic than assuming that intakes were from only a few unknown incidents at unknown times. If many acute intakes occur throughout the exposure period, this assumption is still appropriate because a chronic intake will approximate a series of small acute intakes and is sufficiently accurate for most cases, particularly when no information about possible intakes is known.

It is possible that bioassay results below the detection limit could be the result of no intake. However, it is also possible that a chronic intake will result in a true bioassay value just below the detection limit. The probability of this is significantly reduced as more results below the limit are obtained. To account for these possibilities, a triangular distribution is used in IREP to represent the probability distribution of the potential missed doses. This distribution allows all the possibilities between zero and the detection limit to be accounted for while providing little weight to the extreme possibilities. Details of the data distribution types used in IREP are discussed elsewhere in this issue (Kocher et al. 2008).

Unmonitored dose

Unmonitored dose is defined as the potential dose that could have been received by an employee but for which no monitoring was performed or no monitoring data are available. As previously noted, regulations for occupational protection programs require monitoring when a worker's dose is expected to exceed a specific value, while EEOICPA does not provide for a minimum value—all dose contributes to the probability that a cancer occurred. This requires inclusion of dose that may have been unmonitored. When there is uncertainty as to whether an intake may have occurred, the "benefit of the doubt" goes to the claimant and a dose is assigned. As might be anticipated, unmonitored dose can arise through several scenarios.

Monitoring for selected radionuclides. Because occupational protection programs often control intakes to the most dosimetrically significant radionuclide—significant being relative to the limits in place at the time of monitoring—some nuclides can go unmonitored. The relative contributions of these nuclides to the total effective dose can be quite different from their contribution to the dose in a specific organ. For example, in a facility with both plutonium and tritium intake potentials, the effective dose from a type S plutonium intake may far outweigh any dose received from tritium. However, the effective dose from an intake of type S plutonium is largely driven by lung dose while the tritium dose is distributed uniformly throughout the body. This can cause the tritium to be the larger dose contributor to some organs that do not concentrate plutonium, such as the

brain. This phenomenon can be demonstrated by comparing selected organ dose coefficients, normalized to a unit effective dose, for ^{239}Pu (type S) and ^{3}H in the form of tritium oxide (HTO) (Table 3). A further complication arises when the annual organ doses are factored into the evaluation. Because, as noted previously, only the dose through the date of cancer diagnosis is of interest, the plutonium can contribute a very small dose to some organs if the date of diagnosis is not long after the start of intake (Table 4). The intake used to generate the annual doses in Table 4 is the same as that used in Table 3.

Monitoring for nuclides categorized as a contaminant in a material. A prime example of this is recycled uranium, which is uranium that has been irradiated in a reactor, chemically processed, and returned to a facility that processes uranium for materials preparation, purification, enrichment, and fabrication. Such uranium contains trace quantities of transuranics and fission products that in some cases will contribute sufficient dose to warrant inclusion in an evaluation; this is typically dependent on the organ of interest. Radiation protection personnel at DOE sites historically did not monitor individuals for these contaminants because it was believed that controlling exposure to the uranium would provide adequate protection of the worker.

Lack of bioassay data. In some cases, most frequently in the early decades of the weapons complex, information (such as job title or interview responses) in the worker's file appears to indicate a potential for intakes, but no bioassay results are available. This could be due to a number of reasons: the worker's employment precedes the implementation of a bioassay program; the data could not be located or were not easily retrievable; or monitoring was not believed to be necessary at the time.

Table 3. Comparison of normalized dose distributions in various organs due to inhalation of HTO (vapor) and ^{239}Pu (type S).[a]

Organ	Normalized dose distribution[b]	
	HTO (vapor)	^{239}Pu (type S)
Effective	1.0	1.0
Bone surfaces	1.0	11
Brain	1.0	0.018
Breast	1.0	0.018
Lower large intestine	1.0	0.022
Kidneys	1.0	0.047
Liver	1.0	2.3
Muscle	1.0	0.018
Red marrow	1.0	0.54
Lungs	1.0	5.7
Testes	1.0	0.14

[a] ICRP Publication 68 (1994b).
[b] Ratio of equivalent dose to the effective dose.

Table 4. Annual doses to selected organs from an acute inhalation intake of ^{239}Pu (type S) that delivers an effective dose of 1 Sv.

