OCAS-IG-002

INTERNAL DOSE RECONSTRUCTION IMPLEMENTATION GUIDELINE

Rev. 0
August 2002

Department of Health and Human Services (DHHS)
Centers for Disease Control and Prevention (CDC)
National Institute for Occupational Safety and Health (NIOSH)
Office of Compensation Analysis and Support (OCAS)
4676 Columbia Parkway
Mail stop R-45
Cincinnati, Ohio 45226

Preface

The purpose of this guide is to provide basic information on the methods employed in reconstructing doses under the Energy Employees Occupational Illness Compensation Program Act of 2000. The intent of this guide is to assist a qualified health physicist in determining annual organ dose from exposure to various sources of internal radiation. Because not all possible exposure scenarios can be foreseen, this guide does not provide step by step instructions for how the dose reconstruction should be performed. It is recognized there will be situations for which the methods outlined in this guide result in underestimates or overestimates of a claimants actual dose. In these cases, care must be exercised that the doses are conservative (claimant friendly) but reasonable for the claimant's exposure scenario.

Office of Compensation Analysis and Support (OCAS)	Document No. OCAS-IG-002 Effective Date: August 2002 Revision No. 0
Internal Dose Reconstruction Implementation Guideline	Page 1 of 48
Approval: J. W. Neton, Health Science Administrator Date: 8/9/02 Concurrence: L. J. Elliott, Director Date: 8/12/02	Supersedes: None

Subject Expert: David Allen

Table of Contents

Record of Issue/Revisions

Issue Authorization Date	Effective Date	Revision No.	Description
August 2002	August 2002	0	Initial Issue

1.0 INTRODUCTION

The purpose of this document is to provide guidance on the methods and approaches that can be used to reconstruct occupational radiation dose from internally deposited radionuclides in support of the Energy Employees Occupational Illness Compensation Program Act (EEOICPA, 2000). The responsibilities of NIOSH toward this goal are included in the Act and Executive Order 13179 (2000). The end result of the internal dose reconstruction will be the dose, expressed in cSv (rem), received in individual calendar years to the organ of interest along with the uncertainty associated with the dose. 42 CFR part 82 (2002) governs the process of reconstructing doses to individuals. This dose will be used as input into the NIOSH-IREP program to determine the Probability of Causation (PC) that the cancer was contracted as a result of the individual's radiation exposure from DOE sources. 42 CFR part 81 (2002) governs the process of determining the individual's Probability of Causation.

This process differs from traditional internal dosimetry in a number of aspects. Some of the more important aspects include:

1. Internal dosimetry has traditionally been concerned with radiological protection. As such, only the most exposed organs and effective whole body doses are of concern. Under EEOICPA only the dose to a specific organ is calculated. That organ is often not one of the most exposed organs. This means that the approach to identifying "worst-case" conditions will differ from that used in traditional radiological protection programs.
2. Traditionally, analytical sensitivity is a program issue which affects the design of a dosimetry program. No dose is assigned unless the results are detectable. In reconstruction for compensation, analytical sensitivity must be treated on an individual basis in order to determine the amount of intake (and thus dose) that may not have been measured. This missed dose must be quantified and applied to the specific organ dose.
3. Current radiological protection practices determine the "committed" dose received from internally deposited radionuclides. The committed dose is the dose the organ will receive over the 50-year period following an intake. Under EEOICPA, annual doses are calculated. This is necessary to allow the appropriate relative risk to be used based on the time between exposure and diagnosis.

Section 2 of this guide provides background information pertaining to the internal dosimetry models that will be utilized by NIOSH in reconstructing internal doses.

Sections 3 through 7 describe the actual dose reconstruction process itself. Sections 3, 4, and 5 pertain to the information-gathering phase from the various potential sources of information. Section 6 provides details for utilizing the efficiency process described in 42 CFR part 82. This process allows for limited research and analysis to be performed for cases in which the Probability of Causation (PC) is clearly greater than or less than 50%. This guide demonstrates how NIOSH intends to utilize that process in

reconstructing internal doses as well as the necessary coordination with the individual's external dose. Section 7 describes the detailed internal dose estimation process. Since this process is very individualized and specific to each claim being reconstructed, no set procedure was described, only additional considerations are provided.

Section 8 provides an example of an internal dose reconstruction. The example includes preliminary estimates for two different organs. The example is then expanded into a detailed dose estimate.

2.0 MODELS

The most current models and recommendations, as deemed appropriate for dose reconstruction purposes, from the International Commission on Radiological Protection (ICRP) will be used to assess dose from internally deposited radionuclides. These recommendations describe how various internally deposited radionuclides enter, transfer, and leave the body. From this information, the dose to specific organs over specific time frames can be determined.

2.1 General Models

The ICRP has recommended different biokinetic models for various radionuclides. While these models vary, they all derive from the general model depicted in Figure 1. As indicated, the primary routes of entry to the body are inhalation, ingestion, absorption and injection (wounds). The ICRP has separately published more detailed models that govern inhalation and ingestion. These models describe the rate and amount that enters the transfer compartment. For the purposes of dose reconstruction, the transfer compartment shall be considered the blood stream. It is a compartment that transfers radionuclides from the entry point to various organs and eventually out of the body.

The deposition and clearance of inhaled radioactive material is governed by the lung model. The current lung model, described in ICRP Publication 66 (ICRP 66, 1994), accounts for deposition of the inhaled radionuclide into various regions of the lung (Figure 2). The size of a particulate is the primary variable in determining the extent of deposition. Once deposited, the chemical form of the radionuclide determines the rate that the material is cleared from the various regions of the lung. Physiological clearance mechanisms are also considered. The normal physiological lung clearance function results in some of the material being swallowed, at which point, the material is treated as an ingestion intake.

The gastrointestinal (GI) tract model (Figure 3) predicts how ingested material is incorporated into the body and how a portion is eventually eliminated. This model also describes the dose to the various regions of the GI tract from an ingestion intake. Material can enter the GI tract by direct ingestion, indirectly by transfer from the respiratory tract, or by transfer from other body organs via the transfer compartment. The current GI tract model is described in ICRP Publication 30 (ICRP, 1979).

Figure 1
General Biokinetic Model

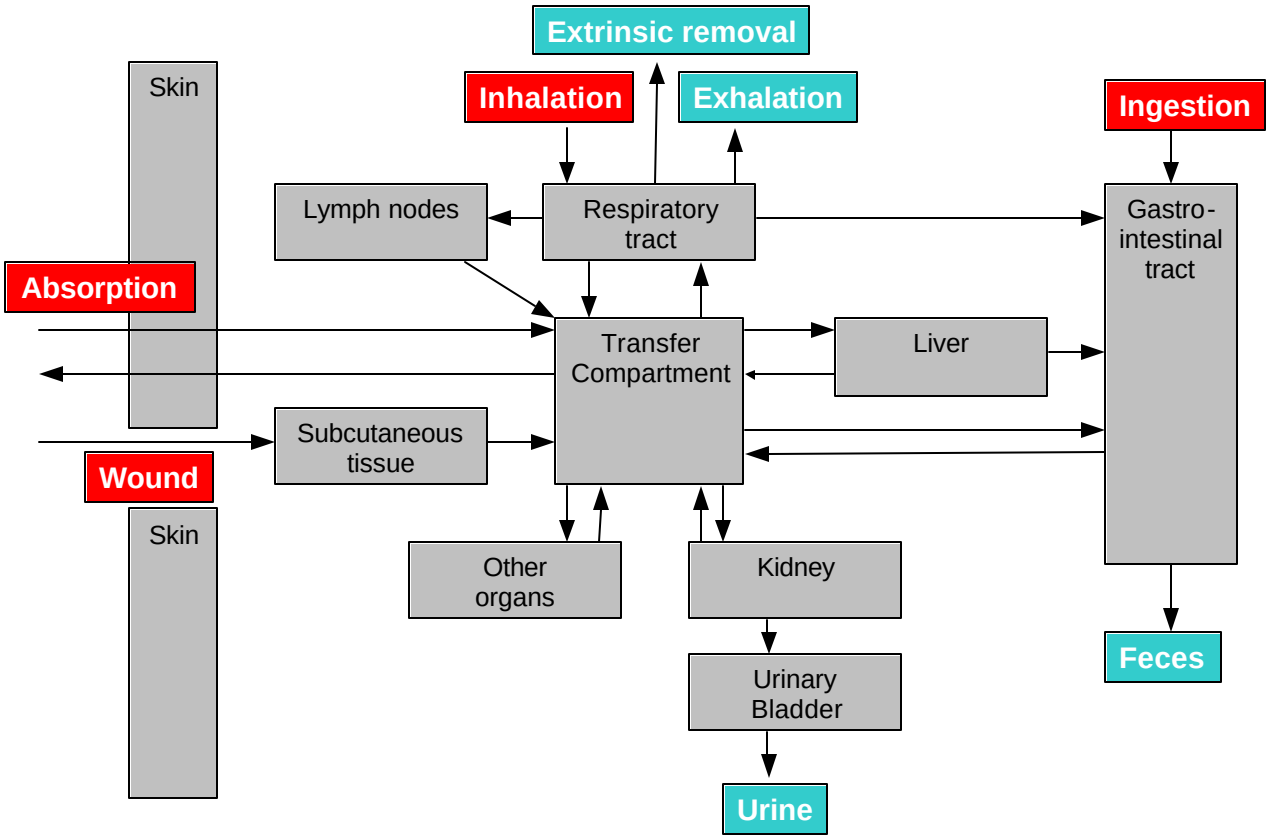
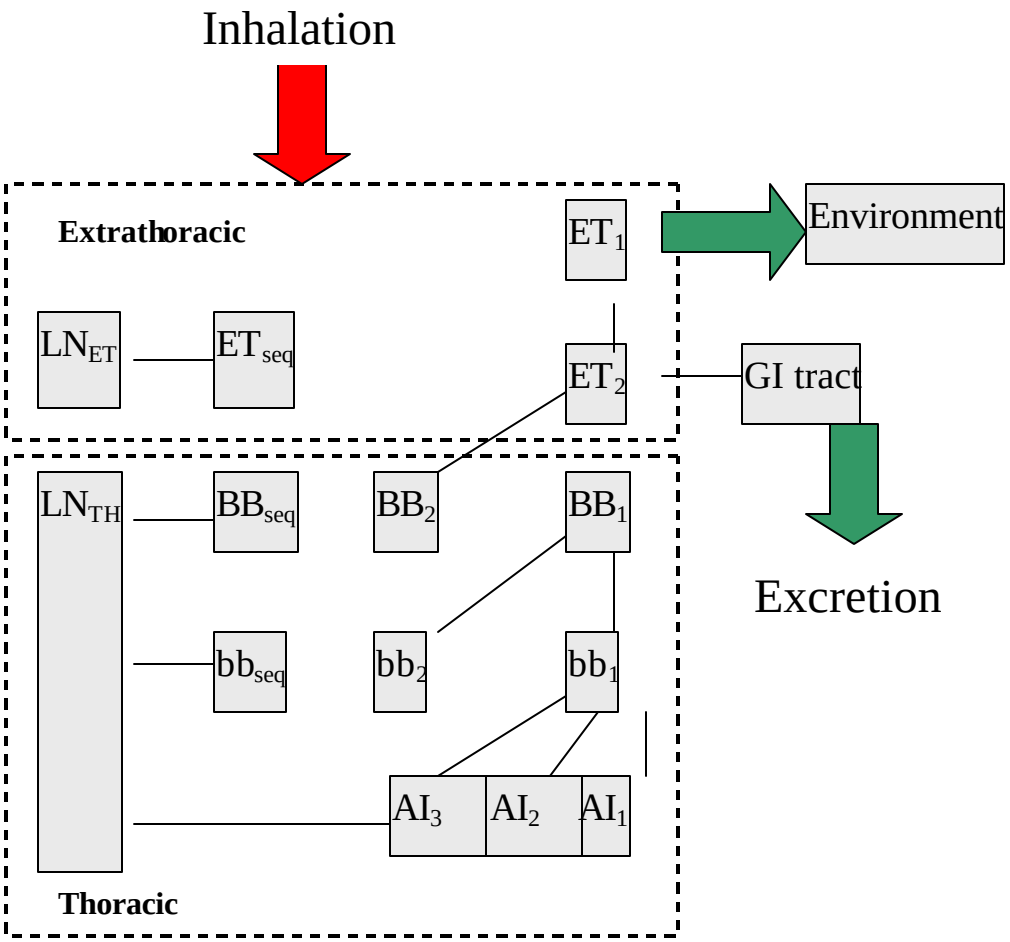
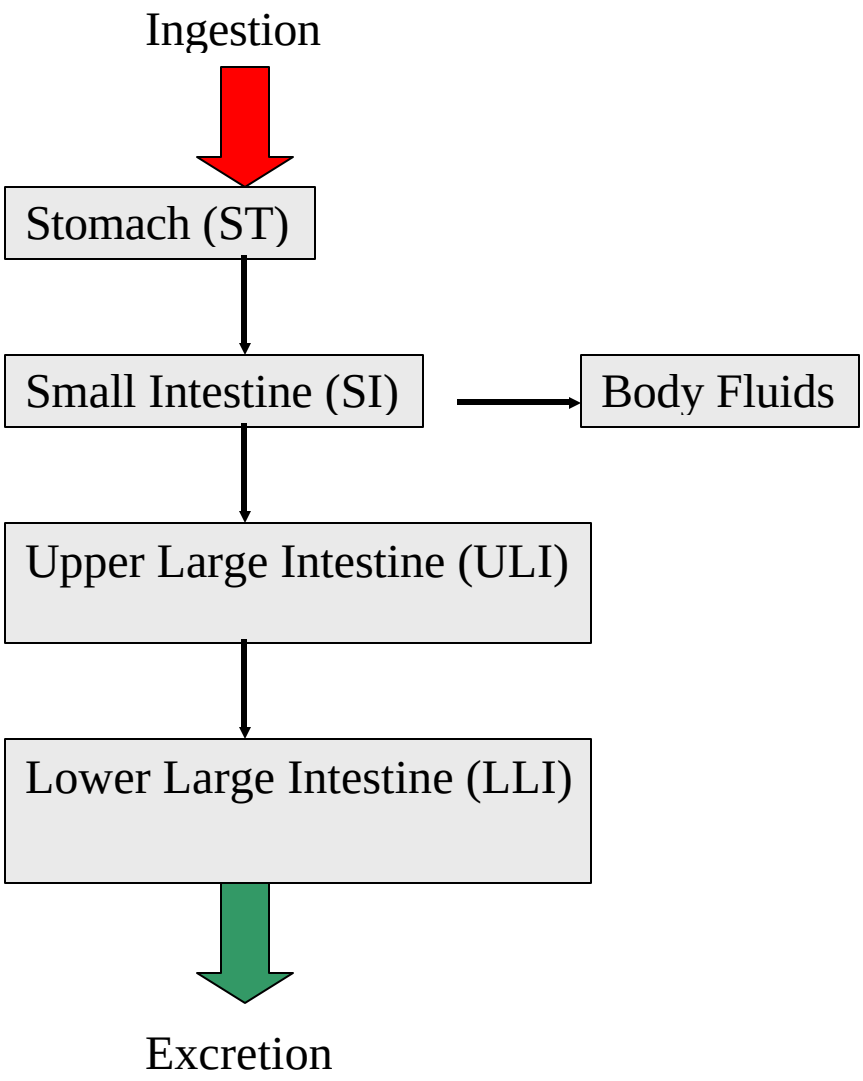


Figure 2
ICRP 66 Lung Model



AI = Alveolar-interstitial region
bb = Bronchiolar region
BB = Bronchial region
ET1 = Anterior Nasal Passage
ET2 = Posterior Nasal Passage, Pharynx, Larynx
LN_{ET} =Lymph nodes associated with the Extrathoracic region
LN_{TH} =Lymph nodes associated with the Thoracic region

Figure 3
ICRP 30 Gastrointestinal Tract



Absorption and injection are both considered to directly enter the transfer compartment. In the case of injection, it is possible that material will lodge in the subcutaneous tissues and then be cleared to the transfer compartment over time.

