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Executive Summary

The internal dosimetry program at the Savannah River Site (SRS) consists of radiation protection programs and activities used to detect and evaluate intakes of radioactive material by radiation workers. Examples of such programs are

- air monitoring
- surface contamination monitoring
- personal contamination surveys
- radiobioassay
- dose assessment

The objectives of the internal dosimetry program are to demonstrate that the workplace is under control and that workers are not being exposed to radioactive material, and to detect and assess inadvertent intakes in the workplace.

The Savannah River Site Internal Dosimetry Technical Basis Manual (TBM) is intended to provide a technical and philosophical discussion of the radiobioassay and dose assessment aspects of the internal dosimetry program. Detailed information on air, surface, and personal contamination surveillance programs is not given in this manual except for how these programs interface with routine and special bioassay programs.

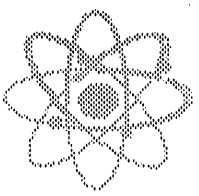
The TBM is divided into four parts:

- In Part I intake and dose assessment methods are discussed. Minimal routine bioassay programs are also developed for specific radionuclides.
- In Part II the radionuclide specific routine bioassay programs given in Part 1 are used to develop routine bioassay programs for specific facilities.
- In Part III the methods and capabilities of *in-vitro* bioassay techniques are discussed.
- In Part IV the methods and capabilities of *in-vivo* bioassay techniques are discussed.

The TBM is intended to give some insight into the reasoning behind the development and application of internal dosimetry procedures. The TBM is not intended to be a procedure, and not all procedures necessarily have their technical basis in its pages.

PART I - INTERNAL DOSIMETRY PRINCIPLES, MODELS, AND METHODS

PART IV – IN VIVO BIOASSAY



Part I

Part I - Internal Dosimetry Principles, Models, and Methods

Chapter 1

Introduction

Chapter 1 Preview

- Overview of Part I
- Glossary

1

Introduction

The objectives of the internal dosimetry program are to demonstrate that the workplace is under control and that workers are not being exposed to radioactive material, and to detect and assess inadvertent intakes in the workplace.

Part I provides an overview of dose assessment aspects of the internal dosimetry program at SRS, including

- definition of dosimetry terms
- performance criteria for internal dosimetry programs
- methods of calculating dose
- methods of calculating intakes
- designing routine bioassay programs
- missed dose tables for designing bioassay programs
- followup to incidents
- medical procedures for mitigation of dose following intakes

The internal dosimetry program outlined in this manual is intended to meet or exceed the requirements of DOE Order 5480.11[1] for monitoring the workplace and for assessing internal radiation doses to workers. The Draft Internal Dosimetry DOELAP[2] provides the basic requirements for the SRS Internal Dosimetry Program. This manual should be considered to be complementary to the DOELAP Manual, providing detailed information specific to SRS; it is not intended to be used as a procedure, but rather, to provide justification in support of site procedures.

Glossary

The words shall, should, and may are used as follows in this manual: shall denotes a requirement; should, a recommendation; and may, a permissible practice. Other verbs are used only in a tutorial sense.

Absorbed Dose - is the energy deposited in matter per unit mass by ionizing radiation. The SI unit is the Gray (1 Gy = 1 J/Kg) and the traditional unit is the rad (1 rad = 0.01 Gy). Unless specified the material in which the energy is deposited is soft tissue.

Activity Median Aerodynamic Diameter (AMAD) - is the diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

ALARA - is an acronym for As Low As Reasonably Achievable. It is the objective of current radiation protection efforts to maintain exposures to radiation as low as reasonably achievable, with limiting economic and social factors being taken into account.

Annual Effective Dose Equivalent - is the effective dose equivalent from both external and internal irradiation received in a calendar year. The annual effective dose equivalent is expressed in the same units as dose equivalent.

Annual Limit on Intake (ALI) - is the quantity of a single radionuclide which, if inhaled or ingested, would irradiate a person, represented by Reference Man (ICRP Publication 23) to the limiting value for control of the workplace.

Assimilation - means the same as intake.

Bioassay - is the measurement of the amount or the concentration of radioactive material in the body or in biological material excreted or removed from the body.

Biokinetic Model - is a series of mathematical relationships formulated to describe the intake, uptake and retention of a radionuclide in various organs of the body and the subsequent excretion from the body by various pathways.

Breathing Zone Air Sampler (BZA) - is an air sampler that draws air close enough to the nose so as to be considered representative of the air a person breaths.

Committed Dose Equivalent - is the calculated dose equivalent projected to be received by a tissue or organ over a 50-yr period after an intake of radionuclide into the body. It does not include contributions from external dose. Committed dose equivalent is expressed in the same units as dose equivalent.

Committed Effective Dose Equivalent - is the sum of the committed dose equivalents to various tissues in the body, each multiplied by its weighting factor. Committed effective dose equivalent is expressed in the same units as dose equivalent.

Confirmed Assimilation - is an intake of radioactive material that will deliver an effective dose equivalent of 100 mrem in the 12 months following the intake. A confirmed assimilation requires a special hazards investigation.

Confirmed Deposition - radioactive material in the body verified by bioassay measurements.

Cumulated Activity - is the integral activity of a source organ over a specified period of time.

Cumulative Effective Dose Equivalent - is the sum of the annual effective dose equivalents received for all years of employment at SRS. Cumulative effective dose equivalent is expressed in the same units as dose equivalent.

Decision Level - is the amount of material in a sample corresponding to a 5% chance of a false positive. If the result of an analysis is above the decision level then material is deemed to be present.

Deposition - is the fraction of an intake retained in the body.

Derived Air Concentration (DAC) - is the quantity obtained by dividing the ALI for any given radionuclide by the volume of air breathed by an average worker during a working year (2400 cubic meters).

Direct (*In Vivo*) Bioassay - is the assessment of radioactive material in the body by detection of radiations emitted by the material.

Dose Equivalent - is the absorbed dose multiplied by the quality factor for the types of radiation absorbed. The SI unit of dose equivalent is the Sievert (1 Sv = 0.01 J/Kg) and the traditional unit is the rem (1 rem = 0.01 Sv). It is assumed in radiation protection that 1 Sv of any type of radiation will produce the same biological damage as any other type of radiation.

Effective Dose Equivalent - is the sum of the weighted dose equivalents to all significantly irradiated organs. The units of effective dose equivalent are the same as those for dose equivalent. An effective dose equivalent of 1 Sv is deemed to pose the same stochastic risk as a uniform whole body dose equivalent of 1 Sv.

Exposure Route - is a pathway by which radioactive material enters the body. The main exposure routes are inhalation, ingestion, absorption through the skin, and entry through a cut or wound in the skin.

Gastrointestinal Tract Model - is a mathematical representation used to stylize the behavior of radionuclides in the contents of the human gastrointestinal tract.

Indirect (*In Vitro*) Bioassay - is the measurement or analysis of radioactive material in excreta or other biological samples removed from the body.

Intake - is the amount of radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion or through wounds. Part of the intake may be exhaled, excreted or excised; part of the intake may be deposited into the respiratory tract, GI tract or wound; part of the intake may translocate to extracellular fluid.

Metabolic Model - is a mathematical representation of the behavior in the metabolic processes of cells, tissues, organs and organisms. It is used to describe distribution among tissues and excretion.

Minimum Detectable Amount (MDA) - is the smallest amount of a material in a sample that will be detected with a 5% probability of false detection while accepting an 5% probability of false non-detection. The MDA is used in the design of analytical systems and is not to be used as a criteria to decide if material is present in a sample (see decision level).

Nonemployee Occupational Worker - is an individual who is either a subcontractor to DOE or a DOE contractor who performs work for or in conjunction with DOE and whose occupational exposure records are maintained by the DOE or DOE contractor.

Occupational Worker - is an individual who is either a DOE or DOE contractor employee; an employee of a subcontractor to a DOE contractor; or an individual who visits to perform work for or in conjunction with DOE or utilizes DOE facilities.

Organ Content - is the amount of a radionuclide present in the organ of reference at a specific time and means the same as organ burden.

Personal Air Sampler (PAS) - is a portable breathing zone air sampler that is carried by a worker.

Positive Result - (see definition for Decision Level).

Radiation Worker - is an occupational worker whose job involves operating radiation producing devices or working with radioactive material, or who is likely to be routinely occupationally exposed above 0.1 rem (0.001 sievert) per year, which is the sum of the annual effective dose equivalent from external radiation and the committed effective dose equivalent from internal radiation.

Reference Man - is a representative human model with the anatomical and physiological characteristics defined in the report of the ICRP Task Group on Reference Man (ICRP Publication 23).

Respiratory Tract Model - is a mathematical representation of the behavior of particles and gases in the human respiratory tract.

State-of-the-Art - refers to the most advanced technology that has been commercially available for at least 5 years.

Special Bioassay Monitoring - refers to a bioassay measurement not part of the routine program, such as measurement made for prompt follow-up to a potential intake.

Task Group Lung Model - is the model that describes the behavior of particles in the respiratory tract of man developed by the ICRP Task Group on Lung Dynamics.

Uptake - is the amount of a radionuclide absorbed into extracellular fluid or taken up by the systemic compartment of the body, (e.g. by injection into blood, by absorption from compartments in the respiratory tract or the GI tract or by absorption through the skin or through wounds in the skin).

Valid Bioassay - is an analysis that accurately determines the activity in a sample that contains only metabolized material.

Visitor – is a nonemployee not classified as a "nonemployee occupational worker" visiting a facility that is operated by DOE or a DOE contractor. This includes any occupational worker who performs work for or in conjunction with DOE or utilizes DOE facilities and whose occupational exposure records are maintained by their employer.

Weighted Dose Equivalent – is the dose equivalent to an organ multiplied by the stochastic risk weighting factor for that organ. Weighted dose equivalent has the same units as dose equivalent. A weighted dose equivalent of 1 Sv is deemed to pose the same stochastic risk as a uniform whole body dose equivalent of 1 Sv.

Weighting Factors – are fractions used in the calculation of annual and committed effective dose equivalent to equate the risk arising from the irradiation of a tissue to the total risk when the whole body is uniformly irradiated. The weighting factors as defined by ICRP Publication 26 are

Organ or Tissue	Weighting Factor
Gonads	0.25
Breasts	0.15
Red Bone Marrow	0.12
Lungs	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder(a)	0.30

(a) Remainder means the five other organs with the highest dose (e.g., liver, kidney, spleen, thymus, adrenals, pancreas, stomach, small intestine, upper large intestine or lower large intestine, but excluding skin, lens of the eye, and extremities). The weighting factor for each such organ is 0.06.

Whole Body Dose Equivalent – is the dose equivalent that results when the whole body is irradiated. If the irradiation is uniform whole body dose equivalent is the same as effective dose equivalent. Whole body dose equivalent is expressed in the same units as dose equivalent.

Working Level – is a unit of air concentration of potential alpha energy released from radon and its daughters. One working level is any combination of short-lived radon daughters in one liter of air that will result in the emission of 1.3×10^5 MeV of potential alpha energy ($1 \text{ WL} = 2.08 \times 10^{-5} \text{ J m}^{-3}$).

Working Level Month (WLM) – is the cumulative exposure equivalent to exposure to one working level for a working month (170 hours). $1 \text{ WLM} = 0.0035 \text{ J h m}^{-3}$.

References

(1) *Department of Energy Order 5480.11*, Effective 1-1-89.

(2) *Department of Energy Draft Performance Standard for Internal Dosimetry*, Draft 8.0.

Chapter 2

Design Criteria for Internal Dose Monitoring Programs

Chapter 2 Preview

- Routine Monitoring Programs
- Special Bioassay Programs

2

Design Criteria for Internal Dose Monitoring Programs

An internal dosimetry monitoring program may consist of either breathing zone air monitoring, bioassay, or both. The two basic types of monitoring programs used are

- routine monitoring program
- special monitoring program

A routine monitoring program consists of breathing zone air monitoring and bioassay, which are performed at prescribed times. The purpose of the routine monitoring program is to confirm that workplace air monitoring is working properly and that workers are not being exposed unintentionally to radioactive material. Routine monitoring programs are also used to assess dose from controlled exposures to radioactive materials, for example, chronic exposure to tritiated water.

A special bioassay program is initiated in response to a specific incident, for example, an unintentional personal contamination. The purpose of the special bioassay program is to determine if an intake has occurred, and if it has, to calculate the dose.

To design an adequate monitoring program we need to define the goals of the program, for example, what we expect the program to achieve. The objectives of the monitoring programs will be given in this section.

Routine Monitoring Programs

There are two types of routine monitoring programs

- workgroup
- worker

A workgroup monitoring program is used in facilities where the workers have a reasonable potential for exposure to radioactive materials but are unlikely to receive a committed effective dose equivalent (CEDE) in excess of 100 mrem from intakes during the year. A worker monitoring program is required if workers are likely to receive in excess of 100 mrem CEDE.

Potential for Exposure

DOE policy states that facilities should, where feasible, be designed and operated with engineered controls that prevent intakes of radioactive material by workers. A facility that has effective containment and contamination control, where workers are unlikely to receive 100 mrem CEDE during the year, is called a "clean" facility. The routine monitoring program in a clean facility concentrates on monitoring the workplace environment rather than the worker, and only a minimal bioassay program is required.

If effective containment is lost and workers are likely to be exposed to radioactive material that could deliver more than 100 mrem CEDE in the year, enhanced monitoring programs are needed to identify the exposed individuals. In this situation the worker is also monitored. Thus, the monitoring requirements for a clean facility are less complex than those for a facility that is not clean.

Monitoring Programs for Clean Facilities

A clean facility is one where radioactive material is contained to the extent that, although there is some reasonable potential for exposure to radioactive materials, workers are unlikely to receive an intake that would deliver a dose in excess of 100 mrem CEDE in any calendar year. The purpose of the monitoring program in a clean facility is to verify that the air and contamination monitoring are working properly and that the facility is clean. The monitoring program for a clean facility consists of

- an air monitoring program that samples air that is representative of the worker's breathing zone and is capable of detecting intakes of 40 DAC-h or more in a calendar year
- a bioassay program that can detect intakes by a workgroup that would deliver a dose of 100 mrem CEDE in a calendar year

Air Monitoring Program for a Clean Facility

The air monitoring program requires that concentrations of radionuclides in the air of less than 0.02 DAC be measured throughout the year, assuming continuous occupancy for 2000 hours a year. This requirement is for each independent source of radioactive material.

Workgroup Bioassay Program

The bioassay program for workers in a clean area focuses on the workgroup, which is a collection of workers who perform approximately the same tasks in approximately the same locations. A fraction of the workers in the workgroup should be monitored at a frequency where an intake that would deliver 100 mrem CEDE would be detected; note that this program is not designed to identify individuals who exceeded 100 mrem CEDE, but rather to sample a group of workers that is unlikely to be exposed to radioactive materials. If any member of the workgroup has an intake of radioactive material then all members of the workgroup should be placed on a special bioassay program.

The NCRP[1] recommends that 10% of a workgroup composed of 100 or more workers should be sampled at a frequency of not less than once a year. Workgroups composed of 11-100 workers should have 10 workers monitored, and all workers in workgroups composed of 10 or less workers should be monitored.

Who Should not be in a Workgroup Bioassay Program
Any worker who

- has radioactive material in his body that interferes with detecting and assessing additional intakes
- uses any form of respiratory protection
- enters Airborne Radioactivity Areas

should not be part of a workgroup bioassay program.

These workers should be placed on a worker bioassay program. In practice, workers who have significant potential for exposure to transuranics are not placed on workgroup bioassay programs.

