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Technical Basis Document for Internal Dosimetry at Sandia National Laboratories

Revision 3

Sandia National Laboratories
Radiation Protection Internal Dosimetry (RPID) Program
Department 7578
P. O. Box 5800, MS-0651
Albuquerque, NM 87185-0651

Prepared by: _____ _____
C. A. Potter, RPID Project Leader Date
Personnel Monitoring and Laboratory Services

Reviewed by: _____ _____
I. J. Frazier, QA Engineer Date
Personnel Monitoring and Laboratory Services

Approved by: _____ _____
D. J. Thompson, Manager Date
Personnel Monitoring and Laboratory Services

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Introduction

Project Mission

The mission of the Sandia National Laboratories, (SNL) Radiation Protection Internal Dosimetry (RPID) Project is to detect, assess, document, and provide appropriate response to occupational exposures from internally deposited radioactive materials originating from SNL activities. This technical basis document (TBD) is designed to meet these objectives by providing guidance on the following:

- define the project participation requirements,
- describe the use of defensible monitoring methods,
- determine appropriate analytical techniques and associated quality control/assurance measures,
- describe a defensible approach to determine the internal dosimetry of detected exposures,
- develop reporting levels and define appropriate responses to unexpected exposures,
- describe the documentation control and quality control/assurance requirements, and
- provide a technical basis for the generation of procedures.

Project Scope

The RPID Project is implemented at all SNL facilities for activities involving the processing and/or storing of radioactive materials. Reference to SNL throughout this document includes all SNL facilities and activities.

Internal Radiation Protection Elements and Strategy

The information in this section is taken from "Evaluation of Radiation Protection Internal Dosimetry Program Manual & Procedures for Sandia National Laboratories" by K. W. Skrable (Skrable 1996).

Radiation Protection Program Elements

- Adequate primary radiation protection limits [10CFR835]
- Qualified staff
- Identification of potential exposure pathways and engineering controls to maintain exposures ALARA
- Adequate design, construction, operation, and maintenance
- Methods and procedures to maintain the combined dose from internal and external sources ALARA
- Monitoring and survey programs for the facility and surrounding environment
- Prospective job specific evaluations of potential external and internal exposures and selection of protection measures that maintain the collective total effective dose equivalent ALARA (TEDE ALARA)
- Retrospective assessments of radiation exposures and doses
- Investigation, action, recording, and other reference levels
- Emergency plans and procedures
- Adequate records
- Quality assurance and audit programs

The effectiveness of a radiation protection program depends on the type of primary radiation protection limits chosen for the first element because all other elements are influenced by this most important element of the program. The 8th item includes operational, routine, and termination bioassays, the final quality

control procedures used to assure that workers have received adequate protection from internal radiation sources, and special bioassays used for estimating intakes, exposures, and doses.

Elements of an Internal Radiation Protection Program

- Evaluating types and quantities of radionuclides representing potential or real exposures of workers
- Determining physical and chemical forms of radionuclides, including their specific activity, which can greatly influence the potential of generating airborne radioactivity
- Establishing requirements based upon potential of real exposures for:

  (1) area monitoring,

  (2) personal air sampling, and

  (3) bioassay, including criteria for selection of individuals and frequencies, for timely detection and accurate assessment of both acute and chronic exposures.

- Establishing performance criteria for the area monitoring and bioassay programs including:

  (1) required minimum detectable concentrations, exposures, intakes, and committed doses.

  (2) required precision and accuracy.

- Establishing the area monitoring program for the working environment in conjunction with and in support of the program including:

  (1) CAMs that provide adequate warning [of danger from radioactive particulate or gaseous releases (requirements specified in 10CFR835)]],

  (2) air sampling and analysis,

  (3) contamination control surveys,

(4) personal air samplers for monitoring of exposures received by individual workers, and

(5) action levels for modifying or initiating routine, operational, and special bioassay procedures.

- Selecting routine and operational bioassay protocols [ICRP, 1988] and monitoring frequencies that provide the sensitivity and accuracy needed to assure that workers are receiving adequate protection that warrant corrective actions and accurate assessments of intakes, exposures, and committed doses including:

  (1) the physically significant activity (PSA) above which a bioassay result is considered to be statistically significant:

    (a) for a specified probability of a false positive, and

    (b) based on measurements of background and appropriate blanks, including baseline data for the worker population when plant specific radionuclides in bioassay samples include contributions from naturally occurring radionuclides in the diet, etc.;

  (2) the minimum detectable activity for a specified probability for detection above the PSA;

  (3) the intake retention fraction (IRF) of fraction of an intake expected to be present in the bioassay measurement, and

  (4) the minimum detectable intake (MDI) the minimum detectable exposure (MDE) and the minimum detectable committed effective dose equivalent MDH_E:

- Selecting appropriate reference levels for the area monitoring and bioassay programs including levels for:

  (1) recording of exposures, intakes, and doses,

  (2) control actions for the work environment,

(3) initiating investigations,

(4) implementing baseline, operational, routine, termination and special bioassay procedures,

(5) work restrictions for diagnosis, and

(6) medical intervention.

- Establishing criteria and or procedures for:

  (1) selection if individuals who will participate, and

  (2) determination of worker's baseline, background, and burden of radionuclides that could cause interferences.

- Training management, workers, and staff regarding purposes, scope, and requirements of internal radiation protection program including:

  (1) methodology and capability of personal air sampling and various bioassay procedures, including a comparison of their timeliness, physically significant quantities, and minimum detectable quantities

  (2) possibility of medical intervention and work restrictions for diagnosis

  (3) use of special bioassay procedures even when magnitude of an exposure might not be warranted by regulatory requirements.

  (4) interpretation of bioassay and PAS data in terms of intakes, exposures, average airborne concentrations, and doses

- Developing requirements for records, calibration, quality assurance, and audit program

Project Implementation Strategy

Activities at SNL facilities involve processes handling radioactive materials which may result in occupation internal exposures. This document provides the technical basis for implementing the RPID Project which is designed to monitor these exposures. Because SNL operations do not commonly involve long-term,

continuing operations with the same materials, usual internal dosimetry practices may not apply. Therefore, internal dosimetry requirements are implemented through Radiation Work Permits (RWP).

Project Requirements

10 CFR 835 (U.S. DOE 1993) provides specifications and requirements for internal dosimetry projects developed for DOE facilities. 10 CFR 835 states occupational annual dose limits, requires dose assessment, and defines participation, reporting, recording, and quality control requirements. Reference to radiological workers in this technical basis document includes SNL employees and the employees of companies under contract to SNL. Mandatory requirements in 10 CFR 835 are enforceable by law and are identified through the use of shall. Requirements contained in the RCM are mandatory to the extent they are incorporated by a contract or through administrative means.

Monitoring Techniques

Personal Air Sampling

Personal air sampling (PAS) is the primary indicator of intakes at SNL. This is accomplished by use of a lapel or breathing-zone air sampler, which is a device consisting of a sampling pump usually attached to a belt, with a hose connecting to a filter assembly. The filter assembly is attached to the individual's clothing such that it is in or near the worker's breathing zone. These units commonly operate at 2 to 10 lpm flow rate. Since this device provides a real-time measurement of the worker's exposure, it is an excellent indicator of intake and whether further actions need to be taken. For some radionuclides where detection by routine bioassay is not possible at DOE imposed monitoring levels (U.S. DOE 1993) personal air sampling is used to assess intake and the resulting dose equivalent.

Analysis of the filter itself depends on the characterization of the environment in which the individual is working. If the environment is well-characterized, gross-alpha/beta counting is adequate, even for radionuclides with low annual limits on intake (Eckerman et al. 1998). However, if this is not the case, other analytical methods are possible. The filters can be analyzed by gamma-spectroscopy, or digested, electroplated, and subsequently analyzed by alpha-spectroscopy. Direct alpha-spectroscopy is also possible by taking into account self-absorption in the filter itself.

Personal air sampling measures the actual exposure that an individual has received. If using these measurements to infer intake of radioactive materials, error introduced will be that from the difference between Reference Man's (ICRP 1975) breathing rate and that of the actual exposed individual.

In-Vivo Bioassay

In-vivo measurements are possible for radionuclides which emit photons with sufficient energies to penetrate the body and, therefore, be detected outside of the body (e.g., ^{60}Co). These measurements provide direct values of the content of radionuclide in the organ or group of organs being measured. The sensitivity and accuracy of in-vivo techniques may be limited by the following:

- Background radionuclide interference within the subject (e.g., ^{40}K, ^{137}Cs, etc.),
- Variability of photon attenuation and radionuclide deposition patterns between the measured worker and the calibration geometries,
- Potential errors caused by the presence of skin contamination on the subject,
- Background interferences from environmental sources (e.g., cosmic rays, radon progeny, etc.), and
- Use of medical radionuclides for diagnosis or therapy including use of those radionuclides commonly found at SNL.

These limitations are less significant for higher energy emitting radionuclides (e.g., ^{60}Co, ^{131}I). In-vivo measurements may also be used for monitoring insoluble forms of low-energy photon emitters deposited in the lungs (e.g., oxides of transuranics, etc.) which are often difficult to measure using in-vitro techniques.

In-vivo techniques are primarily used for routine surveillance of workers to detect potential exposures. Organ-specific contents sometimes can be evaluated when the counting equipment is properly calibrated for the geometry of the organ. In some cases, the corresponding dose and retention patterns can be determined in these situations. In-vivo counting systems are generally classified as whole-body counters (WBC) and other type counters.

Whole-Body Counting

Several counting systems are available for WBC, ranging from a single, unshielded detector to multi-detector arrays used in shielded laboratories designed to minimize the influence of environmental background radionuclides. Detectors available for in-vivo measurements include inorganic crystals (e.g., sodium iodide, phoswhich, etc.), solid state (e.g., high purity germanium), gas-flow proportional, and organic scintillators. Selection of the appropriate equipment is dependent on the required sensitivity, number of radionuclides potentially present, and the photon emission characteristics.

The Personnel Monitoring and Laboratory Services (PMLS) Department maintains an in-vivo bioassay capability using a high-purity germanium (HPGe) shadow-shield detection system. This system is capable of reliably quantifying body burdens of high energy photon emitting radionuclides (i.e., photon emissions greater than 250 keV). In addition, the PMLS system provides a qualitative spatial distribution of internal depositions. The use of this system is emphasized in the RPID Program.

Wound Counting

Another important use of organ-specific counters is the assessment of wound exposures. Depositions of soluble materials would be quickly transported from the wound site and may be quantified using in-vitro techniques. However, insoluble depositions may remain at the wound site and can be assessed using an appropriate detector. The information obtained is useful to evaluate the need for medical intervention (e.g., tissue excision) since insoluble materials will be eventually absorbed if permitted to remain. SNL maintains an HPGe detector which is used to assess wound depositions.

Other

Organ counters can be used to detect and quantify radionuclide depositions in specific organs. Examples of these systems are lung counters and thyroid probes. Lung counters are designed to determine body burdens of insoluble particles deposited within the lungs. Such depositions are slowly eliminated from the body and may be difficult to detect using in-vitro bioassay methods. The specialized facilities required for determining lung burdens are not currently available at SNL/NM. Suspected lung exposures will be assessed by an outside.

Radionuclide compounds containing iodine may be selectively collected and retained in subject's thyroid gland. Thyroid burdens are determined by placing a radiation detector close to the subject's neck. The neck measurements can then be compared to the results from a standard thyroid phantom containing the radionuclide of interest (e.g., ^{131}I, ^{125}I, etc.). A thyroid counting station is maintained in the Technical Area V counting laboratory.

Bone seeking radionuclides can be detected, and sometimes quantified, by measuring limb or skull activities. Skull counting may be particularly useful

because of the lack of attenuating tissue, the large radius of curvature allows close approach of detector, and the large skeletal mass associated with skull. However, calibration of these detection systems is difficult, and measurements require specialized facilities which are not available at SNL/NM

In-Vitro Bioassay

In-vitro techniques indirectly determine the radionuclide body burden by measuring biological samples outside of the body. In-vitro techniques are performed under controlled conditions (e.g., low radiation backgrounds, constant measurement geometries, etc.) which result in lower analytical sensitivities compared to in-vivo techniques. In addition, radionuclides which do not emit penetrating photons can be assessed. However, the accuracy of in-vitro assessments is limited by the uncertainties associated with the biokinetic models used. Potential uncertainties include:

- Extrapolation of biokinetic data from animals to humans,
- Variability among individuals (i.e., deviances from "Reference Man" assumptions),
- Effects of the individual's health on biokinetic patterns,
- Establishing the correct chemical and physical form of the radionuclide compound,
- Establishing time of exposure and duration of exposure (i.e., chronic versus acute),
- Using models based on data from similar compounds,
- and, Analytical measurement uncertainties.

Biokinetic models are typically catenary kinetic models with constant coefficients and removal rates. From these models, uptake retention functions can be developed, usually in the form of a sum of exponential terms. Excretion is determined from these models by treating the excretion as a compartment at the end of the catenary chain. By using kinetics, i.e. forming compartmental chains for

each pathway from intake to excretion and deriving equations describing the excretion from each pathway, an intake value can be calculated from the excretion results. Least-squares or some other type of data fitting can be used to give the best-fit intake value when there are multiple data points and associated errors.

Urinalysis

A fraction of the intake that is absorbed into systemic circulation, either directly after removal from the intake compartment (lung, GI tract, etc.) or after absorption by a systemic organ or tissue, is removed by the kidneys and discharged from the body in urine. This removal fraction is dependent on the compound's chemical form and is subject to temporal changes as deposition locations release materials by kinetic processes into the systemic circulation. The magnitude of the intake and its subsequent retention in the body can be determined indirectly from urinalysis results.

Urinalysis is the most common in-vitro bioassay technique since samples can be easily collected and their subsequent analysis can be extremely sensitive. Considerable variability may be observed when using accepted biokinetic models to evaluate urine samples (ICRP 1979). Sources of uncertainty include duration of exposure, retention of radionuclides from previous exposures, and the effects of medical intervention techniques (e.g., chelation therapy, etc.). However, the advantages of urinalysis can be seen when the intake is of a radionuclide that has no mid or high energy gamma emissions. Detection limits for alpha and beta emitting radionuclides by alpha-spectroscopy and proportional counting respectively are usually more adequate than in-vivo counting for low-energy gamma photons or x-rays.

Urinalysis is the primary detection method for intakes of ^{3}H and ^{3}H compounds at SNL.

Fecal Bioassay

Many radionuclides are eliminated from the systemic body within feces as well as within urine. Therefore, fecal analysis provides another means in assessing body burdens. Many of the urinalysis principles are also applicable to fecal analysis. However, radionuclides can be transferred to the feces from direct excretion through the GI tract from deposition in the stomach or from deposition in

lung compartments that is transferred up the trachea by ciliary action and swallowed as well as from systemic organs and tissues. An understanding of these pathways is necessary for interpreting fecal analysis results.

Insoluble compounds, which are not readily absorbed by the small intestine (e.g., oxide forms), will have a large component that is eliminated in the feces from fractions of the initial deposition in lung compartments where particles are removed to the GI tract. Soluble compounds (e.g., tritiated water, ionic iodine and cesium, etc.) may be difficult to detect, unless there is a large fraction that is removed from the systemic body through the biliary excretion pathway. The predominant respiratory elimination mechanism of insoluble compound forms in the early post-exposure period (i.e., 1 to 2 days after exposure) will be from particle transport from the lung to the GI tract. This allows for early fecal samples to be taken as a very sensitive indicator of intake. However, due to uncertainties in the GI tract model at early times after exposure, the early data may not be useful for intake evaluation.

Comparing fecal and urinalysis results may provide information regarding the time of exposure. However, there are several technical and practical limitations of using fecal analyses within a routine bioassay program. These limitations include the following:

- Difficulty in collecting a true 24-hour sample because of the temporal variability of fecal output,
- Inability to adjust output fluctuations to represent 24-hour samples (i.e., fecal samples lack indexing measures),
- and, Collection and analysis of fecal materials is more objectionable in practice compared with other bioassay procedures.

However, if all feces are collected from time of intake to a particular time post intake, these incremental samples can be added together to represent accumulated samples. This eliminates the difficulties in the first two bullets. Also, for highly insoluble (Class Y) compounds, two weeks after intake all short-lived lung compartments from which particles translocate to the GI tract will have emptied themselves of the initial deposition. Therefore, all particles translocating directly to the GI tract will be from the initial deposition in lung compartment "g."

If the fraction of systemic excretion going to feces is low and the fraction of material in the GI tract translocated to the transfer compartment is low for this particular element, fecal excretion is very easily modeled.

At SNL, fecal analysis is used to evaluate inhalation exposures to insoluble radionuclide compounds resulting from an accident situation. The RPID Program maintains a capability for fecal analysis using specialized collection kits and a contract analytical laboratory.

Other

Nasal smears are easily collected by swiping the nasal passages with filter paper wrapped around a cotton swab. Deposition of radionuclide compounds in nasal passages are not well understood and detailed biokinetic models are unavailable to quantify body burdens from nasal smear samples. However, the detection of radioactive materials is useful to initiate follow-up internal dosimetry measures. Nasal smears are used to evaluate detected accidental inhalation exposures in the RPID Program.

Radionuclide compounds may also be exhaled from the body. Body burdens of ^{226}Ra and ^{232}Th can be estimated using breath analysis since gaseous progeny are produced from these radionuclides. Measurement of these noble gases require specialized procedures and equipment which are not routinely available at SNL/NM. Tritiated and ^{14}C labeled compounds metabolized into tritiated water and carbon dioxide can also be detected by breath analysis. However, the collection of samples and subsequent intake estimation of these radionuclides are more easily accomplished using other techniques (e.g., urinalysis). Routine breath analysis is not performed in the RPID Program for these reasons.

The direct measurement of radionuclide materials within the systemic circulation can be made by collecting blood samples. A potential advantage of blood analysis, compared to urinalysis, is that body burdens can be estimated without the uncertainty inherent in standard renal function models. However, radionuclide compounds are rapidly deposited within body organs. These body burdens may not be detected by a single blood sample and may lead to incorrect exposure assessments. 24-hour blood samples would be more representative but are impractical due to the limits on the amount of blood which can be withdrawn from an individual. Consideration must also be given to the potential presence of blood-

borne pathogens (e.g., hepatitis-B, etc.) which present significant biohazards to sampling and analytical personnel. For these reasons, blood analysis is not used in the RPID Program.

After reaching the systemic circulation, some radionuclides will eventually be incorporated in certain body tissues where they can be readily detected. Quantifying uptake from these measurements is generally not possible since biokinetic models have not been developed for many of these in-vitro techniques. However, they may be useful as an indicator of exposure. For example, facial and scalp hair analyses have been useful in detecting exposures from radon and thoron. Fingernails are also integrators of exposure for some radionuclides. Analysis of saliva and perspiration samples may reflect systemic body burdens. These alternate methods are not expected to be used in the RPID Program since other in-vitro techniques (e.g., urinalysis) provide similar, or superior, information.

Timing of In-Vitro Bioassay Samples

At SNL, routine bioassay samples are often collected for analysis of multiple radionuclides. Activity is not expected to be present in routine bioassay samples (except tritium and radionuclides present from fallout and natural sources) because other indicators such as activity on a personal air sample or contamination would indicate if intake was to be suspected. Therefore, for routine and confirmation samples, individuals are instructed to fill the entire 1500 ml urinalysis kit. However, in the case of special monitoring 24-hour samples are recommended. NCRP Report No. 87, *Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition* (NCRP 1987) suggests that samples not collected over a 24-hour period may not be representative due to differences in fluid intake over the period.

When urine samples are being collected for incident purposes, SNL requests true 24-hour samples. These are obtained by beginning the time period immediately after the first void post incident, which is not collected, and collecting for a full 24 hours continuously after that. Fecal samples are obtained on a one stool per fecal kit basis for as long as is required.

Monitoring Programs

Bioassay monitoring programs for internal dosimetry are defined and implemented using terms from ICRP Publication 54, *Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation* (ICRP 1988) and ICRP Publication 35, *General Principles of Monitoring for Radiation Protection of Workers* (ICRP 1982). These categories are operational monitoring, routine monitoring, special monitoring, and confirmatory monitoring. Implementation of reference levels are used as defined in ICRP-54 except where DOE requirements are different than the international standard. At SNL, operational monitoring and routine monitoring are used only for ^{3}H. This is because personal air sampling is better for other radionuclides used at SNL. The two documents listed above are quoted from liberally in this section.