2.2 Specific Models

Since the publication of ICRP 30, the ICRP has issued updated biokinetic models for selected radionuclides. Table 1 indicates the model that will be used for reconstructing doses for each of these selected radionuclides. The remaining radionuclides will be based on the models contained in ICRP Publication 30 (1979).

TABLE 1. ICRP Publication Containing the Models for Selected Radionuclides

Element	**ICRP publication**		**Element**	**ICRP publication**
Tritiated water	56		I	56
H^3 or organically bound tritium	56		Ba	67
Fe	69		Ce	67
Zn	67		Pb	67
Se	69		Po	67
Nb	56		Ra	67
Sr	67		Th	69
Zr	67		U	69
Mo	67		Np	67
Ag	67		Pu	67
Sb	69		Am	67
Te	67			

This table was recreated from ICRP Publication 68, Table 5

3.0 COLLECTION OF DATA FOR THE INDIVIDUAL CASE

The first step in any internal dose reconstruction under EEOICPA is to collect the data associated with the case. The primary sources of this information are the case file sent from the Department of Labor, pertinent information on dose from the Department of Energy and the interview conducted with the claimant.

3.1 Covered Employment

The location that the individual worked is obviously important. Dates as well as location are important since processes changed through the years at a number of sites. It is also not unusual for an individual to be employed at more than one site throughout his career. Some individuals were employed by one site but worked at another. Lastly, employment location is not limited to the site or company at which the individual worked. Employment location can often be determined on a building or area basis from the claimant interview.

3.2 Incidents

Incidents are often recorded in the claimant interview. These are very important to internal dosimetry for three reasons. First, they can document the date of an intake. This is often a critical piece of information when evaluating bioassay samples. Second, an incident report often documents many of the details associated with the event including, radionuclides present, concentrations in the air, and even dosimetry data. This documentation can be very useful in determining the individual's dose for a period when the dose potential may have been very high. Third, even without an incident report, the claimant's report of an incident documents an unusual exposure condition. This is important when data is lacking and an interpolation technique is considered during that time frame.

3.3 Organ of Interest

The organ (or tissue) of interest is the organ that developed a primary cancer. This will be the organ for which the radiation dose is calculated. Documentation from the Department of Labor (DOL) will include a verification of the organs or tissues with primary cancer. Only the DOL verified organs can be used in the dose reconstruction. If inconsistencies are noted between the DOL documentation and the clamant interview, DOL must be contacted to verify any additional primary cancers.

The Department of Labor will normally classify the cancer by the ICD-9 code (International Classification of Diseases, Clinical Modification 9th revision) associated with the cancer (Department of HHS, 1991). This code is a classification system that groups related diseases and procedures for the reporting of statistical information. This code will be provided for each claimant on the Department of Labor referral summary sheet. However, some of the codes can be too specific, while others are not sufficiently specific. For example, the ET2 compartment of the lung model is used to calculate dose to the posterior nasal passages, however, the ICD codes divide this region into a number of types of cancer such as numerous surfaces and structures of the tongue, salivary gland, lips, and gums. On the other hand, ICD-9 code 159.0 describes malignant neoplasm of the intestinal tract without more detail but the ICRP GI tract model specifies 3 separate regions of the intestinal tract, each of which will have a separate dose calculation.

The first case, when the ICD codes are too specific, can be addressed by assigning the dose to the more general region for each of the ICD-9 cancers specified. For example, the organ dose calculated for the ET2 region can be used to describe the dose to a cancer identified as ICD-9 code 141.2, which is specific for cancer of "the tip and lateral border of the tongue".

The second case requires a review of the medical records submitted with the claim. If the records indicate a more exact description of the cancer location, then use this description to choose the appropriate region for which to calculate the dose. If no specific location can be determined, use the highest organ dose among the possible regions associated with the cancer. As discussed above, ICD code 159.0 is described as the "intestinal tract"

while the ICRP GI tract model calculates a dose for 3 separate regions of the intestinal tract. In this case, if the medical records describe the cancer as being in the small intestine, the dose should be calculated for the small intestine. However, if the description does not mention the exact location, the dose to all three regions should be calculated and the region with the highest dose assigned to the claimant.

3.1.1 Organs Not Included in ICRP Models

An additional problem arises when the organ that developed cancer is clearly specified but the ICRP model does not calculate a dose to that specific organ. For example, there is no ICRP model for ICD-9 code 190.5, which describes cancer of the retina. In these situations, the dose assigned to the organ should be the highest dose among the other organs that are not part of the ICRP metabolic model. The ICRP metabolic models always calculate doses to several different regions of the lung, and the GI tract, but the metabolically modeled organs vary with radionuclide. Organs that do not metabolically concentrate a radionuclide will, however, receive photon exposure because of their proximity to a source of radiation (the concentrating organs). The newer ICRP biokinetic models also consider exposure from beta and alpha radiation to these other organs by defining them as a "soft tissue" compartment and describing uptake and clearance rates for this compartment. Using these techniques, many of these other organ doses are calculated. Since these organs are all considered soft tissue, and thus are all similarly exposed, all these doses are relatively equal. This implies that choosing the highest of these doses is claimant friendly. However, it is possible for one of the organ doses to be much higher than the others due to a close proximity to a concentrating organ emitting photon radiation. In this case, the location of the cancer must be evaluated to ensure the estimate is not unrealistically high. If it is, the next highest organ dose should be used.

As a final note, the only lymph node dose specifically calculated by the ICRP models is that for the lymph nodes associated with the lungs. A number of ICD codes describe cancers of the lymph system without specifically describing the location in the body. It might appear reasonable to assign this calculated lymph node dose to the individual without further consideration. However, insoluble compounds often cause the lymph nodes associated with the lungs to receive high doses, often the highest dose of any organ. Because lymph nodes in the lung are considered to retain radioactive material almost indefinitely, the material is not transferred throughout the lymphatic system. It would be a gross exaggeration to assign this dose to lymphatic cancer associated with a lymph node located in a different part of the body. This means that lymphatic cancers not associated with lymph nodes in the lungs must be treated in the manner described above. That is, the dose to the highest exposed organ that is not described by the ICRP metabolic models should be assigned as the appropriate dose.

Table 2 summarizes the decision process discussed in this section.

TABLE 2. Correlation of ICD-9 Codes to ICRP Models

	Scenario	Resolution
1	More than one ICD code describes organs associated with only one region calculated by ICRP models.	Calculate the dose to the ICRP described region and assign that dose to the organ.
2	One ICD code describes organs associated with more than one region calculated by ICRP models.	Attempt to reconcile the location from medical records; if not possible, assign the highest dose from the appropriate ICRP regions.
3	Organ described by ICD code is not described by ICRP models.	Use the dose from the highest exposed organ not associated with the ICRP metabolic model.
4	ICD code describes a type of lymphatic cancer.	Use lymph node dose calculated from ICRP lung model only for lymph cancer associated with these lymph nodes. Otherwise, use same resolution as number 3 above.

4.0 COLLECTION OF WORK AREA DATA

Collecting work area data pertains to evaluating the material to which the individual could have been exposed. Much of this data can be obtained in the interviews conducted with the claimant or co-workers. In addition, the DOE site profile databases assembled by OCAS can also provide information useful to characterize the workplace exposure conditions. The various parameters important to internal dose estimates include:

- Routes of entry
- Radionuclide
- Solubility class
- Particle size (for inhalation exposures)

4.1 Routes of Entry

The route of entry is the path taken by the radionuclide into the individual's body. The route of entry of a radionuclide into the body has a substantial effect on the manner in which the body transfers and eliminates the deposited material. This in turn affects the dose to individual organs. All intakes of radionuclides can be generally categorized into one of four categories.

- Inhalation
- Ingestion
- Injection
- Absorption

4.1.1 Inhalation

In the workplace, inhalation is perhaps the most common route of internal exposure to radionuclides. This is an important route of entry since almost any operation with uncontained radioactive material involves some chance of the material becoming

airborne. Once the material is airborne, a worker in the vicinity is susceptible to inhaling it. ICRP 66 (Figure 2) describes the deposition and clearance processes that take place in the lung and how they are modeled. This publication also describes the method for determining the dose to various regions of the respiratory system.

Exposure from inhalation depends on a number of parameters such as particle size and solubility. Unless evidence to the contrary exists, default values from the International Commission on Radiological Protection Publication 66 (ICRP 66, 1994) will be used.

4.1.2 Ingestion

Exposure by ingestion is generally not a significant route of entry. Ingestion and clearance of insoluble compounds through the gastrointestinal tract (GI) delivers a dose for only a few days, and soluble compounds that are readily absorbed are eliminated fairly quickly. Also, loose material that could be accidentally ingested could also be inhaled. Unless the fraction that was inhaled or ingested can be determined, the most conservative (i.e. claimant favorable) approach that yields the highest dose (inhalation) should be used. For these reasons, ingestion generally does not need to be considered during a dose reconstruction unless there is some evidence of an unusual event.

While the ingestion pathway typically does not produce a significant dose compared to other pathways, it can play a useful role in the dose reconstruction process. While the fraction of material ingested often results in relatively minimal dose, it can produce bioassay data comparable to a larger inhalation dose. This implies that the erroneous assignment of a fraction of the bioassay data to ingestion can significantly bias the assigned dose. In some cases, this effect can result in doses that are several orders of magnitude low. Because of this, caution must be used before assuming any bioassay data is the result of ingestion. However, what appears to be conflicting bioassay data must be evaluated. For example, a fecal sample for Th-232 indicated a large dose when assumed to be the result of inhalation while an in vivo measurement indicates no detectable Th-232 in the lungs. If both samples are valid, and some evidence exists that indicates ingestion is possible, this dose can be assigned, at least in part, as ingestion since that is the only way to reconcile the two valid measurements.

When evaluating ingestion exposures (or potential exposures) the current IRCP recommendations are to be used. Any evidence that would produce more accurate results than the ICRP default values may be substituted, provided that documentation is available and all assumptions are clearly stated.

4.1.3 Injection

Injection exposures are a result of radioactive material that is taken up directly into the body. These types of exposures normally occur as a result of some sort of accident, such as a plutonium metal splinter being stuck in a hand. Most often, these types of exposures are isolated incidents and there is usually no need to evaluate injection exposure scenarios except in the case of a reported event. When such an event occurs, there is normally some monitoring data to support a dose reconstruction.

The uptake occurs as the material dissolves and is taken up by the bloodstream. Once the uptake is determined, the ICRP model can be used to calculate the dose by assuming the uptake goes directly to the transfer compartment. However, particular attention needs to be paid to the exposure duration. In many of these cases, a portion of the material is physically removed but not always completely eliminated. This can leave the individual with a high rate of uptake initially followed by a step decrease, but not to zero. Some of these cases result in the excision of tissue at a later date causing yet another step change in the rate of uptake. The events, including any medical procedures, should have been documented, so there is normally reasonable data for reconstructing this dose and its subsequent effect on bioassay samples. For cases involving uptake by injection, every effort should be made to obtain all incident reports, associated monitoring data and medical procedure reports.

4.1.4 Absorption

Absorption through the skin is another potential route of entry. Since absorption occurs with exposure to soluble compounds, the material is usually eliminated relatively quickly from the body. However, if the quantity of material to which the individual is exposed is large, the resultant doses can be significant.

Absorption is limited to only a few compounds. Tritium compounds (and gas) are the most likely encountered in the weapons complex, however, other exotic chemical compounds have been produced in national laboratories that could result in an absorption hazard. If the individual was working with any of these soluble compounds, absorption dose must be considered. Current ICRP recommendations should be used when determining a dose from absorption. Credit can be given to protective clothing, however good documentation must exist to evaluate its effectiveness.

4.2 Radionuclide Present in the Work Area

Obviously knowledge of the radionuclides that are present in the work area is important to internal dosimetry. The primary radionuclides are generally well known for most areas, however, consideration must be given to other aspects of the particular facility.

4.2.1 Primary Radionuclide

While the primary radionuclides are generally well known, this knowledge is based on exposure to the most exposed organs or the effective dose to the whole body. Under EEOICPA, organ doses must be calculated to organs that often are not the most exposed. This may change the evaluation as to which radionuclides present are the primary source of exposure for a particular case.