Independent Sources

The criteria specified in the previous section are for each independent source of radioactive material. The air monitoring and workgroup bioassay programs for one source of radioactive material do not have to account for any other independent sources. For example, if workers routinely enter multiple clean facilities, the air monitoring program for each facility should be able to detect 40 DAC-h and the workgroup bioassay program should be able to detect intakes by the workgroup that would deliver 100 mrem CEDE. The detection capability of the programs does not have to be reduced because of multiple independent sources. Likewise, if one glovebox in a laboratory contains plutonium and another separate glovebox contains tritium, the air monitoring program should be able to detect 40 DAC-h of plutonium and 40 DAC-h of tritium. If the plutonium and tritium are in the same glovebox, the air monitoring program should be able to detect 40 DAC-h of the mixture.

Monitoring Programs for Workers

If workers are likely to be exposed to radioactive material that would deliver a committed effective dose equivalent in excess of 100 mrem CEDE in any calendar

year, the worker should be placed on a worker monitoring program. The purpose of this program is to detect exposures of the worker to radioactive material so that dose may be assessed and special bioassay programs initiated if necessary.

Once a worker monitoring program is required, certain performance objectives must be met. First and foremost the minimum monitoring program must be able to demonstrate compliance with the 5 rem annual effective dose equivalent (AEDE) limit and the 50 rem annual organ dose equivalent (AODE) limit. Note that these limits are the sum of internal and external dose and that they apply to a specific person. This means that the minimum monitoring program required for each person will vary depending on the internal and external dose they receive and our ability to assess future intakes of radioactive material.

Rather than define a custom monitoring program for every worker, conservative performance objectives were established that may be used to develop generic monitoring programs for workers who are expected to receive less than 2 rem AEDE. These performance objectives are

- The monitoring program shall be capable of assessing intakes of radioactive material in a year that will deliver an effective dose equivalent of 100 mrem in the 12 months following the intake.

- The monitoring program should be capable of assessing 1/10 of the above, that is, 10 mrem in the 12 months following the intake.

Note that a constant time interval of 12 months after the intake is used so the calendar date of the intake does not need to be considered in the design of the monitoring program.

Monitoring Programs

The internal dosimetry monitoring program for a worker may consist of bioassay alone or bioassay augmented with breathing zone air sampling. Bioassay is the preferred method to assess intakes, but breathing zone air sampling data may be used in situations where adequate bioassay data are not available.

Independent Sources

The 100 mrem and 10 mrem performance objectives apply to each independent source of radioactive material in the workplace. For example, if a worker handles tritium in one building and plutonium in another (for example, they are independent sources), the monitoring program for the worker should be able to detect 100 mrem from intakes of tritium and 100 mrem from intakes of plutonium. If the sources are not independent, the 100 mrem performance objective applies to the mixture. Note that the 5 rem AEDE and 50 AODE rem limits apply to all occupational sources of radiation and radioactive material and the independent source rule may not be used.

Inadequate Programs

With the use of personal air samplers (PAS), feces bioassay, and urine bioassay the 100 mrem 12-month EDE design objective can be met for all radionuclides. One complication occurs when an excretion curve is relatively flat and high

frequencies (for example, every week) are required to strictly meet the 100 mrem objective while convient frequencies (for example, annual) are only slightly over the objective. In these situations, considering the conservative nature of the missed dose calculations, missed doses of up to 200 mrem will be deemed acceptable.

If worker monitoring programs can not ensure compliance with the 5 rem AEDE and 50 AODE rem limits then work restrictions should be imposed to prevent intakes of radioactive materials.

Workgroup monitoring programs are designed for individuals who are not being exposed to radioactive material, and for this reason PAS and feces bioassay are not routinely used for workgroup monitoring. If the workgroup monitoring program can not meet the 100 mrem CEDE design objective, then the following actions should be taken:

- Steps should be taken to verify that the air monitoring program is working properly and that it can detect exposures in excess of 40 DAC-h.
- The minimum dose that can be detected with the current bioassay programs should be documented.
- If a bioassay type has a missed dose of less than 100 mrem CEDE at some impractical frequency (every day urine samples for example), then a bioassay program consisting of that bioassay type at a reasonable frequency should be instituted. The theory behind this program is that statistically, some fraction of any intakes that occur will be detected. This is reasonable considering the workers are not being exposed to radioactive material.
- Operational studies employing personal air samplers and fecal sampling should be considered for workers with the highest potential for exposure.

Missed Dose

Design objectives for routine bioassay programs were given in terms of effective dose equivalent in the previous sections. Now we need a method of describing the ability of a given bioassay program to detect dose to see if a program meets these objectives. In this manual we use what is called "missed dose" to describe the abilities of a bioassay program. Missed dose is the maximum effective dose equivalent associated with a less-than-MDA result from a given bioassay program.

Let us consider a bioassay program for tritiated water to illustrate missed dose. The MDA for tritium in the urine is 0.02 μCi/L. Assuming the following:

- a 10 day biological halflife for water in the body
- a dose-rate to concentration factor of 0.2 mrem/day per μCi/L of body water

the missed dose D for urine bioassay at 15 days after intake is

$$D = 0.2 \cdot (0.02\ \mu Ci/L) e^{15 \cdot 0.693/10} = 0.011\ \text{mrem}$$

This missed dose may be compared to the design objective of 100 mrem CEDE to determine the adequacy of collecting urine samples every 15 days. Tables of

missed dose are presented in Chapter 9 for single acute inhalation intakes of pure radionuclides. These tables are for the design of routine bioassay programs and may not be appropriate for other purposes.

Missed dose is only one of several factors that go into the design of a bioassay program. We must also consider the following:

- monitoring the buildup of long-halflife material in the body
- placing upper limits or caps on intakes
- cost
- time away from the job
- aesthetics

In addition, bioassay is sometimes used to monitor and control exposures of workers to radioactive materials. This can lead to a higher bioassay frequency than indicated by a missed dose analysis alone.

Special Bioassay Programs

A special bioassay program is initiated in response to an unintentional intake of radioactive material. The special bioassay program has the same design goals as the routine worker monitoring program, for example, the special bioassay program

- shall ensure compliance with 5 rem AEDE and 50 rem AODE limits
- shall detect and assess intakes that could deliver 100 mrem in the 12 months following the intake
- should detect and assess intake that could deliver 10 mrem in the 12 months following the intake

Inadequate Special Bioassay Programs

A special bioassay program that is inadequate to comply with the 100 mrem design objective warrants an investigation to determine the cause of the inadequacy. Steps should be taken to correct any problems identified by the investigation.

Programs that can not ensure compliance with the 5 rem and 50 rem limits warrant immediate work restrictions for the involved individuals and an investigation to determine the cause of the inadequacy.

References

(1) *Use of Bioassay Procedures for Assessment of Internal radionuclide Deposition*. NCRP Report Number 87, 1987.

Chapter 3

Designing Routine Monitoring Programs

Chapter 3 Preview

- Elements of a Routine Monitoring Program

3

Designing Routine Monitoring Programs

Design objectives for routine internal dosimetry monitoring programs were given in Chapter 2. In this section a practical means of implementing these design objectives in a routine monitoring program is given. Recommended minimum monitoring programs for various radionuclides are given in Chapter 10. The routine programs for a specific facility are given in Chapters 12 through 17.

Elements of a Routine Monitoring Program

One objective of the Technical Basis Manual is to derive monitoring programs for facilities at SRS. To accomplish this we first identify source terms and radionuclides of concern, and then worker and workgroup monitoring programs are recommended for the facility. Assignment of specific workers to appropriate worker or workgroup programs is the responsibility of Health Protection Operations in the facility.

Source Terms

Efforts should be made to identify major source terms in a facility

- streams of radionuclides coming into and out of a facility
- areas where radionuclides are produced in a facility
- areas where the chemical or physical form changes significantly
- areas where the potential for release increases significantly

Radionuclides present, chemical and physical forms, particle size distributions, and biological solubility should be determined if feasible.

Radionuclides of Concern

There may be many radionuclides present in a facility; some have the potential for delivering significant doses or may be used as tracers for a mixture of radionuclides. Such radionuclides are referred to as radionuclides of concern because air monitoring and bioassay programs are designed to detect them.

Radionuclides of concern are determined as follows. All radionuclides in a facility to which workers could be exposed are identified from contamination survey records, safety analysis reports (SAR), technical reports, the open literature, personal interviews, etc. The radionuclides whose radiotoxicity and exposure potential combine to deliver the majority of the dose, say 90%, are deemed to be the radionuclides of concern. All other radionuclides may be ignored unless they are suitable for use as a tracer.

Radon

A radon survey of 360 buildings at SRS was completed in 1987[1]. Radon levels at all locations were below the EPA action level of 4 pCi/L. Because of this survey and the nature of work currently done at SRS, all radon exposures are deemed to be non-occupational and are therefore not within the scope of this manual. The situation will be reassessed if future operations at SRS involve exposure to radon.

Air Monitoring

Air monitoring capable of detecting 40 DAC-h over a year should be performed in locations were workers have the potential for exposure to airborne radioactive material.

Potential for Exposure of Workers

Historical records concerning contamination of workers, airborne radioactivity, and surface contamination are useful for determining the potential for exposure of workers to radioactive material. Safety evaluations for a facility can be consulted to augment the historical records or to provide the primary information for new facilities. If the air monitoring programs in a facility are not adequate to detect 40 DAC-h in a year then workers in the facility should be assumed to be likely to receive intakes.

Workers with No Potential for Exposure

Air monitoring and bioassay programs are not required for workers with no potential for exposure to radioactive materials. In general, if radioactive material is not processed or stored in a facility then workers are assumed to have no potential for exposure in that facility.

Workgroup Bioassay Program

If workers have the potential for exposure to radioactive materials but are unlikely to receive intakes in excess of 40 DAC-h then air monitoring and workgroup bioassay programs are required. At a minimum, workers who enter Radiological Control Areas (RCA) where protective clothing is required, should be on a workgroup monitoring program. Some fraction of workers in workgroups with the highest potential for exposure should be sampled (see Chapter 2, page 4) so that samples from the workgroup are submitted uniformly throughout the year. In practice, all workers in the workgroup are usually placed on the program because it makes the program easier to administer.

The minimum bioassay frequency is determined by feeding an intake of the radioactive material that would deliver a 100 mrem CEDE into a Reference Man biokinetic model and observing the time after intake when the quantity of material in the bioassay goes below the Minimum Detectable Amount (MDA); this is the maximum time between bioassays for the workgroup. For design purposes the workgroup should be assumed to have 12 workers who are monitored, which conveniently translates annual worker frequency to a monthly workgroup frequency. Once actual workgroups are identified, the number of workers actually monitored may be used. If the 100 mrem objective can not be reasonably achieved then the actions outlined in Chapter 2, page 6 should be taken.

If a mixture of radionuclides is present and its fractional composition is known and relatively constant, the 100 mrem performance objective may be based on the dose delivered by the mixture and one radionuclide can be used as a tracer. Alternatively, if the fractional composition is not known, the performance objective may be based on the most limiting single radionuclide detected by each bioassay technique. For example, in a mixture of U-238, Cs-137, Ce-144, and Nb-95, the frequency of whole body counts can be based solely on detecting a 100 mrem dose from Ce-144, and the urine bioassay frequency is based solely on detecting a 100 mrem dose from U-238.

If chronic exposures are anticipated, then chronic biokinetic models should be used for the missed dose analysis, otherwise, acute biokinetic models should be used.

Worker Bioassay Program

The bioassay type and frequency for a worker bioassay program are selected with a missed dose analysis, as was done with the workgroup bioassay program, with the following exceptions:

- the design objective is a 12-month effective dose equivalent of 100 mrem in place of a committed effective dose equivalent of 100 mrem
- the design objective applies to a specific worker, not a workgroup
- bioassay frequency for a worker should be not less that once a year

Workers exposed to transuranics, thorium, or insoluble uranium are required to have urine bioassay, chest counts, and either personal air samplers (PAS) or feces bioassay. Feces bioassay is required for workers using any form of respiratory protection (except for plastic suits used for protection from tritium).

If convenient, the acute intake that would give a 12-month dose of 100 mrem may be used for chronic intakes. Also, the committed effective dose equivalent may be used in lieu of the 12-month effective dose equivalent for designing worker programs.

If the dose objectives given above can not be achieved then the actions outlined in Chapter 2, page 6 should be taken.

Workers can have radioactive materials in their body that interfere with the detection and evaluation of intakes, for example, medical radionuclides and previous occupational intakes. In these situations a rough estimate of missed dose should be performed for the worker to ensure that doses in excess of 5 rem annual effective dose equivalent can be detected.

References

(1) *Results of the U.S. Department of Energy Indoor Radon Study.* DOE/ID/1258 U-75, Volume 2. August 1990.

Chapter 4

Designing Special Bioassay Programs

Chapter 4 Preview

- Special Bioassay Programs
- Survey of Medical Procedures

4

Designing Special Bioassay Programs

Design objectives for special bioassay programs were given in Chapter 2. A general discussion of special bioassay programs is given in Appendix 4A of the Internal Dosimetry DOELAP. Special bioassay programs for SRS are recommended in DPSOL 193-212 for a number of scenarios. This section will provide some general philosophy for special bioassay programs and mention some important specifics. Medical procedures for accelerating the removal of radioactive materials from the body will also be reviewed.

Special Bioassay Programs

Special bioassay programs are initiated to determine if an intake has occurred, and to quantify the intake and calculate the dose. Such evaluations are basically an iterative process of postulating and verifying models.

For example, a contamination incident triggers a conservative intake estimate based on the available data and standard assumptions. Thus, we have postulated a model that describes all relevant aspects of the intake such as time of intake, the mode of intake, biokinetic models, etc. Key assumptions in the model are then identified and bioassay data are collected to justify more accurate (usually less conservative) assumptions in the model.

This process is repeated until we are comfortable with the answer, which could mean

- additional information will not justify any significant changes in the model
- a point of diminishing returns has been reached, i.e., it is not worth fooling with any more

Thus, the uncertain aspects of our model dictate the data needed, which dictates the experiments that should be performed (the type and frequency of bioassay).

When Special Bioassay Programs Are Required

A special bioassay program is required whenever a worker is suspected to have had an intake of radioactive material. For example:

- personal contamination
- airborne radioactivity
- positive routine bioassay

Whether or not a worker is suspected to have had an intake can be very subjective. The best policy is to err on the safe side - if in doubt, sample, because you can only collect today's data today.

Chest and Whole-body Counts

Chest and whole-body counts should be performed as soon as possible after a suspected intake of any photon emitter. If there is no external contamination the counts will provide valuable information on the initial deposition of material in the body. If there is external contamination, the counts still provide an upper limit of the initial deposition, which may be used to plan the special bioassay program.

Chest counts should be performed following a suspected intake of thorium, uranium, or any of the transuranics. Note that this applies to soluble uranium compounds like uranium hexafluoride.

Fecal Samples

Fecal samples should be collected after a suspected intake of strontium, thorium, uranium, or any of the transuranics. Note that this applies to soluble uranium compounds like uranium hexafluoride. At least three days of excretion or two voids, which ever takes longer, should be collected. For larger intakes, fecal samples should be collected for 7 to 10 days following the intake.

Urine Samples

True 24-hour urine samples are preferable for all special bioassay programs except for those assessing intakes of tritiated water. Onsite samples are acceptable in lieu of 24-hour samples if they represent 8 hours of excretion and are submitted promptly for analysis.

Nasal Swipes

A positive nasal swipe indicates an intake may have occurred. A negative nasal swipe does not indicate anything - it especially does not indicate that an intake did not occur.