ICRP-35 (ICRP 1982) defines Working Conditions A and B where Working Condition A describes conditions where individuals could receive a dose equivalent greater than 3/10 of the "relevant annual limits." Working Condition B describes conditions where it is unlikely that individuals could receive that level of dose. Routine monitoring is recommended by ICRP in the case of workers in Working Condition A, setting a monitoring threshold of 3/10 ALI intake. DOE's occupational monitoring requirement, as stated in 10CFR835.402(c) is "0.1 rem or more committed effective dose equivalent, and/or 5 rems or more committed dose equivalent to any organ or tissue, from all occupational radionuclide intakes in a year…(U.S. DOE 1993)" In contrast to the ICRP recommendation, DOE actually gives two monitoring requirements, the first being 0.02 ALI intake for stochastic effects and the second being 0.1 ALI intake for non-stochastic effects. When comparing these two limits, it is found that most, if not all, radionuclides will be limited by the stochastic limit. This requires recalculation of ALIs and DACs for radionuclides where the ALI is based on the non-stochastic limit.

General Descriptions and Philosophies of Monitoring Programs

Radiological workers, most likely those working frequently with large amounts ^{3}H, may be required to participate in either routine or operational monitoring in the RPID Program. Routine monitoring is designed to confirm that radiological controls are adequate and to detect unknown radionuclide intakes. The

purpose of operational monitoring is to detect intakes occurring during a particular operation.

The internal dosimetry monitoring requirement as stated in 10CFR835.402(c) is as follows: "For the purpose of monitoring individual exposures to internal radiation, internal dose evaluation programs (including routine bioassay programs) shall be conducted for:

"(1) Radiological workers who, under typical conditions, are likely to receive 0.1 rem (0.001 sievert) or more committed effective dose equivalent, and/or 5 rems (0.05 sievert)or more committed dose equivalent to any organ or tissue, from all occupational radionuclide intakes in a year;

"(2) Declared pregnant workers likely to receive an intake resulting in a dose equivalent to the embryo/fetus in excess of 10 percent of the limit stated in 835.206 [0.5 rem]; or

"(3) Minors and members of the public who are likely to receive, in 1 year an intake resulting in a committed effective dose equivalent in excess of 50 percent of the limits stated in 835.207 [0.1 rem] or 835.208 [0.1 rem] respectively (U.S. DOE 1993).

Bioassay monitoring and personal air sampling at SNL is designed to detect internal radionuclide exposures at the monitoring levels given above. Individuals who have the potential to exceed these dose levels participate in one of these depending on the radionuclide to which they are exposed.

Routine Monitoring

The general description of a routine monitoring program is taken from ICRP-35 as follows:

> Routine individual monitoring constitutes regularly repeated or continuous measurements made on an individual worker. If the estimated annual dose equivalents or intakes are well below the relevant dose equivalent limits or annual limits of intake recommended by the Commission, it will be sufficient to assess the upper limits of the estimate rather than the actual values and to assess their importance by the use of, for example, investigation levels. Since this may lead to an overestimate of the dose equivalent to the group of workers involved, this limitation in accuracy should be borne

in mind when routine monitoring results are to be used for purposes of optimization of radiation protection. In instances where routine individual monitoring techniques or instrumentation are not capable of facilitating the estimates of dose equivalent or intakes for individuals with the necessary confidence, programs of monitoring of the workplace may have to be instituted to provide estimates of the relevant values…

The most important factors influencing the frequency of routine measurements are the distribution pattern in time of the intake of radioactive material, the residence time of the contaminant in the body and the sensitivity of detection in relation to the appropriate derived investigation and recording levels. If routine measurements are needed and are initiated to assess personal intakes or dose equivalents, they should be spaced so that all significant intakes will be detected. In some situations routine estimation of intakes and associated dose equivalents is not necessary once satisfactory working conditions have been established, which can be confirmed by routine monitoring of the workplace. Furthermore, slow deterioration of conditions can be conveniently detected by making infrequent measurements of appropriate samples. These measurements are not intended to assess personal dose equivalents and their frequency is determined by the working conditions. For materials of very short effective half-life, such as insoluble materials in the gastrointestinal tract, routine monitoring would have to be so frequent as to be impracticable. Fortunately, adequate control can be maintained over such materials by monitoring the workplace…

For materials of long effective half-life in the body, the amount retained after an intake of one ALI is usually much less than can be measured either by instruments external to the body or by the assay of body fluids. The interval between measurements is then governed by the need to make a periodic check of the long-term build-up of radioactive material in the body. There is no practicable system of routine individual monitoring for materials of long effective half-life that can be used to provide early detection of unsatisfactory conditions

in the workplace. For this purpose, monitoring of the workplace is necessary (ICRP 1982).

ICRP-54 adds: "Routine monitoring involves regular measurements on individual workers. Routine individual monitoring for intakes of radionuclides is only required in conditions of essentially continuous risk of contamination of the workplace as a result of normal operations. Since routine monitoring is conducted at predetermined times not related to known intakes, it is necessary to assume a pattern of intake in order to interpret the measurement in terms of intake or committed dose equivalent. (ICRP 1988)"

Operational Monitoring

The general description of a routine monitoring program is taken from ICRP-35 as follows: "Operational individual monitoring, that is individual monitoring during particular operations, for example by the provision of additional dosimeters, may contribute significantly to operational monitoring programs, especially if the devices used incorporate direct-reading or alarm functions. However, such monitoring is limited in time in that it will be implemented for a particular operation or a series of operations. In certain instances it may be useful to undertake individual operational monitoring to establish whether routine individual monitoring is required. (ICRP 1982)"

ICRP-54 adds: "Operational monitoring is intended to provide information about a particular operation. As regards individual operational monitoring for intakes of radionuclides, there is likely to be more specific information than in the case of routine monitoring. In particular, the time and duration of potential exposure is known and there will be some information about the physical and chemical nature of the potential contaminant and the likely route of intake. (ICRP 1988)"

Special Monitoring

The general description of a routine monitoring program is taken from ICRP-35 as follows:

> Special individual monitoring should be carried out in actual or suspected abnormal conditions including accidents…

> Special individual monitoring should be initiated when the results of monitoring in the workplace indicate that significant intakes may have occurred or when workers have been associated with known accidents possibly involving significant intakes of radioactive materials. Simple tests of skin contamination and nose blows will sometimes eliminate the need for further immediate detailed studies. Only experience can tell what circumstances require a special monitoring program, and it is therefore extremely important to review the results of such programs in relation to the situation initiating them.
>
> Special monitoring may be used with advantage for studying human metabolism of radionuclides. This will often justify measurements at lower activity levels in the body than those of direct concern in individual monitoring. This type of study should be encouraged (ICRP 1982).

The general description of a routine monitoring program is taken from ICRP-54 as follows: "Special monitoring refers to monitoring carried out in actual or suspected abnormal conditions. In the case of monitoring for intakes of radionuclides, special monitoring may be triggered by a known event, such as a recognized loss of contaminant by an alarm such as that given by an automatic air sampler, by recognized unusual conditions in the working environment or by an unusual result obtained during individual monitoring. (ICRP 1988)"

Confirmatory Monitoring

The general description of a routine monitoring program is taken from ICRP-54 as follows: "Confirmatory monitoring refers to monitoring carried out to verify satisfactory working conditions of workers who are not thought likely to be exposed to significant intakes of radionuclides (ICRP 1988)"

Implementation of Monitoring Programs

Routine Monitoring

Baseline Sampling

The purpose of baseline sampling is to identify and quantify any possible interferences that may affect future bioassay measurements. Many of the radiological workers at SNL/NM have been previous employed at other facilities involved with production, processing, and storage of radioactive materials. The knowledge of all previous exposures is essential for correct assessment of dose in the RPID Program. In addition, quantifying the chronic intake of naturally occurring radionuclides (e.g., members of the uranium and thorium decay series) may be needed to discriminate between occupational and background exposures. Individuals obtaining dosimeters at SNL are requested to fill out whether they know of any previous occupational intakes they may have had. If they respond in the affirmative, baseline sampling is performed.

Baseline samples may include urinalysis or whole-body counting, depending on the radionuclides involved. Urinalysis provides sufficient sensitivity to detect prior exposures to most radionuclides. When possible (i.e., radionuclides with < 250 keV gamma emissions), individuals are also given whole-body counts.

Routine Sampling

At this time, SNL operations involving the use of radioactive materials consist of widely varying operations and experiments that use many different radionuclides. This makes development of routine sampling programs extremely difficult. Therefore, personal air sampling is used both as an indicator of intake and a bioassay method when the exposure received is below the minimum detectable concentration for the particular radionuclide. The technical basis for this is discussed in detail in Section V. Personal air sampling is not useful for ^{3}H, however, so routine sampling by bioassay is performed at SNL and discussed here.

Bioassay for ^{3}H is carried out as recommended in American National Standard HPS N13.14-1994, *Internal dosimetry Programs for Tritium Exposure—Minimum Requirements* (HPS 1994). This standard recommends a tritium bioassay program "unless it can be demonstrated that the total intake in one year by any worker would rarely exceed 0.1ALI…" While this conforms with NRC requirements for radiological workers, it does not comply with 10CFR835 requirements.

Therefore, the program is designed using philosophies in the standard, with action levels modified to conform with 10CFR835 requirements. Table 1 below is modified from Table 1 in HPS N13.14-1994. The standard states that working under conditions and with tritium amounts less than what is stated in the original table will not result in an intake of more than 100 μCi. Assuming a working year of 50 weeks and modifying the intake to that corresponding to a 100 mrem dose gives the routine monitoring requirements listed in Table 1.

The monitoring frequency for tritium based on the standard is biweekly. A frequency of this length minimizes dose equivalent that may be missed over the course of a year due to detection limits while allowing for relatively timely detection of intakes.

Table 1. Tritium Levels Requiring Bioassay (HPS 1994)

Type of Operation	HTO and tritiated organics including DNA precursors	Tritium Gas in sealed process vessels	HTO mixed with more than 10 kg of inert H_2O or other material
Processes in open room on bench top	0.03 Ci	30 Ci	3 mCi/kg
Processes in a fume hood with adequate face velocity and performance reliability	0.3 Ci	0.3 kCi	30 mCi/kg
Processes in a glove box	3 Ci	3 kCi	0.3 Ci/kg

Operational Monitoring

While personal air sampling is the preferred method for both routine and operational monitoring of individuals for internal exposure, operational bioassay may also be implemented. Bioassay requirements are indicated on Radiological Work Permits (RWPs). This is based on the potential for intake of 100 mrem over

the course of the job. There are two types of RWPs - general and short-term. General RWPs are used to control routine or repetitive activities such as, tours and inspections, in areas with well-characterized and stable radiological conditions. General RWPs may be approved for periods up to one year. Job-specific RWPs are used to control non-routine operations or work in areas with changing radiological conditions. Job-specific RWPs remain in effect for the duration of the job and shall not be written initially for a period to exceed three months.

Operations performed under either type of RWP have the potential of exceeding the monitoring limit. When reviewing operations for internal dosimetry requirements, the Brodsky criteria are used (Brodsky 1980):

> Empirical experience indicates that with facilities, equipment, and procedures currently in effect for protection against radiation, the following probabilities (or fractional amounts) may be assumed to usually remain $<10^{-6}$.
> (a) The fractional amount of material handled that is inhaled by a worker in an accident or explosion.
> (b) The fractional amount of radioactivity placed into process in routine operations that will enter the body of any worker, averaged over an extended period (e.g. 1 yr)
> (c) The fractional amount of contamination on 1 m^2 of floor or ground that will enter 1 m^3 of air and be respirable by any person (over an extended period of time) either outdoors within large contaminated areas, or indoors with smaller contaminated areas.
> (d) The fractional amount of material released from a building that will be inhaled by someone 800 m away, even under the most severe hypothetical conditions.
> (e) The fractional amount of material released from a building or stack that will deposit on 1 m^2 of ground 800 m or more away.
> (f) The fractional amount of material released into a reactor containment building that will be released outside the building in the first place (except for the rare gases).
> (g) The fractional amount of material in transit that will enter the body of any person, even in the event of a serious transportation accident and/or fire.

Another consideration is existing airborne conditions. When determining requirements, respiratory protection (i.e., protection factors) are taken into account. If respiratory protection is worn, there may be no internal dosimetry requirements except that if other indicators show potential for intake, RPID is contacted for further action. This would be treated as special monitoring (see below). Indicators that may exist showing possible intake are positive nasal wipes and swipes of the inside of respirator masks after use showing contamination.

If an evaluation of the job shows potential for intake over 100 mrem over the duration of activities on the RWP, operational monitoring may be required in addition to personal air sampling. This is up to the discretion of the Radiological Control Technician, the RP Team Supervisor, and the RPID Project Leader. If personal air sampling results show that individuals have had exposures indicating the possibility of intake greater than the monitoring limit, those individuals will be monitored as required.

Special Monitoring

Guidelines for When an Intake is Suspected

Guidelines below for special monitoring are taken from *The Savannah River Site Internal Dosimetry Technical Basis Manual* (SRS 1994). These airborne levels, contamination levels, etc. are based on experience with incidents involving radioactive contamination at SRS. An intake is suspected if any of the following situations has occurred:

- Positive nasal smear or contamination inside a respirator mask.
- A worker is exposed to airborne radioactivity in excess of 8 DAC-h in a day or the indicated air concentration could greatly underestimate that to which the worker was exposed. The values for air concentration and exposure assume the protection factor for any respiratory protection in use will be applied.
- Contamination is measured on a single-layer protective clothing in excess of 10,000 d/m per 100 cm^2 alpha or 100,000 d/m per 100 cm^2 beta-gamma if respiratory protection is not in use. Respiratory protection refers to any filtered-air or supplied-air respirator. Protective clothing refers to coveralls, shoe covers, labcoats, etc.

- Contamination is measured on the inner layer of multiple-layer protective clothing in excess of 10,000 d/m per 100 cm^2 alpha or 100,000 d/m per 100 cm^2 beta-gamma if respiratory protection is not in use.

- An unplanned release of radioactive material produces contamination on accessible surfaces in excesses of 1500 d/m per 100 cm^2 alpha or 15,000 d/m per 100 cm^2 beta-gamma if respiratory protection is not in use.

- Any detectable personal contamination is measured on the hair, face, neck, chest, arms, or hands, or anywhere else on the body in excess of 1000 d/m per 100 cm^2 alpha or 10,000 d/m per 100 cm^2 beta-gamma if respiratory protection is not in use.

In addition to the above guidelines there may exist additional factors determining if bioassay is or is not necessary. Therefore, administration of bioassay is left up to the discretion of the dosimetrist.

Actions for Suspected Intakes

The following describes the types of monitoring and methods of evaluation of early data for intakes of radioactive material. After immediate actions are performed, the dosimetrist may prescribe continuing sampling over a protracted period both to monitor the removal of the radionuclide from the body and to refine the intake and dose estimates.

Using Work Area Information

At early times after an incident has occurred involving possible intake of radioactive material, the only information available for any inference of possible magnitude of intake may be work area information. This may come in the form of airborne radioactivity levels, surface contamination levels, or personal contamination levels. An intake may be inferred directly from airborne radioactivity levels depending on the situation. Other types of contamination values may not prove useful. However, nasal and/or mouth swabs may be useful as indicators of intake.

The ICRP-30 lung model (ICRP 1979) assumes 30% deposition in the nasal passages. For all lung classes translocation out of the nasal passage occurs relatively quickly. Depending on lung class, a fraction is removed from the nasal passage with a clearance half-time of 0.01 days (14.4 min) and an additional

fraction may be removed with a clearance half-time of 0.4 days (9.6 hr). This information may be used to infer the relative magnitude of exposure and intake.

Table 2 lists intake retention fractions for the nasal passage for times up to 8 hr after intake. These values can be used to infer an exposure from nasal swipe values. One must keep in mind that because of the uncertainty involved in this nasal model including the fact that intake through the mouth is not considered. Therefore, any exposure inferred from these values should be used only to consider further actions, and not to assess intake and dose.

Table 2 - Intake Retention Fractions for Nasal Passages

Time After Intake (h)	Class D	Class W	Class Y
0	**1E0**	**1.00**	**1.00**
1	**5.6E-2**	**0.84**	**0.92**
2	**3.1E-3**	**0.78**	**0.86**
3	**1.7E-4**	**0.72**	**0.80**
4	**9.6E-6**	**0.67**	**0.74**
5	**5.4E-7**	**0.63**	**0.69**
6	**3.0E-8**	**0.58**	**0.64**
7	**1.7E-9**	**0.54**	**0.60**
8	**9.2E-11**	**0.51**	**0.56**

Sampling for Alpha Emitters

Detection of pure alpha emitters using in-vivo methods is difficult, but possible for radionuclides that reside for long periods of time in the lung. Lung counting is commonly done for individuals having intakes of Class Y uranium or plutonium. This type of in-vivo counting detects low-energy X-rays that penetrate the chest wall. When performing this type of measurement, chest wall thickness must be taken into account carefully since small changes in thickness can result in relatively large changes in efficiency. MDAs for this type of measurements are usually in the range of 10s of nanocuries. If excreta measurements are taken in a timely way, these can be more sensitive, especially fecal bioassay. At this time,

SNL does not have a facility to perform lung counting and does not recommend this type of analysis, except in special cases.

A suspected intake of an alpha emitter will trigger bioassay in the form of both urine and fecal bioassay. Without other indicators of intake less than 50 mrem, continuous sampling should continue until preliminary results are received on early samples. When an initial estimation of intake can be determined, decisions involving further sampling can be made.

Sampling for Beta/Gamma Emitters with Gamma Energy > 200 keV

Beta emitters with associated high energy gamma rays can be easily detected using the SNL whole-body counter. Individuals who have possibly had an intake of one of these radionuclides should be whole-body counted as soon as possible. Results of this analysis can be used to make an immediate determination of the probable magnitude of intake. This can then be used to designate whether further whole-body counting and/or bioassay sampling is necessary.

Sampling for Pure Beta Emitters or Beta/Gamma Emitters with Gamma Energy <200 keV)

At SNL, radionuclides in this category are usually the pure beta emitters strontium and yttrium. Normally these two radionuclides are detected in secular equilibrium. Detection of Sr is quite difficult and takes typically 14 days to quantify accurately. Preliminary counts can estimate concentrations within an order of magnitude. Sr is easily detected in both urine and feces. Sampling should be performed as for alpha emitters above.

Strontium can be detected by counting by liquid scintillation. SNL is currently developing a procedure to directly detect Sr in urine using liquid scintillation. In the absence of this procedure, the following method may be used to infer a concentration of Sr in the sample. Gamma spectroscopy can be performed on a sample which is then counted for gross beta using liquid scintillation. Assuming that ^{40}K is the only other beta-emitter in the urine at that time, the portion of the gross beta activity from ^{40}K (inferred from gamma-spectroscopy results) can be subtracted from the gross beta value to get a total strontium value. The uncertainty in this value is very large, and the value must be used with care.

Strontium analysis is a two-count process. After the Sr is chemically extracted from the urine sample, it is deposited on a filter which is then counted on a proportional counter for gross-beta. After several days the filter is recounted to

account for ^{89}Sr decay that has occurred. In this way both major isotopes of Sr can be accounted for. Total strontium results may be available after the first beta count and can be used to make decisions regarding further monitoring.

Sampling for Tritium

Tritium is easily detected in urine. An intake of tritium is uniformly distributed throughout the body in as little as two hours. A single-void urine sample can be taken and analyzed to determine concentration of tritium which, after equilibrium in the body has been reached, is the same as the concentration of tritium in the body water. By knowing this concentration, the volume of body water (Reference Man - 42l (ICRP 1975)), and the biological half-life for tritium of 10 days, one can quite easily calculate the intake. Successive single-void samples after intake can be used to refine the intake and dose estimate and to determine a subject-specific biological half-life if necessary.

Medical Intervention

Medical intervention techniques are discussed in NCRP-65, *Management of Persons Accidentally Contaminated with Radionuclides* (NCRP 1980). These techniques are medical procedures and will not be discussed here. This publication does not give guidelines as to when medical intervention is appropriate. Guidelines can be found in *Guidebook for the Treatment of Accidental Internal Radionuclide Contamination of Workers* (Bhattacharyya 1992), which was cosponsored by the US Department of Energy Office of Health and Environmental Research. The ALI is suggested for use for decision making in this article as follows for transportable radionuclide compounds that are inhaled:

> "Treatment is not a consideration when the estimated intakes are below one ALI…
>
> "When the intake is likely to be between 1 and 10 times the ALI (20-200 mSv), treatment should be considered, even though clinical consequences from the incorporation are unlikely to occur. Under these situations, short-term administration will usually be appropriate…When the estimated intake exceeds 10 times the ALI, then extended or protracted treatment, depending on the severity of exposure should be implemented"

Lung lavage is recommended only if the inhalation intake is of poorly transportable forms of radionuclides and greater than 100 ALI. This is due to the severity of the treatment. The *Guidebook* also notes that "Similar considerations apply for ingestion as for inhalation." Wound contamination is also addressed, but note is given to the fact that contamination can be excised if necessary, and treatment should be considered on a case-by-case basis.