4.2.2 Radionuclide Impurities

Many materials handled at weapons complex facilities have additional radionuclide impurities associated with them. While these normally account for minimal doses when compared to the material itself, some chemical processes can concentrate these impurities. Familiarization with these processes is an important part of gathering work area data.

4.2.3 Radionuclide Progeny

Some of the radionuclides encountered in DOE facilities decay to progeny which have long residence times in the body. Because of this, it is important to account for the build-up of these progeny in the body. While this is performed under the current ICRP recommendations, this only accounts for the progeny that grow in after an intake of parent radionuclide. When inhaled as a mixture, the intake of progeny must be accounted for separately. Many internal dosimetry programs concentrate on the major radionuclides only and the progeny are considered to be negligible or are assumed to be related to the primary intake by some factor. These evaluations while useful, cannot be universally accepted since most of these programs were designed under either the ICRP Publication 30 (ICRP, 1979) or ICRP 2 (ICRP, 1959) dosimetry models. Progeny deemed as having negligible contribution to internal dose in 1955 may not be so evaluated under current models. Also, historical air sample data are largely based on gross alpha or beta measurements and will often assign all of the activity to the most dosimetrically significant radionuclide. Prior to using air sample data, any historical program assumptions should be reviewed to ensure they consider decay series progeny.

4.2.4 Radon

Occupational exposure to radon and its progeny presents a number of unique issues. A discussion of these issues and their effect on assessing radon exposure is included in section 7.4.

4.3 Solubility Class

Solubility of a given radionuclide is one of the most important parameters in determining the internal radiation dose. This parameter is highly dependent on the chemical form of the material. The current ICRP recommendations for these solubility classes will be used as the default values. Some solubility studies have been done by various facilities that may provide more process specific data. Where available, these studies should be evaluated and, if appropriate in the context of the dose reconstruction, applied to the dose reconstruction.

The most accurate means of evaluating the solubility class is by examining multiple bioassay samples after an intake. This has the potential of providing an accurate determination of the solubility for the particular material. However, inhaled material often exhibits more than one solubility class. A plot of multiple bioassay samples can produce a curve that appears to show a soluble compound when in fact it is only the soluble portion of the inhaled material that is actually being followed. The slowly changing insoluble portion may not be noticeable. Therefore, consideration must be given to the potential presence of more insoluble compounds whenever bioassay samples are used to determine solubility. Figure 4 demonstrates this effect. As can be seen, a mixture of solubility class S and M plutonium produces a clearance curve with virtually the same slope as that of pure class M material.

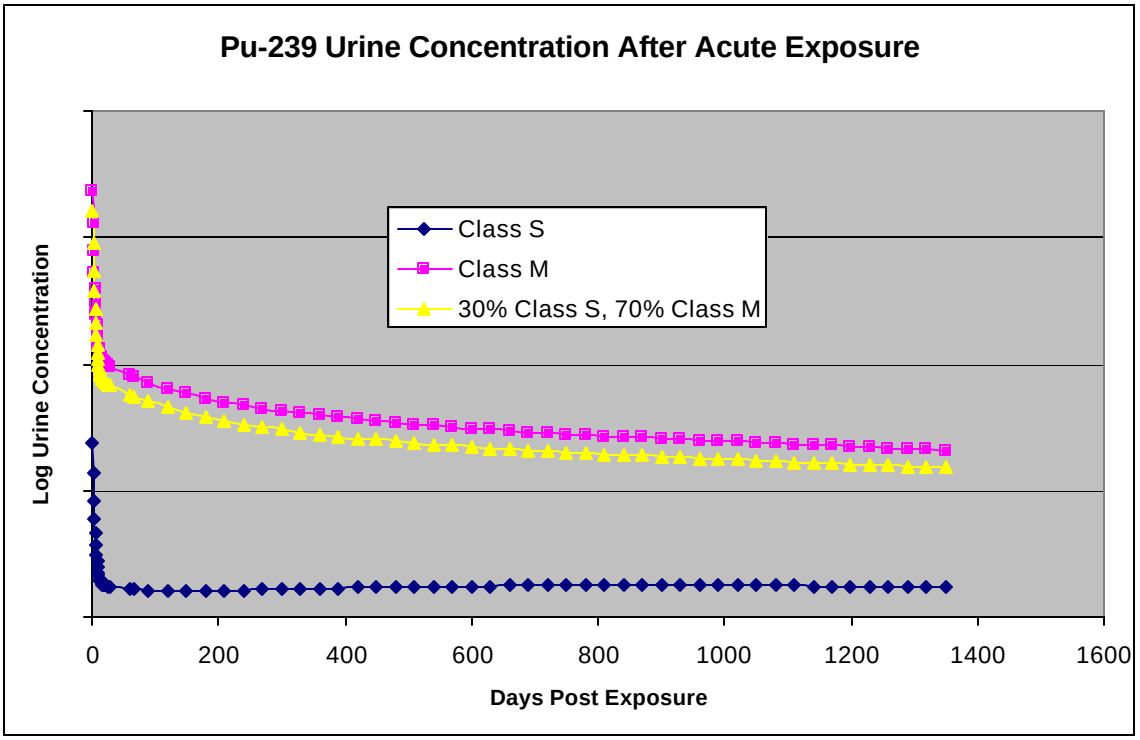


Figure 4. Example Urine Concentrations for Mixtures of Solubility Classes

4.4 Particle Size

Particle size is an important parameter in determining internal dose from inhalation of radionuclides. Particle size is typically material and process related. A number of facilities have measured particle sizes for various processes in the past. Some of these measurements may be transferable to similar processes at other facilities utilizing similar material. In the absence of any measurements or studies, default values from the International Commission on Radiological Protection Publication 66 (ICRP 66) will be used.

5.0 COLLECTION OF THE INDIVIDUAL DOSIMETRY DATA

Once the exposure pathways and workplace characteristics are evaluated, the individual's dosimetry record should be reviewed. The internal dosimetry information is normally in the form of urinalysis and in vivo counts, however, other types of information may also be present, including personal air sample results and incident reports. In general, the individual's dosimetry data can be categorized into three categories: bioassay data, work-place monitoring data, and source term data. The last two types of information are typically not in the individual's dosimetry record and may require the dose reconstructionist to revisit the work area data.

5.1 Bioassay Data

Bioassay measurements are generally the most reliable data available for assessing internal exposures. This is the result of the fact that all other methods must estimate the actual intake of radionuclides based on an assessment of the environmental conditions. To insure accuracy, however, intake assessments based on bioassay measurements must also consider some of the environmental exposure conditions (particle size, solubility, etc.) but not all (airborne concentration, breathing rate). As such, the dose reconstruction process will rely on these data when available. These data must, however, be evaluated to ensure that they are valid.

Bioassay data is applicable to all routes of entry and almost all radionuclides. For the purposes of this implementation guide, bioassay is considered to be any means of measuring the actual intake or uptake of radionuclides by an individual. These measures include but are not limited to:

- Urinalysis;
- fecal samples;
- In Vivo measurements; and,
- breath radon and or thoron results.

From these measurements, the appropriate biokinetic model is used to determine the actual intake or uptake based on the amount and rate of elimination. Using multiple points, the rate of elimination can also be used to help determine when the intake occurred, whether the intake was acute or chronic and, possibly, the solubility of the material.

One of the most important considerations when evaluating bioassay data is the extent to which the sampling scheme would detect the radionuclide of concern. While bioassay programs are typically designed to detect a particular radionuclide, exposure to mixtures of radionuclides is not uncommon. The bioassay program must be evaluated for its ability to detect each radionuclide being considered.

When a dose reconstruction is performed using this data, the detection limits and uncertainty of the analyses will be used. If this information is available in the dosimetry record, it is important to note that. However, this will normally be information that will have to be obtained from the work area data in the form of a site dosimetry technical manual or other documentation.

Assuming an adequate bioassay program exists, the next step is the evaluation of any positive results. Positive results in this context are results indicating the presence of the radionuclide above the detection limit. If an analysis of these positive results establishes a dose that results in a probability of causation of $\geq 50\%$, there is no reason to further refine the dose estimate.

The one caveat to this is that the positive results must be valid. In any type of analysis there is a potential for a false positive result. False positive bioassay samples can be due

to a number of factors, including contamination of the sample from contamination on the individual's hands or clothing. An attempt should be made to verify the reasonableness of any unusually high bioassay results. This does not have to be a quantitative verification, but simply a check to determine if the result was anomalous. As one example, a very high air sample or a report of some sort of unusual release during the time frame that yielded the positive bioassay result would verify the data. In the absence of a valid reason for a very high result, an evaluation of co-worker data, air sample data, or any other reliable data should be used to determine if the sample is anomalous or reasonably realistic. Follow-up samples can be very useful in this determination. If a very high result is followed the next day by a non-detectable result, one of the samples must be considered suspicious and anomalous. It is important to keep in mind that the emphasis of this evaluation is not to produce small refinements of the estimated dose but to identify gross errors in sampling, analysis, or transcription of data.

5.2 Workplace Monitoring Data

If bioassay data is not adequate to evaluate the individual's internal dose, workplace monitoring data can be used. Workplace monitoring data consists of any type of sample that assesses the conditions in the workplace. Some examples of this type of data include:

- Breathing zone air samples
- General area air samples
- Surface contamination surveys

While workplace monitoring may be useful in the evaluation of ingestion or absorption cases, it is primarily applicable to the inhalation route of entry.

When used appropriately, workplace monitoring data is a viable alternative when bioassay data is not available. This type of data tends to be less reliable than bioassay, since it is an indirect measurement of an individual's uptake. However, with due care, this data can be a substitute for bioassay data (Ritter et al, 1984).

The general approach to using workplace monitoring data is to determine as closely as possible the airborne radioactivity concentration in the individual's breathing zone. This concentration, along with the associated exposure time, particle size, solubility, respirator use, etc. can be used to estimate the individual's intake of radionuclides. The best data for determining airborne concentrations are from job specific air samples. Since the individual's breathing zone is the location of interest, lapel type breathing zone air samples are preferred. In the absence of breathing zone samples, general area air samples can be used, but consideration must be given to any factors that could create a difference between general area and breathing zone concentrations (NRC, 1992). Some of the factors that should be considered are: the amount and direction of ventilation, the location of the airborne sources in relation to the individual and the air sampler, and whether the individual is mobile or stationary in the course of the work.

In the absence of air sample measurements, contamination surveys can provide a quantitative indication of the amount of dispersible material in the work area available to create airborne contamination. Consideration should be given to types of work activities, the type of material, and ventilation or other forces that could cause the material to be suspended in air. Resuspension factors can be used, provided enough information is available to properly classify the material and conditions. Some references for resuspension factors are available (NRC, 1993) but the basis of these references must be reviewed to ensure the factors apply to the particular situation.

Once the radionuclide concentration in the breathing zone is established, the individual's intake and deposition of radionuclides must be estimated. When no other information is available, the ICRP 66 defaults for a "reference worker" will be used for deposition fractions, particle size, etc. It will also be necessary to estimate the individual's exposure time. For a normal workday, the average airborne concentrations and average worker exposure time should be acceptable with minimal error. For unusual or abnormal conditions that created much higher than normal airborne concentrations, a more rigorous examination of the exposure time should be conducted. Typical exposure time and abnormal events can often be obtained from the claimant interview.

An additional factor that must be considered is respiratory protection. Measured and documented fit factors should not be used since they are not typically indicative of the protection afforded in the work environment. Prior to giving credit for respiratory protection, the respirator program should be evaluated to determine its protection effectiveness. This is not an audit of the program but rather an evaluation to determine if quantitative fit testing was performed and whether it is likely a respirator was worn during the times that credit is given for the protection. This evaluation may rely on any source of information, including a comparison of airborne results to bioassay, interviews, or written documentation. It should not rely solely on a written administrative requirement unless there is some evidence of the enforcement or normal compliance with that requirement.

5.3 Source Term Evaluation

Without bioassay or air sample data, the last resort is to attempt a determination of the airborne concentrations using source term evaluations. Besides the factors previously mentioned, the key ingredients of this evaluation are the dispersible quantity of material available, and the fraction of this quantity that actually produces airborne contamination in the individual's breathing zone. The distinction between dispersible and non-dispersible material is important. For example, it would not be realistic to assume the entire mass of a large piece of uranium metal produces airborne contamination. It would however, be realistic to assume the bare piece of metal might corrode and produce some oxides that could create airborne contamination.

If only limited information is known about the material, published values for resuspension factors can be used, however, an effort must be made to ensure the most appropriate factor is chosen for the given situation. If no information is available about

the material type, the entire quantity can be considered to be dispersible. If this assumption creates an airborne concentration estimate that is unreasonable or inconsistent with other information (anecdotal, photographic, etc.) then the assumption should not be used.

6.0 PRELIMINARY DOSE ESTIMATES

A complicated dose reconstruction could require days, weeks, or even months to complete, even after all the data is available. While accuracy is an important parameter for dose reconstruction, the dose reconstruction analyst must keep in mind that the ultimate purpose is to determine whether the covered exposure to radiation is "at least as likely as not" to have caused a particular cancer. This implies that the dose reconstruction only be sufficiently refined to ensure that the decision for compensibility is correct. This allows the use of some very conservative assumptions (either high or low) for initial estimates which require further refinement only if the likelihood of compensibility is not clear. For example, if the upper limit of a possible exposure scenario is unrealistically high but still results in a low probability of causation, no refinement is necessary. Likewise, if only the recorded bioassay results (once validated and without accounting for missed dose) are sufficiently large to result in the individual having a high probability of causation, no refinement to the dose estimate is necessary.

The dose reconstruction efficiency process is described in 42 CFR part 82 paragraph 82.10 (k). This process allows for the degree of research and analysis to be limited to that which is necessary to determine if the radiation dose will reach a compensable level (i.e. a dose producing a probability of causation of 50% or greater at the 99% credibility limit). The first step in the efficiency process is to determine whether the radiation dose is clearly high or low when compared to this criterion. This criterion is not simply one number for a dose of radiation or even one number based on the type of cancer. The exposure dates, the age of the individual, the types of radiation as well as other factors affect this decision. For the purposes of an initial estimate, the bioassay data, the type of cancer and the age of the individual should be sufficient to determine the appropriate starting point. If the covered cancer is not in a metabolic organ for the particular radionuclide (e.g. spleen cancer for a plutonium intake), the internal radiation dose to that organ is likely low. Also, if the analyses do not indicate any detectable results, the radiation dose is likely (but not necessarily) low. This evaluation should be performed for each radionuclide to which the individual is potentially exposed. If the organ of interest is a metabolic organ for any of the alpha emitting radionuclides in which there was detectable bioassay result, assume the radiation dose is high.