Year after intake	Annual dose (Sv)			
	Brain	Lung	Bone surface	Liver
1	0.00015	3.2	0.062	0.011
2	0.00026	0.47	0.12	0.021
3	0.00031	0.34	0.16	0.029
4	0.00034	0.25	0.18	0.035
5	0.00035	0.19	0.20	0.040
6	0.00035	0.15	0.21	0.043
7	0.00035	0.12	0.21	0.045
8	0.00035	0.094	0.22	0.048
9	0.00035	0.078	0.22	0.049
10	0.00035	0.066	0.22	0.050

Fitting bioassay results

A fit to the data may be performed when there are positive bioassay results in the worker's file. This occurs far less frequently than was anticipated at the outset of the program; the majority of workers were either unmonitored or had results less than the detection levels, or their doses can be assessed using efficiency methods. This aspect of EEOICPA comes closest to standard methods used in an occupational setting. Any known information about the intake, such as its date of occurrence, particle size, or material type, is used in the fit. When the intake date is unknown, as it is in the majority of cases, it is initially assumed to have occurred midpoint between the date of the positive result and that of the previous result. This intake date is modified if the resulting fit is in disagreement with the data. The default particle size distribution is 5 μm activity median aerodynamic diameter (typically referred to as AMAD), as specified in ICRP Publication 66 (1994a). The departure from an occupational program, aside from the models, comes when there are insufficient data to distinguish between multiple possibilities; in such cases, the claimant-favorable options are selected.

Complicating factors

Radiation weighting factors, previously referred to as quality factors, represent the effectiveness of different types of radiation in causing biological damage. In the IREP software, these factors have been replaced with radiation effectiveness factors (Kocher et al. 2005); these are distributions rather than a single, constant value. The models incorporated into IREP are based on the assumption that the input dose was calculated using ICRP Publication 60 (1991) radiation weighting factors. These values are removed and replaced with the radiation effectiveness distributions programmed into IREP. Radiation effectiveness distributions are assigned to alpha particles, two ranges of beta radiation energies, three energy ranges of photon radiation, and five ranges of

neutron energies. Therefore, all doses must be separated into their individual radiation components, rather than by radionuclide, when compiling the data. This is a significant departure from traditional internal dosimetry and presents several challenges. The calculation of such doses is quite complex and caused a significant increase in the time required to compute them using the Integrated Modules for Bioassay Analysis (IMBA) program, which is further discussed below. With internal doses already divided into annual doses, further separation of each annual dose into radiation type can quickly lead to many lines of input into the NIOSH-IREP program, causing a large increase in computational time as well. In cases in which the time between start of exposure and date of diagnosis is long, the maximum number of input lines for the program can be exceeded. These issues are avoided by choosing a single, favorable radiation type to represent the dose from a given radionuclide.

The requirement for the doses to be calculated on a calendar year basis complicates the assessment, particularly in the development of efficiency tools, because the annual doses following a given intake will vary dependent on the day of the year in which it occurred or began (Table 5). Calendar year doses resulting from unit intakes of ^{239}Pu (type M) and ^{137}Cs (type F) are given for selected organs; intake dates of 1 January and 1 September are compared. ^{239}Pu is a long-lived, long-retained nuclide, so once deposited in an organ it remains there for an extended period. The type M material clears from the lungs relatively rapidly and continues to be cleared, but once material has reached the systemic organs, the annual doses tend to converge. ^{137}Cs distributes throughout the body without concentrating in a specific organ and clears rapidly; 90% of the dose is delivered within the first year and approximately 99% is delivered within the first 2 y. In such cases, the selected date of intake strongly influences the dose distribution.

Efficiency methods

As noted previously, efficiency methods are employed whenever possible to expedite the completion of cases. There are two general types of expediting methods that can be applied: overestimation and underestimation.

Because the organ of interest is that in which the cancer originates rather than that receiving the largest dose, it is possible to postulate a relatively large intake

Table 5. Comparison of annual (calendar year) doses (Sv) to selected organs due to a single acute inhalation intake of either 1 Bq of ^{239}Pu (type M) or of 1 Bq of ^{137}Cs (type F) on two different dates (1 January or 1 September) within a year.