Confirmatory Monitoring

Confirmatory monitoring is performed on a random sampling of individuals who sign in on Radiation Work Permits in contamination, high contamination, or airborne radioactivity areas for whom bioassay is not required. Urinalysis or whole-body counting is performed as indicated by the radionuclide(s) stated on the RWP.

Dosimetry Models

ICRP-30 General Model

The basic models used to calculate intake are those taken from ICRP Publication 30, *Limits of Radionuclides by Workers* (ICRP 1979), and its supplements. Part 1 includes the lung and GI tract models which change only in the values of certain parameters with radionuclide and compound type. Metabolic models for each radionuclide (except radon) are included in the latter parts of the publication. These models describe metabolism of the radionuclide after leaving the lung or GI tract.

Lung Model

The ICRP-30 lung model consists of four regions and 10 compartments connected by catenary pathways and clearing to either the body fluids or GI tract (Fig 1.). Deposition following inhalation occurs in three of the regions, those being the nasal passage region (N-P), trachea and bronchial tree (T-B) and pulmonary parenchyma region (P). The magnitude of this deposition is dependent on particle size. A particle size of 1 μm AMAD is assumed unless particle size is actually known. The 1 μm AMAD particle size results in 30% deposition in the N-P region, 8% in the T-B region and 25% in the P region. The rest (37%) is assumed exhaled. Translocation rate constants and individual compartment deposition can be found in ICRP Publication 30 Part 1.

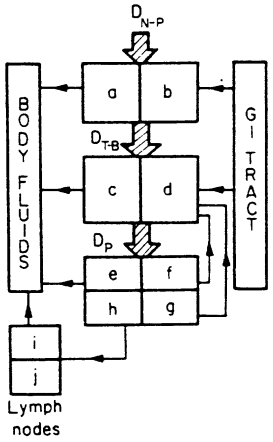

Figure 1: ICRP-30 Lung Model (ICRP 1979)

GI Tract Model

The ICRP-30 GI tract model (Fig. 2) consists of four compartments representing the stomach and large and small intestines. Parameters provided with the model include mean times in each compartment. These mean times can be translated to half clearance times by multiplying them by ln(2). While using half clearance times does not accurately represent the true bolus flow mechanism of the GI tract, it makes the mathematics easier and is fairly accurate for longer times after intake. Parameters for the GI tract model can be found in ICRP Publication 30, Part 1.

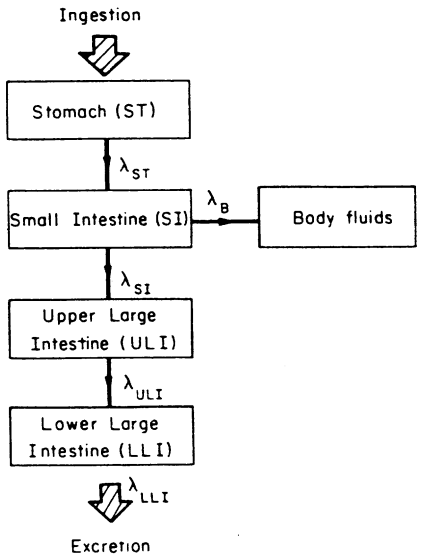

Figure 2: ICRP-30 GI Tract Model (ICRP 1979)

Metabolic Models

Metabolic models for relevant radionuclides are presented in ICRP-30. Models for some radionuclides are presented in this document in later sections. The general format of these models is presented in Fig. 3. Parameters for these models can be found in ICRP Publication 30, Parts 1-3 (ICRP 1979, 1980, 1981). The half clearance time out of the transfer compartment, unless otherwise stated in the specific metabolic model, is 0.25 days. Models are also available from other ICRP publications such as ICRP Publications 56, 67, 69, 71, and 72 (ICRP 1989, 1993, 1995a, 1995b, 1996). In addition, "Pseudo Uptake Retention Functions for

the Systemic Whole Body for Estimating Intakes from Excretion Bioassay Data" by Skrable, et. al (Skrable et al. 1987b) describes a method where actual data can be used to derive a retention function.

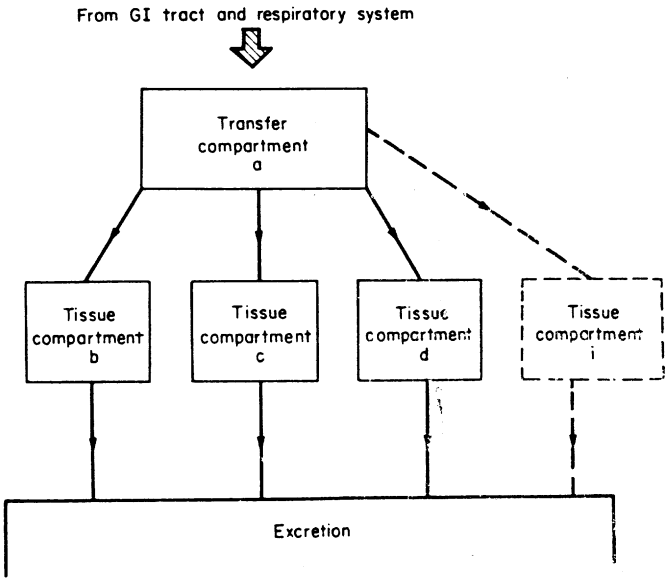

Figure 3: ICRP-30 Metabolic Model (ICRP, 1979)

INDOS[1] General Model

The following was obtained from the *INDOS* Technical Reference Manual (Skrable et al. 1987a).

The use and limitations of the various programs used in this manual can be better appreciated from a description of the metabolic models and parameter values used in the *INDOS* programs. The description here will help users of the *INDOS* programs to distinguish those input parameter values that must be entered when called by the various menus from those parameter values embedded in the various programs. A more detailed description of the models and mathematics is given in the Technical Reference Manual.

The models used in the *INDOS* programs are based upon a multi-compartmental description of the metabolism of radioelements as depicted in Figure 4 by various one-way catenary pathways from intake to excretion. The word

[1] INDOS. Skrable Enterprises.

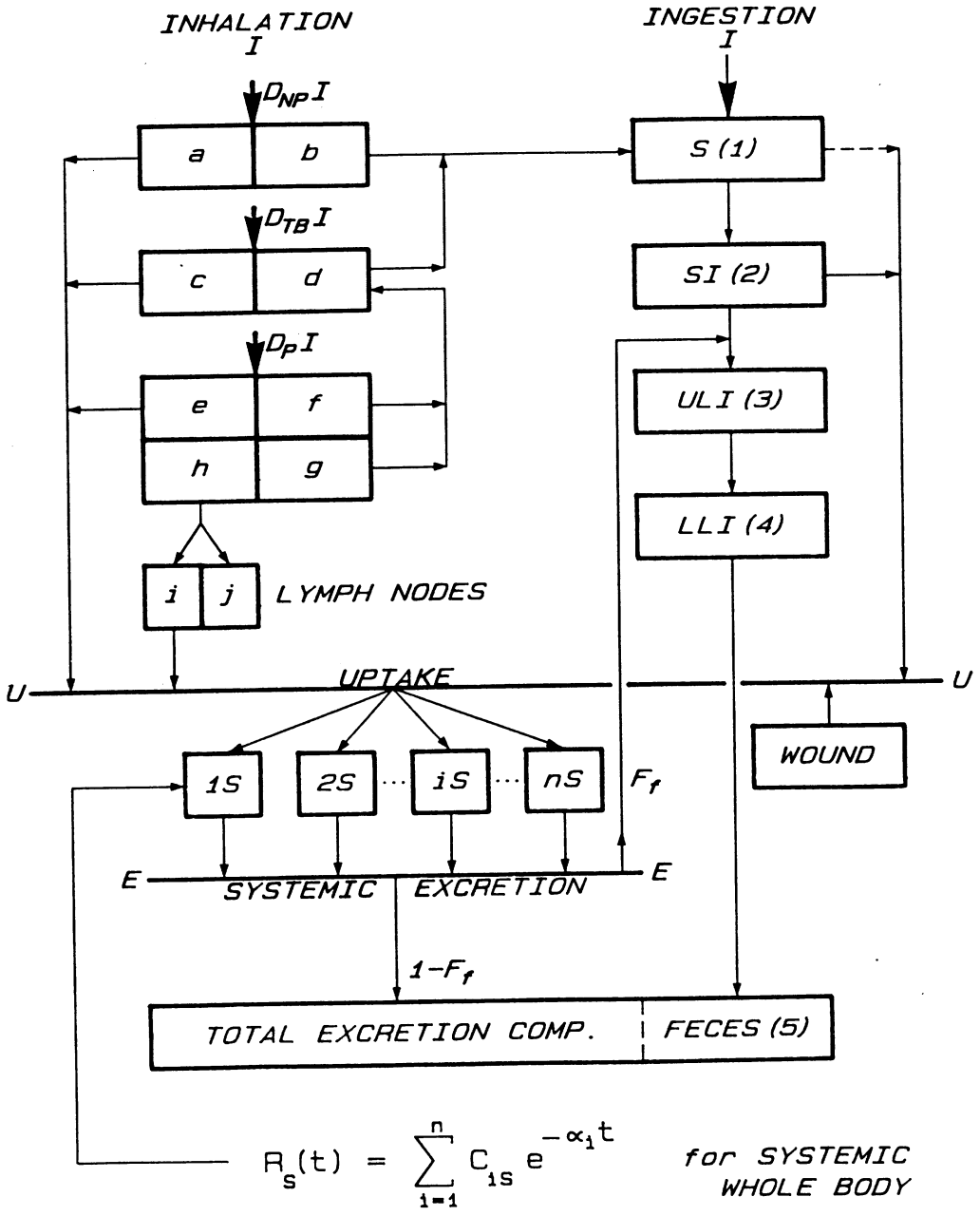

Figure 4: INDOS General Metabolic Model (Skrable 1987b)

catenary refers to a chain of compartments, and a one-way catenary system means one in which the radioelement is modeled to move only in one direction. The last compartment of all catenary systems is shown and designated in Figure 4 as the 'total excretion compartment', which conceptually may be considered as a

'bucket' where all excretion is collected. Atoms of a radionuclide are removed from this compartment only by radioactive decay. Although only one-way transfers between compartments are depicted, the model does in fact account for the recycling of elements in systemic compartments by the use of a systemic whole body retention function whose parameter values incorporate this recycling. This function is shown at the bottom of Figure 4. The word compartment is used in its mathematical sense and it may or may not represent a real structured physiological entity in the body.

Removal of a radioelement from a compartment may occur either by radioactive decay or by transfer to the next compartment in the catenary system. This transfer is described by a 'biological' clearance half-time (i.e., half-life), which determines the biological translocation rate constant. These transfers are shown by arrows in Figure 4. Removal of atoms of a radioelement from a compartment by radioactive decay is implicit in Figure 4, and is automatically accounted for by the programs in INDOS. Because removal of a stable or radioactive element from a compartment is directly proportional to the content of that compartment, simple exponentials give the fraction of a single deposition of the stable or radioactive element expected to be present in the compartment as a function of time post deposition.

To understand the metabolic model, it is important to distinguish the terms intake, deposition, uptake, and content. Intake is the quantity of a radioelement taken into the body, e.g., by inhalation or ingestion. Deposition is the quantity deposited, e.g., a certain fraction of an inhalation intake in various compartments of the respiratory tract. Uptake is the quantity absorbed into the systemic circulation, e.g., by injection into the blood, through a wound, or by absorption from compartments in the respiratory and GI tracts. Content is the quantity of a radioelement present in some compartment or combinations of compartments, e.g., the compartments c through j that comprise the lungs in the ICRP Publication 30 respiratory tract model. The time dependent quotients of the expected contents by the amounts of a single acute intake, deposition, or uptake define what are called intake retention functions, deposition retention functions, and uptake retention functions, respectively. These retention functions are determined by the metabolic models and associated parameter values. The numerical value of intake retention functions, i.e., the fractions of an intake expected to be present in a compartment at a chosen time, are obtained by application of a simple recursive kinetics equation to the various compartments of the catenary pathways depicted in Figure 4. In other

words, an intake retention function (an equation) is evaluated at a certain time to give the intake retention fraction (a number referred to by the acronym 'IRF').

As shown in Figure 4, pathways for intakes can be specified for inhalation intakes, ingestion intakes, instantaneous uptake, or exponential uptake from a wound site. There are a number of catenary systems or chains for each type of intake. Each chain begins with an intake compartment and ends with an excretion compartment. Pathways involving inhalation intakes are shown on the upper left, and pathways involving ingestion intakes are shown on the right in Figure 4. Uptake from a wound is depicted in Figure 1 as occurring from a single compartment.

Inhalation involves, in general eight compartments of deposition. The activity median aerodynamic diameter (AMAD) of the distribution of inhaled aerosol particles determines the deposition fractions in the three regions of the respiratory tract. The fraction of a regional deposition cleared by a particular pathway and the associated clearance half-time are designated by the ICRP and are embedded in the INDOS programs according to three chemical compound classifications: 1) (days), NV (weeks), and Y (years), which are representative of the clearance half-times from certain pulmonary lung compartments. These have the respective clearance half-times of 0.5, 50, and 500 days. The user of the INDOS programs must specify the AMAD and the chemical compound class or mixture of classes for an inhalation intake. The programs in INDOS then calculate the intake retention fractions (IRFS) based upon the expected deposition fractions in the three regions, fractions and clearance half-times for the respiratory tract compartments, and parameter values describing the metabolism for the GI tract and systemic compartments of the body. The user has the option of changing the embedded or default parameter values specified for the ICRP respiratory tract model.

Shown on the right hand side of Figure 4 are the four segments of the gastrointestinal tract. A radioelement that is contained within an ingestion intake or is translocated from certain respiratory tract compartments is first deposited in the contents of the stomach. The radioelement then may be translocated from one segment of the GI tract to another and then finally to the feces, which is designated as compartment 5 and considered as part of the total excretion compartment. The transfer of a radioelement from one segment to another with the contents is characterized by translocation rate constants obtained from ICRP Publication 30.

These are used as default parameters and are embedded in the INDOS programs. The user, however, can modify these parameter values.

Absorption of a radioelement from the GI tract into the blood is normally considered to occur from the small intestine. If the fraction fl of an ingested stable element that is absorbed into the blood is given as unity in ICRP Publication 30 for a given compound, then absorption is considered to occur directly from the stomach, which pathway is shown by a broken arrow to the upper right of Figure 4. The user of the INDOS programs must specify the value of fl only; all other GI tract parameter values are either embedded or calculated in the programs.

Absorption of a radioelement into the systemic circulation via all pathways is shown to lead to a horizontal line designated by the upper case letter U, which represents uptake, i.e., absorption into the systemic circulation. All catenary pathways combine at this point and then divide into n compartments represented by the n exponential terms in the systemic whole body uptake retention function for the stable element shown at the bottom of Figure 4. In other words, each exponential term is treated as a deposition retention function for one of the n catenary compartments of the systemic whole body. The effective fraction of an uptake that enters each of these systemic compartments is simply the coefficient of the particular exponential term input by the user.

The systemic whole body uptake retention must be input by the user as a sum of exponential terms with constant coefficients. The user will be asked in certain menus to input the number of exponential terms and values for the coefficients and 'biological' half-times for all of these terms. In some cases, the user may be interested in predicting the content of a specific systemic organ or tissue, e.g., the thyroid gland for radioiodines. In such case, the uptake retention function for the thyroid for stable iodine would have to be input by the user instead of the uptake retention function for the systemic whole body. It is to be emphasized that no single term of the systemic whole body uptake retention function can be associated with any particular systemic organ or tissue if there is any significant recycling of the element between systemic compartments. Although the ICRP does make such assignments of exponential terms for the purpose of estimating the 50 year committed doses to organs and tissues and the annual limits on intake (ALIs), such assignments may not reflect the short term kinetics and may not be appropriate for the estimation of intakes from bioassay data.

Removal of a radioelement from one of the compartments of the systemic whole body is shown in Figure 1 as a direct pathway to systemic excretion designated by a horizontal line identified by the upper case letter E. The translocation rate constants that describe this effective removal to excretion from the catenary compartments of the systemic whole body are determined by the clearance half-times specified by the user when the stable element uptake retention function for the systemic whole body is input. The horizontal excretion bar, i.e., the line identified by the upper case letter E, is necessary for designating what fraction of systemic excretion leaves the systemic whole body via the fecal excretion pathway and what fraction leaves by all other pathways, e.g., via urine, sweat, exhalation, etc. The fraction of systemic excretion via the fecal pathway is shown to enter the top of the upper large intestine, which effectively bypasses the small intestine where absorption into the blood is assumed to take place. Thus, the fraction of systemic excretion via the fecal pathway should be considered as an effective value. The user will be asked to input values for the fraction of the systemic excretion by the fecal and urinary pathways in menus of some of the programs, e.g., for excretion bioassay or for bioassay based on whole body counting where the GI tract contents might contribute to the response of the counter.

Further details and warnings regarding the metabolic models are given in the discussions for the use of each of the programs in INDOS. Some of the warnings are purposely redundant to assure the correct use of the INDOS programs. We recommend that the metabolic models and parameter values in ICRP Publication 30 be used if no other specific information is available. When other models are known to be more representative of the metabolism of workers at a specific site or when more accurate models and parameter values are reported in the literature, they should be used. We have constructed INDOS to allow this flexibility. When non-standard models and parameter values are chosen, they will automatically be printed along with other outputs from the INDOS programs.

Determination of Monitoring Action Levels

Personal Air Sampling Monitoring Levels

Items in this section, except sections between square brackets are taken from *Evaluation of Radiation Protection Internal Dosimetry Program Manual & Procedures for Sandia National Laboratories* by K. W. Skrable (Skrable 1996).

Material in Process

Of particular significance to the evaluation of potential internal exposures from measured removable surface contamination is Section 1.1 of the Nuclear Regulatory Commission (NRC) Regulatory Guide 8.25, where it is quoted partly (U.S. NRC 1992): "As a general rule, any licensee who handles or processes unsealed or loose radioactive materials in quantities that during a year will total more than 10,000 times the ALI for inhalation should evaluate the need for air sampling… When quantities handled in a year are less than 10,000 times the ALI, air sampling generally is not needed. (The basis for this value is that experience has shown that worker intakes are unlikely to exceed one-millionth of the material being handled or processed, as discussed in NUREG-1400)." (Hickey 1993)

This statement of the NRC is equivalent to saying that air sampling should be considered when workers may have unlikely intakes in any year that exceed 1% of an ALI or 10% of the NRC monitoring reference level of 0.1 ALI. The NRC unlikely intake fraction of $1x10^{-6}$ for loose radioactive material is used to calculate a *derived activity level* (DAL) for the loose activity that could be processed in any one work day without likely exceeding the DOE monitoring reference level of 40 S-DAC-h or intake of 0.02 S-ALI for any control year of practice, i.e., 2% of the S-ALI. Although this is less demanding by a factor of 2 than the NRC guidance, it is conservatively assumed in all of the calculations below that a quantity of loose radioactive material equal to the DAL is processed each of the 250 working days in any control year. The DAL is then calculated:

$$DAL = 1x10^{-6}\left(\frac{0.02(S-ALI)}{250 days}\right) = 80\, S-ALI/day \qquad (1)$$

Thus, if a worker does not handle or process more than 80 S-ALI of any radionuclide in any work day, then that worker is not likely to exceed the DOE monitoring reference level of 40 S-DAC-h in any control year even if this activity is processed in each of the 250 work days in that year. For tritiated water, which has a S-ALI of 80,000 μCi, the DAL is calculated as 6.4 Ci day^{-1}. This derived action level of 6.4 Ci day^{-1} for tritiated water is considerably larger than the previously used NRC bioassay reference levels of (1) 100 mCi $month^{-1}$ and (2) 10 mCi kg^{-1} when quantities in excess of 10 kg of contaminated H_2O are exposed to the air in a room (NRC, 1988). If the DAL of 6.4 Ci day^{-1} calculated here is exceeded, then continuous monitoring of the workplace should be used to provide warning of elevated levels of airborne tritium. Urinalysis should be used for accurate assessments of workers' exposures and timely detection of significant exposures that might warrant corrective actions. Urinalysis programs for tritium are discussed in other sections.

The optimum action for correct PAS usage is that DALs be calculated for the mixture of radionuclides at each SNL facility or job site to determine the need for using personal air samplers for monitoring the exposures of workers to radioactive aerosols. If the frequency F or number of days per year of processing loose radioactive material is less than 250 days per year, then the DAL could be increased by the ratio of 250/F. However, for conservatism and other reasons mentioned above and discussed below, the DAL should be calculated from 80 S-ALI.