Once this decision is made, the dose reconstructionist can follow the general steps outlined below to perform a preliminary internal dose estimate. The steps below are described to evaluate inhalations of low solubility compounds of actinides with bioassay data available for the radionuclide through the majority of the individual's employment. While this may appear to be a very limited situation, it should be the largest category of the many possible categories that will be encountered. Also, the dose reconstruction for many of the remaining categories may be based largely on this process with only minor

modifications. The remaining situations should follow a similar philosophy in arriving at a preliminary dose estimate.

6.1 Preliminary Dose Estimate – Low Dose Potential

If the analyst has determined that the radiation dose to the organ of concern is likely low, the next step is to perform a preliminary dose estimate that reasonably and realistically maximizes the dose to the organ. This preliminary estimate is performed as follows:

1. Choose a radionuclide from the radionuclides for which the individual was monitored.
2. Recalculate the bioassay values using the higher of the MDA or the actual result plus two standard deviations.
3. Choose a solubility class from the credible classes given the radionuclide and the individual's work area.
4. Assume an acute inhalation on the first day of employment and determine the highest intake that will produce a predicted bioassay value that will equal at least one of the recalculated bioassay values from step 2 above.
5. Assume a constant chronic exposure throughout the individual's entire employment and determine the highest intake that will produce a predicted bioassay value that will equal at least one of the recalculated bioassay values from step 2 above.
6. Repeat steps 4 and 5 for all potential solubility classes.
7. Determine the scenario that produces the highest 50 year committed dose to that organ. (If the time between exposure and diagnosis is <10 years, use the first and last year doses instead)
8. Using the scenario that produces the highest dose, determine the annual doses to the organ of concern from this scenario. This will be used in a preliminary probability of causation (PC) analysis later.
9. Choose the next radionuclide for which the individual was monitored and repeat steps 2 through 8.
10. Once all radionuclides have been accounted for, group them by major radiation type emitted (i.e. alpha, beta, gamma, etc.)
11. Sum the annual doses by radiation type from each isotope.
12. Use the total annual dose input along with the external annual dose input to run the NIOSH-IREP program. Use the "constant" distribution since the dose determined is the upper bound.
13. If the PC is below 50%, no further refinements to the internal dose estimate is required. However, if the external dose estimate is preliminary, the entire PC calculation must be performed again once the external values are finalized.
14. If the PC is greater than or equal to 50%, refinements to the estimate are required. If the PC is much higher than 50%, perform a preliminary estimate assuming a high dose potential (see section 6.2). Sources of refinement that should be explored include:

A. Running the PC calculation for internal and external separately. If one mode produces a much higher PC, concentrate efforts on refining that mode.
B. Adjusting the intake scenario to a more reasonable representation of the intakes. Dates, chronic vs. acute, and amounts can all be adjusted provided all but one of the predicted bioassay values are above the values calculated in step 2. The potential of several scenarios capable of giving the same results must be explored. If more than one is found, the scenario that produces the highest dose must still be used.
C. If one sample is driving the analysis and it appears to be anomalous, remove the sample from the data set and recalculate the dose and the subsequent PC. If that greatly changes the PC, evaluate the potential of permanently removing that sample as anomalous.
D. Evaluate the interrelationship of the various samples to determine if the individual results used are possibly real. For example, if gross alpha analysis is used for one radionuclide and a specific chemical extraction is used for a different alpha emitting radionuclide, activity from the second radionuclide will be accounted for in both analyses. The amount of the gross alpha analysis attributed to the second radionuclide should be subtracted from the alpha analysis thus lowering the highest potential concentration of the first radionuclide.

15. At some point it becomes counterproductive to continue refining a preliminary estimate and a detailed dose reconstruction must be undertaken. Before reaching that decision, both a high and low estimate of the individual's internal dose should have been performed.

6.2 Preliminary Dose Estimate – High Dose Potential

1. If sample data exists for the individual for more than one radionuclide, use professional judgment to choose a radionuclide to start. This judgment can be based on the radionuclide that will deliver the most dose per unit intake or the one with the highest bioassay results.
2. If there is a clear increase in activity at some date, use that date for an inhalation.
3. Choose a solubility class from the credible classes given the radionuclide and the individual's work area.
4. Assume an acute inhalation on the first day of employment and determine the highest intake that will not exceed any of the measured bioassay values. Do not use the MDA values or subtract the uncertainty from the measured values.
5. If the acute scenario does not produce a realistic curve, attempt to find a chronic scenario that more reasonably depicts the measured data. Insure the predicted bioassay results do not exceed any measured values.
6. Repeat steps 4 and 5 for all potential solubility classes.

7. Determine the scenario that produces the lowest 50 year committed dose to that organ. (If the time between exposure and diagnosis is <10 years, use the first and last year doses instead)
8. Using the scenario that produces the lowest dose, determine the annual doses to the organ of concern from this scenario.
9. Use the annual dose input along with the completed external annual dose input (if available) to run the NIOSH-IREP program. Use the "constant" distribution since the dose determined is a lower bound.
10. If the PC is >50%, no further refinements to the internal dose estimate are required. It is also not necessary to perform an external dose estimate. Additional dose from any source will only cause a higher PC.
11. If the PC is <50%, use professional judgment to refine the estimate. If the PC is greatly below 50%, perform a preliminary dose estimate assuming a low dose potential. Sources of refinement that should be explored include:

    A. Use the most credible solubility class instead of the one that produces the lowest dose.
    B. Use a more credible exposure scenario rather than the one that produces the lowest dose.
    C. Repeat the process for other radionuclides for which the individual was monitored and add this dose to that already calculated.
    D. If bioassay results that are below MDA are driving the intakes scenario down a great deal, use the lower of the MDA value or the result plus 2 standard deviations and reevaluate the scenario.
    E. If one bioassay sample is driving the analysis and appears to be anomalous, remove it from the data set. If that greatly changes the PC, evaluate the potential of permanently removing that sample as anomalous.
    F. If not already done, add the external dose to the probability of causation analysis.
    G. If refinements fail to result in a PC >50%, run the PC calculation separately for internal and external exposure. If one is clearly higher, attempt to refine that estimate first.

12. At some point it becomes counterproductive to continue refining a preliminary estimate and a detailed dose reconstruction must be undertaken. Before reaching that decision, both a high and low estimate of the individual's internal dose should have been performed.

6.3 Modifications to the Preliminary Dose Estimate Process

As stated in the beginning of this section, the process outlined above applies to only a limited number of cases. However, most cases can be evaluated using this approach with only minor modifications. Consider the situation where the individual was not monitored for all the potential radionuclides to which they were exposed. If monitored radionuclides cause the PC calculation to exceed 50% then the estimation process works. On the other hand, if the monitored radionuclides cause a very low PC the estimation

process needs to be modified. In this situation, the process can be followed for all the radionuclides for which the individual was monitored. Then an upper bound must be determined for the unmonitored radionuclide. This may be done using as a fraction of the monitored radionuclides or by other means available to the analyst. The important point to remember is that if the potential of this exposure is known, the information used to establish that potential can normally be used to determine some bounds for the exposure.

Uncertainty can be added to a preliminary estimate that produces a PC that is barely under or over the PC of 50%. For example, assume an individual has a relatively high dose received from intakes of radionuclides. The preliminary dose estimate based on the high dose potential produces a PC of 48% after several refinements. Since this is a lower bound, it may be possible to perform two new estimates to determine an upper bound and a most likely dose. The upper bound relying on the results plus 2 standard deviations and the most likely based on only half the samples being above the predicted results instead of all the samples. These three points can then be used to establish a triangular distribution. This causes the original estimate to be the lower bound of a distribution instead of a point estimate and should raise the PC (possibly beyond 50%) by more accurately describing the individual's potential dose distribution.

Ideally, internal dosimetry program data will exist that encompasses the individual's entire exposure history. However, it is very likely that gaps in the information will be encountered. When this occurs, there are several options. The first option is to interpolate between existing bioassay data. For this option to be effective, the period of missing data must be "bounded" by periods of valid data that are representative of the missing period. For example, interpolation would be most appropriate for a period of missing data in which data exists before and after the period and all three time frames represent the same type of work, with the same type of material, in the same location. This interpolation applies to bioassay, air sampling, and any other type of data that is used. Options for interpolating data points has been previously published (Crawford-Brown et. al., 1989)

Another method for filling in data gaps is the use of co-worker data. If the individual had co-workers in the same area, any data from these co-workers could theoretically be used to estimate the individual's dose. Since this information is not actually from the particular individual, it must be judged for applicability. When possible, it is best to use data from several co-workers. If data from several co-workers are available but of varying applicability, appropriate weighting factors can be assigned to each data point so that the most relevant data is weighted more heavily. For example, if a co-worker was performing the same job in an area with airborne concentrations twice as high as the individual in question, a weighting factor of 0.5 could be assigned to the co-worker's dose. This allows for several doses to be used to determine a more representative value, even when few co-workers closely matched the individual's exposure. If weighting factors are used, the basis for these factors must be documented.

Many other modifications are possible in order to estimate the internal dose for a particular case. The process used, even if the preliminary process is followed, must be

documented in the dose reconstruction report. Unless a detailed dose reconstruction is performed, the dose utilized must be clearly high (for PC<50%) or low (for PC>50%).

6.4 The Probability of Causation Calculation

The primary product of any dose reconstruction is the input for the probability of causation calculation. This calculation is performed by the computer program NIOSH-IREP which utilizes Monte Carlo techniques to perform its calculations. It is important to realize that the values produced by this program may vary slightly due to the nature of a Monte Carlo calculation. Varying the numbers of trials or the input "seed" value can cause the calculated PC to change by several percentage points. Therefore, while the process outlined above utilizes a PC of 50% as a decision level, care must be utilized when performing these calculations. A PC between 40% and 60% should be re-evaluated with various numbers of trials and seed values before any decisions are made.

6.5 Refining Preliminary Estimates

The process of performing a preliminary dose estimate includes steps for refining the estimate. The process steps list a number of potential refinements. When a refinement is necessary, a more rigorous approach to the dose reconstruction must be adopted. Since initial estimates often rely on very conservative assumptions, the refinement process attempts to find more valid values for these parameters. This may require a search for additional data. Like the initial estimates, the refinement need not attempt to be 100% accurate, only more accurate than the original estimate. As previously stated, the degree of accuracy required is that sufficient to render an accurate decision for compensation.

A useful technique for refining dose reconstructions is to compare estimates from different methods. For example, gaps in an individual's bioassay data could be estimated by interpolation, by co-worker data or by air sample data. By evaluating each dataset, there may be only a small band of possible answers that fits all three methods. This evaluation could then lead to a calculation of the average of the results with an uncertainty distribution. Since uncertainty is an input into the NIOSH-IREP program, this will be reflected in the probability of causation outcome. This method may also help to recognize anomalies. If there is no answer that fits all three methods, at least one of the methods must be in error. Finding the erroneous assumption or sample could change the assumptions elsewhere in the dose reconstruction and eventually produce a more accurate result.

Comparing estimates from different methods does not have to be limited to periods when gaps exist in the data. In vivo counts or air sample measurements could be used to place an upper or lower limit on an intake indicated by urinalysis. As discussed earlier, multiple urinalyses could be used to evaluate the elimination of an acute uptake resulting in more accurate solubility parameters. Also, numerous air sample data in a particular area could be used to determine a pattern of airborne activity. This pattern could indicate an overestimate or underestimate of the intake calculated by other means.

Other refinements include obtaining studies or data that more accurately define some of the parameters used, such as solubility class or particle size. This is especially possible when some default parameters are used. Particle size studies, solubility studies, as well as ventilation tests, are all sources of potential refinements.

7.0 DETAILED DOSE ESTIMATES

The preliminary dose estimates described above should minimize the analysis and research necessary to complete many dose reconstructions in accordance with 42 CFR part 82. This allows for a more efficient process that will finalize dose reconstructions in a more timely manner. However, this efficiency process only works in cases where the decision for compensation can be shown to be clear. Some cases will likely be too close to determine a clear decision for compensation without a detailed dose estimate.

The dose reconstructionist will use professional judgment in determining the point at which preliminary dose estimates are counterproductive and a detailed dose reconstruction must be undertaken. Information obtained and calculations performed during the preliminary estimates may be used to the fullest practical extent during the detailed dose reconstruction process. However, the desire for efficiency should not interfere with the need for accuracy of the detailed dose reconstruction.

In performing a detailed dose reconstruction, the dose reconstructionist is attempting to find the best estimate of the individual's dose rather than find the upper or lower bound of the dose. Because of this, the detailed dose reconstruction requires the uncertainty in the analysis to be quantified. It is still important, however, to keep in mind the purpose of the dose reconstruction. Efforts should still be directed to the parameters that make the largest difference in the individual's dose. Worst-case (claimant favorable) assumptions can still be used for parameters that produce little change in the estimated dose.

Although individual cases vary too much for an all inclusive step-by-step instruction to be developed, some additional considerations when performing a detailed dose reconstruction are included below.

7.1 Estimate of Intake Date

The time of intake is an important parameter in assessing bioassay data. Based on one positive sample, the intake could have occurred anytime since the last non-detectable sample. The difference in a calculated intake, based on assuming the intake occurred at either the beginning or the end of this period, can vary by orders of magnitude. Without any additional information, the standard approach should be to assume the exposure occurred midway between the two sample dates. The rationale and logic behind this assumption is discussed below.