Composition of Radioactive Materials

There are two things to remember about the composition of radioactive materials at SRS

- most radionuclides exist as mixtures
- the importance of each radionuclide in a mixture depends on who you ask

For example, a worker may be exposed to highly enriched uranium reported by the production department as being essentially pure (>98%) U-235. Further investigation reveals that the material is pure U-235 by mass but essentially pure U-234 by activity.

Survey of Medical Procedures

Various medical procedures are used at SRS to mitigate the dose from intakes of radioactive materials. NCRP Report Number 65, Management of Persons Accidentally Contaminated with Radionuclides[1], is the basic technical resource used by the Medical Department, but the professional judgement of the physician plays an important part of determining the actions to be taken for any particular case. This section will review some of the more important procedures used at SRS to mitigate dose from intakes radioactive materials.

Treating Intakes of Transuranics

Calcium or zinc salts of diethylenetriaminepentaacetic (DTPA) acid have been used at SRS for over twenty years to chelate plutonium and the higher transuranics. DTPA is not used to chelate neptunium because the chelate is not stable[2]. This limitation is not of great importance because an intake of neptunium large enough to require chelation has never occurred at SRS. Calcium salts of DTPA are administered typically by inhalation to males and to females who are known not to be pregnant. The zinc salts of DTPA are proported to have a lower embryotoxicity and are therefore used on females who may be pregnant. Zn-DTPA is administered by injection because it has a foul taste.

Treating intakes of Uranium

Intakes of uranium that could be nephrotoxic are treated with oral doses of sodium bicarbonate, which causes the formation of a less toxic uranyl bicarbonate complex in the kidneys. There are no treatments in use at SRS for highly enriched uranium.

Treating Intakes of Radioiodine

Potassium iodide (KI) is used to block uptake of radioiodine by the thyroid. To be effective the KI must be administered as soon as possible after intake.

Treating Intakes of Tritiated Water

Intakes of tritiated water are routinely treated by increased intake of water. Very large intake would be treated with diuretics prescribed by the physician.

Treatment of Contaminated Wounds

Tissue contaminated with transuranics is removed from wound sites to prevent uptake to the blood and deposition in systemic organs. Chelation therapy is usually used in conjunction with excision of tissue. Silver nitrate is occasionally administered topically to wounds to cauterize the site and lockup any radioactive material that may still be present.

Use of Cathartics

Cathartics are routinely administered to reduce the residence time of ingested material in the gastrointestinal tract. A shorter residence time reduces uptake of the the material to the blood and reduces the dose to the GI tract. The usual cathartic administered is Fleets phosphate of soda.

Use of Lung Lavage

Lung lavage is a risky procedure that would only be considered for very large intakes of insoluble uranium or transuranics. This procedure has never been used at SRS.

Skin Contamination

Skin contamination is usually dealt with by Health Physics personnel. If they are unsuccessful at removing contamination the physician may resort to more drastic measures such as bleach and sandpaper.

References

(1) *Management of Persons Accidently Contaminated with Radionuclides.* NCRP Report Number 65, 1979.

(2) *Treatment of Incorporated Transuranium Elements.* IAEA Technical Report Series Number 184, 1979.

Chapter 5

Calculating Intakes

Chapter 5 Preview

- Iterative Methods of Calculating Intake from Bioassay
- Statistical Techniques Used in Iterative Intake Evaluations
- Evaluation of Multiple Types of Bioassay Data
- Noniterative Methods of Calculating Intake from Bioassay
- Calculating an Intake from Air Concentration Measurements
- Decision Rules for Intakes
- Assumptions Used for Calculating Intakes
- Reported Precision for Intakes
- Uncertainties in Intake Estimates
- Computer Codes
- Types of Bioassay
- Evaluating Americium-241 Chest Counts

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Calculating Intakes

An intake is the quantity of radioactive material that passes through the nares, the mouth, or the skin. For practically all occupational exposures intakes can not be measured but rather must be inferred from other measurements such as bioassay and air monitoring. This inference requires a biokinetic model of some sort that can relate intake to air concentration or bioassay data such as urine and fecal excretion. Intake calculations can become quite complex, but they have the advantage of at least attempting to account for all radioactive material that enters the body, whether it can be measured or not.

Intake calculations may be based on

- air concentration measurements
- bioassay
- both air concentration measurements and bioassay

Bioassay-based intake calculations may be classified as iterative or non-iterative. This section will give an overview of each technique.

Iterative Methods of Calculating Intake from Bioassay

Assume a biokinetic model has been selected that completely describes the intake, retention and excretion of a radioactive material in an idealized Reference Man. This model specifies

- how material enters the body, how much is deposited, and the rate at which it leaves the deposition site
- the rate at which material will feed into the bloodstream and the gastrointestinal tract

- where the material will reside in the systemic organs and for how long
- where the material will be excreted and at what rate

The biokinetic model is selected with the hope that is adequately describes what is happening to material in a real person, who we will call John Doe.

Reference Man is given an intake of radioactive material and expectation bioassay data are generated that match the types and times of John Doe's empirical bioassay data.

Working on the assumption that biokinetics of the material in John Doe and Reference Man are the same, any difference in the expectation and empirical bioassay data may be attributed to the magnitude of the intake. The intake is therefore adjusted to produce the "best" match between the expectation and empirical data (exactly what "best" means is discussed later).

In its simplest sense, the process just described is iterative bioassay evaluation. In this case one parameter, the intake, was changed to make the expectation and empirical bioassay data match. Frequently, there are systematic differences between the expectation and empirical bioassay that can not be reconciled by adjusting the intake alone. In these cases the assumption that the biokinetics of Reference Man and John Doe are the same is considered incorrect and the biokinetic model is modified to obtain better agreement. Modifications can range from changing the particle size of inhaled material to changing half-lifes in systemic compartments.

The problems with modifying biokinetic models are that several different modifications can cause the same effect in the expectation bioassay data and the empirical data from an occupational exposure may be woefully inadequate for adjusting parameters in a biokinetic model. The end result may be that even though the expectation bioassay data and empirical bioassay data match, we may still have the wrong model, the match being fortuitous.

To summarize the problem, on one hand we have a strong desire to eliminate systematic differences between the expectation and empirical bioassay data, which may require modifying biokinetic models. On the other hand, we may not know what parameter in the biokinetic model to adjust. The general rule we will use for iterative bioassay evaluations is to begin with standard models. If there is an obvious parameter that can be adjusted to make the data fit better, such as biological elimination rate of tritiated water, then that parameter should be adjusted. If the parameter is not obvious, then modify parameters that are not well known first, and then, if necessary, modify parameters that are relatively well known. The respiratory tract model is assumed to be less well known than the systemic model. Thus, in an iterative evaluation of bioassay data, modifications to the standard biokinetic model should be tried in the following order:

- use variations of standard lung model
  - different Particle AMAD
  - mixture of classes, for example, 50% class W and 50% class Y

- modify standard lung model
  - change retention times for compartments in respiratory tract, NP and TB regions first, then P
  - change fraction of deposition in compartments of respiratory tract, NP and TB regions first, then P
- modify systemic retention model or excretion model, short half-life compartments first and then the long half-life compartments

In vitro lung solubility studies and particle size studies can provide information on inhaled material. This information should be incorporated into the evaluation if possible.

Modifications to parameters can be constrained by using several different types of bioassay data, for example, feces, urine, and chest count bioassay data.

The major problem with iterative methods are that a biokinetic model, which includes exposure pathways and patterns, must be fully specified. Multiple over-lapping acute and chronic intakes can make it impossible to specify the model and perform an iterative evaluation. In many instances non-iterative methods can be used to evaluate these cases.

Statistical Techniques Used In Iterative Intake Evaluations

Statistics are used in iterative intake evaluations to help answer two questions.

- Do I have the right biokinetic model?
- What is the best estimate of the intake (assuming I have the right biokinetic model)?

The Right Biokinetic Model

As discussed previously, the right biokinetic model is a modification of a standard model that adequately predicts all the available bioassay data. Statistical techniques are used to aid the evaluator in deciding whether or not the fit is adequate, but it must be stressed that the evaluator is the final judge of adequacy. Two statistical methods used are:

- examination of residuals
- runs test

A residual e_i is the weighted difference between the observed bioassay data o_i and the expectation bioassay data $\hat{o}_i$:

$$e_i = \sqrt{w_i}\,(o_i - \hat{o}_i).$$

A plot of the residuals versus time that has the residuals evenly and randomly distributed about the zero line shows that the variance of the data was properly handled and the there is no systematic bias. Draper[1] and Thakur[2] may be consulted for further information on residual plots.

The Runs Test is a non-parametric test of the assumption that the residuals are randomly distributed about the zero line. Draper[1], Thakur[2], and Mendenhall[3] discuss the test in detail. A model is assumed to be correct if the probability associated with the Runs Test is greater than 5%. This means that we are assuming the residuals can not be too randomly distributed (i.e., we can not have too many runs) and that we are rejecting fits where the residuals have less than a 5% chance of being derived from a random parent distribution (i.e., too few runs).

If a model passes the Runs Test and has a good residual plot we can say there is no reason to believe that the model is wrong. Note that we are saying the model is not wrong, not that it is correct. The residual plot and Runs Test may indicate that a model has a systematic bias and yet the model may still be acceptable. The evaluation of tritiated water intakes provides a good example. The urinary excretion following an intake of tritiated water frequently follows a slight but distinct sinusoidal pattern about the expectation line. This pattern does not significantly impact the intake estimate but it will cause a perfectly good fit to be rejected. This example reinforces the the fact that statistics are only an aid to the evaluator, who must make the final decision as to whether or not the model is acceptable.

The Best Estimate of Intake

The best estimate of an intake I is the one that minimizes the difference between the observed contents o of a bioassay compartment and the expected contents $\hat{o}$ at time t. The sum of the squares SS is given by

$$SS = \Sigma\, w_i(o_i - \hat{o}_i)^2$$

where o_i and $\hat{o}_i$ are understood to be a function of time and w is the weighting factor. For a given biokinetic model $\hat{o}$ is calculated using f, the fraction of a unit intake expected to be present at time t after an intake I:

$$\hat{o}_i = I \cdot f_i$$

Substituting $I \cdot f_i$ for $\hat{o}_i$, differentiating with respect to I, setting the expression equal to zero, and solving for I gives the following expression:

$$I = \frac{\Sigma\ (w_i \cdot o_i \cdot f_i)}{\Sigma\ (w_i \cdot f_i^2)}$$

The weighting factor is usually the inverse of the total variance of the bioassay measurement. The total variance is composed of the variance of the radiometric technique, the biological variance of the individual, and the variance between the biokinetic model and the individual. This overall variance is seldom known, so various assumptions are made for the weighting factor in order to calculate the intake with the above equation:

- The weighting factor is a constant k for all measurements

$$w_i = k$$

- The weighting factor is inversely proportional to the measurement

$$1/w_i \ \alpha\ o_i = k \cdot o_i$$

- The weighting factor is inversely proportional to the expectation measurement

$$1/w_i \ \alpha\ \hat{o}_i = k \cdot \hat{o}_i = k \cdot I \cdot f_i$$

The first assumption leads to an unweighted fit

$$I = \frac{\Sigma\ (o_i \cdot f_i)}{\Sigma\ (f_i^2)}$$

The second assumption leads to a point weighted fit

$$I = \frac{\Sigma\ (f_i)}{\Sigma\ (f_i^2 / o_i)}$$

The third assumption leads to a group weighted fit

$$I = \frac{\Sigma\ (o_i)}{\Sigma\ (f_i)}$$

Another method is an "eye-ball" fit, whimsically referred to as an eye-chi fit.

The question of which fitting technique is the best depends on how much confidence is placed on the estimates of the variance. If the overall variance of each datum is known, then a weighted fit using these variances as weighting factors should provide the best fit (note that this is different than the second assumption discussed above). If the overall variance is not known, then the group weighted fit should provide the best fit because it is more resistant to outliers than the point weighted fit. The eye-chi fit is provided because it will sometimes provide the best "looking" fit, but the user should be aware that changes in scale will greatly affect what looks good. If the correct biokinetic model is used, the expected and observed values will match rather well and the calculated intake will not be greatly affected by the type of fit used. On the other hand, if an incorrect model is used, then statistics alone will not provide the correct answer.

The variance V in the intake I is given by

$$V = \sum_{i=1}^{N} (o_i / f_i - I)^2 / (N - 1),$$

where N is the number of data. The error E in the intake is the square root of the variance

$$E = \sqrt{V}.$$

Note that one or more outliers can produce a high error for a fit that is otherwise quite good.

In summary

- The goal is to select a standard model that matches all observed data, i.e., the residual plot is uniform and random and the Runs Test result has a probability of >5%.
- If necessary, modify the biokinetic model to achieve an acceptable fit.
- The group-weighted fit is recommended as the default fitting method.
- If several models fit the data, the single best model is assumed to be the one with the most runs. If two or more models have the same number of runs, then the model with the highest runs probability is the best. If two or more models have the same number of runs and the same runs probability, then the model with the lowest error is the best.

Less-Than Data

Less-than data are data reported as less than some reporting level, for example, plutonium urine results of <0.1 d/m/L or a Ce-144 whole body count of <20 nCi.

Less-than data are used as a constraint on a fit, that is, the predictions of a model should agree with the less-than data. For example, if a model predicts a urine concentration of 0.002 d/m/L and the measured concentration is <0.1 d/m/L, the empirical and expectation results are in agreement. Less-than data are not used for residual plots, the Runs Test, or least squares fitting procedures; however, several codes in use at SRS accept less-than data and plot it along with positive measurements to allow easy comparison.

Outliers

An outlier is a datum that does not seem to belong with the rest of the data, that is, it is significantly higher or lower than its neighboring data. In general, we do not reject or "throw out" any data because it does not seem to belong. The preferred method of handling outliers is to collect sufficient data to "dilute" the influence of the suspect result. Sometimes we can not obtain sufficient data to dilute the outlier. In these cases the outlier should be included in the evaluation but its impact on the fit may be reduced by giving it a relatively small weight or by using an eye-chi fit. By using these techniques the desired fit is achieved, but unlike data rejection, it is clear what is being done.

An outlier can also be a datum that does not agree with a model that agrees well with all other data. If good reasons can be given for outliers that do not match a model, the datum may be given little weight in the evaluation. For example, the excretion on the first day of an intake typically does not agree with models because the short-term excretion has the most uncertainty in a model. For this reason we do not have to be overly concerned with excretion on day 1 that does not match a model that otherwise looks good.

An Example of Iterative Evaluation of Bioassay Data

A relatively simple intake of tritiated water will be evaluated to illustrate the statistical procedures discussed.

The urine bioassay data collected following an acute intake of tritiated water is given below:

Concentration of tritium in urine (μCi/L)	Time after intake (days)
47.3	1.29
46.4	1.34
44.0	1.99
41.3	2.31
41.0	2.94
34.8	4.00
32.8	4.56
31.1	5.02
29.7	5.40
29.0	5.96
26.3	6.92
23.1	7.96
22.6	8.96
19.4	9.96
17.5	10.96
14.3	13.48
11.7	15.00
7.6	17.81
7.0	20.00
6.2	21.52
5.4	21.97
4.2	23.79
4.0	25.08
3.2	27.46
2.4	28.29
2.8	29.29
2.1	30.27
0.7	41.29
0.4	52.00
<0.2	73.00
<0.2	93.00

Evaluation of Bioassay Data

The computer code CAIBEC was used to evaluate the bioassay data. The standard biokinetic model for tritiated water assumes a 10 day biological halflife for water in the body. The best fit of the data to this biokinetic model is shown in Figure 5-1 and the residual plot in Figure 5-2. This fit gives an intake of 1801 μCi ± 35% at 1σ, and 4 runs (12 positive residuals and 17 negative residuals), which has a probability of $1.860 \cdot 10^{-5}$. Four significant digits are given in the intake to facilitate comparisons. As discussed on page 19 of this chapter, intakes should be reported with two significant digits.