Example calculations of the unlikely internal dose and the actual external exposure from working at distance of 1 meter from an activity of 1 DAL of Class Y ^{60}Co are provided as follows. The DAL is calculated from the ^{60}Co S-ALI of 30 μCi:

$$DAL = 80S - ALI = (80)(30\mu Ci) = 2400\mu Ci = 2.4mCi \tag{2}$$

The Γ constant of ^{60}Co is 1.32 mR h^{-1} mCi^{-1} m^2; so, the total external gamma exposure X over an 8 hour work day at a distance of 1 meter is calculated:

$$X = (2.4mCi)(1.32mR \cdot h^{-1} \cdot mCi^{-1})(8h) = 25.3mR \tag{3}$$

which approximately equals an external deep dose equivalent H_d of 25.3 mrem. The internal committed effective dose equivalent (CEDE) is not likely to exceed:

$$CEDE < \left(\frac{0.02\,ALI}{250}\right)\left(\frac{5000mrem}{S-ALI}\right) = 0.4mrem \tag{4}$$

Thus, the ratio of the external deep dose equivalent H_d to the CEDE is 63.3. Therefore, it can be concluded that the internal exposure to ^{60}Co will automatically be controlled well below the DOE monitoring reference level of 40 S-DAC-h provided that the external dose is controlled below the regulatory limit of 5000 mrem even when all of the ^{60}Co is classified as loose radioactive material. In reality, most of the external radiation exposure at nuclear power plants arises from activated or contaminated material that is not readily dispersed into the air. Experience also shows that most inhaled radioactive aerosols have a particle size that is non-respirable. Such non-respirable aerosols deposit in the upper respiratory tract and clear rapidly to the gastrointestinal tract and fecal excretion. Therefore, this non-respirable portion of an inhalation intake can be treated as effective ingestion intake, which generally has a much smaller dose consequence than the 1 μm activity median aerodynamic diameter (AMAD) aerosol assumed in the derivation of the ALI and DAC.

The actual likelihood of intake of loose radioactive material depends on many factors such as the specific activity, the chemical forms, the volatility of the radioactive materials, the use of any local ventilation or containment systems, the amount of room ventilation, the proximity of the worker to the source of any release into the ambient air, and the potential for work activities to cause releases into the air. Obviously, the potential for generating radioactive aerosols must be evaluated for each job, e.g., for specific decontamination and decommissioning activities. If practical, mock up tests should be performed to evaluate the potential for generating airborne activity before any large scale decommissioning or decontamination projects are initiated. In cases where the amounts and types of radioactive materials are unknown, then constant air monitors (CAMs) should be used to provide warning of elevated levels, and personal air samplers should be used to monitor the exposures of individual workers.

Evaluations Based on the Measurements of the Specific Activity of Materials

The exposure pathway of greatest concern is inhalation of aerosols, especially for the alpha emitting actinides. The potential for the generation of significant airborne activity from contaminated surfaces or soil depends on both the level of contamination, the characteristics of the surfaces and soil (e.g., dry versus wet situations), and the existence of forces that cause the generation of radioactive aerosols (e.g.. wind, air turbulence, the handling and movement of highly contaminated materials in open areas, or in the case of the ^{230}Th exposure case discussed in Attachment 1, the decontamination of a surface with water and wire brushes). The total specific activity S_A of the soil or material contaminating a surface determines the mass of material per cubic meter of air that must be present in the air to reach a given airborne activity concentration. Thus, the smaller the specific activity S_A, the smaller will be the potential for significant exposures to airborne radioactivity.

If the mass concentration C_m in the air is maintained at or below the respirable dust loading limit of 0.005 gm^{-3} specified by the Occupational Safety and Health Administration (OSHA), then a given specific activity S_A of respirable dust must be exceeded in order to exceed the stochastic effect-based derived air concentration or S-DAC of the radionuclide. For 1 μm AMAD class Y aerosols of ^{230}Th which has a calculated S-DAC of 7.96 pCi m^{-3} or 8 pCi m-3, the OSHA limiting specific activity S_{OLA} of respirable dust is calculated when the mass concentration C_m in air is at the OSHA respirable dust limit C_{Om} of 0.005 g m^{-3} for example:

$$S - DAC = S_{OLA} C_{Om} \tag{5}$$

$$S_{OLA} = \frac{S - DAC}{C_{Om}} = \frac{8\,pCi \cdot m^{-3}}{0.005 g \cdot m^{-3}} = 1600\,pCi \cdot g^{-1} \tag{6}$$

Thus, when the airborne concentration C_m of respirable dusts is maintained below the OSHA limit C_{Om} of 0.005 g m^{-3} and the specific activity S_A is less than 1600 pCi g^{-1}, the airborne activity concentration of ^{230}Th will automatically be controlled below the S-DAC of 8 pCi m^{-3}. Class Y ^{239}Pu has a calculated S-DAC of 6.76 pCi m^{-3} or 7 pCi m^{-3} and a limiting specific activity S_{OLA} of 1400 pCi g^{-1} (EPA, 1988). If the concentration C_m of respirable dusts is controlled at a fraction F of the

OSHA limit, then the limiting specific activity SLA would be increased by the factor 1/F, e.g. to 16,000 pCi g^{-1} for ^{230}Th when the concentration of respirable dusts is limited to 0.0005 g m^{-3}. In such case, a specific activity S_A of 1600 pCi g^{-1} would limit the airborne concentration to 10% of the S-DAC. which determines an airborne radioactivity area in 10 CFR 835.

When respirable dusts in the air are maintained at the OSHA limiting concentration C_{Om} of 0.005 g m^{-3}, the activity concentration U of the radionuclide can be calculated from the specific activity S_A. For the highest specific activity of 160,000 pCi g^{-1} for ^{230}Th shown in Attachment 11 for example, the activity concentration U is calculated:

$$U = S_A C_m = (160{,}000\,pCi \cdot g^{-1})(0.005 g \cdot m^{-3}) = 8000\,pCi \cdot m^{-3} = 1000 S - DAC \qquad (7)$$

For the maximum specific activity S_A of 160,000 pCi g^{-1} of ^{230}Th shown in attachment 11, the mass concentration Cm of respirable dusts that corresponds to the S-DAC of 8 pCi m^{-3} is calculated:

$$C_m = \frac{S - DAC}{S_A} = \frac{8\,pCi \cdot m^{-3}}{160{,}000\,pCi \cdot g^{-1}} = 5x10^{-5}\, g \cdot m^{-3} \qquad (8)$$

which is only 50 μg m^{-3} and only 1% of the OSHA dust limit of 0.005 g m^{-3}.

The calculations shown above in equations 7 and 8 imply a potential for significant intakes if exposures to high ^{230}Th specific activity shown in Attachment I[2] were to take place at the OSHA dust loading limit of 0.005 g m^{-3}. However, even the highest specific activity of 160,000 pCi g^{-1} would not appear to represent a high probability of having an acute intake exceeding the class Y S-ALI of 20,000 pCi for ^{230}Th. An intake of 1 S-ALI would require an intake of 0.125 g or 125 mg of this highest specific activity ^{230}Th. Based on the NRC intake fraction discussed above, a worker is unlikely to inhale more than a fraction of $1x10^{-6}$ of the loose radioactive material that he processes. Thus, to have an intake of 1 S-ALI, a worker would presumably have to handle 125,000 g or 275 pounds of the highest specific activity

[2] This refers to Attachment I of *Evaluation of Radiation Protection Internal Dosimetry Program Manual & Procedures for Sandia National Laboratories* by K. W. Skrable.

of ^{230}Th shown in Attachment 1. This processed quantity corresponds to a total activity of 2xl0^{10} pCi (20 mCi), which is many orders of magnitude higher than the actual activity workers processed in the decontamination activity described in Attachment 1.

A derived specific activity level (DSAL) based upon the DOE monitoring reference level of 40 S-DAC-H for any control year can be calculated on the basis of an assumption about the mass concentration C_m of respirable dusts in the air, e.g., a concentration C_m of respirable dusts at some fraction of the OSHA limit of 0.005 g m^{-3}. For conservatism, the calculation below assumes a mass concentration C_m of respirable dusts in the air equal to the OSHA limit of 0.005 g m^{-3}. When measurements of the gross specific activity S_A are made, the measurements include activity of non-respirable materials, and a reasonable assumption can be made to take this into account. For the purpose of calculating a DSAL for such measurements, the following assumptions are made:

1. A worker is exposed continuously 8 hours per day, 250 days a year or for a total time of 2,000 hours to an activity concentration of 0.02 S-DAC, which corresponds to the DOE monitoring reference level of 40 S-DAC-H in any one year.

2. The activity of respirable dusts is 0.1 of the total activity measured.

3. The mass concentration Cm of respirable dusts in the air is maintained at the OSHA limit or at 0.005 g m^{-3} for the entire 2000 h exposure period.

The derived specific activity level (DSAL) for determining the need for monitoring is then calculated from these assumptions:

$$DSAL = (10)\left(\frac{0.02S - DAC}{C_m}\right) = (10)\left(\frac{0.02S - DAC}{0.005g \cdot m^{-3}}\right) = 40m^3g^{-1}S - DAC \quad (9)$$

For Class Y ^{230}Th and Class Y ^{239}Pu, which have S-DACs of 8 pCi m^{-3} and 7 pCi m^{-3} respectively. the DSALs are calculated respectively as 320 pCi g^{-1} and 280 pCi g^{-1}.

Evaluations Based Upon a Resuspension Factor of $1x10^{-6}$ m^{-1}

A resuspension factor Of $1x10^{-6}$ m^{-1} has been applied to the estimation of potential airborne contamination from measured surface contamination levels (Healy, 1980 and Brodsky, 1980). This factor represents the expected steady state activity concentration U per unit surface activity A_s:

$$< \frac{U}{A_s} >= 1x10^{-6} \frac{Ci \cdot m^{-3}}{Ci \cdot m^{-2}} \tag{10}$$

The derived surface activity level (DA_sL) required for an intake of 0.02 S-ALI and exposure equal to the DOE monitoring reference level of 40 S-DAC-H for continuous occupational exposure to 0.02 S-DAC in one control year (2,000 h) is calculated for any radionuclide for a resuspension factor <U/As> of $1x10^{-6}$ m^{-1}:

$$DA_sL = \frac{0.02S - DAC}{< \frac{U}{A_s} >} = \frac{0.02S - DAC}{1x10^{-6} m^{-1}} = 20{,}000 m^{-1} S - DAC \tag{11}$$

The S-DACs for Class Y ^{60}Co and Class D ^{137}Cs are respectively $1x10^{-8}$ Ci m^{-3} and $6xl0^{-8}$ Ci m^{-3}, which yield respective DA_sL values of 200 μCi m^{-2} and 1200 μCi cm^{-2}. These contamination levels correspond to 4,440,000 dpm per 100 cm^2 for ^{60}Co and 26,600,000 dpm per l00 cm^2 or ^{137}Cs. Despite these seemingly high surface contamination levels, the worker's total internal exposure for the year would not exceed 40 S-DAC-h for the assumed resuspension factor. In this case in which both DACs are based upon the whole body limit of 5,000 mrem, the internal doses would be limited to 100 mrem CEDE. The external doses H_E, however, are much higher as shown in the calculations below for these derived surface activity levels.

Internal and external dose conversion factors to three digits can be obtained from the respective computer programs "DFint" and "DFext" provided by Keith Eckerman. External dose factors apply to the effective dose equivalent rate to the whole body from an infinite planar source. The actual size of the planar source does not have to be very large to yield a dose rate at the center which approximates that from an infinite planar source. Internal dose conversion factors in the program DFint include the committed effective dose equivalent per unit intake. Values provided in the program DFint also are available in abridged form in Federal

Guidance Report II [Eckerman et al. 1988]. Values for the ^{60}Co and ^{137}Cs external dose rate factors are:

$<D_E/A_s>$ = 25,800 mrem h^{-1} per Ci m^{-2} of surface contamination for ^{60}Co.

$<D_E/As>$ = 6,570 mrem h^{-1} per Ci m^{-2} of surface contamination for ^{137}Cs.

For a surface activity A_s of any radionuclide, the external effective dose equivalent H_E is calculated for 2,000 hours of occupational exposure in one control year of practice:

$$H_E = A_s < \frac{D_E}{A_s} > 2000h \tag{12}$$

From the external dose conversion factors and the derived surface activities levels of respectively $2x10^{-4}$ Ci m^{-2} and $1.2x10^{-3}$ Ci m^{-2} obtained by equation 12 for ^{60}Co and ^{137}Cs for limiting the annual internal exposures to 40 S-DAC-h the annual external doses H_E are calculated by equation 13 as follows.

For ^{60}Co, the annual external dose is calculated:

$$H_E = (2x10^{-4} Ci \cdot m^{-2})\left(25{,}800\frac{mrem \cdot h^{-1}}{Ci \cdot m^{-2}}\right)(2000h) = 10{,}300mrem \tag{13}$$

which is 103 times the internal dose of 100 mrem calculated for an exposure of 40 S-DAC-h.

For ^{137}Cs, the annual external dose is calculated:

$$H_E = (1.2x10^{-3} Ci \cdot m^{-2})\left(6570\frac{mrem \cdot h^{-1}}{Ci \cdot m^{-2}}\right)(2000h) = 15{,}800mrem \tag{14}$$

which is 158 times the internal dose of 100 mrem calculated for an exposure of 40 S-DAC-h.

The calculations shown above for planar sources using a resuspension factor of $1x10^{-6}$ m^{-1} show that the external dose potential is 103 times and 158 times greater than the internal dose potential from respectively ^{60}Co and ^{137}Cs. Thus. when the external dose is at the regulatory limit of 5000 mrem the internal doses will be about 50 mrem and 30 mrem for the respective planar sources of ^{60}Co and ^{137}Cs. When the annual internal exposures are limited to 40 DAC-h, the external dose rates are 5.15 mrem h^{-1} and 7.90 mrem h^{-1} for ^{60}Co and ^{137}Cs respectively. The derived high surface contamination levels of 200 μCi m^{-2} and 1200 μCi in needed to yield annual exposures of 40 S-DAC-h for respectively ^{60}Co and ^{137}Cs certainly would not be tolerated on the basis of these external radiation levels alone. When it is unrealistically assumed that a worker is exposed for 2000 hours in one control year of practice to a planar source of either ^{60}Co or ^{137}Cs that would yield an external dose of 5 rem, the internal exposure potential would involve intakes equal to only about 1% and 0.6% of the respective S-ALIs for ^{60}Co and ^{137}Cs. These calculations show that internal exposures to ^{60}Co and ^{137}Cs are very unlikely to result in intakes greater than 1% of the S-ALIs because such intakes would require overexposures to external radiation. The major radiation hazard from these radionuclides is from external radiation. Even when transuranics might be present along with beta-gamma emitting fission and activation products in the working environment of nuclear reactors, the major radiation hazard will most likely be from external gamma radiation as shown in Attachment 11.

The TEDE ALARA example shown above also could repeated here of ^{60}Co for the hypothetical planar sources of 4,440,000 dpm ^{60}Co per 100 cm^2 and 26,600,000 dpm of ^{137}Cs per 100 cm^2. Again it would be concluded that the use of respiratory protection would violate the requirement for TEDE ALARA. Instead of showing this calculation. a calculation will be made of a derived skin surface activity of ^{60}Co, DA_{skin}, for demonstrating the undue concern that is often expressed when skin contamination is detected on a worker. This derived limit will be based upon the DOE skin dose monitoring reference level of 5 rem in any control year of practice (10 CFR 835.402(a)(ii)) [U.S. DOE 1993]. The following unduly conservative assumptions are made for calculating this skin dose derived monitoring reference level:

1. The derived skin surface contamination level, DA_{skin}, is deemed to be present on the skin continuously for 1 year or for 8,760 h.

2. The value for the DA_{skin} is calculated for the skin monitoring reference level of 5 rem.

3. A beta-gamma skin dose rate conversion factor of 4 rem h^{-1} per μCi cm^{-2} of ^{60}Co calculated from the NRC computer code VARSKIN-MOD2 is used for the calculation.

With the above assumptions, the value for the derived skin contamination level is calculated:

$$DA_{skin} = \frac{5rem/8760h}{4rem \cdot h^{-1}/\mu Ci \cdot cm^{-2}} = 1.43x10^{-4}\frac{\mu Ci}{cm^2} = 31{,}700\frac{dpm}{100cm^2} \quad (15)$$

Despite this seemingly high level of contamination on the skin, the 5 rem skin dose monitoring reference level would not be exceeded. This calculation using the unduly conservative assumption that this contamination level of 31,700 dpm per 100 cm^{-2} is continuously present on the skin, even during off-hours at home and elsewhere, demonstrates the undue concern often expressed when much lower levels of contamination are actually detected on a worker's skin. Some of this undue concern has arisen at nuclear power stations from the detection of single hot particles on a worker's protective clothing. It is interesting to note that this derived skin activity DA_{skin} of 31,700 dpm per 100 cm^2 is not much less than the ^{60}Co derived surface activity level (DA_sL) of 44,400 dpm per 100 cm^2 calculated above for the 40 S-DAC-h exposure monitoring reference level calculated for continuous occupational exposure for 1 year (2,000 h) to this planar source.

[While the above technique protects the individual from the effects of resuspension, it does not protect from activity disturbed and made airborne. This is a consideration at high surface activities like those described for ^{137}Cs above. Therefore, upper levels of 10,000 dpm/100 cm^2 and 100,000 dpm/100 cm^2 have been set for International Labor Organization Toxicity Classifications Groups 1 and 2 (Shleien et al. 1998), and 3 and 4 respectively. Where these upper levels are above the High Contamination Area trigger levels (U.S. DOE 1993), personal air sampling is required in High Contamination Areas.]

Technical Basis for Personal Air Sampling

The effectiveness of personal air sampling can be seen by using radionuclides expected to be the worst case and determining critical levels (L_c), minimum detectable activities (MDA), and other quantities. PAS filters are commonly counted on a proportional counter which determines both alpha and beta activity simultaneously. Spectroscopic data is not available from this type of detection; however, radionuclide data can be obtained by further analysis if necessary.

Radionuclides that are considered worse case alpha and beta emitters at SNL are ^{239}Pu and ^{90}Sr respectively. The worse case compound class for Pu is class D, and that for Sr is class Y. However, because class Y Sr only applies to $SrTiO_2$ which is not expected to be found at SNL, class D is used. Typical count times for PAS filters are 30 minutes with a 20 minute background. Efficiency of the proportional counter is 43% for alpha and 54 % for beta. Quantities below are either classified as "physically significant" (PS), i.e., relating to the critical level and decision points, or "minimum detectable" (MD) relating to instrument capability using the lower limit of detection. A PAS flow rate of 2 lpm was assumed which is typical at SNL. The physically significant concentration value assumes a 4 hour stay time. This results in an IRF of 0.1. Values for ^{90}Sr and ^{239}Pu are listed in Table 3 below:

All of these quantities are for each filter. If 500 filters were used per year for ^{239}Pu (2 per day for 250 working days), the maximum missed dose over the year would be over 800 mrem. This may seem considerable. However, this is most certainly better than any bioassay method for ^{239}Pu detection due to the extremely small IRF values associated with bioassay methods. The maximum missed dose for ^{90}Sr is less than 2 mrem. SNL does not typically expose individuals to environments where a PAS is required often. Those jobs tend to be short term operations. When an individual has worn a PAS enough times that the maximum missed dose is greater than 100 mrem, a bioassay program for that individual is required as per 835.402(c) (U.S. DOE 1993) and is instituted. Other action levels are calculating dose, recording, and reporting at 1 DAC-h, and bioassay instituted at 10 DAC-h. While these values are considerably below the 10CFR835 monitoring requirement, they are appropriate for the low level of hazard expected at SNL.