Since bioassay samples are correlated, a large intake detected in a bioassay sample will likely continue to be detected in the next several samples. Eventually, the pattern of subsequent sample results provides a means of estimating the intake date, and thus the

quantity. Also, multiple small intakes (such as a chronic exposure) will eventually reach a level where subsequent samples are detectable. Once this level is reached, the total intake from the multiple intakes can be estimated fairly accurately. The assumed intake date of each individual intake may be in error but the overall intake estimate will be accurate. This implies the midpoint of sampling dates will only be used in the case of a few small intakes. In that case, the overall dose is likely to be small. The exception to this are radionuclides that quickly clear from the body. In these cases, the residence time in the body is so short that even a large intake does not produce a significant dose. This implies the largest errors occur with the smallest doses and therefore the midpoint estimate between two dates should not significantly affect the decision for compensation.

Even without the correlation of multiple samples, the midpoint estimate will likely yield reasonable results. While it is possible that an individual received an intake just after leaving one routine sample, it is somewhat unlikely. The possibility that such an event occurs sequentially multiple times is even more unlikely. In fact, if a more rigorous Monte-Carlo calculation is performed, assuming an equal chance of an intake on each day between samples, the mean value is the midpoint. Using Monte-Carlo calculation in this approach it is a valuable tool that can be used to determine the midpoint and to estimate the uncertainty associated with the intake.

It is also important to keep in mind that the midpoint is only used in situations when there is no other information. Incident reports or air sample results, as well as other sources of information, can be very useful in determining the date the intake occurred.

7.2 Uncertainty

The uncertainty of the internal dose calculations has a number of components that can be difficult to quantify. However, the largest uncertainty associated with internal dose calculations will predominately be associated with determining the intake. This implies that the method used to assess the uncertainty depends on the method used to assess the intake. For non-correlated techniques (air sample measurements, injections of known quantities, etc.) the uncertainty of a single sample is usually understood and readily calculated. Combinations of results from these methods (such as averaging air sample results) are readily dealt with using standard propagation of errors techniques. The standard equation for this is:

$$s_f^2 = \sum \left(\frac{\partial f}{\partial a} \right)^2 s_a^2$$

Where σ is the standard deviation of the function (f) or the independent variable α. The summation (Σ) must be performed for all independent variables. It is important to note that this equation is only applicable if all the variables are independent. When variables are not independent, correlation coefficients must be applied.

Bioassay samples are correlated by their very nature. The correlation coefficients depend on the length of time between the individual intakes as well as a number of other factors.

Since the date of the intake can be somewhat arbitrary, this coefficient can be very difficult to calculate.

However, the correlation among samples does provide a more accurate result for total intake. Consider the following example. An individual is working with Pu-239 with a solubility class of S. He unknowingly receives an intake of 1 µCi of Pu-239 on January 25th. He then submits a routine urine sample on January 31 which contains 0.37 pCi of Pu-239. With no other information, the analyst assumes the midpoint of January 15th for the date of intake (based on the date of the last sample) and thus calculates an overestimate of the intake of 2.02 µCi. On February 24th the individual is involved in an incident in which he inhaled Pu-239. A bioassay sample submitted on February 28th contains 0.42 pCi. Still relying on the midpoint estimate, the analyst would calculate that 0.34 pCi of that sample is attributed to the first intake. The remaining 0.08 pCi is attributed to the new intake on February 24th which predicts a 0.135 µCi intake on February 24th. However, in reality, the first intake was 1.0 µCi on January 25th so only 0.171 pCi of Pu would be left in the urine on February 28th. Thus, 0.249 pCi (0.42 pCi – 0.171 pCi) of the February 28th sample is due to the February 24th intake. Therefore, the February 24th intake was actually 0.42 µCi.

In this scenario, the real intake was 1.42 µCi (1.0 µCi + 0.42 µCi) while the estimated intake was 2.155 µCi (2.02 µCi + 0.135 µCi). The initial intake was overestimated by 1.02 µCi (102%) but after the 2nd intake, the total overestimate dropped to 0.735 µCi (52%). As subsequent intakes are evaluated, the estimate of the total intake becomes increasingly accurate. Since the most accurate estimate in this analysis will be the total intake, the best uncertainty value to use is the relative error associated with the total intake.

To calculate the relative error it is first necessary to determine the error associated with each intake. This can be done by applying the relative error of the bioassay sample on which the intake is based. Next, propagate the errors of all the intakes to determine the absolute error associated with the total intake. Finally, divide this error by the total intake to obtain the relative error. This relative error will be the error applied to the calculated doses.

It should be noted that this procedure does not accurately reflect the uncertainty of the initial intake. This is because it does not account for the accuracy of the date chosen or the correlation between samples. However, it can be considered an accurate representation of that component of the total intake. Just as the dates can be arbitrary, the size of this component can be considered arbitrary. The important number is the uncertainty of the total intake. This number is found by propagating the error of the individual components of the total intake.

Therefore, if an individual receives only one intake, the error associated with the total intake will be equal to the error associated with that one intake. Conversely, if an individual receives multiple intakes, the relative error associated with the total intake will be less than any one of the individual relative errors.

This approach has the advantage of easily allowing the incorporation of other methods of intake assessment into a single uncertainty analysis. For example, if an individual received several intakes that were evaluated using either bioassay or air monitoring data, the error of each intake can first be determined separately, then all the intakes can be summed and the errors propagated as discussed above.

It is important to note that, while the uncertainty of an internal dose estimate can be dominated by the uncertainty in determining the intake, this is not always the case. The intakes for individuals that submit many detectable bioassay samples may have their total intake calculated fairly accurately. However, this intake is based on a particular biokinetic model. Any inaccuracies or biases produced by this model must be considered.

Uncertainties associated with the biokinetic models are difficult to assess. While some attempts have been made to evaluate the uncertainty of the overall models, (NCRP, 1998; Till et. al, 2000), it is important to tailor the uncertainty assessment to the specific situation at hand.

For example, an uncertainty assessment for PuO_2 inhalation was performed by Radiation Assessment Corporation (Till et al, 2000). This assessment listed values for specific organs and particle sizes. The report listed the uncertainties as lognormal distributions with geometric standard deviations (GSD) that varied between 1.9 and 4.5 depending on the organ and the particle size. However, it appears that the dominant factor in the uncertainty was the solubility class. If the solubility for a particular compound is well known, the uncertainty associated with this compound must be lower than that described by the report. Also, this assessment was based on the inhalation of a known (or calculated) amount of material. If the intake is determined from bioassay data, a very different result is obtained, especially for non-metabolic organs.

An acceptable approach, when feasible, is to determine the lowest possible, most likely, and highest possible doses given the data set used for the particular individual. Once these values are determined, a triangular distribution can be assumed using these three points as the parameters of the distribution. This approach gives credit for the parameters that are known while accounting for the parameters that are not well known. When properly performed, this method also inherently accounts for correlated parameters. Figure 5 shows a typical triangular distribution with a minimum value of zero, a maximum value of three and a most likely value of one.

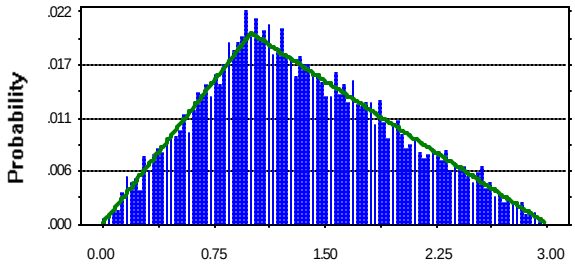


Figure 5. Example of a Triangular Probability Distribution

7.3 Missed Dose

Missed dose is the quantity of dose that could have been received with all measurements remaining below the detection limit of the sampling method employed. It may appear that assigning all non-detectable samples a value equal to the detection limit would be appropriate. However, this approach will, in most instances, significantly overestimate the missed dose.

The first problem with assigning the detection limit to the samples it that this process assumes that a person received an intake that resulted in a bioassay sample that was just under the detection limit. While this is possible, it is extremely unlikely to occur each and every time a sample is taken.

Missed dose from airborne activity samples are usually small since these samples are normally counted long enough to detect activity at very low concentrations. However, if this does become a problem, it can be easily overcome if the actual results were recorded as something other than "<MDA" (< minimum detectable activity). Since the individual samples will be used collectively (i.e. averaged) the detection limit of the batch (the average) is the important quantity, not the individual detection limit. By propagating the errors of the individual samples, the standard deviation of the overall average will decrease. Because of this, the detection limit of the overall average is lower than the detection limit for an individual samples. Once the actual results are averaged, that average can be compared with the detection limit associated with the batch average.

If the results were recorded as "<MDA" then other means must be used to estimate the actual airborne activity. If there are a number of samples with detectable results, one option is to evaluate the distribution of the samples. From that distribution, an estimate of the central tendency along with its associated uncertainty could be determined. This distribution can then be used to describe the sample results that are not detectable.

The second problem in determining missed dose is the correlation between bioassay samples. For bioassay samples, if a person receives an intake of an insoluble compound that produces a bioassay sample just under the detection limit, some portion of this intake will continue to be present in the next sample. This makes it unlikely for a person to receive this maximum uptake in two successive sample periods without producing a

detectable bioassay result. After several sample periods the likelihood becomes increasingly small. This effect is most pronounced when inhaling fairly insoluble compounds (class S solubility). For very soluble compounds, the correlation among samples is less important but the missed dose associated with such intakes is also relatively low.

The most appropriate method for determining the missed dose associated with a bioassay sample will vary depending on the bioassay program itself, as well as other factors surrounding the individual exposure scenario. Missed dose could be assigned based on factors such as co-worker data or interpolation. For example, if the individual tours a facility for approximately one hour each day and had no detectable bioassay samples, it could be helpful to evaluate the exposure of others working in the facility all day. If they also had no detectable bioassay samples, it would not be appropriate to assign the same missed dose to both individuals. The missed dose of the individual with limited exposure time (from a tour) should be proportional to the other individual who was in the work area for the whole time. This prorating could apply even if the worker submitted detectable bioassay samples. Provided the exposure potentials are equal (other than exposure time) the touring individual's missed dose should be based on the other individual's detectable dose.

When there is no other information on which to base a decision, the missed dose should be determined using the following protocol:

1. Determine the standard deviation of the bioassay method at the detection limit. If this cannot be readily determined, assume that it is 0.3 times the value of the detection limit. This factor is derived from the fact that the sample analysis probably consists of a gross sample result minus a blank result. It assumes that the standard deviation of the gross result is at least as high as the blank result and therefore propagating the error dictates that the standard deviation of the detection limit must be at least $\sqrt{2}$ times the standard deviation of the blank. The factor also assumes some variation of the standard Currie equation ($LD = 2.71 + 4.65\ \sigma_b$) (Currie, 1968) was used to determine the detection limit. If the 2.71 is ignored, σ_b (the standard deviation of the blank) becomes LD/4.65 (where LD is the detection limit) and the standard deviation of the detection limit then becomes $\sqrt{2}$*LD/4.65 or 0.3*LD. Even though this is not an exact value, this should provide a reasonable approximation in the situations when no other information is available.
2. Subtract 1.645 times the standard deviation from the detection limit to achieve a new target value. This provides a target value that 95% of the samples at this level will not exceed the detection limit.
3. Assume a constant chronic exposure over the entire period in question and determine that intake based on the highest bioassay sample equaling this target value from number 2 above.
4. The uncertainty of this estimate will be considered to be normally distributed with a relative error equal to the standard deviation determined in step number 1 divided by the detection limit times 100%.

It must be noted that this procedure should be used only when there is a large deficiency of information. It should also be noted that this is likely a high estimate. This is because a number of bioassay samples that all indicate less than the detection limit are more likely closer to zero than to the detection limit itself. This method produces a target value that is ½ the detection limit (LD-1.645*0.3*LD = 0.5*LD). If the standard deviation is known, the target value could be higher or lower but it would be based on more information.

7.4 Radon

Radon 222 and its progeny is an exception to much of the discussion provided in this document. It is a naturally occurring source of radiation exposure as well as an occupational source under certain conditions. Radon and its short-lived progeny (decay products) are continuously produced from the decay of Ra-226, a member of the naturally occurring U-238 series. As an inert gas, radon emanates from soils containing natural levels of U-238. Chemical separation of the uranium from its ore will strip out the majority of the Ra-226 as well as other progeny. While the decay process will again produce Ra-226, this process takes thousands of years for the Ra-226 to re-achieve equilibrium. Therefore, not all uranium will be a significant source of radon exposure. In fact, many DOE facilities only handled uranium after it had undergone a chemical separation. The Ra-226 removed during the process was at times used in other processes or experiments. At other times, it was stored as waste. Any of these processing or storage locations are potential sources of occupational radon exposure.

Since radon is a naturally occurring radionuclide, it can be difficult to distinguish environmental radon exposure from occupational sources of radon. For the purposes of dose reconstruction under EEOICPA, occupational radon exposure is considered to be radon emanating from sources other than those naturally occurring in the area.

The natural (background) level of radon in the area must be assessed and subtracted from any measured values whenever estimating occupational levels of radon. Since by definition, natural radon is not a result of DOE activities, current background levels of radon can be assessed and applied to prior years. The background however, must be measured without interference from the current or past DOE activity.

The dose from radon will not be determined directly because the probability of causation risk coefficients are based on working level months (WLM) of exposure which is a direct input into the NIOSH-IREP program. A working level month is the exposure to one working level of Rn-222 progeny for 170 hours (40 hours per week for one month). A working level is defined as follows:

> "Any combination of short-lived radon daughters in one liter of air that result in the emission of 1.3×10^5 Mev of potential alpha energy." (NCRP 78, 1984)

Therefore, one working level has an equilibrium equivalent concentration of 100 pCi of Rn-222 in one liter of air.

Since radon-222 and its progeny are very short-lived, air sampling is the only type of monitoring that is feasible on a routine basis. Radon was typically not considered an occupational concern or at least a very low concern compared to other sources of radiation exposure. For this reason, air sample data may be limited, especially in the earlier years (1945-1980s). However, there may be information that can be used to relate interference to other air monitors.

Respiratory protection factors afforded by HEPA filtered respirators can be taken into account in accordance with section 9 of this guide. The efficiency of HEPA type filtration is nearly 100% for radon progeny due to its electrostatic nature. However, since radon is an inert gas, no protection is provided by HEPA filters for the Rn-222 gas itself. Because it is chemically inert, the majority of the gas is exhaled without depositing in the lungs. Some of the gas will however diffuse into the body tissues. With the majority of the gas being exhaled, the dose from the gas is small in comparison to the progeny. In these cases, when HEPA filtered respirators are used, an assumption that the gas saturated the soft tissues in the body is usually a high estimate of the dose that produces very little exposure. Since the risk factors in NIOSH-IREP are based on WLMs, and a WLM is defined only for progeny, any exposure to Rn-222 gas (without its accompanying progeny) will have to be calculated as a dose and input into NIOSH-IREP as a dose instead of a WLM.