^{239}Pu (type M)								
	Annual dose (Sv)							
Year after intake	Brain		Bone surface		Lung		Liver	
	1 Jan	1 Sept	1 Jan	1 Sept	1 Jan	1 Sept	1 Jan	1 Sept
1	5.9×10^{-8}	1.8×10^{-8}	2.5×10^{-5}	6.9×10^{-6}	1.9×10^{-5}	1.7×10^{-5}	4.3×10^{-6}	1.1×10^{-6}
2	5.9×10^{-8}	6.2×10^{-8}	2.9×10^{-5}	2.8×10^{-5}	3.5×10^{-7}	1.6×10^{-6}	5.4×10^{-6}	4.9×10^{-6}
3	5.2×10^{-8}	5.7×10^{-8}	2.8×10^{-5}	2.9×10^{-5}	8.6×10^{-8}	2.0×10^{-7}	5.6×10^{-6}	5.5×10^{-6}
4	4.6×10^{-8}	5.0×10^{-8}	2.7×10^{-5}	2.8×10^{-5}	5.0×10^{-8}	6.7×10^{-8}	5.6×10^{-6}	5.6×10^{-6}
5	4.2×10^{-8}	4.5×10^{-8}	2.6×10^{-5}	2.7×10^{-5}	4.2×10^{-8}	4.7×10^{-8}	5.6×10^{-6}	5.6×10^{-6}
6	3.9×10^{-8}	4.1×10^{-8}	2.5×10^{-5}	2.6×10^{-5}	3.9×10^{-8}	4.1×10^{-8}	5.6×10^{-6}	5.6×10^{-6}
7	3.6×10^{-8}	3.8×10^{-8}	2.4×10^{-5}	2.5×10^{-5}	3.6×10^{-8}	3.8×10^{-8}	5.6×10^{-6}	5.6×10^{-6}
8	3.5×10^{-8}	3.6×10^{-8}	2.4×10^{-5}	2.4×10^{-5}	3.5×10^{-8}	3.6×10^{-8}	5.6×10^{-6}	5.6×10^{-6}
9	3.4×10^{-8}	3.4×10^{-8}	2.4×10^{-5}	2.4×10^{-5}	3.4×10^{-8}	3.4×10^{-8}	5.5×10^{-6}	5.5×10^{-6}
10	3.3×10^{-8}	3.3×10^{-8}	2.3×10^{-5}	2.3×10^{-5}	3.3×10^{-8}	3.3×10^{-8}	5.4×10^{-6}	5.5×10^{-6}
^{137}Cs (type F)								
	Annual dose (Sv)							
Year after intake	Brain		Bone surface		Lung		Liver	
	1 Jan	1 Sept	1 Jan	1 Sept	1 Jan	1 Sept	1 Jan	1 Sept
1	5.1×10^{-09}	3.1×10^{-09}	6.0×10^{-09}	3.6×10^{-09}	5.6×10^{-09}	3.3×10^{-09}	5.9×10^{-09}	3.5×10^{-09}
2	4.5×10^{-10}	2.4×10^{-09}	5.8×10^{-10}	2.7×10^{-09}	5.4×10^{-10}	2.5×10^{-09}	5.7×10^{-10}	2.7×10^{-09}
3	4.9×10^{-11}	2.3×10^{-10}	5.7×10^{-11}	2.7×10^{-10}	5.2×10^{-11}	2.5×10^{-10}	5.6×10^{-11}	2.6×10^{-10}
4	4.8×10^{-12}	2.3×10^{-11}	5.6×10^{-12}	2.6×10^{-11}	5.1×10^{-12}	2.4×10^{-11}	5.5×10^{-12}	2.6×10^{-11}
5	4.7×10^{-13}	2.2×10^{-12}	5.4×10^{-13}	2.6×10^{-12}	5.0×10^{-13}	2.4×10^{-12}	5.4×10^{-13}	2.6×10^{-12}
6	4.6×10^{-14}	2.2×10^{-13}	5.3×10^{-14}	2.5×10^{-13}	4.9×10^{-14}	2.3×10^{-13}	5.3×10^{-14}	2.5×10^{-13}
7	4.5×10^{-15}	2.1×10^{-14}	5.2×10^{-15}	2.5×10^{-14}	4.8×10^{-15}	2.3×10^{-14}	5.1×10^{-15}	2.4×10^{-14}
8	4.4×10^{-16}	2.1×10^{-15}	5.1×10^{-16}	2.4×10^{-15}	4.7×10^{-16}	2.2×10^{-15}	5.0×10^{-16}	2.4×10^{-15}
9	4.3×10^{-17}	2.0×10^{-16}	5.0×10^{-17}	2.4×10^{-16}	4.6×10^{-17}	2.2×10^{-16}	4.9×10^{-17}	2.3×10^{-16}
10	4.2×10^{-18}	2.0×10^{-17}	4.8×10^{-18}	2.3×10^{-17}	4.5×10^{-18}	2.1×10^{-17}	4.8×10^{-18}	2.3×10^{-17}