Table 3: Example Quantities Relating to PAS Effectiveness

Quantity	^{239}Pu	^{90}Sr
Critical Level (L_c)	0.150 cpm	0.938 cpm
Lower Limit of Detection (L_d)	0.391 cpm	1.97 cpm
Physically Significant Activity (PSA)	0.346 dpm	1.72 dpm
Minimum Detectable Activity (MDA)	0.900 dpm	3.61 dpm
Physically Significant Intake (PSI)	3.46 dpm	17.2 dpm
Minimum Detectable Intake (MDI)	9.00 dpm	36.1 dpm
PSI/ALI	$2.44x10^{-4}$	$4.18x10^{-7}$
MDI/ALI	$6.34x10^{-4}$	$8.75x10^{-7}$
Physically Significant Concentration (PSC)	$3.25x10^{-13}$ µCi/ml	$1.62x10^{-12}$ µCi/ml
Minimum Detectable Concentration (MDC)	$8.45x10^{-13}$ µCi/ml	$3.39x10^{-12}$ µCi/ml
Physically Significant Exposure (PSE)	0.488 DAC-h	$8.35x10^{-4}$ DAC-h
Minimum Detectable Exposure	1.27 DAC-h	$1.75x10^{-3}$ DAC-h
Physically Significant CEDE (PSH_E)	0.668 mrem	$1.86x10^{-3}$ mrem
Minimum Detectable CEDE (MDH_E)	1.74 mrem	$3.90x10^{-3}$ mrem

TEDE ALARA Considerations

Various arguments could be made about the assumptions used to calculate the derived monitoring reference levels in the three previous sections. When all of the assumptions are noted in the calculations of the derived reference levels, including the assumption of continuous occupational exposure for 250 days, it may be concluded that the final calculated values are conservative with respect to the internal exposure reference level of 40 S-DAC-h. Because of the requirements for

TEDE ALARA, it is important that reasonable assumptions be used in the evaluation of potential external and internal exposures when specific data are unavailable for making choices of protective measures. When undue concern for internal exposures leads to the choice of protective measures that decrease the efficiency for completing a given job and when external radiation dominates the total exposure, then the TEDE will not be ALARA. Despite the lack of specific data in a given job situation, protective measures should be chosen for each job activity that are thought to maintain TEDE ALARA. When potential internal radiation exposures are deemed to significantly exceed the 40 S-DAC-H monitoring reference level for any year, CAMs should be used to provide alarms. Alarm set points can be chosen that prevent exposures of 1 S-DAC-h of most, if not all, radionuclides. In addition to the timely detection of significant exposures that might warrant corrective actions and the implementation of follow-up bioassay procedures, data measures that are thought to maintain TEDE ALARA. I recommend that consideration be to provide additional data needed in the evaluation of potential internal radiation exposures and the choices made to maintain TEDE ALARA.

Monitoring Levels for Tritiated Water

As another example of the use of specific activity in the evaluation of potential exposures, consider exposure to tritiated water vapor. The maximum absolute concentration C_m of water in the air cannot exceed the saturation concentration limit for a given temperature which is 17.3 g m^{-3} at 20 degrees centigrade. The S-DAC for tritiated water vapor is 20 µCi m^{-3}. Even if all of the water vapor in the air of an unventilated room were to be drerived from water contaminated with tritium at a specific activity S_A, the airborne activity concentration U could not exceed:

$$U = S_A C_m = 17.3 g \cdot m^{-3} S_A \tag{16}$$

The limiting specific activity SLA is calculated the S=DAC of 20 µCi m^{-3} and the saturation concentration C_m of 17.3 g m^{-3} for 20 degrees centigrade:

$$S_{LA} = \frac{S - DAC}{C_m} = \frac{20 \mu Ci \cdot m^{-3}}{17.3 g \cdot m^{-3}} = 1.16 \mu Ci \cdot g^{-1} \tag{17}$$

which is equivalent to a limiting activity concentration of 1160 μCi L^{-1} in the water. Thus, an activity concentration of 116 μCi L^{-1} in the water will automatically control the airborne concentration to 10% of the S-DAC.

The derived activity level (DAL) was previously calculated from equation 1 as $6.4x10^6$ μCi day^{-1} for tritiated water. If a worker were to process in each working day a volume V of 5520 L of water contaminated at the level of 1160 μCi L^{-1}, which was calculated from equation 10 for maintaining a concentration of 1 S-DAC, then the total activity processed would equal the DAL of $6.4x10^6$ μCi day^{-1}. However, for this DAL the worker's total exposure for the year would not likely exceed 40 S-DAC-h. Yet if a worker stayed 2000 h in a room with 1 S-DAC, the worker's exposure would be 2000 S-DAC-H which is 50 times the worker's unlikely exposure calculated on the basis of the NRC's unlikely intake fraction of $1x10^{-6}$. This calculation shows the undue conservatism in the assumption that the air is saturated with water vapor entirely comprised of the contaminated water being processed in a room with no ventilation. Based upon this factor of 50, a derived specific activity level (DSAL) of 58 μCi g^{-1} or 58 mCi L^{-1} would require the processing of (5520 L day^{-1})/(50) or 110 L day^{-1} to reach the DAL of 6.4 Ci day^{-1}, which would have to take place over 250 days for an unlikely exposure of 40 S-DAC-h per year. It is recommended that a derived specific activity level (DSAL) of 50 mCi L^{-1} be used for determining the need for a tritium bioassay program at SNL, which is only 5 times the previous NRC guidance discussed above.

Intake Assessment

Data Evaluation

Detection Limit Issues

Detection limits originally were set at critical levels for bioassay analysis. The critical level equation assumes a 5% Type I error, that is, 5% of results showing a result greater than the critical level will have no activity present. This statistic includes uncertainties involving counting data only, as shown below:

$$L_c = k_\alpha \sigma_b \left(1 + \frac{T_b}{T_{s+b}}\right)^{1/2} \tag{18}$$

where:

L_c = critical level,
σ_b = standard error in the background,
T_b = background count time, and
T_{s+b} = sample count time.

However, it can be shown by observation of actual bioassay data that use of the critical level for all analyses results in greater than a 5% false positive rate. However, for radionuclides not normally expected in humans from environmental sources, the critical level is still the statistic used at SNL. For radionuclides which may exist in the population at SNL from environmental sources, baseline data has been analyzed using a log-normal distribution, and decision levels have been derived at a 95% probability. Personal air sample results are analyzed using the critical level statistic.

In addition, detection levels for tritium are extremely low, on the order of one pCi/ml. This could correspond to doses on the order of 0.01 mrem. Because of this, a decision level has been derived that corresponds to an acceptable reporting level.

Positive Result Decision Levels

Decision levels for uranium , strontium, and thorium are based on a log-normal distribution of baseline values. When plotted in this fashion, all three data sets reasonably follow the distribution. A value at 95% has been obtained from

each distribution for comparison with results to determine if the possibility of an intake has occurred. These values are:

Table 3: Decision Levels for Naturally Occurring Radionuclides

Radionuclide	Concentration (pCi/l)
Sr-90	1.5
Th-232	0.046
Natural Uranium	0.022 (0.32 μg/l)

There is also a natural background of other isotopes of thorium, particularly ^{230}Th and ^{234}Th. These are part of the natural uranium decay chain and are treated accordingly.

In the case of tritium, a decision level is set based on a recording level of 2.5 mrem (1 DAC-h) and assuming a 7 day interval between exposure and bioassay. The time interval is reasonable, because tritium sampling is done on a per-use basis as indicated on a radiological work permit (RWP). The resulting decision level is 500 pCi/ml.

Dose Calculation

Dose Calculation Principles

Radionuclide intakes can be estimated from bioassay data using appropriate metabolic models. The corresponding dose is calculated from the estimated intake using the appropriate dose conversion factor. The following information should be obtained to facilitate intake and dose assessments in the RPID Project (NCRP 87):

- Radionuclide species from bioassay, air sampling, or contamination measurements,
    - Physical form of the radionuclide from air sampling measurements, retention patterns, or process knowledge,

- Chemical form of the radionuclide from air sampling, contamination measurements, retention patterns, or process knowledge,
- Route of exposure from incident investigations or process knowledge,
- Previous exposure history, including environmental exposures,
- Sampling time after exposure from incident investigations,
- Bioassay information available including routine and/or special bioassays,
- Age and health status of exposed individual,
- Appropriate metabolic model from ICRP or NCRP recommendations and/or retention patterns, and
- Appropriate model to estimate intake, dose equivalent rate, or committed dose equivalent from ICRP or NCRP recommendations and/or retention patterns.

Most of the required information will be ascertained from the radiological incident investigation process described in RPID procedures. However, there may be instances when an exposure incident occurs that is undetected until bioassay results are received. Assumptions would then be based on facility process knowledge. Conservative assumptions about the intake are made in the absence of knowledge of appropriate parameters. These include:

- Inhalation exposure pathway,
- Conservative pulmonary clearance class, and
- One micron AMAD particle size.

Occupational intakes, i.e., intakes resulting in measurements greater than the corresponding decision levels as stated in Section A of this part, are assessed and dose is calculated. In addition, suspected intakes may be evaluated to ascertain the need for exposure assessment bioassays, exposure mitigation, and/or medical intervention.

Dose Calculation from Air Monitoring Data

Personal air sample results greater than the corresponding critical levels are entered into the PAS database. Exposure in DAC-h is automatically calculated and recorded. The equation used in calculating exposure is:

$$\mathrm{E(DAC-h)} = \frac{A}{F_{PAS}/F_{RM}} ECF \tag{19}$$

where: E(DAC-h) = exposure in DAC-h,
A = measured activity on filter,
F_{PAS} = flow rate of personal air sampler,
F_{RM} = breathing rate of reference man (20 l/m), and
ECF = exposure conversion factor in DAC-h/activity units.

The exposure conversion factor is calculated by obtaining a dose conversion factor from Federal Guidance Report 11 and dividing by 2.5 mrem/DAC-h.

Dose Calculation Using INDOS

Calculating dose using INDOS is explained in detail in the INDOS User's Guide (Skrable 1986). INDOS derives an analytical solution to the ICRP-30 models using catenary kinetics. The solution is in the form of an intake retention fraction for the bioassay compartment of interest and time after intake. The simple solution for one bioassay measurement is just that measurement value divided by the intake retention fraction. If a stochastic annual limit on intake is input into INDOS, the program will use that as a dose conversion factor and calculate a committed effective dose equivalent. More up-to-date dose conversion factors exist in Federal Guidance Report 11 and are used along with the calculated intake value to estimate dose.

Dose Calculation for Tritium

Equations for Single Acute or Chronic Intakes

This section shows derivations of equations used for the assessment of single acute intakes and how to do the actual assessment.

To create a dose conversion factor for an inhalation intake of tritium, the ICRP 30 (ICRP 1979) model is used. This is a single compartmental model where it

is assumed that inhalation and absorption through the skin contribute to the intake. The ICRP lung model is not used in this case. Tritium is assumed to be instantaneously and homogeneously spread throughout the body fluids. It is also assumed that the concentration of tritium in the excreta is the same as the concentration of tritium in the body. For the purposes of dose assessment, the source organ for tritium is the body fluids (42,000g) and the target organ is the whole body.

Figure 5 shows a schematic diagram for a single-compartmental model. In the above illustration, *N(t)* is the number of atoms of the radionuclide in the compartment at time *t*, k_1 is the total translocation rate constant out of the compartment, K_1 is the stable element translocation rate constant out of the compartment, and λ is the radiological half life. This shows how there are two removal processes out of the compartment, physical removal through excretion and radioactive decay.

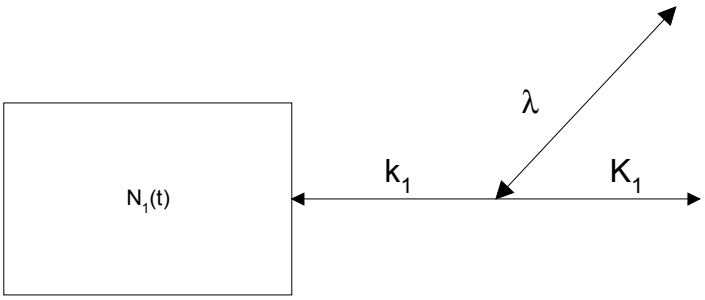

Figure 5: Single-compartment model for tritium.

The equation that the ICRP uses to calculate committed effective dose equivalent for a system with one target organ and one source organ is:

$$H_{50}(T \leftarrow S) = 1.6x10^{-10}\ U_s\ SEE(T \leftarrow S) \text{ Sv.} \quad (20)$$

where: $H_{50}(T \leftarrow S)$ = committed effective dose equivalent from source organ *S* to

target organ T,

U_s = number of disintigrations in the compartment over 50 years, and

$SEE(T \leftarrow S)$ = *specific effective energy*, the energy per unit mass of target organ per transformation that is transferred from the source organ to the target organ.

This equation can be used to calculate a dose conversion factor by assuming an intake of a reference amount, such as 1 Bq, and calculating a dose in some units, such as Sv. The dose equivalant calculated will be a dose conversion factor in Sv/Bq. The derivation follows:

$$A = N\lambda,$$

$$N = \frac{A}{\lambda} = \frac{1\ \text{Bq}}{\ln 2 / 12..262\ \text{y}} x \frac{3.154x10^{7}\ \text{s}}{1\ \text{y}}, \tag{21}$$

$$N = 5.579x10^{8}\ \text{atoms}$$

The number of transformations that occur in the compartment can be calculated by multiplying the number of atoms entering the compartment by a *branching fraction* of those removed from the compartment by radioactive decay:

$$U_s = N\frac{k_r}{k_1} = \left(5.579x10^{8}\ \text{atoms}\right)\frac{1.549x10^{-4}}{k_1},$$

$$U_s = \frac{8.624x10^{4}}{k_1}\ \text{transformations.} \tag{22}$$

This value plus the specific effective energy can be plugged into equation (20) to obtain the dose conversion factor:

$$DCF = 1.6x10^{-10}\left(\frac{8.624x10^{4}}{k_1}\right)\left(9.0x10^{-8}\ \text{MeV/ g} \bullet \text{transformation}\right)\text{Sv / Bq},$$

$$DCF = \frac{1.244x10^{-12}}{k_1}\ \text{Sv/ Bq.} \tag{23}$$

Using the ICRP (ICRP 1979) value for the physical removal rate constant of 10 days in this equation gives:

$$DCF = \frac{1.244x10^{-12}}{\ln 2/10 \text{ days}},$$

$$DCF = 1.795x10^{-11}\text{Sv/ Bq},$$

$$DCF = 0.06640 \text{ mrem/ } \mu\text{Ci}. \tag{24}$$

These dose conversion factors must be used in conjunction with an equation which calculates the dose from the concentration of tritium in the urine. This equation is:

$$C_n = C_0 e^{-k_1 t_n}, \tag{25}$$

where: C_n = result of urine concentration (Ci/l),
C_0 = maximum concentration at time of acute intake (Ci/l),
k_1 = total translocation rate constant, and
t_n = time of sampling after acute intake.

This equation can be rewritten to solve for the peak concentration:

$$C_0 = \frac{C_n}{e^{-k_1 t_n}} = C_n e^{k_1 t_n}. \tag{26}$$

Since the intake of tritium can be calculated by multiplying the peak concentration of tritium in the body by the volume of body fluids, the dose equivalent can then be calculated by:

$$H_{50} = (C_n e^{k_1 t_n} \mu\text{Ci/ l})\,(42\text{l})\left(\frac{0.004603}{k_1}\text{mrem/ }\mu\text{Ci}\right),$$

$$H_{50} = 0.1933\ C_n\ \frac{e^{k_1 t_n}}{k_1}\ \text{mrem}. \tag{27}$$

Using the accepted value of k_1 of ln2/10 days, equation 27 reduces to:

$$H_{50} = 2.789\ C_n e^{\ln\frac{2}{10} t_n}\ \text{mrem}.$$

$$H_{50} = 2.789\ C_n e^{0.0693 t_n}\ \text{mrem}. \tag{28}$$

Dose Calculation from Bioassay Data Using Other Methods

Dose calculation can be performed using least-squares fitting for several data points or from available intake retention fractions for one data point if necessary.

Single Data Point

Intake retention fractions are available in NUREG CR-4884 (Lessard et al. 1987). Intakes are calculated from an individual data point by obtaining the appropriate intake retention fraction either directly from the corresponding list or by interpolation between the values. If using interpolation, exponential interpolation is recommended due to the exponential nature of the retention functions. The intake is calculated by dividing the bioassay result by the intake retention fraction.

Least-squares Fitting

Least-squares fitting is useful for determining the intake when there are multiple bioassay methods used or there are multiple intakes. Intake retention fractions for use in least-squares fitting are available from NUREG CR-4884 (Lessard et al. 1987) as indicated above, or INDOS can be used to calculate them from the appropriate metabolic model. Matrices are set up with the intake retention fractions in as many columns as there are intakes and the results in a column matrix. The results in the form of intake or intakes can be multiplied by the appropriate dose conversion factors to obtain committed effective dose equivalents.

Source Term Summary

SNL/NM activities may potentially expose workers to a variety of radionuclide compounds. However, the majority of internal exposures at SNL/NM are expected to result from a relatively few radionuclides. The primary radionuclides used at SNL/NM are tritium, the isotopes of uranium, and the isotopes of plutonium. The proper selection and interpretation of bioassay techniques requires an understanding of the physical, radiological, and biologic properties of these radionuclides.

The deposition, absorption, distribution, retention, and elimination of radioactive materials in the body involve complex processes which can substantially differ between individuals. The 10 CFR 835 (U.S. DOE 1993) and RCM requirements are based on biokinetic models reported in International Commission on Radiological Protection (ICRP) Publication 30 *Limits for Intakes of Radionuclides by Workers* (ICRP 1979). This TBD recognizes the ICRP 30 recommendations as the best practicable methodology to assess internal radionuclide exposures of large worker populations. Therefore, this TBD adopts all of the ICRP principles for use in the RPID Program. The ICRP has published revised methodologies and philosophies for monitoring internal exposures in ICRP 60 (ICRP 1991b) and ICRP 61 (ICRP 1991a). This TBD will be revised, as necessary, to incorporate new recommendations from these publications once they are recognized in the 10 CFR 835 and the RCM.

Tritium

Tritium (^{3}H or T) is the only radioactive species of the three major isotopes of hydrogen (e.g., protium (H), deuterium (D), and T). Tritium decays by beta emission with an average energy of 5.7 KeV and a radiological half-life of 12.3 years. Tritium may be encountered in numerous physical and chemical forms since its chemical properties are almost identical to H which is ubiquitous in the environment. Tritium is normally encountered at SNL/NM as tritium gas, in the form of HT, DT, or T_2, and tritiated water, in the form of HTO, DTO, or T_2O.

Facilities Using Tritium

Radiological workers at the following facilities which process, store, or are potentially contaminated with tritium should be considered for participation in the RPID Program:

TA I: Building 891 - Facilities containing erbium tritide (ErT_3)

TA II: Entire Area - Tritium may be contained within test components which can be converted to DT or HTO when exploded.

TA III: Entire Area - Tritium is a potential contaminant at some environmental restoration sites (e.g., Mixed Waste Landfill, etc.)

Radioactive and Mixed Waste Management Facility (RMWMF) – Any radionuclides used at SNL including ^3H, U, Th, Pu, Am, Sr and tracer quantities of others.

TA IV: PBFA-II and SABER facilities use accelerator targets containing tritium

TA V: ACRR, Hot Cell Facility, SPR facilities are potentially contaminated by tritium compounds.

CTF: (none known)

TTR: (none known)

Biokinetics of Gaseous Tritium

Tritium gas (HT) can be directly absorbed into the systemic circulation through the lungs following inhalation exposures. According to the laws of partial pressures, the absorbed HT remains in the body with a biological half-life of about two minutes (MLM 91). Most of the HT is eliminated through exhalation. However, a small percentage (e.g., 0.003 to 0.005 percent) is converted to tritiated water (HTO) within the GI tract and remains in the body for longer time periods. The fraction of HT converted to HTO behaves identically to exposures to HTO. Skin absorption of HT has been found to be negligible (MLM 91).

The DAC of HT is 2 x10^{10} Bq/m^3 which is 25,000 times less than the HTO DAC (i.e., 8 x10^5 Bq/m^3) (ICRP 78). The HT DAC is based on lung dose limits since the short residence time of HT in the body limits the whole-body exposure. In fact, the dose contribution from the HTO converted from HT is approximately equal to the lung exposure from HT. Inhalation exposures to tritium at SNL/NM is expected to involve both HT and HTO, although exposure to HT is much more likely because of the greater source term. In most applications, the exposure from HTO will be the limiting factor since its DAC is more than 4 orders of magnitude lower than the HT DAC. Therefore, bioassay techniques for HTO are considered appropriate for HT monitoring of workers participating in the RPID Program. Special bioassay procedures (e.g., breath analysis) may be performed if air sampling indicates the potential for chronic occupational exposures to HT.

Biokinetics of Tritiated Water

HTO can be absorbed into the body through multiple pathways. HTO is efficiently and rapidly absorbed into the systemic circulation after inhalation exposures (e.g., up to 99 percent efficient). Ingested HTO is almost entirely absorbed by the GI tract and is distributed throughout the whole-body in minutes. The skin absorption pathway can also be important due to the normal movement of water through the skin, especially in hot weather. The measured percutaneous absorption may be equivalent to approximately 50% of the HTO inhalation exposure under certain conditions (ICRP 1979).