8.0 EXAMPLE DOSE ESTIMATES

This example internal dose reconstruction is provided to demonstrate the general approach to dose reconstruction, as well as many of the other concepts discussed in this document. It is a fictitious example designed to help illustrate several points. The example is based on the following scenario.

8.1 Scenario

The individual was born in 1931 and worked from November 7, 1969 to August 9, 1973 in an area where he had the potential to inhale plutonium. The plutonium was pure Pu-239 with about half in the form of insoluble oxides and the other half being unspecified compounds. The only dosimetry data that exist are bioassay (urine) samples. The analytical technique that was used to measure the samples was capable of detecting approximately 0.07 pCi/L or about 0.1 pCi/day at a nominal daily excretion rate of 1400 ml of urine per day. The bioassay data available is normalized to daily excretion. Results indicating less than the detection limit were recorded “as read”. The worker’s records indicate he was involved in an incident on 1/20/73 that resulted in the inhalation of Pu-239 by several people. The individual was diagnosed with liver cancer in late December 1998.

As indicated in Table 3, the bioassay data consists of 51 samples with 38 of them indicating greater than the detection limit.

TABLE 3. Bioassay Data for Example Dose Reconstruction

Date	pCi/Day (Pu-239)		Date	pCi/Day (Pu-239)
11/7/1969	0.0		11/26/1971	0.07
12/6/1969	0.36		1/5/1972	0.19
1/23/1970	0.075		1/28/1972	0.18
2/20/1970	0.0825		2/25/1972	0.2919
3/28/1970	0.09		3/31/1972	0.2071
5/1/1970	0.1593		5/26/1972	0
5/29/1970	0.1083		6/23/1972	0.37
6/26/1970	0.02		9/1/1972	0.66
7/24/1970	0.087		11/25/1972	0.3431
8/15/1970	0.03		12/1/1972	0.09
8/21/1970	0.0741		1/12/1973	0.4156
9/11/1970	0.252		1/22/1973	0.3926
10/17/1970	0.02		2/1/1973	0.85
10/23/1970	0.2575		2/8/1973	1.38
12/4/1970	0.1201		2/15/1973	1.36
1/8/1971	0.2794		2/22/1973	1.22
2/5/1971	0.3475		3/1/1973	1.2635
3/5/1971	0.38		3/8/1973	1.42
4/13/1971	0.3		3/22/1973	1.42
5/16/1971	0.2		4/12/1973	1.4
6/11/1971	0.26		4/26/1973	1.39
7/16/1971	0.4278		5/3/1973	1.34
8/15/1971	0.2401		6/7/1973	1.26
9/10/1971	0.08		7/12/1973	1.33
10/24/1971	0.12		8/9/1973	1.25
11/7/1969	0.1704			

8.2 Case Evaluation

No information is available on particle size so the ICRP 66 default of 5 micron will be used. From the information available about the type of compounds used, the solubility of the material will be considered to be ½ class M and ½ class S.

With so many samples, it may be best to start with an overall impression of the data by creating a graph as seen in Figure 6.

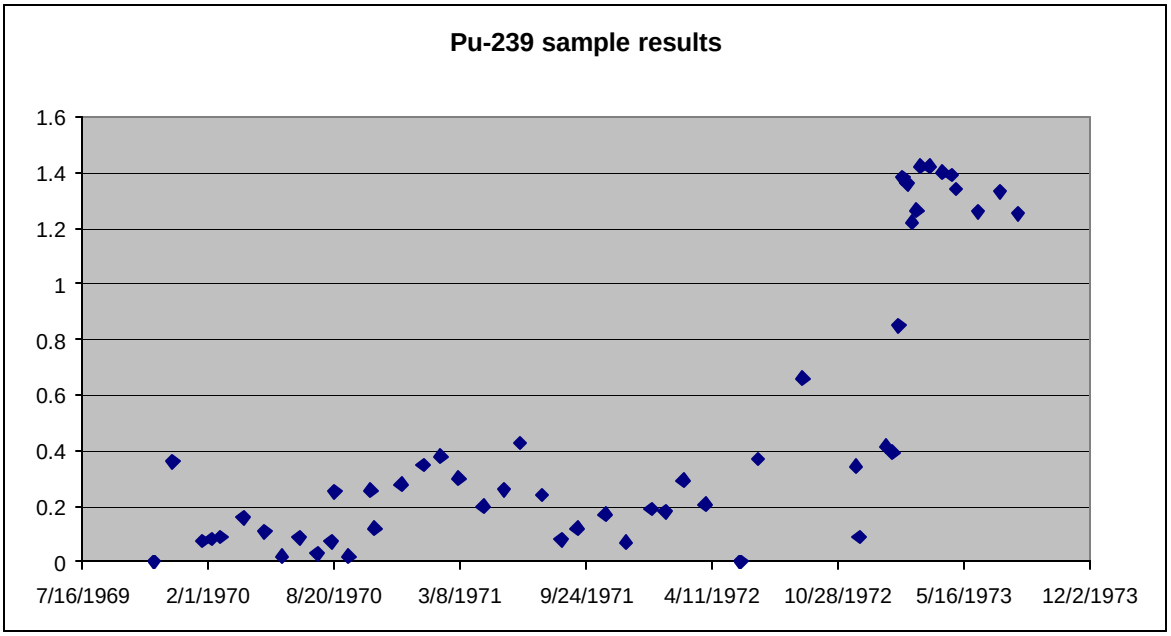


Figure 6. Graph of Bioassay Samples for Example

From Figure 6, it is clear that the individual had an intake in January of 1973. This corresponds well with the report of an incident on 1/20/1973. That information alone is considered sufficient to verify that the elevated results after January of 1973 are valid results, not an analytical problem or anomalies. In this case the individual had several high sample results, which would discount the possibility of a contaminated sample. Since they are all relatively consistent, the possibility of cross contamination of the sample is remote. Also a number of the results indicate greater than the detection limit prior to the 1/20/73 incident. Again, with so many samples elevated and the results being consistent, they can be considered to be true values and not an analytical anomaly. Therefore, the overall impression of this data is that the individual received several intakes prior to a larger intake on 1/20/73.

The liver tends to concentrate plutonium. Combining this with the fact that the liver is fairly radiosensitive, and the fact that the individual is believed to have received several real intakes, it would be prudent to perform a preliminary dose estimate using the data at hand to create an underestimate of dose to the liver. This underestimate could be done quickly to determine if the individual will easily fall into a high probability category.

8.3 High Dose Potential Preliminary Estimate

An underestimate can be accomplished by estimating the dose received from the large intake during the 1/20/73 incident alone. In determining the amount of the acute intake on 1/20/73, it is indicated that the concentration in the urine should drop off much faster than the data show. While the cause of this is unclear, all that matters is that there are credible reasons for the observations, such as the incident contaminated the work area enough to deliver a chronic intake each day for some time. With that assumption stated,

the underestimate was performed by modeling the intake as two back-to-back chronic exposures. The first of 2000 pCi per day from 2/7/73 to 4/17/73 and the second of 1000 pCi per day from 4/18/73 to 8/7/73. Figure 7 shows the predicted bioassay results of this exposure scenario compared to the actual bioassay samples. It can be seen clearly from the graph that the predicted values are lower than the reported values, so this is a conservative underestimate of the exposure. This underestimated exposure resulted in the minimum organ doses presented in Table 4. In this case, the probability of causation determination for this underestimated dose to the liver produced a probability of causation of 83%. Since a detailed reconstruction would only increase the dose, there is no reason to perform a detailed dose reconstruction.

Had the probability been below the compensation level, the estimate could be refined easily by adding another chronic exposure to account for the time frame when he received several detectable intakes. This would increase the preliminary dose estimate while still being an underestimate.

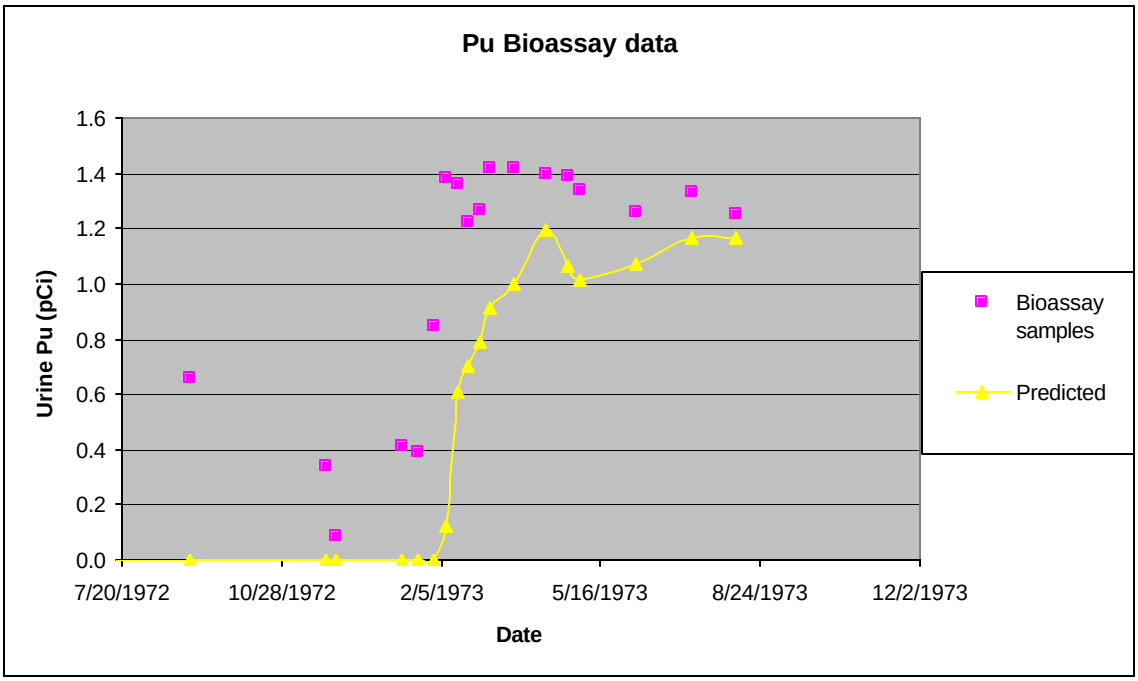


Figure 7. Example of Underestimated Intakes from Bioassay Samples

8.4 Low Dose Potential Preliminary Estimate

In the interest of furthering this example, assume that the cancer developed by this individual is bladder cancer. The bladder does not concentrate plutonium so the dose to the bladder is likely small. In this situation, the analyst would want to perform an overestimate of the dose to determine if there is any need to continue. One very simple and very conservative technique to overestimate the dose would be to assume the individual received an acute intake on his first day of employment that was so large, all

of the sample results fall at or below the predicted urine concentration. This scenario would result in an intake estimate of $6x10^5$ pCi of class M Pu-239 and $6x10^5$ pCi of class S Pu-239. A graph of the predicted bioassay samples compared to the actual samples is provided Figure 8. This figure clearly shows that this estimate is a gross over-estimate of the intake. As indicated in Table 4, the annual dose to the bladder was usually below one rem per year. The probability of causation associated with his bladder dose would be 31%. Since this is a worst-case over-estimate, there would be no need to perform a detailed dose reconstruction.

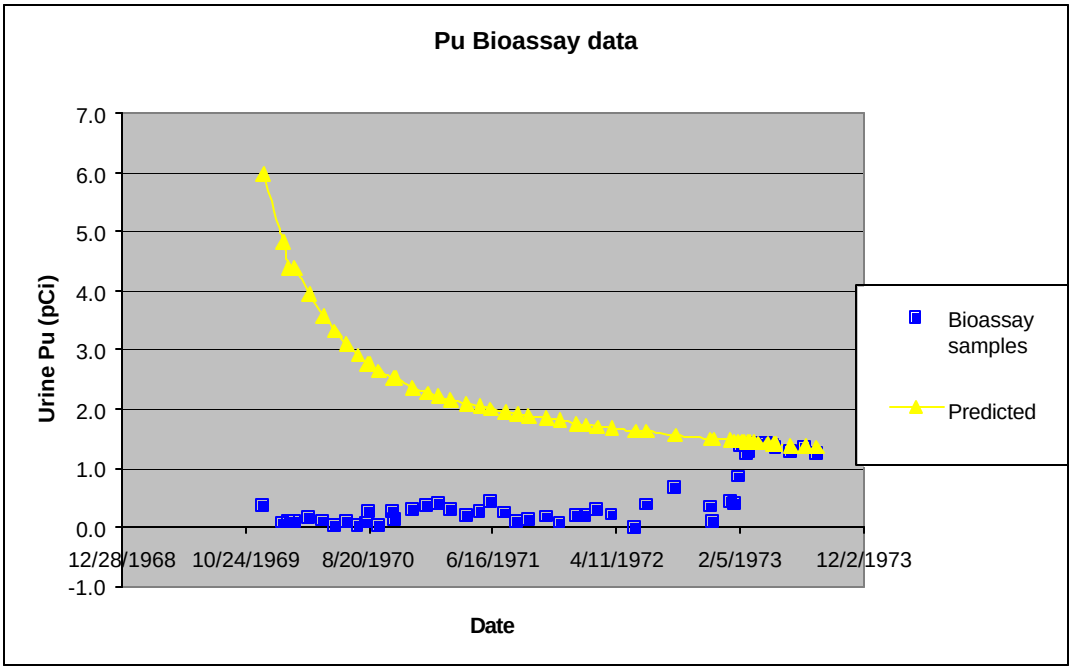


Figure 8. Example of Overestimated Intakes from Bioassay Samples

If this analysis resulted in a value close to 50% probability of causation, the overestimate could be refined slightly by assuming a chronic exposure that would predict results higher than all the actual results. This would still overestimate the intake, but to a lesser extent.