that will yield a small dose. For example, an intake of 1×10^4 Bq of ^{239}Pu (type S) yields a committed equivalent dose of 0.91 Sv to the red bone marrow and 0.47 Sv to the lungs, but delivers only 1.6×10^{-3} Sv to the soft tissue compartment organs. This difference can be used to advantage in the development of techniques for expediting the processing of claims. Upper bound intakes, based on airborne activity limits, process knowledge, worker information, or a combination of these, can be developed to apply to some organs where a large intake will not produce a comparably large dose. If this overestimate does not yield a favorable compensation decision, further refinement of the dose estimate is not required because the dose can only become smaller and will, therefore, not change the compensation decision. Although developed on the basis of assessing doses to individual workers, these procedures can be applied to groups of workers with similar job categories or expected exposure potentials.

An underestimate is the assignment to a worker of an intake or dose that is less than that which potentially would be assigned under this program. If the resulting PC yields a favorable decision, further refinement is not necessary because it will only increase the assigned dose. An underestimate typically is based on a partial assessment of dose (e.g., a single incident, missed dose only) or the under-prediction of all or most positive bioassay results. An underestimate is most likely to be successful when applied to organs specified in the biokinetic model, particularly in cases where the detection level for the nuclide is large. This frequently is the case with actinides in the earlier decades of the weapons complex. In such cases, a missed dose calculation alone might be adequate for determining compensability. Because this method is dependent on an individual's bioassay data, the details are case-specific and do not lend themselves to a generic approach that can be documented in a procedure.

Supplements to the ICRP models

On occasion, there is a need to supplement the ICRP models because issues arose that were not covered by existing models. For example, there are documented cases (Carbaugh and La Bone 2003) where bioassay measurements have demonstrated that some forms of plutonium may be retained in the lungs for much longer times than described by the standard models; this form is sometimes referred to as type "Super S." It was believed that there were insufficient data available on which to base the development of a new model, but it was clear that this form would need to be considered when reconstructing doses. Adjustment factors to be applied to the type S intake rate, annual equivalent dose, or both were developed from several cases previously identified as exhibiting this behavior. These factors, which are for use in this application only, are a function of the length of exposure, the organ of interest, and the bioassay method used to determine the dose.

Another example in which there is a lack of ICRP guidance is in estimating the doses from the metal tritides. A technical information bulletin (ORAUT 2007) providing guidance on this material was developed.

Intakes via wounds were common occurrences at some facilities and happened on occasion at many others. Prior to the recent guidance (NCRP 2007) on the assessment of wounds from the National Council on Radiation Protection and Measurements, a technical information bulletin (ORAUT 2005) was issued to provide guidance to the dose reconstructors.

Uncertainty models

As noted elsewhere in this issue (Merwin et al. 2008b), the calculation of the PC takes into account not only the uncertainty in the radiation risk factors, but also the uncertainty in the dose. Information required for the triangular distribution, used in the assignment of missed dose, is the minimum, mode, and maximum doses. In the particular case of missed dose, the mode dose is calculated based on half the detection limit, using collection dates of bioassay samples and worker-specific employment information. The minimum dose is assumed to be zero and the maximum dose twice the mode (i.e., the dose based on the detection level).

When positive results are present, a fit to the results is performed as previously described. A dose calculated from an individual's bioassay results is assumed to be distributed lognormally, and a geometric standard deviation (GSD) of 3 is applied to account for uncertainty in the biokinetic models (Boecker et al. 1991). A lognormal distribution is also applied to coworker-derived doses; a minimum value of 3 is assigned (i.e., if the calculations yield a GSD less than 3, 3 is used as the input in to IREP) because the same uncertainty in the models would apply.

A constant value is assigned for most overestimates and underestimates; no distribution is assumed. This "distribution" is used to provide a boundary for the dose reconstruction so there is no associated uncertainty. For partial dose estimates, a lower boundary is used; the dose is known to be at least this large. For overestimated doses, the dose could not have reasonably exceeded this amount.

DATA HIERARCHY

A general hierarchy of data has been developed for determining intakes during periods where there was no monitoring. Because individual monitoring data, such as

bioassay results or breathing zone samples, are most representative of the individual's exposure, these are used whenever possible. The specific method that is preferred is dependent on the radionuclide, its chemical form, and the primary route of intake. For much of the early history of the DOE complex, urinalysis is the only available option.