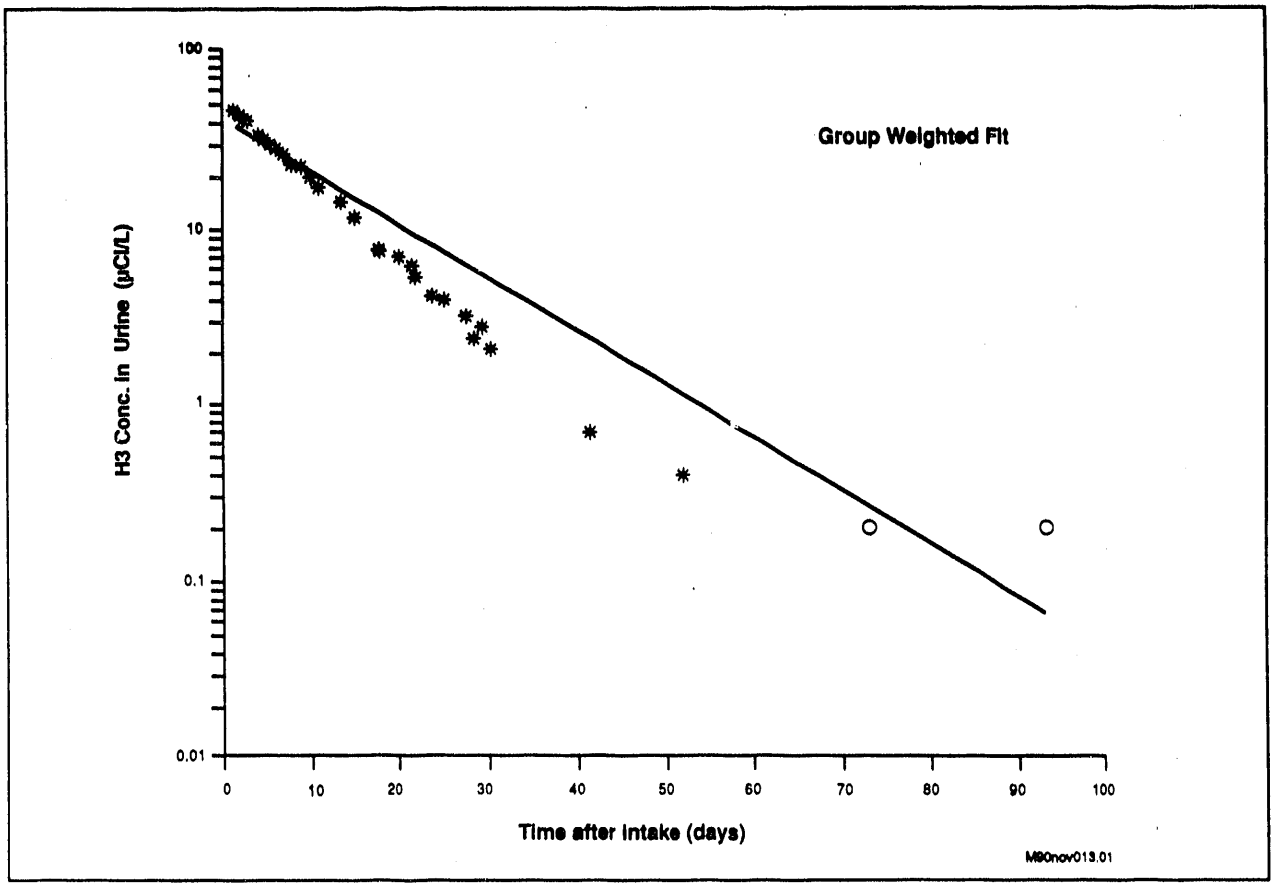

Figure 5-1. Tritium Urine Data Modelled with a 10-Day Elimination Half Life

A cursory examination of Figure 5-1 show that this is the wrong biokinetic model; both the residual plot and Runs Test reinforce this conclusion. The residual plot in Figure 5-2 shows a clear pattern and the Runs Test indicates that the probability of 4 runs occurring randomly in this data is very low.

The less-than data at day 73 and day 93 are represented in Figure 5-1 as circles whereas the rest of the data are presented as stars. The less-than data are not used in the Runs Test, are not plotted on the residual plot, and are not used to calculate the intake or its error.

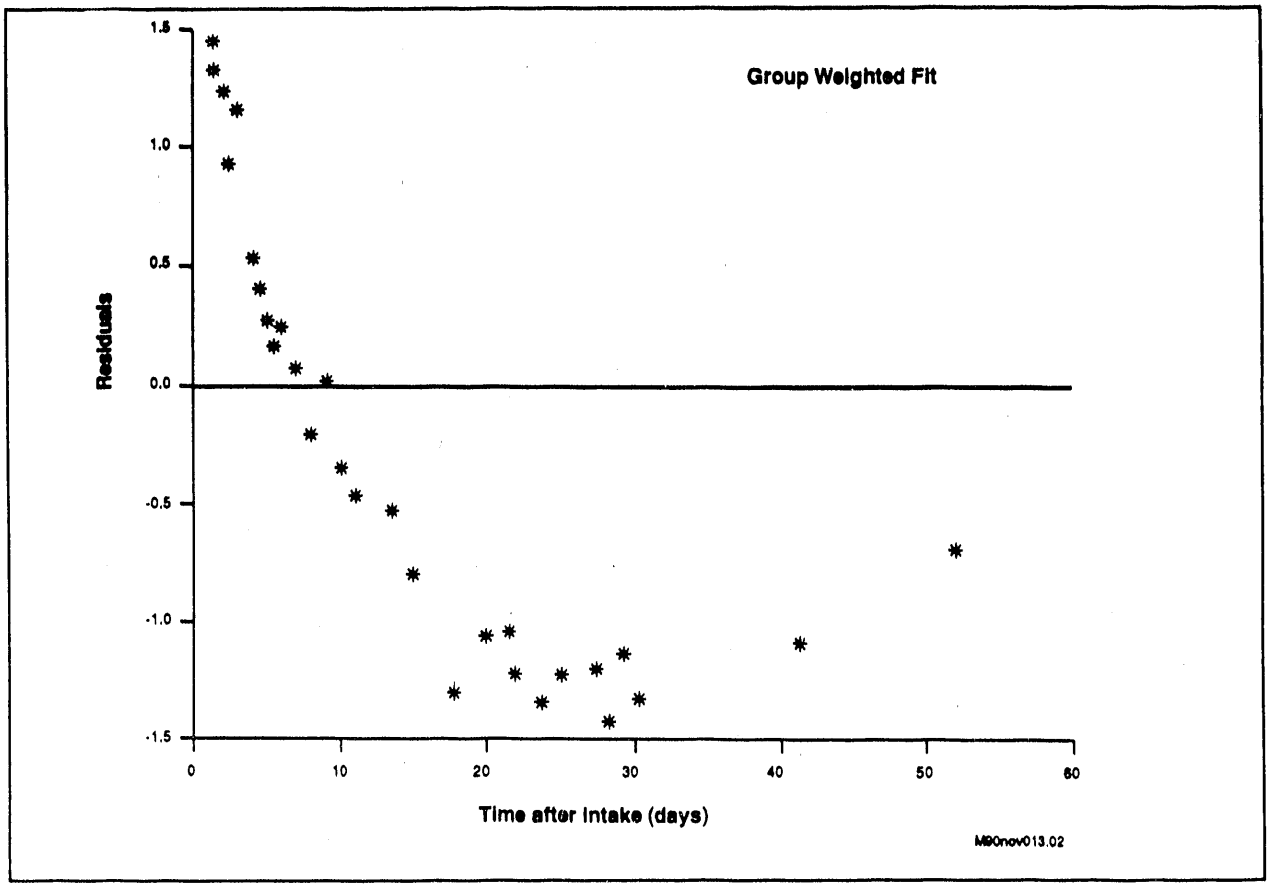

Figure 5-2. Residual Plot for Figure 5-1.

The fit was evaluated with various values of the biological halflife of water in the body. The following table presents the results of the iterations:

Biological Halflife (days)	Number Runs	Probability	Intake and Error (μCi)
10.0	4	$1.860 \cdot 10^{-5}$	1801 ± 35%
9.0	4	$1.860 \cdot 10^{-5}$	1723 ± 29%
8.0	6	$5.142 \cdot 10^{-4}$	1645 ± 20%
7.0	13	$3.999 \cdot 10^{-1}$	1568 ± 11%
6.9	13	$2.264 \cdot 10^{-1}$	1560 ± 11%
6.8	13	$2.264 \cdot 10^{-1}$	1552 ± 12%
6.7	17	$7.951 \cdot 10^{-1}$	1545 ± 13%
6.6	16	$6.707 \cdot 10^{-1}$	1537 ± 15%
6.5	14	$3.733 \cdot 10^{-1}$	1529 ± 18%
6.4	10	$4.721 \cdot 10^{-2}$	1522 ± 21%
6.0	2	$5.434 \cdot 10^{-7}$	1491 ± 39%

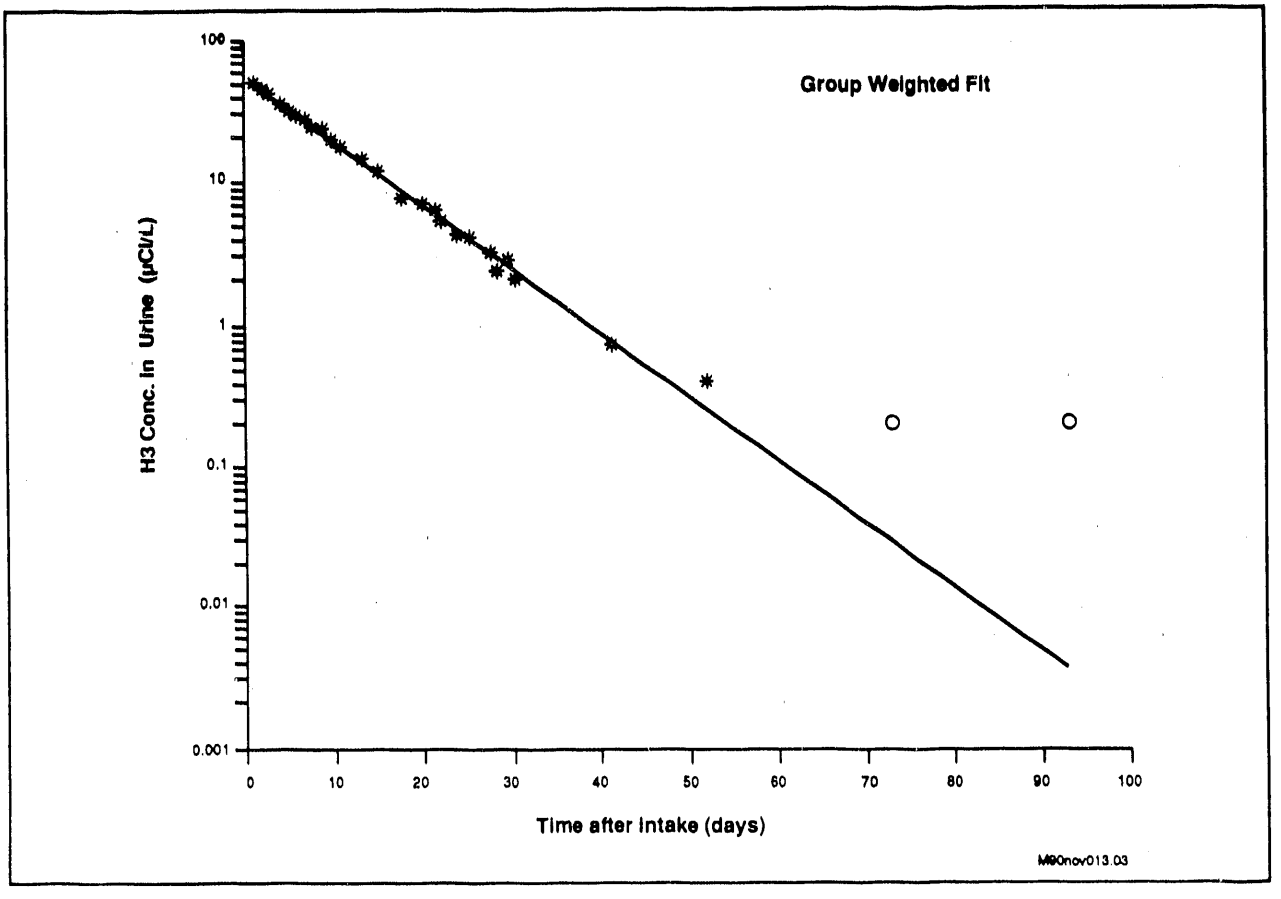

Figure 5-3. Tritium Urine Data Modelled with a 6.7 Day Elimination Half Life.

By the criteria given on page 7, a biological halflife of 6.7 days produces the best fit to the data. The plot of data is given in Figure 5-3 and the residual plot in Figure 5-4. In Figure 5-3 we see that the expectation line is consistent with the less-than data. The residual plot shows that the residuals are symetrically distributed about zero (remember that this is a group weighted fit, so each residual is weighted with the square root of its expectation measurement).

What was gained by modifying the standard biokinetic model? The initial intake estimate using standard models gives a committed effective dose equivalent (CEDE) of

$$\frac{1801\ \mu Ci}{81000\ \mu Ci}\quad 5000\ \text{mrem} = 110\ \text{mrem},$$

where 81000 µCi is the ALI for tritiated water which delivers a CEDE of 5000 mrem. The dose calculated with the modified biokinetic model is

$$\frac{6.7\ \text{days}}{10\ \text{days}} \cdot \frac{1545\ \mu Ci}{1801\ \mu Ci} \cdot 110\ \text{mrem} = 63\ \text{mrem}.$$

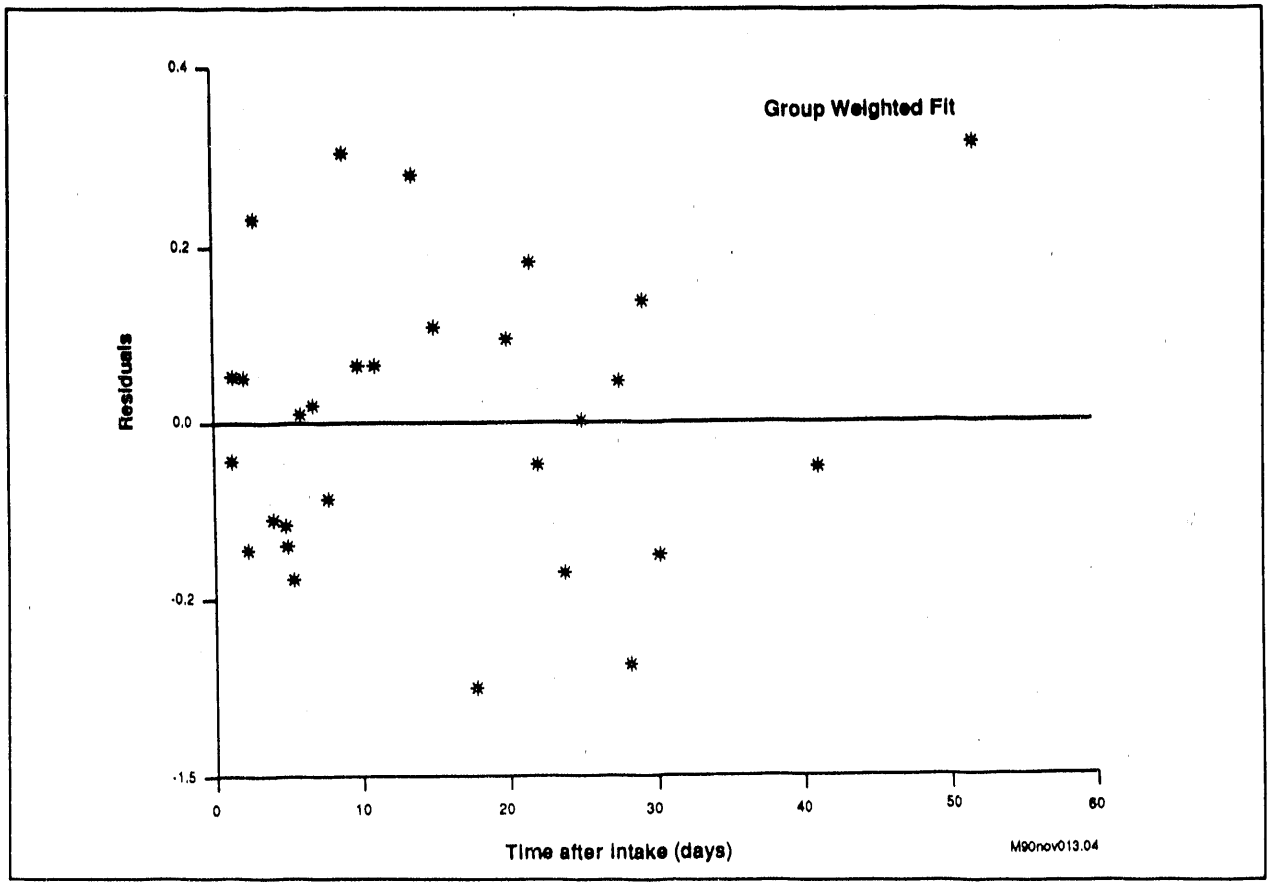

Figure 5-4. Residual Plot for Figure 5-3.