Regardless of the exposure pathway, a large fraction of the absorbed HTO is uniformly distributed throughout the whole-body in the form of body water. The retention of this fraction is characteristic of the normal turn-over rate of water which can be modeled as simple exponential function. A smaller HTO fraction becomes incorporated into liable (i.e., exchangeable hydrogen bond sites) and stronger, non-labile hydrogen bond sites in organic molecules. This fraction exhibits the longer turn-over rate characteristic of cellular components which can be modeled as the sum of two or more exponential terms. Therefore, the HTO intake retention function (IRF) can be described as follow (ICRP 78):

$$IRF(t)_{HTO} = A e^{-\lambda_1 \bullet t} + B e^{-\lambda_2 \bullet t} + C e^{-\lambda_3 \bullet t} \quad (29)$$

where: A = Fraction of exposure contained within body water

λ_1 = Removal rate of body water (d^{-1})
B = Fraction of exposure contained at labile molecular sites
λ_2 = Turnover rate of labile cellular components (d^{-1})
C = Fraction of exposure contained at non-labile molecular sites
λ_3 = Turnover rate of non-labile cellular components (d^{-1})
t = post acute exposure time interval (d)
$IRF(t)_{HTO}$ = Fraction of initial HTO exposure retained at time t

The overall dose contribution from labile and non-labile site HTO depositions have been found to represent less than 10 percent of the CEDE (ICRP 1979). The ICRP does not include the organically bound tritium fraction in the HTO ALI calculation. Therefore, estimates of CEDE from HTO exposures during routine bioassay monitoring in the RPID Program will include only the body water component. The removal rate of HTO in body waters can be calculated as follows:

$$\lambda_1 = \frac{0.693}{T_R} + \frac{0.693}{T_B} \qquad (30)$$

where: TR = Radiological half-life of tritium (d)
TB = Biological clearance half-life of body water (d)
λ_1 = Removal rate of HTO in body water (d^{-1})

The biological clearance half-life of body water (T_B) is closely related to fluid intake patterns. Individual patterns can fluctuate widely depending on such factors as ambient air temperature, perspiration rates, and personal lifestyles (e.g., exercise patterns). Values of T_B have been observed to range between 4 and 18 days with an average of about 10 days. The average value is in agreement with the approximate half-life obtained based on standard man assumptions (i.e., Daily water intake rate of 3,000 grams and a constant 42,000 grams total body water mass). Therefore, the TB value for body waters will be assumed to be 10 days for routine applications of the RPID Program. Inserting the TB (i.e, 10 days) and TR (i.e. 4490 days) values yields the following HTO IRF:

$$IRF(t)_{HTO} - \frac{C_t}{C_o} = e^{-0.0695 \cdot t} \qquad (31)$$

where: C_t = Concentration of HTO in body water at time t
C_o = Initial concentration of HTO in body water

t = time post acute exposure (d)
$IRF(t)_{HTO}$ = Retention of HTO

HTO is assumed to be totally eliminated from the body in urine. Therefore, urine HTO concentrations is representative to body water HTO concentrations. If necessary, radiological incident evaluations (e.g., accident evaluations) may include the organically bound components in the HTO retention function for more precise CEDE estimates.

Biokinetics of Other Tritiated Compounds

Tritiated vacuum pump oil may be a significant exposure source at SNL/NM accelerator facilities. Tritium specific activities may range between a few mCi/L to a few tens of Ci/L. The primary radiological hazard from tritiated pump oils remains HTO since it has greater skin permeability. Routine bioassay monitoring of HTO should be sufficient to monitor exposures to tritiated oils (MLM 91).

Tritium may also be encountered in the form of metal tritides (e.g., titanium tritide and erbium tritide) during neutron generator research at SNL/NM. Current health protection for guidelines for metal tritides are based on the assumption that the biological behavior of these compounds is similar to HTO. However, recent research suggests that retention times of metal tritides may significantly exceed HTO. The RPID Program will continue to use the HTO models to represent metal tritides until new models are developed and confirmed by animal tests.

Another source of HTO contamination is tritiated solvents. Exposure pathways from solvents include skin absorption and inhalation of the volatile components. Solvents may be deposited in organs which are not normally effected by HTO. The limiting consideration of these exposures may be the chemical toxicity of the solvent. Exposure assessment techniques are expected to be variable depending on the solvent form. Facilities using tritiated solvents will require site-specific consideration in the RPID Program.

Skin contact with tritium contaminated glass or metal surfaces have been shown to result in exposures to organic tritium forms. Such exposures can result in elevated tritium concentrations at the exposure site and other tissues in addition to large amounts of organic tritium in urine. The tritium dose contribution from these compounds are often insignificant compared to the concurrent absorbed HTO dose.

Therefore, surface contamination sources which generate organic tritiated compound exposures will not be considered during routine bioassay monitoring in the RPID Program. Individual exposure assessments following radiological incidents may include consideration of these exposures.

Techniques and Considerations for Tritium Bioassay

In-vivo bioassay techniques to assess T body burdens are not possible since the beta emission can not be detected outside of the body. Tritium is quickly distributed throughout the whole-body within 2 hours post exposure to HTO. Once equilibrium is achieved, the body burden of T can be assessed using one of several in-vitro bioassay techniques (e.g., blood, saliva, and urine). Fecal analyses is not practiced for monitoring tritium exposures since HTO is assumed to be totally absorbed from the GI tract. Because the collection and interpretation is relatively simple, urinalysis will be used to assess T body burdens in the RPID Program.

Table 4: Tritium Intake Limits (ICRP, 1979)

Physical/Chemical Form	Parameter	Ingestion Pathway (Limiting Factor)	Inhalation Pathway (Limiting Factor)
Tritiated Water (HTO)	ALI (Bq)[1]	3×10^9 (Stochastic)	3×10^9 (Stochastic)
	DAC (Bq/m3)[2]	N/A	8×10^5
Elemental Tritium (HT)	ALI (Bq)	N/A	N/A
	DAC (Bq/m3)	N/A	2×10^{10} (Non-stochastic - Lungs)

[1] 1 Bq = 2.7×10^{-5} uCi

[2] 1 Bq/m^3 = 27 uCi/ml

Urine samples collected shortly after exposure may not have had sufficient time to establish equilibrium within body waters. Subsequent analysis of these samples would not be representative of the true T body burden. The preferred method is to wait 2 hours followed by voiding the non-representative urine sample. This sample may be used to represent pre-accident, baseline conditions.

Tritium Intake Limits

Because a critical organ has not been identified, the ingestion and inhalation pathway ALI for HTO are determined from the stochastic limit to the whole-body (i.e., 5 rem CEDE limit). The ALI for HTO skin absorption and wound exposures are also assumed to be equivalent to these values. Table 4 summarizes the ALI and the DAC for exposures to HT and HTO.

Uranium

Isotopes of uranium may be encountered at several facilities or activities at SNL/NM. However, occupational exposures mainly occur from natural uranium (U-Nat), depleted uranium (DU), and enriched uranium (EU) containing different proportions of the naturally occurring uranium isotopes (i.e., ^{234}U, ^{235}U, and ^{238}U).

Table 5: Significant Uranium Decay Series Radionuclides at SNL/NM

Uranium Decay	Radionuclide	Energies (MeV) and % Abundances of Major Emissions		
Series Member	Half-Life	Alpha	Beta	Gamma
^{238}U	$4.51x10^{9}$ y	4.15 (25%) 4.20 (75%)	(none)	(none)
^{234}Th	24.1 d	(none)	0.103 (21%) 0.193 (79%)	0.063 (3.5%) 0.093 (4%)
^{234m}Pa	1.17m	(none)	2.29 (98%)	0.765 (0.3%) 1.001 (0.6%)

Progeny of these isotopes are radioactive and form decay chains. Uranium-238 and ^{234}U are members of the uranium decay series, while ^{235}U is a member of the actinium decay series. Several of these progeny may have significant internal dosimetry implications when secular equilibrium is maintained. However, most of the uranium forms encountered at SNL have been chemically extracted from the virgin feed materials. Progeny with long half-lives (e.g., ^{234}U in the uranium series,

and ^{231}Pa in the actinium series) effectively prevent secular equilibrium along the entire decay chain in these cases. The resultant radionuclide progeny which occur in significant abundance to impact radiological controls are ^{234}Th and ^{234m}Pa in the uranium series, and ^{231}Th in the actinium series (EGG 88). However, other decay progeny may be present from incomplete chemical separations and from naturally occurring deposits of uranium. Tables 5 and 6 present the radiological characteristics of common uranium isotopes and major progeny.

Table 6: Significant Actinium Decay Series Radionuclides at SNL/NM

Actinium Decay	Radio-nuclide	Energies (MeV) and % Abundances of Major Emissions		
Series Member	Half-Life	Alpha	Beta	Gamma
^{235}U	7.04x10^{8} y	4.37 (18%) 4.40 (57%) 4.58 (8%)	(none)	0.144 (11%) 0.185 (54%) 0.204 (5%)
^{231}U	25.5 h	(none)	0.14 (45%) 0.22 (15%) 0.305 (40%)	0.026 (2%) 0.084 (10%)

All of the uranium isotopes decay by alpha emission while the non-uranium decay progeny are primarily beta emitters. Thus, significant internal doses result from exposures to the uranium isotopes while the dose contribution from the decay series progeny are often ignored. In-vivo bioassays are possible due to the presence of the uranium progeny, with the exception being ^{235}U. The biokinetics of uranium compounds are dependent on the exposure pathway (e.g., inhalation, ingestion, etc.) and on the chemical solubility of the compound in bodily fluids.

SNL/NM Facilities Using Uranium Compounds

Radiological workers at the following facilities which process, store, or are potentially contaminated with uranium compounds should be considered for participation in the RPID Program:

TA I: (none known)

TA II: (none known)

TA III: Entire Area - Uranium compounds are potential contaminants at some environmental restoration sites. Uranium may also be found in the Radioactive and Mixed Waste Management Facility (RWMWF).

TA IV: (none known)

TA V: ACRR, Hot Cell Facility, SPR facilities process or are contaminated by uranium compounds.

CTF: Entire Area - Uranium compounds are potential contaminants at some environmental restoration sites.

TTR: Entire Area - Uranium compounds are potential contaminants at some environmental restoration sites.

Manzano: Waste storage containing uranium.

Biokinetics of Inhaled Uranium Compounds

The principle exposure pathway for uranium compounds is inhalation in most occupational situations. Table 7 contains the clearance classes for uranium compounds established by the ICRP (ICRP 1988):

Table 7: ICRP Pulmonary Clearance Classes for Uranium Compounds

Uranium Compound	Pulmonary Clearance Class
UF_6, UO_2F_2, $UO_2(NO_3)_2$	D
UO_3, UF_4, Ucl_4	W
UO_2, U_3O_8	Y

In general, the biokinetics of inhaled particles are characterized by an initial rapid clearance, due to mucociliary transport into the GI system, followed by long-term clearance attributed to particle dissolution and mucociliary actions. Table 9

illustrates the ultimate biokinetic fate of inhaled 1 micron AMAD particles containing uranium based on ICRP models (Rich et al. 1988):

Table 8: Biokinetic Fate of 1 μm AMAD Uranium Compounds

Clearance Class	Percent Exhaled from the body	Percent Absorbed into Systemic Circulation	Percent Removed to the GI Tract
D	37	47.6	15.4
W	37	12	51
Y	37	5.4	57.6

Biokinetics of Ingested Uranium Compounds

Depending on its solubility, ingested uranium compounds can be absorbed into the body via the small intestine or eliminated from the body within fecal materials. Table 9 provides the ICRP GI absorbed fraction (f_1) which is related to the clearance class of the uranium compound:

Table 9: ICRP Absorption Fractions (f_1) for Ingestion Exposures (ICRP, 1979)

Clearance Class	Absorbed Fraction of Activity into Systemic Circulation (f_1)
D	0.05
W	0.05
Y	0.002

A maximum of 5 percent of the total ingested uranium is expected to reach systemic circulation. The remaining portion remains in the GI tract and is ultimately eliminated within fecal materials. Since there are relatively few penetrating radiation emissions from the uranium compounds, internal dosimetry implications of the unabsorbed portion are minimal. The *INDOS* computer program is based on the ICRP 30 GI Tract Model and accurately models ingestion exposures from uranium compounds. Therefore, analytical results from fecal

samples can be used to determine uranium body burdens using the *INDOS* program.

Biokinetics of Absorbed Uranium Compounds

Once introduced into the systemic circulation, via inhalation, ingestion, skin absorption, or wound exposures, all uranium compounds are assumed to be transported and retained similarly in the various body organs. Approximately 20 percent of absorbed uranium compounds are deposited in the skeleton, 12 percent within the kidneys, and 12 percent distributed throughout the whole body. Uranium IRF of various body organs are described as follows (ICRP 1979):

$$IRF_B = 0.2\,e^{\left(\frac{-0.693 \bullet t}{20}\right)} + 0.023\,e^{\left(\frac{-0.693 \bullet t}{5000}\right)} \quad (32)$$

$$IRF_K = 0.12\,e^{\left(\frac{-0.693 \bullet t}{6}\right)} + 0.00052\,e^{\left(\frac{-0.693 \bullet t}{1500}\right)} \quad (33)$$

$$IRF_O = 0.12\,e^{\left(\frac{-0.693 \bullet t}{6}\right)} + 0.00052\,e^{\left(\frac{-0.693 \bullet t}{1500}\right)} \quad (34)$$

where: t = Time post acute intake (d)
IRF_B = Bone uranium retention function
IRF_K = Kidney uranium retention function
IRF_O = All other body organs uranium retention function

The balance of the uranium compounds absorbed into the systemic circulation (i.e., approximately 54%) is assumed to be directly eliminated from the body within urine. The IRF for this component is defined as follows (ICRP 1979):

$$IRF_U = 0.54\,e^{\left(\frac{-0.693 \bullet t}{0.25}\right)} \quad (35)$$

where: t = Time post acute intake (d)
IRF_U = retention function for fast urinary clearance fraction

All uranium removed from these mathematical compartments is assumed to eliminated within urine. Therefore, urinalysis can be used to determine uranium systemic body burdens using the *INDOS* program. The bone uranium retention

function is of additional interest since the bone surface is considered the critical organ from uranium exposures. The kidney retention function is useful in determining kidney uranium burdens for comparison with regulatory limits concerning uranium nephrotoxicity.

Chemical Toxicity of Uranium Compounds

Health effects from exposures to uranium compounds include chemical toxicity hazards (i.e., nephrotoxicity) in addition to radiological hazards. In certain situations, acceptable radiological exposures may exceed regulatory thresholds for chemical exposures. The determination of the limiting hazard (i.e., radiological or chemical) is dependent on the solubility of the uranium compound. Based on a radiological limit of 5 rem and chemical limits derived from the Occupational Safety and Health Administration standards, chemical toxicity effect thresholds are more limiting for exposures to DU and U-Nat compounds of transportability class D and W (Rich et al. 1988). In contrast, class Y compounds are always limited by radiological concerns, regardless of the uranium enrichment. The majority of uranium exposures at SNL are expected to be in the form of oxides (i.e., class Y chemical forms). Therefore, the governing regulatory constraints are expected to be based on radiological hazards.

Techniques and Considerations of Uranium Bioassay

In-vivo bioassay of uranium exposures are possible due to the radiation emissions of uranium progeny and of ^{235}U (Tables 3-3 and 4-3). However, the SNL WBC is not considered sufficiently sensitive to detect small uranium body burdens. Therefore, the WBC will not be used for routine bioassays. More sensitive in-vivo bioassay techniques, such as lung counting, can be performed at a contract laboratory to evaluate significant exposures to these compounds. It should be noted that bioassays based on uranium progeny assume that the progeny is retained at the same location within the body as the parent isotope. This is an assumption used in ICRP Publication 30 (ICRP 1979).

Fecal bioassay can be performed in the event of an identified significant exposure event. However, routine monitoring is not recommended in the RPID Program since uranium compounds are rapidly removed via feces and may not be detected. Thus, urinalysis will be the in-vitro technique of choice for routine monitoring applications in the RPID Program.

The uranium isotopes are ubiquitous throughout the environment. Non-occupational exposures occur from dietary uranium intake through foods and drinking water. As a result, workers routinely ingest approximately 2 micrograms of uranium each day which may be misinterpreted-interpreted as occupational exposures (ICRP 1975). Because these body burdens result from chronic exposures which result in fairly uniform elimination rates, baseline bioassays (i.e., urinalysis) are invaluable to discriminate occupational and environmental exposures. Special bioassay studies of non-radiological workers may prove useful in determining average environmental uranium body burdens of SNL/NM workers. For the purposes of this TBD, chronic uranium exposures from food and water consumed at SNL/NM are considered part of the worker's environmental exposure (i.e., all occupational exposures to uranium result from acute exposure events).

The time of in-vitro bioassay sample collection may be useful in determining the source of exposure. Collections at the end of the work week would be more sensitive for occupational exposures, while samples collected after extended leaves of absence (e.g., vacations) would result in more accurate assessments of environmental exposures.

Intake Limits of Uranium Compounds

Table 10: ICRP Intake Limits of U

(derived from information in Rich et al. 1988)

% Enrichment	Assumed Class	DAC (μg/m^3)	ALI (mg)
0.2 (Depleted)	Y	40	100
1	D	700	2000
2	D	500	1000
5	D	250	600
10	D	100	200
20	D	60	100
50	D	20	50
100	D	8	20

The ALI of uranium compounds which are quickly absorbed into the systemic circulation (e.g., inhalation of class D compounds, ingestion of class D and W

compounds, and wound exposures) are determined from the non-stochastic dose limit (i.e., 50 rem) to the bone surfaces. The ALI from other exposures are limited by the stochastic dose limit to the whole-body. Table 10 summarizes the intake limits for several enrichments of uranium.

Plutonium and Americium

Plutonium exposures at SNL/NM are expected to be rare compared to other exposure sources. The biokinetics of plutonium isotopes are included in this document because of the high degree of public sensitivity to these compounds and their low ALI (i.e., high radiotoxicity). The principle isotopes of plutonium found in non-production DOE facilities are ^{238}Pu and ^{239}Pu. The radiological properties of these isotopes are summarized in Table 13. Compounds of these isotopes potentially contain other radionuclides. These contaminants are typically found in minute quantities and are subsequently ignored in routine bioassay programs. The exception is ^{241}Am, which is significant for in-vivo bioassay application and is also noted in Table 11.

Table 11: Plutonium Radionuclides at SNL (Faust et al. 1988)

Radio-nuclide	Radio-nuclide	Energies (MeV) and % Abundances of Major Emissions		
	Half-Life	Alpha	X-ray	Gamma
^{238}Pu	87.7 y	5.50 (71.6%) 5.46 (28.3%)	0.011 to 0.021 (10.5%)	(none)
^{239}Pu	2.41x10^{4} y	5.156 (73.8%) 5.143 (15.2%) 5.105 (10.7%)	0.0116 to 0.0215 (4.8%)	(none)
^{241}Am	432 y	5.486 (85.2%)	0.0119 to 0.0222 (37.6%)	0.0595 (35.7%) 0.0263 (2.4%)

All of the relevant plutonium isotopes decay by alpha particle emission which cannot be detected using normal in-vivo techniques. However, in-vivo bioassay is possible by assessing the uranium L x-rays from plutonium decay. Detection of these x-rays are difficult due to their low energies which can be easily attenuated within the subject. The 59.5 KeV gamma ray emission from ^{241}Am is easier to detect and can be used to quantify smaller exposures of plutonium when the ^{241}Am/Pu ratio is known. The biokinetics of plutonium compounds are dependent on the exposure pathway (e.g., inhalation, ingestion, etc.) and on the chemical solubility of the compound in body fluids.

SNL/NM Facilities Using Plutonium

Radiological workers at the following facilities which process, store, or are potentially contaminated with plutonium compounds should be considered for participation in the RPID Program:

TA I: (none known)

TA II: (none known)

TA III: (none known)

TA IV: (none known)

TA V: ACRR, Hot Cell Facility, SPR facilities process or are contaminated by plutonium compounds.

CTF: (none known)

TTR: Entire Area - plutonium compounds may be present at some environmental restoration sites.

Biokinetics of the Plutonium Exposure Pathways

Deposition patterns of inhaled plutonium compounds are assumed to adequately described by the ICRP Lung model (ICRP 1986). Retention in the lungs

is a function of the compound's chemical solubility in body fluids. Table 12 contains the pulmonary clearance class for plutonium compounds.

Table 12: Pulmonary Clearance Classes of Americium and Plutonium Compounds (ICRP 1986)

Compound Type	Pulmonary Clearance Class
Plutonium	
All except oxides	W
Oxides	Y
Americium	
All compounds	W

Ingestion exposures are typically insignificant due to the low absorption rate of plutonium compounds into systemic circulation. The ICRP GI tract model is assumed to adequately model plutonium ingestion exposures. Table 13 contains the f_1 fractions for plutonium.

Table 13: GI Absorption Fractions for Plutonium Compounds (Faust et al. 1988)

Compound Type	f_1
Plutonium	
Oxides (except polydisperse oxides)	0.00001
Nitrates	0.0001
Other compounds or unknown mixtures	0.001

The skin absorption pathway is also considered insignificant due to the low solubility of plutonium compounds. Wound exposures often result in the most significant plutonium exposures at DOE facilities. All of the plutonium compound is assumed to be directly introduced into systemic circulation unless proven otherwise by wound counting in-vivo bioassay techniques.