Situations frequently arise where there are several radionuclides in a mixture, such as the previously discussed recycled uranium, mixed fission and activation products, or weapons-grade plutonium. Under these circumstances, it is not always practical or cost-effective to perform bioassay monitoring for all components of the mixtures. Fortunately, a subset of them can provide adequate information, provided the ratios of the unmonitored to monitored radionuclides are known. These ratios provide the best information for the unmonitored components.

When there are no bioassay results for an individual for a specific source term, data from monitored coworkers are preferred for estimating intakes. Because bioassay collection and analyses are often expensive, time-consuming, and inconvenient for the worker, such procedures are typically performed only on those most likely to have intakes. Therefore, it is believed that the distribution of intakes derived from these data is representative of the potentially exposed population. At the same time, it is assumed that it is unlikely that an unmonitored worker would have received a larger dose than the most highly exposed monitored worker at a site. Ideally, one would perform a statistical analysis of the intakes of individuals working in the same area or performing the same job function as the unmonitored worker for a given period. However, two major issues must be considered:

- Because previous assessments performed by the sites are not applicable for this work, calculations must be based on bioassay results. Time constraints make it impractical to assess intakes to the thousands of monitored individuals at each site; and
- Information is lacking, particularly for those without claims, regarding employment period, work location, and function. This makes it difficult to identify true coworkers.

Therefore, the general approach is to determine the statistical distribution and parameters of all bioassay data for all individuals, by radionuclide, in a given period; this is typically a calendar year but may be longer or shorter depending on the volume of data. A lognormal distribution is assumed, and the 50^{th} and 84^{th} percentiles of these data sets are used to assess intakes, for each possible material type, throughout the site's history (or during those times when bioassay data are available). The result of these statistical analyses is two data sets: that composed of a single bioassay result for each period, equal to the geometric mean of all of the data in the period; and that composed of the 84^{th} percentile values for the same periods. Each of these data sets is used to fit a series of intakes for assignment of dose to unmonitored workers and to determine the GSD of the distribution, which is equal to the 84^{th} percentile intake rate divided by the 50^{th} percentile intake rate.

Bioassay results for an individual are not independent (i.e., material will be excreted or retained in a given bioassay compartment for some period of time following an intake so multiple samples can be representative of the same intake) and will vary in relation to the length of time between the intake and the measurement. As a result, the aggregated bioassay results for a given time are not necessarily related only to intakes within that period. Another complicating factor is that multiple samples are typically collected from a single individual after a known intake occurs so there can be a bias towards positive results. However, the data set as a whole is generally representative of conditions at the site. For many sites, the typical unmonitored worker is assigned the 50^{th} percentile intake, with an associated GSD determined as described above. The dose reconstructor can assign a larger percentile if the worker's job history indicates a larger potential for intakes. At facilities where few individuals were monitored and there is uncertainty as to whether the most exposed workers were included in the monitoring, the 95^{th} percentile intakes can be assigned.

When neither individual-specific nor coworker monitoring is available, the next tier of data to be used is workplace monitoring. Air monitoring data are preferred; source term information may be used when these are not available. This type of intake is generally assigned when there was no individual-specific monitoring program in place, such as at Atomic Weapons Employer facilities where radiological work may not have been the main focus of the site or at facilities with small-scale campaigns involving materials that were not normally part of their bioassay monitoring program. In these cases, data based on air monitoring and/or area surveys are used in conjunction with process knowledge to estimate intakes based on workers' potential for entry and exposure in these areas.

Finally, if none of the aforementioned data are available, information from a surrogate site may be used to determine an appropriate exposure scenario for the worker. The use of a surrogate site to reconstruct doses must be carefully evaluated to ensure that similar work conditions and processes involving similar radionuclides were employed. Surrogate site data follow the same

hierarchy as outlined above; use of coworker data at a surrogate site is considered before the use of area monitoring at a surrogate site. A summary of this information is presented in Table 6.

DOSE COMPUTATIONAL TOOLS

Integrated Modules for Bioassay Analysis (IMBA)

The complexity associated with meeting the requirements of the internal dose reconstruction portion of EEOICPA necessitated the procurement of sophisticated software. First and foremost, the software was required to incorporate the latest recommendations and models of the ICRP. Time was also of the essence; project schedules did not allow time for the development of new software. Because U.S. regulations were still based on ICRP Publication 30 (1979) methods at the outset of the dose reconstruction program, few options were available. The United Kingdom had already adopted the newer ICRP models, and DOE was working with the U.K. Health Protection Agency to adapt its internal dose assessment software, IMBA, to its needs. After a discussion with the authors of the software, it was determined that IMBA, with some modifications, could also meet the needs of the NIOSH Radiation Dose Reconstruction Program.