For some applications the initial dose estimate of 110 mrem would have acceptable accuracy, for other applications it would not. The accuracy required for an intake and dose estimate depends on the application.

Evaluation of Multiple Types of Bioassay Data

Two or more type of bioassay data may be available for evaluating an intake. For example, chest count, urine bioassay, and feces bioassay data may be available following an intake of plutonium. The biokinetic model used to evaluate the intake should produce expectation bioassay results that agree with all the available empirical bioassay data. In the plutonium example, the biokinetic model should produce expectation urine, feces, and chest results that agree with the empirical bioassay data.

The degree of agreement sought depends on the magnitude of the intake and the relative quality of the different bioassay data. For large intakes we should modify the biokinetic models to achieve a better agreement between the expectation and empirical bioassay data, which we assume produces a more accurate estimate of the intake. Also, if one type of bioassay data is less accurate or less reliable than the others then the model does not have to match it as well as it must match the more accurate data.

Non-Iterative Methods of Calculating Intake from Bioassay

In a non-iterative calculation the area under an empirical retention or excretion curve is determined by numerical or graphical techniques. A biokinetic model is selected and the area under its expectation retention or excretion curve is calculated. Any difference between the expectation and empirical areas is assumed to be due to the difference in intake. No comparison between expectation and empirical bioassay is possible which means that there is no feedback with which to modify the biokinetic model. The advantage of non-iterative methods is that they can be made to analyze just about any set of bioassay data, so long as all the data may be described by one biokinetic model.

The primary application of non-iterative methods is in the evaluation of intakes of tritiated water and some uranium exposures.

Calculating an Intake from Air Concentration Measurements

Intake estimates should be based on bioassay data with consideration given to air monitoring and other workplace monitoring data data when it is available. Intakes may be based on workplace monitoring data alone with either of the following circumstances:

- bioassay is not feasible (radon exposures for example), or
- when workplace monitoring data indicate that a radiation worker received an intake of radioactive material that would deliver an effective dose equivalent in excess of 100 mrem (in the 12 months following the intake), but the bioassay data are insufficient to confirm or refute the intake.

The intake is calculated by multiplying the time-averaged concentration that best describes the worker's exposure times volume of air breathed, assuming a breathing rate of 0.02 m^3 per minute. Appropriate protection factors for respirators may be applied to the calculated intake.

Decision Rules for Intakes

All intakes of radioactive material are evaluated in terms of effective dose equivalent. The convention for defining what is and what is not an intake is therefore important. If bioassay did not have any systematic or random errors, any positive result would indicate that an intake occurred. Because these errors are always present in bioassay, we use certain rules to keep from assigning a large number of false intakes. An intake is assumed to have occurred (it is a confirmed intake) in the following instances:

- Rule A: positive bioassay measurement is associated with a known incident
- Rule B: positive bioassay measurement is followed by two consecutive bioassay measurements, one of which is positive
- Rule C: positive bioassay measurement is obtained and an appropriate confirmatory bioassay measurement is not obtained.

All confirmatory bioassay measurements shall be able to detect an intake that would deliver a 12-month effective dose equivalent of 100 mrem. This means that if a followup bioassay is not performed within a certain time of the first positive result, the intake is assigned even if the followup eventually turns up negative.

Rule A means that approximately 5% of all workers involved in incidents not resulting in an intake will be falsely assigned an intake. The magnitude of a false intake will almost always be small because it is detected immediately. Therefore, the dose erroneously assigned as a result of Rule A should be small.

Rule B means that approximately 0.5% of all workers on a routine annual bioassay program will be assigned an erroneous intake. The more frequent the routine bioassay, the higher the probability of a false intake being assigned.

The 0.5% false positive rate is the accepted norm for SRS but it may be unacceptably high in certain situations. If so, then a fourth bioassay may be analyzed, reducing the false positive error rate to 0.07%. Note that performing a fourth bioassay means that sometimes a fifth bioassay is required in order to break a 2-positive/2-negative tie.

The false positive error rate is easily calculated for these rules, but the false negative error rate is much more difficult to calculate in the general case because excretion or retention data must be known or assumed.

Assumptions Used for Calculating Intakes

Many assumptions may be required for evaluating occupational intakes of radioactive material because the biokinetic models can be very complex and limited data is available. Assumptions used to determine followup action should be very conservative, producing the highest intake consistent with the known facts. Followup bioassay programs should be designed to supply information that will permit conservative assumptions to be replaced with experimentally determined facts. Assumptions used for final intake calculations should be reasonably conservative, considering known facts and previous experience.

When information is not available the following assumptions should be used to evaluate intakes for the purpose of determining followup actions:

- intake pathway - inhalation
- intake pattern - acute
- Aerosol AMAD - 1.0 μm
- time of intake - immediately after the time when the last bioassay measurement was below the detection level or when the potential for exposure to radioactive materials began.
- inhalation class - the class that results in the highest committed effective dose equivalent.

When information is not available, the following assumptions should be used to evaluate intakes for the purpose of calculating intakes:

- intake pathway - inhalation
- intake pattern - acute
- Aerosol AMAD - 1.0 μm
- time of intake - halfway between the time when the last bioassay that was below the level of detection and the time of the positive bioassay or when the potential for exposure to radioactive materials was highest
- inhalation class - the class that was used in the most similar case

Reported Precision for Intakes

An intake is an intermediate between bioassay and dose equivalent. In this sense the number of significant digits with which the intake is reported is not important; however, to promote consistency and to avoid the appearance of unjustified precision, all intakes should be reported with no more than two significant digits.

Uncertainties in Intake Estimates

All intake estimates have an degree of uncertainty caused by random and systematic errors in the biokinetic models, the bioassay measurements, etc. These uncertainties are discused in NCRP Report 87[4] and Traub and Robinson[5]. Uncertainties in intake estimates are typically not calculated or reported for the following reasons:

- they are difficult or impossible in practice to determine and,
- they are not used for anything, for example, the errors are not propagated to dose, there are no acceptable errors to compare with, etc.

The only recommendation that can be made here is a qualitative one - reduce errors to the extent reasonably achievable and give the most accurate intake estimate possible.

Computer Codes

The two principle computer codes used at SRS to estimate intakes from bioassay data are CAIBEC[6] and PUCHEL[7]. CAIBEC is used to evaluate single acute or chronic intakes of any radionuclide and PUCHEL is used to evaluate multiple acute intakes of transuranics treated with chelation. Occasionally, custom evaluations may be performed on bioassay data that can not be evaluated with either of the two standard codes. These evaluations should be thoroughly documented.

Types of Bioassay

Bioassay may be classified as *in vivo* or *in vitro*. In an *in vivo* bioassay the material in the body is quantified by the radiation it emits. With *in vivo* bioassay we directly quantify material in the body, which greatly simplifies the calculation of intakes by allowing us to accomplish the following:

- determine some information on the location of material in the body as a function of time
- place an upper limit on material that could be in the body and yet not be detected

In vivo bioassay can be complicated by the presence of external contamination on the person, materials that do not emit high intensity penetrating photons, and materials that emit photons that interfere with the analysis.

With *in vitro* bioassay we quantify the material in solids and fluids excreted or taken from the body. The principle advantage of *in vitro* bioassay is that it allows us to evaluate intakes of materials that are difficult or impossible to quantify by *in vivo* methods. We must infer what is in the body from what comes out of the body; this inference is usually more difficult and less accurate than direct measurement.

To evaluate bioassay data we make certain assumptions concerning how the measurements relate to our models. This section will discuss these assumptions for the following types of *in vivo* and *in vitro* bioassay:

- chest count
- whole-body count
- incremental and spot urine
- accumulated and spot feces
- nasal irrigation and smears
- wound counts

Chest Count

Chest counts are performed with the use of germanium or phoswich detectors that are placed close to the upper area of the thorax.

Location of Material

The chest count is a count of photons leaving through an area of the chest. The material producing the photons may be in the lungs, the structure of the chest, the GI tract, etc. In addition, the chest count gives no information on the distribution of the material. For example, the material might be one hot particle in one lung or it might be evenly distributed throughout both lungs. The chest counters are calibrated with phantoms that have all the radioactive material uniformly distributed in the lungs and we usually assume the same, that is the number reported for a chest count is the amount of material evenly distributed in the lungs.

Chest Counts for Uranium

Enriched uranium is quantified with the 185 keV photon of U-235 whereas natural and depleted uranium are quantified with the 63 keV photon of Th-234, the short-lived daughter of U-238. Note that U-234 and U-236 are not quantified by chest counts. Thus, whichever photon is used to quantify uranium in the chest, the isotopic composition of the uranium must be known in order to calculate the total uranium content.

Use of Tracers

A tracer is a radionuclide for which we have relatively good detection capability that is mixed with a radionuclide that we have difficulty detecting. The classic example is the presence of an Am-241 tracer in Pu-239. If the ratio of Am-241 to Pu-239 is known, the Am-241 in the chest may be quantified and the Pu-239 content subsequently inferred using the known ratio. If the ratio is not known, the Am-241 merely indicates that an intake of plutonium may have occurred, but additional information is required to decide how much, if any. The tracer method assumes that the tracer and primary materials behave in the same manner in the body. This assumption should be used with caution after the day of the intake.

External Contamination

External contamination fixed to the skin can cause erroneous chest counts, particularly for the transuranics. In general, the final chest count after all decontamination efforts are completed is assumed to be valid unless there is defensible evidence that there is non-removable external contamination on the person.

Whole-body Count

Whole body counts are performed with large NaI detectors in a standing geometry (Canberra FastScan).

Location of Material

The whole body counter measures the total activity in the body but does not give any information on the distribution of material. The distribution is assumed to be that dictated by the biokinetic models. In practice this assumption is satisfactory because the dose resulting from material quantified by whole-body counting (usually fission and activation products) is typically quite small and the distribution of the material does not greatly influence the dose. The distribution of material in the body should be determined experimentally for intakes that deliver committed effective dose equivalents in excess of 1 rem.

External Contamination

External contamination fixed to the skin can cause erroneous whole-body counts. In general, the final whole-body count after all decontamination efforts are completed is assumed to be valid unless there is defensible evidence that there is non-removable external contamination on the person.

Cs-137 in Workers from Non-SRS Sources

Cs-137 is present in the environment from atmospheric weapons tests. The Cs-137 can work its way through the food chain and end up in man, for example, through

the deer to human pathway. This natural background can cause problems in interpreting whole-body count data. The levels of Cs-137 typically observed in workers (5-30 nCi) do not represent a significant dose alone, but if the Cs-137 is used as a tracer, a small body content can represent relatively large doses. The following guidelines should be followed concerning Cs-137:

- If feasible, Cs-137 should not be used as a tracer in designing bioassay programs.
- Cs-137 in the bodies of radiation workers should be assumed to come from occupational sources, and doses assigned accordingly unless there is evidence that the Cs-137 comes from a non-occupational source.
- A positive Cs-137 whole-body count for an individual who has potential for exposure to Cs-137/Sr-90/TRU mixtures (especially individuals who work with waste streams from the separation facilities) should trigger a special bioassay program including analyses for Sr-90 and plutonium.

Urine Bioassay

The following sections outline the specifics of a urine bioassay.

Time Interval

In vivo measurements are usually interpreted as instantaneous measurements at a point in time, for example, we assume the distribution of the material in the body does not change during the time it takes to perform the measurement. For this reason *in vivo* retention functions that are used to evaluate *in vivo* data are expressed as the fraction of an intake expected to be present at a point in time.

On the other hand, most *in vitro* measurements represent the collection of the sample over a time interval. The excretion functions used to evaluate the *in vitro* data are therefore expressed in terms of the fraction of an uptake excreted over a given time period.

Thus, to evaluate *in vitro* bioassay data, for example, a urine sample a time interval must be ascribed to each sample. There are several different methods of doing this in use in the industry today, and in order of preference, they are:

- Method A: The times for each void are recorded and the interval calculated as the difference between the first and last void times. This method will always give the right answers if it is done correctly, but in practice it seems as if it is seldom done correctly.
- Method B: The creatinine in a sample is measured and the time interval calculated assuming a constant creatinine output per day. This method is quite good, but requires fresh or refrigerated urine.
- Method C: The specific gravity of the sample is measured and the time interval calculated assuming a constant output rate of solids in the urine. The accuracy of this method has not been documented.
- Method D: The volume of the sample is measured and the time interval is calculated assuming a constant volume of urine is excreted per day (1.4

liters per day is a popular value). This method is simple, easy, and may occasionally give the correct answer.

For best results the bladder should be emptied at every void, and that is usually what we assume. Methods A and D are used at SRS.

Samples frequently span non-integral time periods, for example, from day 1.1 to day 1.7, which gives a time interval of 0.6 days ending at 1.7 days after the intake. In these cases the excretion function can be tailored to the sample or the sample can be tailored to the time period. The excretion function is readily adapted to the sample by looking at the expectation excretion for the interval from day 1.1 to day 1.7. The problem is that each sample can represent different time intervals which means that there is no expectation line with which empirical results can be compared.

The alternative is to somehow normalize the bioassay results to integral one day values. For example, the result from day 1.1 to day 1.7 could be adjusted to give the result from day 1 to day 2. This method allows all the expectation and empirical results to be readily compared on the same graph. The disadvantage with this method is that additional error can be introduced into the bioassay result in the normalization procedure. Consider, for example, the following urine bioassay results:

Urine Sample	Times of voids		Volume (ml)	Pu Conc (d/m/1.5 L)
1	5-9-86	9:50	900	10
		19:00		
	5-10	2:00		
2	5-11	3:05	950	10
		10:05		
		21:08		

The time of the last void prior to sample 1 was 6:30 on 5-9, which we will assume to be the time of intake. The activity in the first sample is

$$(10 \text{ d/m})(900 \text{ ml}) / 1500 \text{ ml} = 6.0 \text{ d/m}$$

and

$$(10 \text{ d/m})(950 \text{ ml}) / 1500 \text{ ml} = 6.3 \text{ d/m}$$

in the second sample. Thus,

Urine Sample	Time Interval	Time (days)	Activity (d/m per sample)
1	0.00 - 0.81	0.81	6.0
2	0.81 - 2.61	1.80	6.3

These results can be modeled as 6.0 d/m excreted from t = 0.00 day to t = 0.81 day and 6.3 d/m excreted from t = 0.81 day to t = 1.80 days. The alternative method is to normalize the results to integral 1 day intervals:

excretion on day 1 = 6.0 / 0.81 = 7.4 d/m, and

excretion on day 2 = 6.3 / 1.80 = 3.5 d/m.