Biokinetics of Absorbed Plutonium Compounds

Once introduced into the systemic circulation, plutonium is primarily deposited in the liver and skeletal system (Faust et al. 1988). The reported distribution and the biologic clearance times varies between ICRP Publication 30 (ICRP 1979) and ICRP Publication 48 *The Metabolism of Plutonium and Related Elements* (ICRP 1986). The ICRP 30 distribution factors are used in conjunction with the biologic clearance parameters from ICRP 48 to model plutonium retention in the ICRP Publication 54 (ICRP 88). The RPID Program adopts this approach since this philosophy reflects the most current guidance from the ICRP. The following IRF for the bone and liver compartments are derived using the ICRP 54 recommendations:

$$IRF_B = 0.45\, e^{\left(\frac{-0.693 \bullet t}{18250}\right)} \quad (36)$$

$$IRF_L = 0.45\, e^{\left(\frac{-0.693 \bullet t}{7300}\right)} \quad (37)$$

where: IRF_B = Bone intake retention function
IRF_L = Liver intake retention function
t = time post acute exposure (d)

The remaining 10 percent of the systemic body burden is assumed to be distributed to a number of minor deposition sites. The ICRP considers the conservatism built into the liver and bone models (i.e., dose over-estimation) compensates for ignoring the minor deposition sites in dosimetry evaluations (ICRP 86) However, dose assessments to the gonads may be of interest since the biological clearance time of these organs is considered infinite. Partitioning factors vary from 0.035 percent in males, to 0.011 percent in females. The partitioning value for males with a biological clearance half-time of 100 years will be conservatively used for routine bioassay applications in the RPID Program. The remaining fraction of plutonium in systemic circulation is modeled by urine elimination with a biological half-life of 0.25 days. The adoption of these assumptions results in remaining organ-specific IRF for absorbed plutonium:

$$IRF_G = 0.00035\, e^{\left(\frac{-0.693 \bullet t}{36500}\right)} \quad (38)$$

$$IRF_U = 0.09965\, e^{\left(\frac{-0.693 \bullet t}{0.25}\right)} \quad (39)$$

where: IRF_G = Gonad retention function
IRF_U = Early urinary excretion retention function
t = times post acute intake (d)

The plutonium burdens of these organs can be derived from intake estimates using the *INDOS* computer program. In addition, these equations provide an accurate description of total daily plutonium excretion (i.e., urine and fecal output) at 10,000 days post exposure or greater (ICRP 1988). However, these IRF are inadequate in predicting excretion patterns 5 years (1825 d) or earlier post exposure (ICRP 1988). Since the goal of the RPID Program is early detection of internal exposures, alternate methods based on excretion models are required to estimate intakes from plutonium in-vitro bioassay data.

Excretion Patterns of Plutonium Compounds

Exponential excretion functions have been developed that adequately fit plutonium data. The Jones and Durbin equations describes the excretion of plutonium in urine and feces as follows (ICRP 1988):

$$E_U = 0.00475\, e^{(-0.558 \bullet t)} + 0.000239\, e^{(-0.0442 \bullet t)} + 0.0000855\, e^{(-0.0038 \bullet t)} + 0.0000142\, e^{(-0.0000284 \bullet t)} \quad (40)$$

$$E_F = 0.006\, e^{(-0.347 \bullet t)} + 0.0016\, e^{(-0.105 \bullet t)} + 0.00012\, e^{(-0.0124 \bullet t)} + 0.00002\, e^{(-0.00182 \bullet t)} + 0.000012\, e^{(-0.000173 \bullet t)} \quad (41)$$

where: E_U = Fraction of original intake of plutonium in urine at time t
E_F = Fraction of original intake of plutonium in feces at time t
t = time post exposure (d)

These equations are considered reliable for predicting excretion up to 5 years post exposure. Initial intakes will be over-estimated when evaluating later excretion data using these equations. The excretion function are related to the IRF by the following relationship:

$$E(t)_i = -f_i \bullet \frac{d}{dt} IRF(t) \quad (42)$$

where: f*i* = Fraction of systemic burden secreted by route *i*
IRF (t) = The systemic intake retention function
E*i* = Daily excretion function of route *i*

Therefore, a pseudo systemic IRF can be derived by integrating the excretion function and defining the excretion fraction for the respective elimination routes. The "pseudo" designation is important since the IRF inferred does not represent actual biokinetic compartments (i.e., equation is based on observed excretion patterns, not systemic uptake patterns). However, the pseudo IRF can be used to estimate the initial intake of plutonium compounds using the *INDOS* program. The derived pseudo IRF equations are as follows:

$$IRF_U = 0.0159\, e^{(-0.558 \bullet t)} + 0.0101\, e^{(-0.0442 \bullet t)} + 0.0419\, e^{(-0.0038 \bullet t)} + 0.9321\, e^{(-0.0000284 \bullet t)} \quad (43)$$

$$IRF_F = 0.141\, e^{(-0.347 \bullet t)} + 0.124\, e^{(-0.105 \bullet t)} + 0.079\, e^{(-0.0124 \bullet t)} + 0.0897\, e^{(-0.00182 \bullet t)} + 0.5663\, e^{(-0.000173 \bullet t)} \quad (45)$$

where: IRF_U = Pseudo IRF for determining intake from urinalysis
IRF_F = Pseudo IRF for determining intake from fecal analysis
t = time post acute intake (d)

The excretion fractions for use in the *INDOS* program are as:

Plutonium fraction excreted as urine (f_U) = 0.54
Plutonium fraction excreted as feces (f_F) = 0.12

The use of the pseudo IRF in the *INDOS* program will allow assessment of the initial uptake of plutonium compounds from fecal and urine in-vitro analyses.

Biokinetics of Absorbed Americium Compounds

The biokinetics of compounds containing ^{241}Am are considered to be similar to plutonium compounds by the ICRP (ICRP 1988). Therefore, the IRF and pseudo IRF applicable for plutonium intakes will be used to evaluate ^{241}Am exposures in the RPID Program.

Techniques and Considerations in Plutonium Bioassay

In-vivo bioassay of plutonium exposures require specialized facilities to detect and quantify the low-energy x-ray emissions from plutonium isotopes. The sensitivity of the SNL/NM WBC is insufficient to detect significant plutonium exposures (e.g., exposures exceeding the ALI). Therefore, in-vivo plutonium bioassay is not practical for routine monitoring at SNL/NM. Lung counting facilities at LANL will be used when inhalation exposures to plutonium compounds are suspected.

Routine in-vitro bioassay for plutonium exposures at SNL/NM will be primarily performed by urinalysis. Fecal analyses can also be performed to characterized suspected exposures to plutonium. However, routine fecal monitoring is not recommended in the RPID Program since inhaled plutonium compounds are rapidly cleared from the body and may not be detected.

Because of the small ALI for plutonium isotopes, the analytical sensitivity achieved for in-vitro analyses is not sufficient to meet the monitoring requirements in 10 CFR 835 (i.e., detect exposures equivalent to 100 mrem CEDE). In addition, the bioassay frequency necessary to meet these requirements may be prohibitively frequent for worker convenience and cost considerations. The use of personal air samplers may be required when significant exposures to plutonium are possible.

Intake Limits of Plutonium Compounds

The ALI of plutonium and americium compounds are determined from the non-stochastic dose limit (i.e., 50 rem) to the bone surfaces except for class Y ^{238}Pu which is based on stochastic limits (ICRP 1988). Tables 14 through 16 summarize the ALI and the DAC for exposures to ^{238}Pu, ^{239}Pu, and ^{241}Am (ICRP 1979 and ICRP 1988):

Table 14: ICRP Intake Limits of 238Pu

^{238}Pu Pulmonary Clearance Class	Parameter	Ingestion Pathway[1] (Limiting Factor)	Inhalation Pathway[2] (Limiting Factor)
W (f_1 = 0.0001)	ALI (Bq)	$3x10^5$ (Bone surfaces))	$3x10^2$ (Bone surfaces)
	DAC (Bq/m3)	N/A	$1x10^{-1}$
Y (f_1 = 0.00001)	ALI (Bq)	$3x10^6$ (Bone surfaces)	$7x10^2$ (Stochastic)
	DAC (Bq/m^3)	N/A	$3x10^{-1}$

1 Ingestion ALI from ICRP 30

2 Inhalation ALI and DAC from ICRP 54 and Assume a 1 micron AMAD

Table 15: ICRP Intake Limits of 239Pu

239Pu Pulmonary Clearance Class	Parameter	Ingestion Pathway1	Inhalation Pathway2
W (f_1 = 0.0001)	ALI (Bq)	2×10^5	2×10^2
DAC (Bq/m^3)	N/A	1×10^{-1}	
Y (f_1 = 0.00001)	ALI (Bq)	2×10^6	6×10^2
DAC (Bq/m^3)	N/A	3×10^{-1}	

1 Ingestion ALI from ICRP 30

2 Inhalation ALI and DAC from ICRP 54 and Assume a 1 micron AMAD

Table 16: ICRP Intake Limits of 241Am

^{241}Am Pulmonary Clearance Class	Parameter	Ingestion Pathway[1]	Inhalation Pathway[2]
W (f_1 = 0.00005)	ALI (Bq)	5×10^4	2×10^2
DAC (Bq/m^3)	N/A		1×10^{-1}

1 Ingestion ALI from ICRP 30

2 Inhalation ALI and DAC from ICRP 54 and Assume a 1 micron AMAD

The ALI for Class W ^{239}Pu and ^{241}Am are limiting. Therefore, potential exposures to unknown mixtures of plutonium will be assumed to be Class W ^{239}Pu.

Other Radionuclides

Although less frequent, occupational exposures may result from exposures to additional radionuclides not specifically addressed in this technical basis document. Appendix A provides brief descriptions of the properties, biokinetics, and dosimetry of these radionuclides at SNL/NM. The bioassay philosophies used to monitor tritium, uranium, and plutonium exposures are anticipated to be adequate in monitoring exposures to other potential contaminants at SNL/NM. Exceptions will be addressed, as needed, in site-specific technical basis document in the RPID Program.

Quality Assurance

The internal dosimetry project is required to be adequate to demonstrate compliance with the requirements promulgated under 10 CFR 835 (U.S. DOE 1993). Therefore, internal audits of all functional project elements are be conducted. The frequency of project audits should not exceed 3 years and should include project content and implementation. Quality assurance practices will be designed to identify project deficiencies and initiate corrective actions. Audits should be performed by both internal and independent, external reviewers. In addition to the 10 CFR 835 requirements, the RCM includes auditing independent contractor internal dosimetry projects whenever applicable. Auditing frequencies will be, at a minimum, every 3 years.

The RPID Project will contain quality assurance provisions including a project re-evaluation frequency, at a minimum, of three years.

Additional Quality Assurance is provided for the elements of the RPID Project with the greatest potential to impact exposure assessments. This includes software used to calculate exposures, lab processes used during Radiobioassay, and the operation of the whole body counting system.

Internal dose calculations are performed using the software INDOS. This software is commercially available and is validated and verified by the supplier. The INDOS manuals provide an example hand calculation along with results from previous INDOS runs. Software quality assurance is provided by confirmation runs on RPID project computers using the input given in the manual and which yield the expected results. Further assurance of calculation repeatability is provided by the printed output which displays all user controlled parameters used in the calculation. A review of the calculational inputs is sufficient to confirm the results.

Radiobioassay for excreta is performed either onsite at SNL or off-site by a contract facility. In both cases the following elements are in place providing quality assurance of the sample counting results. These same elements are also applicable for in-vivo bioassay(whole body counting).

- Operating procedures,
- Staff training requirements,
- Quality control requirements for counting systems,
- Performance requirements(required MDAs),and
- A Quality Assurance Program.

For SNL facilities, both radiochemistry and whole body counting, these elements are required by the Personnel Monitoring and Laboratory Services Quality Assurance Plan. Contractor laboratories are required to have these program elements in place as stated in the Statement of Work for Radiobioassay Laboratores in order to be awarded a contract.

References

Bhattacharyya, M.H.; Breitenstein, B.D.; Metivier, H.; Muggenberg, B.A.; Stradling, G.N.; Volf, V. Guidebook for the treatment of accidental internal radionuclide contamination of workers. Radiat. Protect. Dosim. 41:1-49; 1992.

Brodsky, A. Resuspension factors and probabilities of intake of material in process (or "is 10^{-6} a magic number in health physics"). Health Phys. 39:992-1000; 1980.

Eckerman, K.; Wolbarst, B.; Richardson, C.B. Limiting values of radionuclide intake and air concentration and dose conversion factors for inhalation, submersion, and ingestion. Washington, DC: U.S. Government Printing Office; Federal Guidance Report No. 11, EPA-520/1-88-020; 1988.

Faust, L.G.; Brackenbush, L.W.; Heid, K.R.; Herrington, W.N.; Kenoyer, J.L.; Munson, L.H.; Selby, J.M.; Soldat, K.L.; Stoetzel, G.A.; Traub, R.J. Health physics manual of good practices for plutonium facilities. Richland, WA: Pacific Northwest Laboratory; PNL-6534; 1988.

Health Physics Society. Internal dosimetry programs for tritium exposure – minimum requirements. McLean, VA: Health Physics Society; American National Standard HPS N13.14-1994; 1994.

Hickey, E.E.; Stoetzel, G.A.; Strom, D.J.; Cicotte, G.R.; Wilbin, C.M.; McGuire, S.A. Air sampling in the workplace. Washington, DC: U.S. Nuclear Regulatory Commission; NUREG-1400; 1993.

International Commission on Radiological Protection. Age-dependent doses to members of the public from intake of radionuclides: part 1. Oxford: Pergamon Press; ICRP Publication 56; Ann. ICRP 20(2):1-122; 1989.

International Commission on Radiological Protection. Age-dependent doses to members of the public from intakes of radionuclides: part 2 ingestion dose

coefficients. Oxford: Pergamon Press; ICRP Publication 67; Ann. ICRP 23(3/4):1-167; 1993.

International Commission on Radiological Protection. Age-dependent doses to members of the public from intakes of radionuclides: part 3 ingestion dose coefficients. Oxford: Pergamon Press; ICRP Publication 69; Ann. ICRP 25(1):1-74; 1995.

International Commission on Radiological Protection. Age-dependent doses to members of the public from intakes of radionuclides: part 4 inhalation dose coefficients. Oxford: Pergamon Press; ICRP Publication 71; Ann. ICRP 25(3-4):1-405; 1995.

International Commission on Radiological Protection. Age-dependent doses to members of the public from intakes of radionuclides: part 5 compilation of ingestion and inhalation dose coefficients. Oxford: Pergamon Press; ICRP Publication 72; Ann. ICRP 26(1):1-91; 1996.

International Commission on Radiological Protection. Annual limits on intakes of radionuclides by workers based on the 1990 recommendations. Oxford: Pergamon Press; ICRP Publication 61; Ann. ICRP 23(4):1-41; 1991.

International Commission on Radiological Protection. General principles of monitoring for radiation protection of workers. Oxford: Pergamon Press; ICRP Publication 35; Ann. ICRP 9(4):1-36; 1982.

International Commission on Radiological Protection. Individual monitoring for intakes of radionuclides by workers: design and interpretation. Oxford: Pergamon Press; ICRP Publication 54; Ann. ICRP 19(1-3):1-315; 1988.

International Commission on Radiological Protection. Limits for intakes of radionuclides by workers. Oxford: Pergamon Press; ICRP Publication 30, Part 1; Ann. ICRP 2(3/4):1-116; 1979.

International Commission on Radiological Protection. Limits for intakes of radionuclides by workers. Oxford: Pergamon Press; ICRP Publication 30, Part 2; Ann. ICRP 4(3/4):1-71; 1980.

International Commission on Radiological Protection. Limits for intakes of radionuclides by workers. Oxford: Pergamon Press; ICRP Publication 30, Part 3; Ann. ICRP 6(2/3):1-124; 1981.

International Commission on Radiological Protection. 1990 Recommendations of the International Commission on Radiological Protection. Oxford: Pergamon Press; ICRP Publication 60; Ann. ICRP 21(1-3):1-201; 1991.

International Commission on Radiological Protection. Report of the task group on reference man. Oxford: Pergamon Press; ICRP Publication 23; 1975.

International Commission on Radiological Protection. The metabolism of plutionium and related elements. Oxford: Pergamon Press; ICRP Publication 48; Ann. ICRP 16(213):1-98; 1986.

Lessard, E.T.; Yihua, X.; Skrable, K.W.; Chabot, G.E.; French, C.S.; LaBone, T.R.; Johnson, J.R.; Fisher, D.R.; Belanger, R.; Lipsztein, J.L. Interpretation of bioassay measurements. Washington, DC: Nuclear Regulatory Commission; NUREG/CR-4884; 1987.

National Council on Radiation Protection and Measurement. Management of persons accidentally contaminated with radionuclides. Bethesda, MD: National Council on Radiation Protection and Measurements; NCRP Report No. 65; 1980.

National Council on Radiation Protection and Measurements. Use of bioassay procedures for assessment of internal radionuclide deposition. Bethesda, MD: National Council on Radiation Protection and Measurements; NCRP Report No. 87; 1987.

Rich, B.L.; Hinnefeld, S.L.; Lagerquist, C.R.; Mansfield, W.G.; Munson, L.H.; Wagner, E.R. Health physics manual of good practices at uranium facilities. Idaho Falls, ID: Idaho National Engineering Laboratory; EGG-2530; 1988.

Savannah River Site. The Savannah River Site internal dosimetry technical basis manual. Aiken, SC: Westinghouse Savannah River Company; WSRC-IM-90139; 1994.

Shleien, B.; Slaback, L.A., Jr.; Birky, B.K. eds. Handbook of health physics and radiological health. Baltimore: Williams & Wilkins; 1998.

Skrable Enterprises. INDOS internal dosimetry computer programs users manual. Chelmsford, MA: Skrable Enterprises; 1986.

Skrable, K.W.; Chabot, G.E.; French, C.S.; LaBone, T.R. Intake retention functions and their applications to biaossay and the estimation of internal doses. In: Technical reference manual internal dosimetry programs. Lowell, MA: University of Lowell; 1987.

Skrable, K.W. Evaluation of radiation protection internal dosimetry program manual and procedures for Sandia National Laboratories. Chelmsford MA: Skrable Enterprises; 1996.

Skrable, K.W.; Sun, L.C.; Chabot, G.E.; French, C.S.; LaBone, T.R. Pseudo uptake retention functions for the systemic whole body for estimating intakes from excretion bioassay data. In: Technical reference manual internal dosimetry programs. Lowell, MA: University of Lowell; 1987.

U.S. Department of Energy. Code of federal regulations. Washington, DC: U.S. Government Printing Office; 10 CFR Part 835; 1993.

U.S. Nuclear Regulatory Commission. Air sampling in the workplace. Washington, DC: U.S. Nuclear Regulatory Commission; Regulatory Guide 8.25; 1992.

APPENDIX A: METABOLIC DATA

Dose assessment at SNL is accomplished by using the computer code INDOS (Skrable 86). The data needed to use this code is contained in this attachment, organized by nuclide. Values for f_1, ALI,and DAC are taken from Federal Guidance Report No. 11(EPA 88). Assignment of chemical compounds to pulmonary clearance class and ingestion absorption class is done using the data from ICRP Publication 30 (ICRP 79).

INDOS models radionuclide retention using a sum of exponential terms. Model parameters for most radionuclides are obtained from ICRP 30(ICRP 79). Retention functions for radionuclides that exhibit recycling (Iodine) or which are described in ICRP Publication 20 (ICRP 72) are obtained from Skrable 93.

Urinary and fecal excretion fractions given here are developed from information contained in several references. This includes ICRP Publication 54 (ICRP 88), ICRP Publication 23 (ICRP 75), and NUREG-4884 (NRC 87). If excretion fractions are listed in ICRP 54 then those values are used here. NUREG 4884 values are used if no data is available in ICRP 54. While ICRP 23 data is used to calculate excretion fractions if data is unavailable in the other two references. The following relationships are used in the calculation:

$$f_u = U / (f_1 \cdot I)$$
$$f_o = O / (f_1 \cdot I)$$
$$f_f = 1 - f_u - f_o$$

where f_u = urinary excretion fraction
f_o = excretion fraction to other pathways(sweat, hair...)
f_f = fecal excretion fraction
U = total daily urinary excretion of the radionuclide
f_1 = fraction of the intake absorbed into systemic circulation
I = total daily intake of the radionuclide
O = total daily excetion of the radionuclide to other pathways

If the initial calculation of the urinary excretion fraction gives a value > 1 then f_u is assumed to be 1.0. For some radionuclides there is insufficient data to accurately calculate excretion fractions. In those cases excretion fractions are given as n.a.