Table 4. Annual Doses Calculated from Preliminary Estimates

	Maximum Dose (Rem)			Minimum Dose (Rem)		
YEAR	U. BLADD	LIVER	LUNG	U. BLADD	LIVER	LUNG
1969	0.096	4.9109	58.7	0.0000	0.0000	0.0000
1970	1.188	89.13	249.4	0.0000	0.0000	0.0000
1971	1.287	116.59	136.38	0.0000	0.0000	0.0000
1972	1.163	123.7	94.44	0.0000	0.0000	0.0000
1973	1.044	125.23	69.06	0.0191	1.3246	20.2700
1974	0.958	126.4	52.278	0.0293	2.5114	2.4270
1975	0.895	127.3	40.603	0.0264	2.6848	1.5187
1976	0.851	128	32.557	0.0236	2.7501	1.0953
1977	0.815	127.5	26.522	0.0214	2.7685	0.8175
1978	0.791	127	22.299	0.0197	2.7728	0.6295
1979	0.773	126.3	19.082	0.0186	2.7837	0.4973
1980	0.764	125.6	16.773	0.0179	2.7828	0.4055
1981	0.756	123.7	14.765	0.0172	2.7587	0.3369
1982	0.751	121.9	13.26	0.0168	2.7342	0.2865
1983	0.749	121	12.058	0.0165	2.7091	0.2492
1984	0.752	119.1	11.06	0.0164	2.6928	0.2213
1985	0.751	117.1	10.079	0.0162	2.6547	0.1974
1986	0.753	115.1	9.3	0.0162	2.6166	0.1791
1987	0.755	113.7	8.602	0.0162	2.5772	0.1636
1988	0.760	112.3	8.006	0.0162	2.5488	0.1509
1989	0.762	110.1	7.427	0.0162	2.4988	0.1389
1990	0.766	108.4	6.92	0.0163	2.4588	0.1290
1991	0.769	106.6	6.453	0.0163	2.4258	0.1200
1992	0.775	105.1	6.048	0.0164	2.3908	0.1123
1993	0.777	103.1	5.639	0.0165	2.3392	0.1047
1994	0.781	101.4	5.292	0.0165	2.3018	0.0982
1995	0.784	99.8	4.955	0.0166	2.2634	0.0922
1996	0.790	98.4	4.67	0.0167	2.2291	0.0868
1997	0.792	96.5	4.381	0.0168	2.1882	0.0815
1998	0.795	94.9	4.123	0.0169	2.1516	0.0769

8.5 Detailed Dose Reconstruction

In the event that a determination can not be made from preliminary estimates, a detailed reconstruction is necessary. The following is a detailed reconstruction for the example case.

Since every intake will result in an increase in Pu-239 concentration in the urine for some time to come, it is necessary to work in chronological order when evaluating bioassay data.

Intake #1
On 12/6/1969 the individual submitted a positive bioassay sample. His previous sample on 11/7/69 was non-detectable. The 12/6/1969 sample indicated 0.36 pCi Pu-239 in the urine. Two subsequent samples on 1/23/70 and 2/20/70 indicated less than the detection limit, however, both indicated high in that range. Subsequently, the two samples were averaged and assigned a date of 2/6/70. Using the 50% Class M and 50% class S assumption, the intake date and amount was adjusted to achieve a good fit with the two points (12/6/69 & 2/6/70). The best fit to this data indicated an acute intake of 18200 pCi of Pu-239 on 12/1/69.

Intake #2
On 3/28/70 the individual submitted another positive bioassay sample followed by one additional positive and four additional negative samples. Two of the negative samples were essential zero while the other two were 87% and 75% of the MDA. The two that were nearly zero could not be reconciled with any possible scenario associated with intake #1. This led to the decision to disregard these near zero samples and concentrate on the remaining samples. When the residual concentrations from intake #1 were considered, the remaining four data points provided a good fit to an acute intake scenario of 9000 pCi Pu-239 on 3/20/70.

Intake #3
On 8/21/70 the individual again submitted a positive sample (0.252 pCi). This sample was followed up by a sample on 9/11/70 that indicated 0.02 pCi (<MDA). Due to the fast clearance, intake #3 was assumed to be acute and to have occurred near the sample date of 8/21/70. The intake amount and date could not be adjusted to align the two points (8/81 & 9/11) but it was adjusted so that the second point was less than the detection limit. This resulted in intake #3 being an acute intake on 8/20/70 of 1500 pCi of Pu-239.

Intake #4
On 10/17/70 the individual submitted a sample indicating 0.257 pCi Pu-239. On 10/23/70 the results were substantially lower but not below the detection limit. An acute intake was assumed and the date and amount adjusted to line up the two points. The results were that intake #4 occurred on 10/13/70 with an intake of 7000 pCi Pu-239.

Intake #5 through Intake #13
The next three samples each increased above the previous sample. While the shape of the curve appeared to fit a chronic exposure, no scenario could be developed that did not result in numerous later samples being much lower than predicted. Therefore, a series of acute exposures was assumed. With no second point to help determine the timing of the intakes, the midpoint between samples was assumed as a starting point. With this date the amount of the intake could not be adjusted to keep samples on 8/15/71 & 11/26/71 below the detection limit. The date and amount were then adjusted to achieve this goal.

The analysis then proceeded with intake #6 in the same manner, each intake adding to the predicted values of the samples on 8/15/71 & 11/26/71. The process eventually required intakes #5 through #13 to all be considered simultaneously to reach the best fit with the

data. The final results have the predicted value for the samples on 8/15/71 & 11/26/71 indicating 0.123 pCi and 0.108 pCi respectively. While these values are above the detection limit, they are close and could easily have produced a sample result below the detection limit.

The results for intakes #5 through #13 are as follows

#5	1500 pCi	12/3/70
#6	2100 pCi	1/7/71
#7	7000 pCi	2/2/71
#8	1500 pCi	3/4/71
#9	650 pCi	4/12/71
#10	1200 pCi	5/15/71
#11	2700 pCi	6/10/71
#12	1000 pCi	7/15/71
#13	500 pCi	10/23/71

Intake #14

Intake #14 was assumed to be an acute intake resulting in positive bioassay samples on 1/5/72 and 1/28/72. The date and amount of the intake were adjusted to line up these two points. The results were later changed as discussed in intake #15 and #16. The final results showed an acute intake of 17000 pCi on 12/16/71.

Intake #15

Intake #15 was assumed to be an acute intake resulting in positive bioassay samples on 2/25/72 & 3/31/72. The date and amount of the intake were adjusted to line up these two points. At this point it was noted that the predicted value of the sample on 5/26/72 & 12/1/72 were considerably higher than the actual result. The date and quantity of intake #14 and #15 were adjusted to minimize these points but no acceptable combination could be found. These intakes were then modeled as chronic exposures using various intake dates and quantities. Eventually a suitable combination was found that allowed the predicted result for 12/1/72 to drop below the detection limit, however, the sample on 5/26/72 was still above. Since the results of the sample on 5/26/72 was 0.0 pCi, it was believed that the sample could be flawed and the results were rejected. The results of intake #14 and #15 were again changed as a result of evaluating the next intake. The details are discussed under intake #16. The final result of intake #15 was an acute exposure of 10500 pCi on 2/18/72

Intake #16, #17, and #18

Intake #16 and #17 were then modeled using various combinations of acute and chronic exposures but all failed to reconcile with the sample on 12/1/72. An additional exposure was then added but it too failed to reconcile the difference. At that point the sample on 12/1/72 was rejected. Since this sample led to several decisions pertaining to intake #14 and #15, the analysis was redone starting with intake #14 but without the sample results for 12/1/72. This led to intakes #14 and #15 being reevaluated as acute exposures with the final values listed above.

Intakes #16 and #17 were then evaluated simultaneously. Since the 12/1/72 sample was rejected, an attempt was made to model these intakes so that the predicted value matched the actual value on 11/25/72. The final results indicated an acute exposure of 43000 pCi and 23000 pCi on 5/11/72 and 8/26/72 for intakes #16 and #17 respectively. This then required an additional intake (#18) to match the data on 1/12/73 and 1/22/73. This was modeled as an acute intake on 12/30/72 in order for the predicted line to match both samples. The intake quantity was 17000 pCi.

Since the samples on 1/12/73 & 1/22/73 were only slightly higher than the sample on 11/25/72, it was realized that these three samples could actually represent statistical uncertainty of one predicted line. This scenario was explored by eliminating intake #18 and adjusting the date and quantity of intakes #16 and #17 in order to minimize the residuals of these three samples. This scenario resulted in no change to intake #16 while intake #17 was 40000 pCi on 8/24/72. Notice that this exactly matches the total of the original intake #17 and #18 (23000 + 17000).

In an attempt to further reduce the residuals by "flattening out" the predicted line through the three samples (11/25/72, 1/12/73, & 1/22/73), intake #16 was moved back as far as possible to the day of the previous sample (3/31/72 since 5/26/72 was rejected). This resulted in intake #16 being a 53000 pCi intake. When only intake #17 was added the residuals were minimized when intake #17 occurred on 8/24/72 with 35000 pCi. The last scenario to explore was to reconsider the two intakes (#17 & #18) with intake #16 consisting of 53000 pci on 3/31/72. This resulted in intake #17 being 23000 pCi on 8/26/72 with intake #18 being 13000 pCi on 12/30/72. This yielded a total of 36000 pCi (23000 + 13000) compared to 35000 pCi for the one intake scenario. This indicated that while it was not clear which scenario was correct, the final outcome was comparable. Also note that the total intake from all four scenarios yielded results of intakes of 89000, 88000, 83000, and 83000 pCi. Even moving intake #16 back 41 days changed the total intake by <10%.

This information implies that the total intake is not very sensitive to the actual intake date. The primary objective is to match data as closely as possible by some non-subjective means (such as residuals) regardless of the chosen scenario.

INTAKE #19 through #30

With the conclusions of the above sensitivity analysis in mind, the remaining intakes were modeled using the midpoint between samples for an intake date and adjusting the quantity to match the data. This resulted in intakes #19 through #30 as shown below.

#19	24000 pCi	1/27/73
#20	31500 pCi	2/4/73
#21	23700 pCi	2/11/73
#22	14700 pCi	2/18/73
#23	15000 pCi	2/25/73
#24	18500 pCi	3/4/73
#25	35800 pCi	3/15/73
#26	47000 pCi	4/1/73
#27	10500 pCi	4/19/73
#28	2000 pCi	4/29/73
#29	21000 pCi	5/23/73
#30	35700 pCi	6/24/73

If additional information were available, the estimated intakes would have been evaluated based on that information. For example, area or personal air sample data could have helped determine the date of intake or whether the intake was acute or chronic.

Below is the graph of the bioassay data presented earlier with the predicted concentrations from the dose reconstruction overlaid.

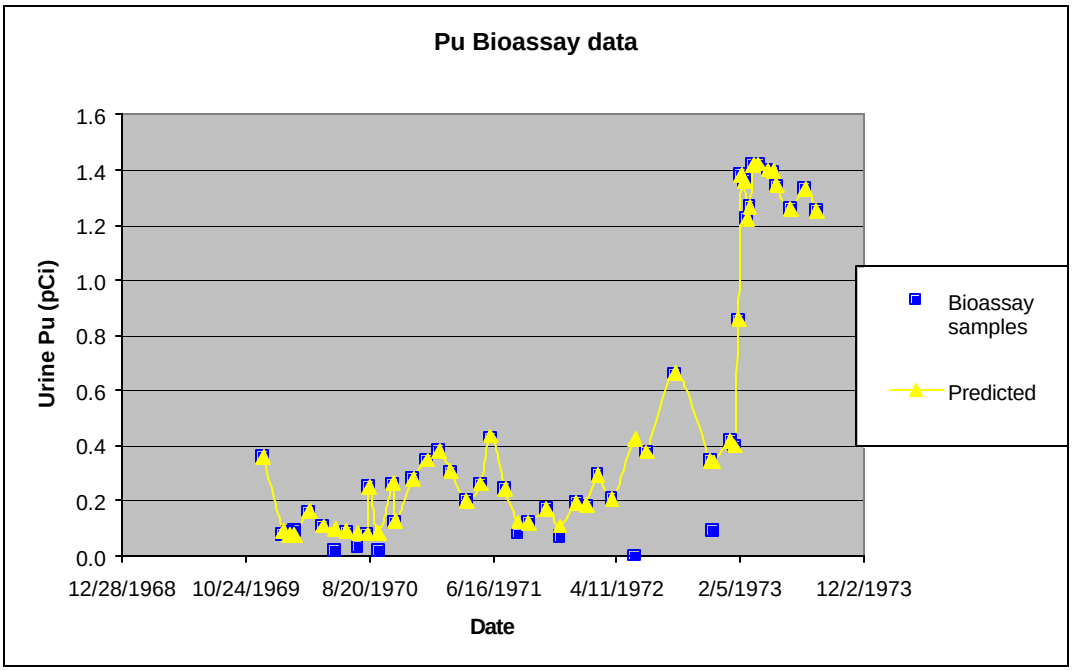


Figure 9. Predicted Bioassay Overlaid with Actual Bioassay

The annual doses to the liver and to the bladder were calculated for these intakes. The results of this calculation are shown in Table 5. The IREP input for liver cancer is included in Appendix A. Since alpha particles deliver the majority of the doses from Pu-239, the entire dose was considered to be from alpha radiation. The probability of causation calculation produces a value of 98.8% for the individual's liver cancer using

the constant probability distribution. If bladder cancer is assumed for this example, the probability of causation calculation produces a value 12.5%.

Table 5. Annual Doses from Detailed Dose Estimate

Calendar Year	Liver Dose (rem)	Bladder Dose (rem)
1969	0.033	0.001
1970	1.774	0.025
1971	4.163	0.051
1972	8.845	0.107
1973	27.873	0.351
1974	42.577	0.464
1975	45.365	0.426
1976	46.498	0.386
1977	46.879	0.353
1978	47.130	0.330
1979	47.188	0.313
1980	47.193	0.302
1981	46.837	0.292
1982	46.497	0.286
1983	46.070	0.282
1984	45.712	0.280
1985	45.051	0.278
1986	44.487	0.277
1987	43.877	0.277
1988	43.369	0.279
1989	42.617	0.279
1990	41.962	0.279
1991	41.305	0.281
1992	40.755	0.283
1993	39.994	0.283
1994	39.340	0.285
1995	38.684	0.286
1996	38.154	0.288
1997	37.424	0.289
1998	36.796	0.290

8.6 Missed Dose

In the preceding example, the intakes were determined for the periods in which many detectable bioassay samples were submitted over a period that covered the entire employment period. However, it is instructive to discuss how this would be done if the employment period were for a longer period of time.