At SRS we typically use the normalization method because it permits us to evaluate all of the data graphically at one time.

Urine samples that are adjusted according to their volume as described in method D are called spot urine samples. For example, routine urine samples are almost always spot samples. At SRS the concentration of spot urine samples is reported in units of d/m per 1.5 liters. The urine output is assumed to be 1.5 liters per day, so the concentration is numerically equal to the daily excretion rate. This method can produce wide variations in the excretion rate and is used only if there is no information concerning the time interval a sample represents.

Urine Bioassay for Uranium

Uranium in urine may be analyzed using some chemical property of uranium, in which case the results will be in units of mass. This is referred to as an elemental analysis for uranium. On the other hand, the uranium may be analyzed using the radiation emitted by radioactive isotopes, in which case the results will be in units of activity. This is referred to as an isotopic analysis for uranium. Elemental analysis is used to evaluate chemical toxicity problems whereas isotopic analysis is used to assign dose.

For all enriched uranium most of the dose comes from U-234. In any event, the dose per unit intake of U-234, U-235, U-236, and U-238 are so close that the precise isotopic composition of uranium in the urine is not needed for dose calculations (note that this is not true for chest counts).

Background Uranium in Urine

Uranium in the urine of SRS workers from natural sources has been observed to be approximately 100 ng per liter.

Total and Isotopic Urine Bioassay for Plutonium

Prior to 1988 plutonium in urine was analyzed by gross alpha counting, which gives results in total alpha-emitting plutonium, for example, the sum of Pu-238, Pu-239, Pu-240, etc. Since 1988 plutonium in urine has been analyzed by alpha spectrometry, which gives results by isotope. Evaluations are frequently performed on mixtures of total and isotopic plutonium urine data. This may be accomplished by summing the isotopic results to produce a total plutonium value. An total plutonium intake is calculated and then partitioned into isotopic plutonium intakes.

Contaminated Urine Samples

Urine samples are occasionally contaminated with extraneous radioactive material. Only the chemists who analyze the urine samples can declare that a positive result is due to extraneous material (and they must document their reasons for thinking so). The internal dosimetrists should not say a sample is contaminated, but rather collect sufficient data to show that a particular result is an outlier and may be legitimately ignored.

Feces Bioassay

Time Interval of a Fecal Sample

The material measured in a fecal sample is the sum of excretion from the systemic body, translocation from the lungs, and unabsorbed ingested material accumulated over a certain time interval. This time interval can be difficult to specify because there can be considerable and variable lag time in the GI tract. To minimize this problem, fecal samples should be collected over a time period that is long compared to the GI tract lag time, a week for example, and the data evaluated with an accumulated feces excretion model. This is particularly important in the first week following an intake. A single isolated fecal sample should be assigned the time interval between voids, and if this time is unknown an interval of 1 day should be used.

Systemic Excretion to the Feces

There is excretion of material from the liver to GI tract via the bile. For plutonium, the quantity of material excreted in the bile is approximately the same as that excreted in the urine[8]. The excretion of plutonium in the bile may be ignored in the first week following an inhalation intake because it is typically small compared to the early translocation of material from the lungs. The impact of excretion via the bile should be determined for other time periods and materials.

Effect of Chelation on Fecal Excretion

Chelation immediately after intake greatly increases the urinary excretion rate of plutonium and other transuranics. Most of this increase is assumed to come from the chelation of free plutonium in the blood, for example, the increase in urinary excretion reflects unincorporated plutonium. Excretion of plutonium in the bile also increases, but reflects the removal of plutonium that was incorporated in the liver[9]. Thus, if a person has a small systemic burden when chelated, there will be little increase in the fecal excretion of plutonium, but if they have a large systemic burden, there will be a large increase in fecal excretion. This effect should be considered when evaluating fecal samples.

Use of Tracers in Fecal Samples

If a mixture of materials with different transport characteristics and solubilities is inhaled, the composition of the mixture may change as it makes its way to the feces. Thus, the composition of material in the feces may not be representative of the composition of the material that was inhaled. The ratio of tracers to other materials should always be determined from samples of the inhaled material rather than excreted material.

Contaminated Feces Samples

Feces samples are occasionally contaminated with extraneous radioactive material. Only the chemists who analyze the feces samples can declare that a positive result is due to extraneous material (and they must document their reasons for thinking so). The internal dosimetrists should not say a sample is contaminated, but rather collect sufficient data to show that a particular result is an outlier and may be legitimately ignored.

Nasal Irrigation and Nasal Smears

The biokinetic models for the nasal region of the respiratory tract are, at their best, very rough approximations of what is actually happening to material in the nasal region. For this reason nasal irrigation is viewed primarily as a therapeutic procedure. Material removed from the nasal region may be analyzed to determine the composition of the inhaled material.

Nasal smears indicate that material was probably inhaled if positive and indicate nothing if negative, hence, a negative nasal smear does not indicate that an intake did not occur.

Wounds

We are interested in how much material is deposited in a wound and the rate at which it is translocated to other parts of the body. These parameters form the source term for evaluating urine bioassay data. The dose to a wound site is seldom calculated.

Problems associated with quantifying the material in wounds are

- the depth of deposition and self absorption of the material can create significant uncertainties in the efficiency
- if material is located, Medical may elect to debride the wound and apply silver nitrate, which strongly absorbs low energy photons

Material excised from a wound should be quantified, and knowing the amount of material that was originally present, an attempt made to determine how much material could be left in the body.

Evaluating Americium-241 Chest Counts

Am-241 is readily detected by germanium and phoswich chest counts (MDA of approximately 0.13 nCi) but the interpretation of the chest count data is complicated by the ingrowth of Am-241 from any Pu-241 that may be present. This section discusses the evaluating of Am-241 chest count data.

Supported and Unsupported Am-241

Am-241 that is present in an aerosol at the time of intake is referred to as unsupported Am-241. Am-241 that grows in after the intake from the Pu-241 present is referred to as supported Am-241. Supported Am-241 is assumed to exhibit the same retention in the lung as the plutonium particle in which it is born. Unsupported Am-241 can exhibit retention in the lung that is different than that of the plutonium that is present.

A measured Am-241 chest content q(t) at a time t after intake is thus given by

$$q(t) = X \cdot g(t) + Y \cdot j(t) \cdot f(t), \quad \text{Equation 1}$$

where

X = intake of unsupported Am-241,

Y = intake of Pu-241,

$g(t)$ = fraction of Am-241 intake present in lungs at time t,

$j(t)$ = fraction of Pu-241 intake present in lungs at time t,

and

$f(t)$ = ratio of supported Am-241 to Pu-241 at time t.

The intake retention fractions g(t) and j(t) may be calculated with INDOS. The ratio f(t) is given by

$$f(t) = \frac{k_2/(k_2-k_1)\ (e^{-k_1 \cdot t} - e^{-k_2 \cdot t})}{e^{-k_1 \cdot t}},$$

where

k_1 = decay rate constant for Pu-241, and

k_2 = decay rate constant for Am-241.

Once the intake of Pu-241 is calculated, the intake of α-Pu (Pu-238, 239,240) is calculated assuming a constant ratio of Pu-241 to α-Pu. This ratio is usually obtained by mass spectroscopy of the material.

Calculating Intakes of Am-241 from Chest Counts

Given the fact that there is Am-241 in the chest, there are four different scenarios we can postulate:

1. There is no Pu-241 in the intake.

2. There is no unsupported Am-241 in the intake.

3. Both unsupported Am-241 and Pu-241 are in the intake and the ratio of unsupported Am-241 to Pu-241 in the intake is known.

4. Both unsupported Am-241 and Pu-241 are in the intake and the ratio of unsupported Am-241 to Pu-241 in the intake is unknown.

Scenario 1 means that the intake was pure Am-241, for example, Y in Equation 1 is equal to zero. Standard codes (CAIBEC and INDOS for example) may be used to calculate the intake in this case.

Scenario 2 means that all the Am-241 present came from the Pu-241, for example, X in Equation 1 is equal to zero. Standard codes may be used to calculate the intake once f(t) is applied to the intake retention fractions calculated for Pu-241.

If Pu-241 and unsupported Am-241 are present g(t), j(t), and f(t) must be calculated and applied to the same chest counts. There are no standard codes that can do this; therefore, a large part of the calculations must be done manually or by non-standard codes. Equations for least squares estimates of intake are given below.

Calculating Intakes When the Am-241/Pu-241 Ratio is Known

If the ratio K of Pu-241 to Am-241 in the intake is known then the intake X of unsupported Am-241 that minimizes the sums of the deviations of the observed and expectation chest counts is

$$X = \sum_{i=1}^{N} (q_i(g_i + K \cdot j_i \cdot f_i)) \; / \; \sum_{i=1}^{N} (g_i + K \cdot j_i \cdot f_i)^2$$

where q, j, and f are understood to be functions of time and N is the number of chest counts. The intake Y of Pu-241 is given by

$$Y = K \cdot X$$

Calculating Intakes When the Am-241/Pu-241 Ratio is Unknown

If the ratio K of Pu-241 to Am-241 in the intake is unknown then the intake X of unsupported Am-241 and intake Y of Pu-241 that minimize the sums of the deviations of the observed and expectation chest counts are

$$X = \frac{\sum_{i=1}^{N}(j_i f_i)^2 \cdot \sum_{i=1}^{N}(q_i g_i) - \sum_{i=1}^{N}(g_i j_i f_i) \cdot \sum_{i=1}^{N}(q_i j_i f_i)}{\sum_{i=1}^{N}(g_i^2) \cdot \sum_{i=1}^{N}(j_i f_i)^2 - \sum_{i=1}^{N}(g_i j_i f_i) \cdot \sum_{i=1}^{N}(g_i j_i f_i)}$$

and

$$Y = \frac{\sum_{i=1}^{N}(g_i^2) \cdot \sum_{i=1}^{N}(q_i j_i f_i) - \sum_{i=1}^{N}(g_i j_i f_i) \cdot \sum_{i=1}^{N}(q_i g_i)}{\sum_{i=1}^{N}(g_i^2) \cdot \sum_{i=1}^{N}(j_i f_i)^2 - \sum_{i=1}^{N}(g_i j_i f_i) \cdot \sum_{i=1}^{N}(g_i j_i f_i)} .$$

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Chapter 6

Calculating Dose

Chapter 6 Preview

- Calculating the Number of Decays
- Calculating Absorbed Dose
- Calculating Effective Dose Equivalent
- Plutonium Isotopes
- Evaluating of Tritium Urine Bioassay Data
- Significant Digits of Dose Estimates

6

Calculating Dose

One purpose of an internal dosimetry program is to calculate the dose a worker receives from internal sources of radioactive material. The basic strategy used to calculate internal dose is to

- determine the number of atoms that decay in relevant organs and tissues of the body over the time span of interest,
- apply factors that relate the number of decays to the energy absorbed in the organs and tissues, and
- convert the energy absorbed in the organs and tissues into some measure of biological effect or risk of biological effect.

This section gives an overview of each step.

Calculating the Number of Decays

Calculating the number of decays is the most complex step in calculating dose because it requires the evaluation of experimental data, for example, bioassay data.

To calculate the number of decays U in a tissue or organ i one must first determine the quantity q (t) of radioactive material that is in the organ as a function of time. There are three ways of determining q(t):

- Method A: Use *in vivo* bioassay to measure q(t) at various t, and with an appropriate biokinetic model calculate q(t) for all times.
- Method B: Use *in vitro* bioassay to measure the excretion e(t) by a given pathway and with an appropriate biokinetic model calculate q(t).

- Method C: Use *in vivo* bioassay, *in vitro* bioassay, and air monitoring data with an appropriate biokinetic model to calculate the intake I. Once I is calculated work backwards to find q(t).

One should notice that in practically all situations biokinetic models of some sort must be used to calculate U. For example, in method A the bioassay measurements can be used to determine q(t) at the times t when measurements were made. Information concerning q(t) at any other times must be obtained by interpolation and extrapolation, which require a model of how q(t) behaves at these times.

Method A, which uses *in vivo* measurements of radioactive material in the organ, has the advantage of being the most direct, least complex, and, when it works, the most accurate method of calculating U. Disadvantages include the problems of quantifying from outside the body the quantity of radioactive material in a particular organ and the difficulties in accounting for the material in organs and tissues which were not measured. This method is used occasionally at SRS for evaluating whole body count data for Cs-137 that is not easily evaluated with method C.

Method B, which uses measurements of radioactive material in excreta, has the advantage of being applicable to radioactive materials that can not be quantified by *in vivo* bioassay, and, in certain cases, provides quite accurate results. Disadvantages include the added complexity of biokinetic models, and problems with accounting for feed compartments such as the lung or organs that do not excrete through the excretion pathway(s) being monitored. This method is used primarily for evaluating urine bioassay data for tritium.

Method C, which calculates an intake as an intermediate, has the advantage of accounting for radioactive material in organs and tissues that can not be directly measured, and it is capable of incorporating all types of bioassay measurements and air monitoring information into the calculation. The primary disadvantage is the great complexity of the biokinetic models, which can require lengthy computer calculations and many assumptions. At SRS this is the primary method for evaluating dose from intakes of materials other than tritium.

In section 4.6 of ICRP 30 Part 1[1] and section 3.3 of ICRP 30 Supplement to Part 1[2] the basic method of calculating the number of decays is given. This basic method is used with two slight modifications.

First, biokinetic models are assumed to apply to any time after the intake, which allows transformations and dose to be computed for any time period. This assumption may not be strictly correct in all cases and caution should be used for large doses calculated over short time periods.

Second, the ICRP assumes that daughter radionuclides follow the metabolism of the parent, except for some iodine and tellurium isotopes. This practice is followed except when more accurate biokinetic models that account for known differences in the biokinetics of the parent and daughter radionuclides are available.

Calculating Absorbed Dose

The committed dose equivalent $H_{50,T}$ to a target organ T from source organs S containing radionuclides j that emit radiations i is given by the following equation from section 4.3 of ICRP 30 Part 1:

$$H_{50,T} = k \sum_{S} \sum_{j} [U_S \sum_{i} SEE(T \leftarrow S)_i]_j$$

The constant k converts from MeV/g to Sv. This method of computing organ dose equivalent is used with the following modifications:

- The whole body is always a source organ and the number of transformations U in the whole body is computed as discussed in section 3.2 of ICRP 30 Supplement to Part 1.

- The target and source organs for dose calculations are given in Table 6-1 and Table 6-2. The dose to target T is summed over all applicable source organs S.

- Transformations that occur in the NP region of the respiratory tract are included in the calculation of the transformations for the lung.

- Complete tables of SEE(T←S) for 36 source organs and 24 target organs specific to males and females can be generated for most radionuclides of interest with the SEEAGE code[3]. Alternatively, the SEE(T←S) given in the supplements to ICRP 30 can be used for dose calculations. If a SEE(T←S) is not available from SEEAGE or the supplements of ICRP 30 the specific effective energy SEE(T←S) for radiation i, source S, and target T should be calculated according to the protocol given in section 4.5 of ICRP 30 Part 1. This value should be summed over all significant radiations emitted by a radionuclide.