A.1 Americium

Inhalation Data

Compounds	Compound Class	f_1
all	W	1e-04

Ingestion Data

Compounds	f_1
all	1e-04

Retention Data (male)

Compartment	Fraction	Half-life (d)
1	0.09965	0.25
2	0.45	14600
3	0.45	36500
4	3.5e-04	1e+08

Retention Data (female)

Compartment	Fraction	Half-life (d)
1	0.09989	0.25
2	0.45	14600
3	0.45	36500
4	1.1e-04	1e+08

Excretion Data

Pathway	Fraction
Urine	n.a.
Feces	n.a.

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{241}Am	1.579e+05	W	6e-03	3e-12	8e-01

A.2 Antimony

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.1
oxides, hydroxides, halides, sulphides, sulphates, nitrates	W	0.01

Ingestion Data

Compounds	f_1
tartar emetic	0.1
all other	0.01

Retention Data

Compartment	Fraction	Half-life (d)
1	0.2	0.25
2	0.76	5
3	0.04	100

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.0

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{122}Sb	2.7	D	2000	1e-06	800
		W	1000	4e-07	700
^{124}Sb	60.2	D	900	4e-07	600
		W	200	1e-07	500

A.3 Barium

Inhalation Data

Compounds	Compound Class	f_1
all compounds	D	0.1

Retention Data

Compartment	Fraction	Half-life (d)
1	0.563	.0127
2	0.262	1.12
3	0.115	2.91
4	0.021	87.5
5	0.0102	381
6	0.0135	1370
7	0.016	5370

Ingestion Data

Compounds	f_1
all compounds	0.1

Excretion Data

Pathway	Fraction
Urine	0.1
Feces	0.9

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{140}Ba	12.74	D	1000	6e-07	500

A.4 Beryllium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	0.005
oxides, halides, nitrates	Y	0.005

Ingestion Data

Compounds	f_1
all	0.005

Retention Data

Compartment	Fraction	Half-life (d)
1	0.4	0.25
2	0.16	15
3	0.44	1500

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.0

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{7}Be	53.44	W	2e+04	9e-06	4e+04
		Y	2e+04	8e-06	4e+04

A.5 Cadmium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.05
sulphides, halides, nitrates	W	0.05
oxides, hydroxides	Y	0.05

Ingestion Data

Compounds	f_1
all inorganic	0.005

Retention Data

Compartment	Fraction	Half-life (d)
1	1.0	9131

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0

Intake Limits

Nuclide	Half-life (d)	Compound Class	Inhalation ALI (uCi)	Inhalation DAC (uCi/ml)	Ingestion ALI (uCi)
^{115}Cd	2.228	D	1000	6e-07	900
		W	1000	5e-07	900
		Y	1000	6e-07	900

A.6 Cerium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	3e-04
oxides, flourides, hydroxides	Y	3e-04

Ingestion Data

Compounds	f_1
all	3e-04

Retention Data

Compartment	Fraction	Half-life (d)
1	1.0	3500

Excretion Data

Pathway	Fraction
Urine	0.1
Feces	0.9

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{141}Ce	32.50	W	700	3e-07	2000
		Y	600	2e-07	2000
^{144}Ce	284.3	W	30	1e-08	200
		Y	10	6e-09	200

A.7 Cesium

Inhalation Data

Compounds	Compound Class	f_1
all	D	1

Ingestion Data

Compounds	f_1
all	1

Retention Data

Compartment	Fraction	Half-life (d)
1	0.1	2
2	0.9	110

Excretion Data

Pathway	Fraction
Urine	0.8
Feces	0.2

Excretion functions

	Urine		Systemic Fecal	
Compartment	Fraction	Half-life (d)	Fraction	Half-life (d)
1	0.028	2	0.0069	2
2	0.0045	110	0.0011	110

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{134}Cs	753	D	100	4e-08	70
^{137}Cs	1.102e+04	D	200	6e-08	100

A.8 Chromium

Inhalation Data

Compounds	Compound Class	f_1
all others	D	0.1
halides, nitrates	W	0.1
oxides, hydroxides	Y	0.1

Ingestion Data

Compounds	f_1
trivalent	0.1
hexavalent	0.01

Retention Data

Compartment	Fraction	Half-life (d)
1	0.3	0.5
2	0.4	6
3	0.25	80
4	0.05	1000

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0

Intake Limits

			Inhalation		Ingestion	
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{51}Cr	27.7	D	5e+04	2e-05	0.1	3e+04
		W	2e+04	1e-05	0.01	3e+04
		Y	2e+04	8e-06		

A.9 Cobalt

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	0.05
oxides, halides, hydroxides, nitrates	Y	0.05

Ingestion Data

Compounds	f_1
organically compexed compounds, inorganic compounds in presence of carrier material except below	0.3
oxides, hydroxides, other inorganics in tracer quantities	0.05

Retention Data

Compartment	Fraction	Half-life (d)
1	0.5	0.5
2	0.3	6
3	0.1	60
4	0.1	800

Excretion Data

Pathway	Fraction
Urine	0.7
Feces	0.3

Intake Limits

			Inhalation			Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{58}Co	70.8	W	1000	5e-07	0.3	2000
		Y	700	3e-07	0.05	1000
^{60}Co	1925	W	200	7e-08	0.3	500
		Y	30	1e-08	0.05	200

A.10 Gallium

Inhalation Data

Compounds	Compound Class	f_1
all others	D	0.001
oxides, hydroxides, carbides, halides, nitrates	W	0.001

Ingestion Data

Compounds	f_1
all compounds	0.001

Retention Data

Compartment	Fraction	Half-life (d)
1	0.3	1
2	0.7	50

Excretion Data

Pathway	Fraction
Urine	n. a.
Feces	n. a.

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{67}Ga	3.261	D	1e+04	6e-06	2e+04
		W	1e+04	4e-06	
^{68}Ga	0.0472	D	4e+04	2e-05	2e+04
		W	5e+04	2e-05	

A.11 Iodine

Inhalation Data

Compounds	Compound Class	f_1
all	D	1

Ingestion Data

Compounds	f_1
all	1

Retention Data

Compartment	Fraction	Half-life (d)
1	0.6999	0.25
2	-0.0347	11.5
3	0.3340	115

Excretion Data

Pathway	Fraction
Urine	0.97
Feces	0.03

Excretion Functions

	Urine		Systemic Fecal	
Compartment	Fraction	Half-life (d)	Fraction	Half-life (d)
1	1.9	0.24	5.1e-06	0.24
2	-0.0019	11	-2.6e-04	11
3	0.0019	120	2.6e-04	120

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{131}I	8.040	D	50	2e-08	30
^{133}I	0.867	D	300	1e-07	100

A.12 Iron

Inhalation Data

Compounds	Compound Class	f_1
all others	D	0.1
oxides, hydroxides, halides	W	0.1

Ingestion Data

Compounds	f_1
all compounds	0.1

Retention Data

Compartment	Fraction	Half-life (d)
1	1.0	2000

Excretion Data(Male)

Pathway	Fraction
Urine	0.2
Feces	0.5

Excretion Data(Female)*

Pathway	Fraction
Urine	0.2
Feces	0.3

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{59}Fe	44.6	D	300	1e-07	800
		W	500	2e-07	

- Iron excretion in females is complicated by losses during the menstrual cycle. This should be considered when calculating intake based on excretion data.

A.13 Manganese

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.1
oxides, halides, hydroxides, nitrates	W	0.1

Ingestion Data

Compounds	f_1
all	0.1

Retention Data

Compartment	Fraction	Half-life (d)
1	0.3	4
2	0.7	40

Excretion Data

Pathway	Fraction
Urine	0.08
Feces	0.8

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{54}Mn	312.7	D	900	4e-07	2000
		W	800	3e-07	

A.14 Molybdenum

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.8
oxides, MoS_2 hydroxides	Y	0.05

Ingestion Data

Compounds	f_1
all not listed	0.8
MoS_2	0.05

Retention Data

Compartment	Fraction	Half-life (d)
1	0.1	1
2	0.9	50

Excretion Data

Pathway	Fraction
Urine	0.6
Feces	0.3

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{99}Mo	2.751	D	3000	1e-06	2000
		Y	1000	6e-07	1000

A.15 Neptunium

Inhalation Data

Compounds	Compound Class	f_1
all	W	0.001

Ingestion Data

Compounds	f_1
all	0.001

Retention Data (Male)

Compartment	Fraction	Half-life (d)
1	0.09965	0.25
2	0.45	14600
3	0.45	36500
4	3.5e-04	1e+08

Retention Data (Female)

Compartment	Fraction	Half-life (d)
1	0.09989	0.25
2	0.45	14600
3	0.45	36500
4	1.1e-04	1e+08

Excretion Data

Pathway	Fraction
Urine	0.5
Feces	0.5

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{237}Np	1e+08	W	4e-03	2e-12	0.5

A.16 Nickel

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.05
carbides, oxides, hydroxides	W	0.05

Ingestion Data

Compounds	f_1
all	0.05

Retention Data

Compartment	Fraction	Half-life (d)
1	0.02	0.2
2	0.68	0.25
3	0.3	1200

Excretion Data

Pathway	Fraction
Urine	0.55
Feces	0.0

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{57}Ni	1.503	D	5000	2e-06	2000
		W	3000	1e-06	

A.17 Phosphorus

Inhalation Data

Compounds	Compound Class	f_1
all others	D	0.8
Phosphates of some elements	W	0.8

Ingestion Data

Compounds	f_1
all compounds	0.8

Retention Data

Compartment	Fraction	Half-life (d)
1	0.15	0.5
2	0.15	2
3	0.4	19
4	0.3	1e+08

Excretion Data

Pathway	Fraction
Urine	0.9
Feces	0.1

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{32}P	14.28	D	900	4e-07	600
		W	400	2e-07	

A.18 Plutonium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	1e-03
oxides	Y	1e-05

Ingestion Data

Compounds	f_1
all others	1e-03
nitrates	1e-04
oxides	1e-05

Urine Retention Data (Jones)

Compartment	Fraction	Half-life (d)
1	0.0159	1.24
2	0.0101	15.70
3	0.0419	182
4	0.9321	22440

Fecal Retention Data (Durbin)

Compartment	Fraction	Half-life (d)
1	0.141	2
2	0.124	6.6
3	0.079	56
4	0.0897	380
5	0.5663	4000

ICRP Retention Data (Male)

Compartment	Fraction	Half-life (d)
1	0.09965	0.25
2	0.45	14600
3	0.45	36500
4	3.5e-04	1e+08

ICRP Retention Data (Female)

Compartment	Fraction	Half-life (d)
1	0.09989	0.25
2	0.45	14600
3	0.45	36500
4	1.1e-04	1e+08

Excretion Data

Pathway	Fraction
Urine	0.54 - Jones
Feces	0.12 - Durbin
Urine	n.a. - ICRP
Feces	n.a. - ICRP

Intake Limits

			Inhalation		Ingestion	
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{238}Pu	3.205e+04	W	7e-03	3e-12	1e-03	0.9
		Y	2e-02	8e-12	1e-04	9.0
					1e-05	90
^{239}Pu	8.814e+06	W	6e-03	3e-12	1e-03	0.8
		Y	2e-02	7e-12	1e-04	8.0
					1e-05	80
^{240}Pu	2.399e+06	W	6e-03	6e-03	1e-03	0.8
		Y	2e-02	2e-02	1e-04	8.0
					1e-05	80
^{241}Pu	5260	W	0.3	1e-10	1e-03	40
		Y	0.8	3e-10	1e-04	400
					1e-05	4000

A.19 Radium

Inhalation Data

Compounds	Compound Class	f_1
all	W	0.2

Ingestion Data

Compounds	f_1
all	0.2

Retention Data

Compartment	Fraction	Half-life (d)
1	0.514	0.133
2	0.204	0.917
3	0.258	4.15
4	9.83e-03	241
5	5.64e-03	1130
6	3.15e-03	3690
7	5.83e-03	8930

Excretion Data

Pathway	Fraction
Urine	0.05
Feces	0.95

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{226}Ra	5.8e+05	W	0.6	3e-10	2
^{228}Ra	2100	W	1	5e-10	2

A.20 Ruthenium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.05
halides	W	0.05
oxides, hydroxides	Y	0.05

Ingestion Data

Compounds	f_1
all	0.05

Retention Data

Compartment	Fraction	Half-life (d)
1	0.15	0.3
2	0.35	8
3	0.3	35
4	0.2	1000

Excretion Data

Pathway	Fraction
Urine	n.a
Feces	n.a

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{103}Ru	39.35	D	2e+03	7e-07	2000
		W	1e+03	4e-07	
		Y	600	3e-07	
^{106}Ru	3.682e+02	D	90	4e-08	200
		W	50	2e-08	
		Y	10	5e-09	

A.21 Silver

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.05
nitrates, sulfates	W	0.05
oxides, hydroxides	Y	0.05

Ingestion Data

Compounds	f_1
all	0.05

Retention Data

Compartment	Fraction	Half-life (d)
1	0.1	3.5
2	0.9	50

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.0

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{110m}Ag	249.85	D	100	5e-08	500
		W	200	8e-08	
		Y	90	4e-08	

A.22 Sodium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	1

Ingestion Data

Compounds	f_1
all	1

Retention Data

Compartment	Fraction	Half-life (d)
1	0.997	10
2	0.003	500

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.01

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{22}Na	9504	D	600	3e-07	400
^{24}Na	0.623	D	5000	2e-06	4000

A.23 Strontium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.3
$SrTiO_3$	Y	0.01

Ingestion Data

Compounds	f_1
soluble salts	0.3
$SrTiO_3$	0.01

Retention Data

Compartment	Fraction	Half-life (d)
1	0.129	0.0271
2	0.0699	1.24
3	0.589	4.36
4	0.0271	56.4
5	0.0665	210
6	0.0580	1860
7	0.0603	8220

Excretion Data

Pathway	Fraction
Urine	0.8
Feces	0.2

Intake Limits

			Inhalation		Ingestion	
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{89}Sr	50.55	D	800	4e-07	0.3	600
		Y	100	6e-08	0.01	500
^{90}Sr	1.04e+04	D	20	8e-09	0.3	30
		Y	4	2e-09	0.01	400

A.24 Sulfur

Inhalation Data

Compounds	Compound Class	f_1
Sulphate and sulphide compounds can be D or W. Check the data for other elements in the compound.	D	0.8
Elemental	W	0.8

Ingestion Data

Compounds	f_1
inorganic compounds	0.8
elemental	0.1

Retention Data

Compartment	Fraction	Half-life (d)
1	0.8	0.25
2	0.15	20
3	0.05	2000

Excretion Data

Pathway	Fraction
Urine	0.9
Feces	0.1

Intake Limits

			Inhalation		Ingestion	
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{35}S	87.44	D	2e+04	7e-06	0.8	1e+04
		W	200	9e-07	0.1	6000
		Vapor	1e+04	6e-06		

A.25 Tantalum

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	1e-03
carbides, oxides, halides, nitrates, nitrides, hydroxides	Y	1e-03

Ingestion Data

Compounds	f_1
all	1e-03

Retention Data

Compartment	Fraction	Half-life (d)
1	0..35	4
2	0..65	100

Excretion Data

Pathway	Fraction
Urine	n.a.
Feces	n.a.

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{182}Ta	114.74	W	300	1e-07	800
		Y	100	6e-08	
^{183}Ta	5.0	W	1000	5e-07	900
		Y	1000	4e-07	

A.26 Tellurium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.2
oxides, nitrates, hydroxides	W	0.2

Ingestion Data

Compounds	f_1
all	0.2

Retention Data

Compartment	Fraction	Half-life (d)
1	0.5	0.8
2	0.25	20
3	0.25	5000

Excretion Data

Pathway	Fraction
Urine	0.75
Feces	0.25

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{132}Te	3.26	D	200	9e-08	200
		W	200	9e-08	

27 Thorium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	2e-04
oxides, hydroxides	Y	2e-04

Ingestion Data

Compounds	f_1
all	2e-04

Retention Data

Compartment	Fraction	Half-life (d)
1	0.1	0.5
2	0.2	700
3	0.7	8000

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.0

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{228}Th	698.8	W	1e-02	4e-12	6
		Y	2e-02	7e-12	
^{230}Th	2.8e+07	W	6e-03	3e-12	4
		Y	2e-02	6e-12	
^{232}Th	1e+08	W	1e-03	5e-13	0.7
		Y	3e-03	5e-13	

A.28 Tin

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.02
sulphides,halides, oxides, nitrates, hydroxides, stannic phosphate	W	0.02

Ingestion Data

Compounds	f_1
all	0.02

Retention Data

Compartment	Fraction	Half-life (d)
1	0.5	0.25
2	0.1	4
3	0.1	25
4	0.3	400

Excretion Data

Pathway	Fraction
Urine	0.25
Feces	0.0

Intake Limits

Nuclide	Half-life (d)	Inhalation Compound Class	Inhalation ALI (uCi)	Inhalation DAC (uCi/ml)	Ingestion ALI (uCi)
^{117m}Sn	13.60	D	1000	5e-07	2000
		W	1000	6e-07	
^{119m}Sn	293.0	D	2000	1e-06	3000
		W	1000	4e-07	

A.29 Tungsten

Inhalation Data

Compounds	Compound Class	f_1
all	D	0.3

Ingestion Data

Compounds	f_1
all	0.3
Tungstic acid	0.01

Retention Data

Compartment	Fraction	Half-life (d)
1	0.95	0.25
2	0.0225	5
3	0.01	100
4	0.0175	1000

Excretion Data

Pathway	Fraction
Urine	n.a.
Feces	n.a.

Intake Limits

		Inhalation			Ingestion	
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{185}W	75.1	D	7000	3e-06	0.3	3000
					0.01	2000

A.30 Uranium

Inhalation Data

Compounds	Compound Class	f_1
UF_6, UO_2F_2, $UO_2(NO_3)_2$	D	0.05
UO_3, UF_4, UCl_4	W	0.05
UO_2, U_3O_8	Y	0.002

Ingestion Data

Compounds	f_1
hexavalent	0.05
tetravalent - UO_4, UO_2, U_3O_8	0.002

Retention Data

Compartment	Fraction	Half-life (d)
1	0.53596	0.25
2	0.24	6
3	0.2	20
4	0.00104	1500
5	0.023	5000

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.0

Intake Limits

			Inhalation		Ingestion	
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	f_1	ALI (uCi)
^{234}U	8.930e+07	D	1	5e-10	0.05	10
		W	0.7	3e-10	0.002	200
		Y	0.04	2e-11		
^{235}U	1e+08	D	1	6e-10	0.05	10
		W	0.8	3e-10	0.002	200
		Y	0.04	2e-11		
^{238}U	1e+08	D	1	6e-10	0.05	10
		W	0.8	3e-10	0.002	200
		Y	0.04	2e-11		

A.31 Vanadium

Inhalation Data

Compounds	Compound Class	f_1
all others	D	0.01
oxides, hydroxides, carbides, halides	W	0.01

Ingestion Data

Compounds	f_1
all compounds	0.01

Retention Data

Compartment	Fraction	Half-life (d)
1	0.7	0.25
2	0.3	1e+04

Excretion Data

Pathway	Fraction
Urine	0.75
Feces	0.25

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{48}V	16.238	D	1000	5e-07	600
		W	600	3e-07	

A.32 Yttrium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	W	1e-04
oxides, hydroxides	Y	1e-04

Ingestion Data

Compounds	f_1
all	1e-04

Retention Data

Compartment	Fraction	Half-life (d)
1	0.25	0.25
2	0.75	1e+08

Excretion Data

Pathway	Fraction
Urine	n.a.
Feces	n.a.

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{91}Y	58.51	W	200	7e-08	500
		Y	100	5e-08	

A.33 Zinc

Inhalation Data

Compounds	Compound Class	f_1
all	Y	0.5

Ingestion Data

Compounds	f_1
all	0.5

Retention Data

Compartment	Fraction	Half-life (d)
1	0.24	20
2	0.76	400

Excretion Data *

Pathway	Fraction
Urine	0.25
Feces	0.75

Intake Limits

			Inhalation		Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{65}Zn	244.4	Y	300	1e-07	400

* Zinc excretion in females is complicated by losses during the menstrual cycle. This should be considered when calculating intake based on excretion data.

A.34 Zirconium

Inhalation Data

Compounds	Compound Class	f_1
all not listed	D	0.002
oxides, nitrates, halides, hydroxides	W	0.002
carbide	Y	0.002

Ingestion Data

Compounds	f_1
all	0.002

Retention Data

Compartment	Fraction	Half-life (d)
1	0.5	7
2	0.5	8000

Excretion Data

Pathway	Fraction
Urine	1.0
Feces	0.0

Intake Limits

		Inhalation			Ingestion
Nuclide	Half-life (d)	Compound Class	ALI (uCi)	DAC (uCi/ml)	ALI (uCi)
^{95}Zr	63.98	D	100	5e-08	1000
		W	400	2e-07	
		Y	300	1e-07	