IMBA Expert is a Microsoft Windows application written in Visual BASIC and is composed of a number of individual modules. Initially, two versions of IMBA were implemented: IMBA-NIOSH and IMBA Expert USDOE. The DOE version was immediately available but did not provide all of the necessary functionality; it incorporated the current recommendations of the ICRP and allowed the simultaneous fitting of up to ten intakes but did not have the capability of calculating annual organ doses. The initial version of IMBA-NIOSH was limited in function so that it could be made available as soon as was possible. The primary purpose of the first version of this software was the calculation of annual organ doses to a limited number of key radionuclides, including several isotopes of uranium and plutonium. Intakes were calculated using the DOE version and the results served as input to the NIOSH version for determining the annual organ doses. The functionality of the DOE version was subsequently incorporated into IMBA-NIOSH and the software has evolved over the course of the dose reconstruction program to its current incarnation as IMBA Expert ORAU-Edition.

Modules were added to calculate annual (on a calendar year basis) organ doses and to separate these doses into their radiation type components (e.g., alpha and electrons) as required under EEOICPA. Additional features of the software include:

- The ability to assess simultaneously ten intakes, both chronic and acute, for a given radionuclide;
- The capability of modeling a variety of input data, including up to 200 data points for each of several bioassay types (urinalysis, fecal analysis, whole body counts, chest counts, blood samples, thyroid counts, liver counts); and
- A user-defined bioassay compartment.

Chronic Annual Dose Workbook (CADW)

Additional tools, primarily in the form of spreadsheets, were developed to ensure both efficiency and consistency in the internal dose calculations. Initially, the IMBA annual dose calculations for a few nuclides with many radioactive progeny, using the early versions of IMBA, required a very long time—on the order of hours—to complete. Also, IMBA is limited by its ability to calculate the annual organ doses from only a single nuclide at a time; this can become inconvenient for cases where the individual might have been exposed to multiple nuclides. Additional applications were therefore developed to assist in the processing of claims.

The CADW spreadsheet is essentially a lookup table containing the annual doses for a unit intake of all of the radionuclide/material type/organ combinations in IMBA. To use this tool, the dose reconstructor first determines the intake quantities for all radionuclides for a given individual. This determination is frequently done using IMBA, although the site profiles or other technical documents can also supply intakes applicable to the case. The intake quantities and associated information, including the radionuclide, organ of interest, and the relevant

Table 6. Hierarchy of data sources for reconstruction of doses from internally deposited radionuclides.

Hierarchy	Data source	Examples
1	Personal monitoring	In-vivo analyses, in-vitro analyses, breathing zone air samples
2	Indicator radionuclides	Known mixtures of materials, such as recycled uranium, where the individual was monitored for some of the nuclides
3	Individual monitoring of coworkers	Information based on in-vivo or in-vitro bioassays of coworkers
4	Workplace monitoring	General work area air samples
5	Work area (source term) data	Identification of radionuclides and quantities available for dispersal
6	Surrogate site	Data from a site where similar processes were being performed

years, are then entered into CADW. The output is a spreadsheet containing the annual organ doses, sorted by radiation type, in a format that is compatible with the required input IREP. CADW and other efficiency tools are described in more detail elsewhere in this issue (Maher et al. 2008).

COMMENTARY AND CONCLUSION

The methods for reconstructing internal doses under Part B of EEOICPA are a significant departure from current internal monitoring programs for regulatory purposes. This and other factors have presented many challenges to the NIOSH Radiation Dose Reconstruction Program. These include assessing dose that might have been undetected or unmonitored and estimating organ dose on an annual basis from the date of employment through that on which the cancer was diagnosed. The responses to these challenges have led to innovative thinking, enabling the many achievements mentioned above. These include the establishment of standard methods for determining the PC using IREP and the development of CADW that, in turn, enabled internal dose computations to be processed on a more timely basis.

These challenges have also resulted in much of the work being counterintuitive for experienced internal dosimetrists. While the committed effective dose exhibits particular relationships to bioassay data or known intake quantities, it is a rather broad brushstroke over a long period and all organs. When the doses are viewed in small time increments over individual organs and radiation types, the behavior is quite variable and presents a considerable challenge in ensuring that all aspects have been considered when trying to determine both what is realistic and what is favorable to the claimant.

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