Calculating Effective Dose Equivalent

Methods for weighting and summing organ dose equivalents to obtain effective dose equivalent are given in section 4.7 of ICRP 30 Part 1 and section 3.4 of ICRP 30 Supplement to Part 1. A more detailed discussion is presented in user's manual to the ICRP code[4]. This method is used with the following modifications:

- All targets and sources are retained for the dose calculation, for example, the 10% significant target or 1% significant source rules are not used.
- In Table 6-1 the weighting factor for each target organ is presented. The effective dose equivalent is the sum of the weighted organ dose equivalent of six risk organs and five remainder organs. The risk organs are those that have a weighting factor other than 0.06. The remainder organs are the five organs with a weighting factor of 0.06 that have the highest 50-year committed weighted organ dose equivalent. The ICRP practice of weighting the highest remainder organ dose equivalent with N x 0.06, where N is the number of remainder organs not eliminated by the 10% rule, and assigning this as the remainder dose is not followed. Remainder organs are selected from the combined dose of a radionuclide and its daughters; however, they are selected separately for each radionuclide in a mixture of radionuclides taken into the body.

Computer Codes

The computer code RADOSE[5] is used exclusively to calculate dose from intakes of uranium and transuranics. TRITDOSE[6] is used exclusively to calculate dose from intakes of tritiated water. GENMOD[7] is a general purpose code that is used primarily for calculating dose from intakes of fission products; however, these intakes can usually be handled with the simplified techniques covered in the next section.

Simplified Method of Calculating Effective Dose Equivalent

In many situations the committed effective dose equivalent received from an intake of radioactive material is quite small, on the order of 100 mrem or less. The annual effective dose equivalent from such intakes can be even smaller by a factor of fifty or more. In such cases it is conveniently to assign the committed effective dose equivalent in the year of the intake in lieu of assigning separate annual effective dose equivalents. The committed effective dose equivalent from an intake may be assigned to an individual in the year of the intake if either of the following conditions apply:

- more than 99% of the committed effective dose equivalent is received within two calendar years
- the committed effective dose equivalent is less than 100 mrem

In practice this simplified method is used only for intakes of fission and activation products. Intakes of uranium and transuranics are more conveniently handled by assigning annual effective dose equivalent with the code RADOSE.

The intake to committed effective dose equivalent conversion factors given in EPA Federal Guidance Report Number 11[8] should be used to calculate dose with this technique.

Plutonium Isotopes

The isotopic composition of the plutonium taken into the body should be determined on a case by case basis and this composition used for dose calculations. In many incidents the composition of the plutonium is not known, in which case the compositions presented below should be used.

Intakes that are assumed to be Pu-239 with an unknown isotopic composition and that are determined from methods that do not identify isotopes of plutonium should be assumed to have the following composition:

- Pu-238 = 0.1071 times the gross alpha plutonium determination
- Pu-239 = 0.8929 times the gross alpha plutonium determination
- Pu-241 = 12.89 times the gross alpha plutonium determination

Intakes that are assumed to be primarily Pu-238 with an unknown isotopic composition and that are determined from methods that do not identify isotopes of plutonium should be assumed to be 100% Pu-238.

Intakes of Pu-239 that are determined from methods that identify isotopes of plutonium should be assumed to have the following composition unless measurements indicate otherwise:

- Pu-238 = 0.1199 times the Pu-239 determination
- Pu-241 = 14.44 times the Pu-239 determination

Intakes of plutonium that are determined from methods that identify isotopes of plutonium should be assumed to contain 100% Pu-238 unless measurements indicate otherwise.

Evaluation of Tritium Urine Bioassay Data

Intakes of tritiated water may be evaluated in terms of intake as shown in Chapter 5[5]. An alternative and very useful method is to calculate the effective dose equivalent directly from the urinary excretion data without calculating the intake. This method will be discussed in this section.

Calculating Effective Dose Equivalent

The effective dose equivalent H_t delivered from the time of uptake to time t after uptake is given by

$$H_t = k \cdot U,$$

where

U = number of tritium atoms that decay in free body water during the time interval ending at t, and

k = a constant.

Derivation of the Constant k

If the free water of the body has a tritium concentration of 1.0 μCi/L then the absorbed dose rate D to the free water is

$$D = \frac{(1.0\ \mu Ci/L)\ (3.7 \cdot 10^4 d/s/\mu Ci)\ (5.7\ keV/d)\ (3600\ s/h)\ (24\ h/day),}{(1000\ g/L)\ (6.242 \cdot 10^{10} keV/d/rad)}$$

$$D = 2.919 \cdot 10^{-4} \text{ rad/day per } \mu Ci/L \text{ in body water}$$

With Q = 1.0 rem/rad, the dose equivalent rate H to body water is

$$H = 2.919 \cdot 10^{-4} \text{ rem/day per } \mu Ci/L \text{ in body water}$$

ICRP 30[1] indicates that the organ is interest is the soft body tissue rather than the free body water. The "density" term in the formula given above changes from 1000 g/L (42000 g of water divided by 42 L of water) to 1500 g/L (63000 g of soft tissue divided by 42 L of water). The dose equivalent rate H_{st} to soft tissue is thus

$$H_{st} = 1.946 \cdot 10^{-4} \text{ rem/day per } \mu Ci/L \text{ in body water} = k$$

Calculating the Number of Decays

A semi-logarithmic plot of tritium concentration in urine versus time after uptake typically yields a straight line. The area under this line is proportional to the number of atoms that decay in the free body water assuming that the concentration of tritium in the urine is the same as that in body water.

One method of calculating the area under the excretion curve is to fit the excretion data with a function consisting of one of more exponentials and then to integrate the function over the time interval of interest. Another method is to simply "connect the dots" and calculate the area of the trapezoids that are formed[9].

An Example

The tritium urine bioassay data given in Chapter 5 were fit to a single exponential function:

$$C(t) = 53.45\ e^{-0.1025t}\ \mu Ci/L,$$

where t is given in days. The number of decays that will occur in the body U over the next fifty years (18250 days) is given by

$$U = \int_0^{18250} C(t)\ dt.$$

$$U = 53.45 / 0.1029 = 519.4\ \mu Ci \cdot days/L$$

The committed effective dose equivalent H_{50} is given by

$$H_{50} = (519.4\ \mu Ci \cdot days/L)(1.946 \cdot 10^{-4}\ rem \cdot L/\mu Ci \cdot day) = 0.101\ rem$$

Also, the intake I of tritiated water is given by the product of the initial concentration of tritium in the body water times the volume of the body water

$$I = (53.45\ \mu Ci/L)(42\ L) = 2245\ \mu Ci.$$

In Chapter 5 the committed effective dose equivalent was reported as 0.063 rem and the intake as 1545 μCi. The difference between the estimates in Chapter 5 and those given here may be attributed to the f_u of 0.47 used in the Chapter 5 calculation. This value is based on Reference Man parameters, which do not adequately describe this person. For example, an f_u of 0.3? used in the Chapter 5 calculation would make it agree with the calculations performed here.

Significant Digits in Dose Estimates

At SRS dose is rounded to the nearest 5 mrem before it is reported. Once rounded the dose becomes an exact integer. A dose of 155130 mrem thus has six significant digits. This is merely a bookkeeping practice and does not imply that we can determine doses with an error of one part in a million.

References

(1) *Limits for Intakes of Radionuclides by Workers.* ICRP Publication 30, Part 1, 1979.

(2) *Limits for Intakes of Radionuclides by Workers.* ICRP Publication 30, Supplement to Part 1, 1980.

(3) *SEEAGE Computer Code User's Manual*

(4) Watson, S. B., Ford, M. R., A User's Manual to the ICRP Code - A Series of Computer Programs to Perform Dosimetric Calculations for the ICRP Committee 2 Report, (Springfield:NTIS) ORNL/TM-6980, 1980.

(5) *RADOSE Documentation ESH-TBD-91-0004*

(6) *TRITDOSE Documentation DPSOL 97-188*

(7) *GENMOD User's Guide*

(8) *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal.* Guidance Report 11, US Environmental Protection Agency, 1988.

(9) American National Standard for Dosimetry - Internal Dosimetry Programs for Tritium Exposure - Minimum Requirements, ANSI N13.14-1983.

Table 6-1. Target Organs for Reference Man

Organ Name	Reference Male Mass (g)	Reference Female Mass (g)	Weighting Factor
adrenals	14	14	0.06
brain	1400	1200	0.06
breasts	26	360	0.15
gall bladder wall	10	8	0.06
LLI wall	160	160	0.06
SI wall	640	600	0.06
S wall	150	140	0.06
ULI wall	210	200	0.06
heart wall	330	240	0.06
kidneys	310	275	0.06
liver	1800	1400	0.06
lungs	1000	800	0.12
muscle	28000	17000	0.06
ovaries	---	11	0.25
pancreas	100	85	0.06
active marrow	1500	1300	0.12
endosteal	120	60	0.03
skin	2600	1790	0.06
spleen	180	159	0.06
testes	35	---	0.25
thymus	20	20	0.06
thyroid	20	17	0.03
bladder wall	45	45	0.06
uterus	--	80	0.06

Table 6-2. Source Organs for Reference Man

Organ Name	Reference Male Mass (g)	Reference Female Mass (g)
adrenals	14	14
brain	1400	1200
breasts	26	360
gall bladder cont	55	50
gall bladder wall	10	8
LLI content	135	135
LLI wall	160	160
SI content	400	375
SI wall	640	600
S cont	250	230
S wall	150	140
ULI content	220	210
ULI wall	210	200
heart content	425	350
heart wall	330	240
kidneys	310	275
liver	1800	1400
lungs	1000	800
muscle	28000	17000
ovaries	---	11
pancreas	100	85
active marrow	1500	1300
cort bone surf	60	60
cort bone vol	4000	2720
trab bone surf	60	60
trab bone vol	1000	680
skin	2600	1790
spleen	180	159
testes	35	---
thymus	20	20
thyroid	20	17
bladder cont	200	200
bladder wall	45	45
uterus	---	80
whole body	70000	58000
soft tissue	63000	53240

Chapter 7

Biokinetic Models and Biokinetic Functions

Chapter 7 Preview

- Respiratory Tract Model
- Gastrointestinal Tract Model
- Systemic Biokinetic Models and Functions
- Elements
- Biokinetic Models for Selected Elements
- Mammillary Biokinetic Models
- Tritium Gas Biokinetic Models

7

Biokinetic Models and Biokinetic Functions

In this manual a biokinetic model is defined as a system of physiological compartments in which the movement of material is governed by first-order linear kinetics. A biokinetic function is an equation, usually a sum of exponential terms, that describes the condition of a compartment or system. For example, an uptake retention function for a system of compartments describes how much material is present in the system at any time after uptake. In ICRP 30 it is common for a term of a biokinetic function to correspond directly to one compartment of the biokinetic model, but this is not always the case, especially for mammillary models.

The biokinetic models used to evaluate bioassay data and calculate dose from intakes of radionuclides are documented in this chapter. Detailed documentations of mammillary and tritium gas biokinetic models are also included.

Respiratory Tract Model

The respiratory tract model described in ICRP 30[1] is used to evaluate bioassay data and calculate dose equivalent following intakes of radioactive material. Retention classes for inhaled radioactive materials are determined from bioassay data, simulated lung solubility tests, and the recommendations of the ICRP. In this report only the retention classes recommended by the ICRP are given. The ICRP 30 respiratory tract model is assumed to apply to intakes of all radionuclides except for some compounds of tritium, sulfur, nickel, and mercury, for which the ICRP uses special respiratory tract models.

Gastrointestinal Tract Model

The gastrointestinal tract model described in ICRP 30[1] is used to evaluate bioassay data and calculate dose equivalent following intakes of radioactive materials. The fraction f_1 of ingested material passing through the wall of the small intestine as reported by the ICRP is given in this report.

Systemic Biokinetic Models and Functions

Systemic biokinetic models may be classified as either cantenary or mammillary. Cantenary models do not explicitly incorporate recycling of material among the compartments whereas mammillary models do. The retention and excretion functions for a cantenary model are easily obtained because of the direct relationship of the compartments to the terms of the functions. Biokinetic functions for mammillary models are more difficult to derive, as discussed in later in this chapter.

Most retention functions presented here are stable element retention functions, i.e., they do not incorporate radioactive decay. Stable element functions are denoted by upper case symbols such as R whereas radio element functions are denoted by lower case symbols such as r.

The computer code CAIBEC[2], which is used to evaluate intakes of radioactive materials, requires retention functions expressed as a sum of seven or less exponentials. Excretion functions are converted to pseudo-retention functions[3] so that they may be used with CAIBEC.

Elements

Biokinetic models and functions for the following elements are given in this report:

- californium
- americium
- curium
- plutonium
- neptunium
- uranium
- thorium
- europium
- cerium
- barium
- lanthanum
- cesium
- iodine
- antimony
- ruthenium
- zirconium
- niobium
- strontium
- zinc
- cobalt
- iron
- manganese
- tritiated water
- tritium gas

Biokinetic Models for Selected Elements

Californium

The respiratory and GI tract parameters for all compounds of californium are class W with an f_1 of 10^{-3}.[4]

Systemic Retention

The systemic retention function for californium given in ICRP Publication 48[5] is

$$R_s(t) = 0.65e^{-3.798 \cdot 10^{-5}t} + 0.25e^{-9.495 \cdot 10^{-5}t} + C,$$

where

t = time after uptake in days, and

C = fraction of uptake going to gonads: $3.5 \cdot 10^{-4}$ for males and $1.1 \cdot 10^{-4}$ for females.

The first term describes the retention of californium on the bone surfaces with a 50 year half-life, the second term describes the retention of californium in the liver with a 20 year half-life, and the constant third term C describes the permanent retention of californium in the gonads. The remainder of the uptake, ~0.1, goes directly to excretion with a 0.25 day half-life. The ICRP 48 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of californium.

Urinary Excretion

The excretion function for americium and curium is used to evaluate urine bioassay data following uptakes of californium.

Americium and Curium

The respiratory and GI tract parameters for all compounds of americium and curium are class W with an f_1 of 10^{-3}.[4]

Systemic Retention

The systemic retention function for americium and curium given in ICRP Publication 48[5] is

$$R_s(t) = 0.45e^{-3.798 \cdot 10^{-5}t} + 0.45e^{-9.495 \cdot 10^{-5}t} + C,$$

where

t = time after uptake in days, and

C = fraction of uptake going to gonads: $3.5 \cdot 10^{-4}$ for males and $1.1 \cdot 10^{-4}$ for females.

The first term describes the retention of Am/Cm on the bone surfaces with a 50 year half-life, the second term describes the retention of Am/Cm in the liver with a 20 year half-life, and the constant third term C describes the permanent retention of Am/Cm in the gonads. The remainder of the uptake, 0.1, goes directly to excretion with a 0.25 day half-life. The ICRP 48 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of americium and curium.