Consider the situation where the individual actually started working at the facility two years prior to his first detectable bioassay sample. Assume he routinely left samples on a monthly basis that all indicated no detectable Pu-239. He left a total of 24 negative samples in this period. He was performing the same type of work and the samples were analyzed in the same manner with the same detection limit. No more information is known about how the value for the detection limit was derived.

In keeping with section 8.2 of this guide, the uncertainty of the bioassay samples is assumed to be normally distributed with a standard deviation equal to 0.3 times the detection limit. Therefore the standard deviation is assumed to be 0.03 pCi/day. Now the missed dose is estimated by assuming a chronic dose for the entire period that will predict a maximum urine concentration of 0.04 pCi/day. This process results in estimating a chronic intake of 23 pCi/day for a 702 day (approx. 2 year) period.

If this intake is then inserted into the detailed model, it is discovered that the predicted line is raised above the actual results for many of the samples. This is due to the fact that this chronic exposure alone will produce some amount of Pu-239 concentration in the urine for years to come. As a result, the entire detailed intake assessment must be re-evaluated. Previously it was mentioned that it is best to work with bioassay samples in chronological order. This example demonstrates the necessity of including missed dose into that approach. Figure 10 graphs the data for this extended employment period example.

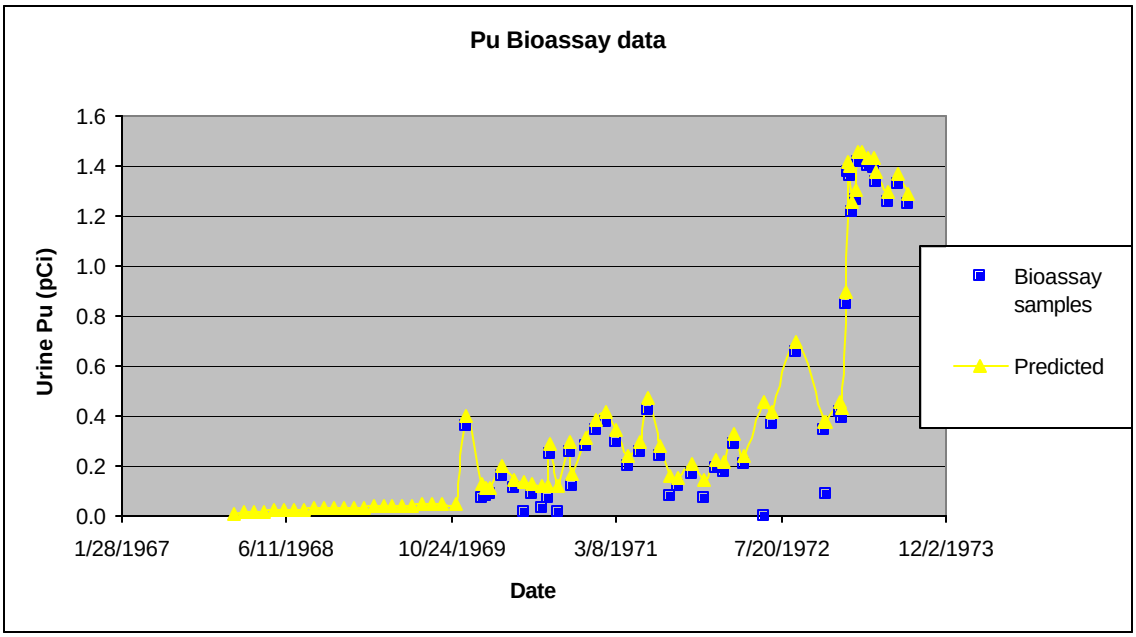


Figure 10. Affect of Missed Dose on Predicted Bioassay Values

This also demonstrates the concept that the most accurate quantity is the total intake. The two years of very small chronic exposure added a total 16146 pCi of intake. However, since the estimated intakes previously determined will all have to be lowered, the overall increase in the total activity may likely be zero.

8.7 Uncertainty

Another parameter that must also be determined is the uncertainty of the intake assessment. It is important to remember at this point that if the preliminary overestimate or underestimate is conclusive, no uncertainty analysis is required since the estimate is already a bounding case.

In cases when a detailed dose estimate is required, the uncertainty of the intake must be addressed. Section 8.1 of this guide describes how this process should work. In the example case, the uncertainty of the individual bioassay samples is unknown. However, the detection limit is known. Therefore, in keeping with section 8.2, the uncertainty of the samples at the detection limit can be estimated by multiplying by 0.3. For our example, this indicates a standard deviation of 0.03 pCi for a sample of 0.1 pCi which yields a relative error of 30%. This relative error can then be assumed to be constant for all of these bioassay samples. As noted before, this is not an exact estimate of the uncertainty but only a reasonable approximation in a situation with many unknowns. This 30% standard deviation is then applied to each intake to obtain an absolute standard deviation for each intake. For example, for the first intake of 18200 pCi, the standard deviation can be estimated as 5460 pCi (18200 x 0.3). This process is repeated for each of the 30 intakes. Then the total uncertainty is propagated by taking the square root of the sum of all the standard deviations squared. This gives an absolute uncertainty of the total intake of 32715 pCi. The total intake itself is just the sum of all the intakes or 450250 pCi. The overall relative error can then be calculated as the standard deviation of the overall intake divided by the overall intake itself or 32715/450250 = 7.3%. This relative error is applied to the dose calculated for each year for the organ of interest.

The error is relatively small as can be expected when a large number of detectable samples are submitted. This relative error is applicable to the intake amount only; it assumes the biokinetic model is accurate. With an intake error this low, it is necessary to assess the uncertainty of the biokinetic model in order to develop a realistic uncertainty for this individual's dose.

Once again, the overall purpose of EEOICPA must be kept in mind. In this case, if the best estimate of the individual's dose with only the intake uncertainty applied produces a probability of causation above the 50% requirement, no uncertainty of the bioassay model need be applied.

In the case of probability of causation values less than 50%, a more detailed analysis will be required. It may be necessary to reproduce the biokinetic model in a Monte Carlo calculation with known values given as constants. The results of this calculation can then be used to describe a detailed probability distribution of the organ dose.

9.0 REFERENCES

1. The Energy Employees Occupational Illness Compensation Program Act, Public Law 106-398, 114 Stat. 1654, 1654A-1231 (October 2000), enacted as Title XXXVI of the Floyd D. Spence National Defense Authorization Act for Fiscal Year 2001.

2. Executive Order 13179, Providing Compensation to America's Nuclear Weapons Workers (65 FR 77487), December 7, 2000.

3. Probability of Causation Guidelines Under the Energy Employees Occupational Illness Compensation Program Act of 2000, Final Rule at 42 CFR 81 (67 FR 22296), May 2, 2002.

4. Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program Act of 2000, Final Rule at 42 CFR 82 (see 67 FR 22314), May 2, 2002.

5. Crawford-Brown et. al. (1989) *Procedures for Assessing Occupational Radiation Monitoring Data for use in Epidemiologic Studies*, ORAU 89/A-127, Oak Ridge Associated Universities, Oak Ridge TN.

6. Currie, L.A. (1968) *Limits for Qualitative Detection and Quantitative Determination. Application to Radiochemistry*, Anal. Chem. 40, 586

7. Department of Health and Human Services (1991) *The International Classification of Diseases Clinical Modification (9th Revision) Vol. I&II,* Department of Health and Human Services Publication No. (PHS) 91-1260, U.S. Government Printing Office, Washington D.C.

8. ICRP, (1959) International Commission on Radiological Protection. *Permissible Dose for Internal Radiation*, ICRP Publication 2, Pergamon Press, Oxford, England.

9. ICRP, (1979) International Commission on Radiological Protection. *Limits for Intakes of Radionuclides by Workers, Part 1*, ICRP Publication 30, Annals of the ICRP Vol. 2 (3/4), Pergamon Press, Oxford, England.

10. ICRP, (1989) International Commission on Radiological Protection. *Age-dependent Doses to Members of the Public from Intake of Radionuclides, Part 1*, ICRP Publication 56, Annals of the ICRP Vol. 20 (2), Pergamon Press, Oxford, England.

11. ICRP, (1994) International Commission on Radiological Protection. *Human Respiratory Tract Model for Radiological Protection*, ICRP Publication 66, Annals of the ICRP Vol. 24 (1-3), Pergamon Press, Oxford, England.

12. ICRP, (1994) International Commission on Radiological Protection. *Age-dependent Doses to Members of the Public from Intake of Radionuclides, Part 2,* ICRP Publication 67, Annals of the ICRP Vol. 23 (3/4), Pergamon Press, Oxford, England.

13. ICRP, (1994) International Commission on Radiological Protection. *Dose Coefficients for intakes of Radionuclides by Workers*, ICRP Publication 68, Annals of the ICRP Vol. 24 (4), Pergamon Press, Oxford, England.

14. ICRP, (1995) International Commission on Radiological Protection. *Age-dependent Doses to Members of the Public from Intake of Radionuclides, Part 3*, ICRP Publication 69, Annals of the ICRP Vol. 25 (1), Pergamon Press, Oxford, England.

15. NCRP, (1984) National Council on Radiation Protection and Measurements, *Evaluation of Occupational and Environmental Exposures to Radon and Radon Daughters in the United Sates*, NCRP Report No. 78, National Council on Radiation Protection and Measurements, Bethesda, Maryland.

16. NCRP, (1998) National Council on Radiation Protection and Measurements, *Evaluating the Reliability of Biokinetic and Dosimetric Models and Parameters Used to Assess Individual Doses for Risk Assessment Purposes.*, NCRP Commentary No. 15, National Council on Radiation Protection and Measurements, Bethesda, Maryland.

17. NRC, (1993) Office of Nuclear Regulatory Research, *Air Sampling in the Workplace*, NUREG-1400, US Nuclear Regulatory Commission, Washington D.C.

18. NRC, (1992) Office of Nuclear Regulatory Research, *Air Sampling in the Workplace*, Regulatory Guide 8.25, US Nuclear Regulatory Commission, Washington D.C.

19. Ritter, P.D., Huntsman, B.L., Novick, V. J. Alvarez, J.L., Rich, B.L., (1984) EG&G Idaho, Inc. *The Role of Personal Air Sampling in Radiation Safety Programs and Results of a Laboratory Evaluation of Personal Air Sampling Equipment*, NUREG/CR-4033, EG&G Idaho, Inc., Idaho Falls, Idaho

20. Till, J. E., Grogan, H. A., Sinclair, W. K. Voilleque, P. G., (2000) Radiological Assessment Corp., *Assessing Risks of Exposure to Plutonium, Rev. 2,* Radiological Assessments Corp., Neeses, SC.

APPENDIX A – IREP-EXCEL INPUT FORMAT

PERSONAL INFORMATION								
Claimant Name	Claim #	Claimant SSN	DOL Claim Center	Gender	Birth Year	Year of Diagnosis	Cancer Model	Should alt model be run?
John Doe	000000	000-00-0000	CL	Male	1931	1998	Liver	No

CLAIMANT CANCER DIAGNOSES						
	Primary Cancer #1	Primary Cancer #2	Primary Cancer #3	Secondary Cancer #1	Secondary Cancer #2	Secondary Cancer #3
Cancer Type	Liver	N/A	N/A	N/A	N/A	N/A
Date of Diagnosis	12/30/98	N/A	N/A	N/A	N/A	N/A

EXPOSURE INFORMATION

Number of exposures					Dose Intensity	Dose Uncertainty	
30							
Exposure #	Exposure Year	Exposure Rate	Radiation Type	Dose Distribution Type	Parameter 1	Parameter 2	Parameter 3
1	1969	chronic	Alpha	Constant	0.032	0.000	0.000
2	1970	chronic	Alpha	Constant	1.770	0.000	0.000
3	1971	chronic	Alpha	Constant	4.160	0.000	0.000
4	1972	chronic	Alpha	Constant	8.840	0.000	0.000
5	1973	chronic	Alpha	Constant	27.800	0.000	0.000
6	1974	chronic	Alpha	Constant	42.500	0.000	0.000
7	1975	chronic	Alpha	Constant	45.300	0.000	0.000
8	1976	chronic	Alpha	Constant	46.400	0.000	0.000
9	1977	chronic	Alpha	Constant	46.800	0.000	0.000
10	1978	chronic	Alpha	Constant	47.100	0.000	0.000
11	1979	chronic	Alpha	Constant	47.100	0.000	0.000
12	1980	chronic	Alpha	Constant	47.100	0.000	0.000
13	1981	chronic	Alpha	Constant	46.800	0.000	0.000
14	1982	chronic	Alpha	Constant	46.400	0.000	0.000
15	1983	chronic	Alpha	Constant	46.000	0.000	0.000
16	1984	chronic	Alpha	Constant	45.700	0.000	0.000
17	1985	chronic	Alpha	Constant	45.000	0.000	0.000
18	1986	chronic	Alpha	Constant	44.400	0.000	0.000
19	1987	chronic	Alpha	Constant	43.800	0.000	0.000
20	1988	chronic	Alpha	Constant	43.300	0.000	0.000
21	1989	chronic	Alpha	Constant	42.600	0.000	0.000
22	1990	chronic	Alpha	Constant	41.900	0.000	0.000
23	1991	chronic	Alpha	Constant	41.300	0.000	0.000
24	1992	chronic	Alpha	Constant	40.700	0.000	0.000
25	1993	chronic	Alpha	Constant	39.900	0.000	0.000
26	1994	chronic	Alpha	Constant	39.300	0.000	0.000
27	1995	chronic	Alpha	Constant	38.600	0.000	0.000
28	1996	chronic	Alpha	Constant	38.100	0.000	0.000
29	1997	chronic	Alpha	Constant	37.400	0.000	0.000
30	1998	chronic	Alpha	Constant	36.700	0.000	0.000