Urinary Excretion

Durbin and Schmidt[6] presented a mammillary model for americium in man. This model was considered by the ICRP when developing the above retention function. The Durbin-Schmidt model may be manipulated to produce the following urinary excretion function (see Chapter 7, page 54 for details):

$$\dot{E}_u(t) = 1.385 \cdot 10^{-1}e^{-2.775t} + 1.729 \cdot 10^{-5}e^{-5.867 \cdot 10^{-3}t}$$
$$+ 5.966 \cdot 10^{-5}e^{-1.059 \cdot 10^{-3}t} + 8.762 \cdot 10^{-6}e^{-2.137 \cdot 10^{-4}t}$$
$$+ 1.083 \cdot 10^{-5}e^{-2.217 \cdot 10^{-5}t} \text{day}^{-1},$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the urine at time t after uptake. The pseudo-retention function derived from this excretion function is

$$\dot{R}_u(t) = 7.820 \cdot 10^{-2}e^{-2.775t} + 4.610 \cdot 10^{-3}e^{-5.867 \cdot 10^{-3}t}$$
$$+ 8.820 \cdot 10^{-2}e^{-1.059 \cdot 10^{-3}t} + 6.419 \cdot 10^{-2}e^{-2.137 \cdot 10^{-4}t}$$
$$+ 7.648 \cdot 10^{-1}e^{-2.217 \cdot 10^{-5}t} \text{day}^{-1},$$

with an f_u of 0.64. Experience has shown that this function underestimates excretion rates in the first few months after uptake. The first two terms in the Durbin-Schmidt function were replaced with those from an empirical curium excretion function presented by Parkinson[7] to produce the Parkinson-Durbin-Schmidt (PDS) excretion function[8]:

$$\dot{E}_u(t) = 5.300 \cdot 10^{-2} e^{-0.935t} + 1.900 \cdot 10^{-3} e^{-7.330 \cdot 10^{-2} t}$$
$$+ 1.600 \cdot 10^{-4} e^{-8.990 \cdot 10^{-3} t} + 5.966 \cdot 10^{-5} e^{-1.059 \cdot 10^{-3} t}$$
$$+ 8.762 \cdot 10^{-6} e^{-2.137 \cdot 10^{-4} t} + 1.083 \cdot 10^{-5} e^{-2.217 \cdot 10^{-5}} \text{ day}^{-1}.$$

The pseudo-retention function derived from this excretion function is

$$R_p(t) = 8.253 \cdot 10^{-2} e^{-0.935t} + 3.776 \cdot 10^{-2} e^{-7.330 \cdot 10^{-2} t}$$
$$+ 2.590 \cdot 10^{-2} e^{-8.990 \cdot 10^{-3} t} + 8.248 \cdot 10^{-2} e^{-1.059 \cdot 10^{-3} t}$$
$$+ 5.973 \cdot 10^{-2} e^{-2.137 \cdot 10^{-4} t} + 7.116 \cdot 10^{-1} e^{-2.217 \cdot 10^{-5}},$$

with an f_u of 0.69. The PDS excretion and pseudo-retention functions are used to evaluate urine bioassay data following uptakes of americium and curium.

Plutonium

The respiratory and GI tract parameters for oxides of plutonium are class Y with an f_1 of 10^{-5}, nitrates are class W with an f_1 of 10^{-4}, and all other compounds are class W with an f_1 of 10^{-3}.[4] Intakes of plutonium have been observed at SRS that are best modelled as class D material.

Systemic Retention

The systemic retention function for plutonium given in ICRP Publication 48[5] is

$$R_s(t) = 0.45e^{-3.798 \cdot 10^{-5}t} + 0.45e^{-9.495 \cdot 10^{-5}t} + C,$$

where

t = time after uptake in days, and

C = fraction of uptake going to gonads: $3.5 \cdot 10^{-4}$ for males and $1.1 \cdot 10^{-4}$ for females.

The first term describes the retention of plutonium on the bone surfaces with a 50 year half-life, the second term describes the retention of plutonium in the liver with a 20 year half-life, and the constant third term C describes the permanent retention of plutonium in the gonads. The remainder of the uptake, ~0.1, goes directly to excretion with a 0.25 day half-life. The ICRP 48 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of plutonium.

Urinary Excretion

Langham[9] presented an empirical urinary excretion function for plutonium

$$E_u(t) = 0.002t^{-0.74}$$

where E_u (t) is the fraction of a unit uptake excreted on day t. The time t must be an integer and greater than or equal to 1. Langham's function underestimates excretion after a few years following uptake so Lawrence[10] added a long-half-life exponential term to Langham's function to correct the problem:

$$E_u(t) = 0.002t^{-0.74} + 8.6 \cdot 10^{-6}e^{-1.9 \cdot 10^{-5}t}.$$

Jones[11] presented the following instantaneous urinary excretion function

$$\dot{E}_u(t) = 4.75 \cdot 10^{-3}e^{-0.558t} + 2.39 \cdot 10^{-4}e^{-4.42 \cdot 10^{-2}t}$$

$$+ 8.55 \cdot 10^{-5}e^{-3.80 \cdot 10^{-3}t} + 1.42 \cdot 10^{-5}e^{-2.84 \cdot 10^{-5}t} \text{ day}^{-1}.$$

The maximum difference observed between the Lawrence and Jones excretion functions over 10,000 days is 34% which occurs on day 1. The pseudo-retention function derived from the Jones excretion function is

$$R_p(t) = 1.590 \cdot 10^{-2} e^{-0.558t} + 1.010 \cdot 10^{-2} e^{-4.42 \cdot 10^{-2} t}$$
$$+ 4.190 \cdot 10^{-2} e^{-3.80 \cdot 10^{-3} t} + 9.321 \cdot 10^{-1} e^{-2.84 \cdot 10^{-5} t},$$

with an f_u of 0.54. The Lawrence and Jones excretion functions, and the Jones pseudo-retention function are used to evaluate urine bioassay data following uptakes of plutonium. The choice of which function to use for a specific application depends primarily on which one is most readily incorporated into the mathematical techniques used.

Fecal Excretion

The ICRP[12] presented Durbin's systemic fecal excretion function

$$\dot{E}_f(t) = 6.0 \cdot 10^{-3} e^{-0.347t} + 1.6 \cdot 10^{-3} e^{-1.1 \cdot 10^{-1} t}$$
$$+ 1.2 \cdot 10^{-4} e^{-1.2 \cdot 10^{-2} t} + 2.0 \cdot 10^{-5} e^{-1.8 \cdot 10^{-4} t}$$
$$+ 1.2 \cdot 10^{-5} e^{-1.7 \cdot 10^{-4} t} \text{ day}^{-1}.$$

Note that this function does not address clearance of plutonium from the respiratory tract through the GI tract to the feces.

Neptunium

The respiratory and GI tract parameters for all compounds of neptunium are class W with an f_1 of 10^{-3}.[4]

Systemic Retention

The systemic retention function for neptunium given in ICRP Publication 48[5] is

$$R_s(t) = 0.75e^{-3.798 \cdot 10^{-5}t} + 0.15e^{-9.495 \cdot 10^{-5}t} + C,$$

where

t = time after uptake in days, and

C = fraction of uptake going to gonads: $3.5 \cdot 10^{-4}$ for males and $1.1 \cdot 10^{-4}$ for females.

The first term describes the retention of neptunium on the bone surfaces with a 50 year half-life, the second term describes the retention of neptunium in the liver with a 20 year half-life, and the constant third term C describes the permanent retention of neptunium in the gonads. The remainder of the uptake, ~0.1, goes directly to excretion with a 0.25 day half-life.

The NCRP presented a slightly different systemic retention function for neptunium in Report 90[13]

$$R_s(t) = 0.50e^{-1.899 \cdot 10^{-5}t} + 0.10e^{-4.748 \cdot 10^{-5}t} + C,$$

where

t = time after uptake in days, and

C = fraction of uptake going to gonads: $3.5 \cdot 10^{-4}$ for males and $1.1 \cdot 10^{-4}$ for females.

The first term describes the retention of neptunium on the bone surfaces with a 100 year half-life, the second term describes the retention of neptunium in the liver with a 40 year half-life, and the constant third term C describes the permanent retention of neptunium in the gonads. The remainder of the uptake, ~0.4, goes directly to excretion with a 0.25 day half-life.

The systemic retention function used to calculate dose and evaluate *in vivo* bioassay data is a combination of the ICRP 48 and NCRP 90 functions:

$$R_s(t) = 0.50e^{-3.798 \cdot 10^{-5}t} + 0.10e^{-9.495 \cdot 10^{-5}t} + C,$$

where

t = time after uptake in days, and

C = fraction of uptake going to gonads: $3.5 \cdot 10^{-4}$ for males and $1.1 \cdot 10^{-4}$ for females.

The first term describes the retention of neptunium on the bone surfaces with a 50 year half-life, the second term describes the retention of neptunium in the liver with a 20 year half-life, and the constant third term C describes the permanent retention of neptunium in the gonads. The remainder of the uptake, 0.4, goes directly to excretion with a 0.25 day half-life.

Urinary Excretion

An excretion function for neptunium is not currently available. The derivative of the ICRP-NCRP retention function with respect to time multiplied by a urinary excretion fraction of 0.5 is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 9.495 \cdot 10^{-6} e^{-3.798 \cdot 10^{-5} t} + 4.748 \cdot 10^{-6} e^{-9.495 \cdot 10^{-5} t} + \\ + 5.564 \cdot 10^{-1} e^{-2.773 t} \text{ day}^{-1}.$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the the urine at time t after uptake.

Uranium

The respiratory and GI tract parameters for UF_6, UO_2F_2, and $UO_2(NO_3)_2$ are class D with an f_1 of 0.05; UO_3, UF_4, and UCl_4 are class W with an f_1 of 0.05; UO_2 and U_3O_8 are class Y with an f_1 of 0.002[1].

Systemic Retention

The systemic retention function for uranium given in ICRP Publication 54[12] is

$$R_s(t) = 0.53596e^{-2.773t} + 0.24e^{-1.155\cdot10^{-1}} + 0.20e^{-3.466\cdot10^{-2}t} + 0.00104e^{-4.621\cdot10^{-4}t} + 0.023e^{-1.386\cdot10^{-4}t},$$

where t is the time in days after uptake of the uranium. The first term describes the direct excretion of uranium. The third and fifth terms describe the retention of uranium in mineral bone. Retention of uranium in the kidney and other tissues are equal and are each described by one-half of the second and fourth terms. The ICRP 54 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of uranium.

Urinary Excretion

The derivative of the ICRP uranium retention function with respect to time multiplied by a urinary excretion fraction of 1.0 adequately describes the urinary excretion of uranium and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 1.486e^{-2.773t} + 2.773\cdot10^{-2}e^{-1.155\cdot10^{-1}t} + 6.931\cdot10^{-3}e^{-3.466\cdot10^{-2}t} + 4.806\cdot10^{-7}e^{-4.621\cdot10^{-4}t} + 3.188\cdot10^{-6}e^{-1.386\cdot10^{-4}t} \text{ day}^{-1}.$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the urine at time t after uptake.

Thorium

The respiratory and GI tract parameters for oxides and hydroxides are class Y with an f_1 of $2 \cdot 10^{-4}$, and all other compounds are class W with an f_1 of $2 \cdot 10^{-4}$.[1]

Systemic Retention

The systemic retention function for thorium given in ICRP Publication 54[12] is

$$R_s(t) = 0.1e^{-1.386t} + 0.2e^{-9.902 \cdot 10^{-4}t} + 0.7e^{-8.664 \cdot 10^{-5}t},$$

where t is the time in days after uptake of the thorium. The first term describes the direct excretion of thorium. The third term describes the retention of thorium in mineral bone. Retention of thorium in the liver is describe by 20% of the second term and retention of thorium in other tissues is described by 80% of the second term. The ICRP 54 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of thorium.

Urinary Excretion

The derivative of the ICRP thorium retention function with respect to time multiplied by a urinary excretion fraction of 1.0 adequately describes the urinary excretion of thorium and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 1.386 \cdot 10^{-1}e^{-1.386t} + 1.980 \cdot 10^{-4}e^{-9.902 \cdot 10^{-4}t} + 6.065 \cdot 10^{-5}e^{-8.664 \cdot 10^{-5}t} \text{ day}^{-1},$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the urine at time t after uptake.

Europium

The respiratory and GI tract parameters for all compounds are class W with an f_1 of $1 \cdot 10^{-3}$.[14]

Systemic Retention

The systemic retention function for europium[14] is

$$R_s(t) = 0.80e^{-1.980 \cdot 10^{-4}t} + 0.06e^{-6.931 \cdot 10^{-2}t} + 0.14e^{-2.773t},$$

where t is the time in days after uptake of the europium. Of the retention described by the first term, 0.5 goes to liver and 0.5 goes to bone. The second term describes retention in the kidneys, and the third term describes direct excretion. This retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of europium.

Excretion

No excretion functions are available for europium.

Cerium

The respiratory and GI tract parameters for oxides, hydroxides, and fluorides are class Y with an f_1 of $3 \cdot 10^{-4}$, and all other compounds are class W with an f of $3 \cdot 10^{-4}$. [1]

Systemic Retention

The systemic retention function for cerium given in ICRP Publication 54[12] is

$$R_s(t) = 1.0e^{-1.980 \cdot 10^{-4}t}$$

where t is the time in days after uptake of the cerium. Of the retention described by this function 0.6 goes to liver, 0.05 to spleen, 0.2 to bone, and 0.15 to all other tissues. The ICRP 54 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of cerium.

Excretion

No excretion functions are available for cerium.

Barium

The respiratory and GI tract parameters for all compounds are class D with an f_1 of 0.1[15].

Systemic Retention

The ICRP alkaline earth systemic retention function is presented in Publication 20[16]. Johnson and Meyers[17] converted this function to a mammillary model which is mathematically less complex yet predicts systemic retention to within +/- 10% of the ICRP function. The Johnson-Meyers model is used to calculate dose following uptakes of barium. Biokinetic functions may be derived from the model (see Chapter 7, page 47 for details)

$$R_s(t) = 5.623 \cdot 10^{-1} e^{-54.6t} + 2.62 \cdot 10^{-1} e^{-6.21 \cdot 10^{-1} t} + 1.15 \cdot 10^{-1} e^{-2.38 \cdot 10^{-1} t} + 2.10 \cdot 10^{-2} e^{-7.92 \cdot 10^{-3} t} + 1.02 \cdot 10^{-2} e^{-1.82 \cdot 10^{-3} t} + 1.35 \cdot 10^{-2} e^{-5.05 \cdot 10^{-4} t} + 1.60 \cdot 10^{-2} e^{-1.29 \cdot 10^{-4} t},$$

where t is the time in days after uptake of the barium. Remember that the individual terms in this function do not correspond to any distinct organ or tissue. The Johnson-Meyers systemic retention function is used to evaluate *in vivo* bioassay data following uptakes of barium.

Urinary Excretion

The derivative of the Johnson-Meyers systemic retention function with respect to time multiplied by a urinary excretion fraction of 0.10[18] adequately describes the urinary excretion of barium and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 3.07 e^{-54.6t} + 1.63 \cdot 10^{-2} e^{-6.21 \cdot 10^{-1} t} + 2.74 \cdot 10^{-3} e^{-2.38 \cdot 10^{-1} t} + 1.66 \cdot 10^{-5} e^{-7.92 \cdot 10^{-3} t} + 1.86 \cdot 10^{-4} e^{-1.82 \cdot 10^{-3} t} + 6.82 \cdot 10^{-7} e^{-5.05 \cdot 10^{-4} t} + 2.06 \cdot 10^{-7} e^{-1.29 \cdot 10^{-4} t} \text{ day}^{-1},$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the urine at time t after uptake.

Lanthanum

The respiratory tract and GI tract parameters for oxides and hydroxides are class W; all other compounds are class D. All classes have an f_1 of 0.001[14].

Systemic Excretion

The systemic retention function for lanthanum given in ICRP Publication 30[14] is

$$R_s(t) = 1.0e^{-1.980 \cdot 10^{-4} t},$$

where t is the time in days after uptake of the lanthanum. Of the retention described by this function 0.6 goes to liver, 0.2 to bone, and 0.2 to all other tissues. This retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of lanthanum.

Urinary Excretion

No excretion functions are available for lanthanum.

Cesium

The respiratory and GI tract parameters for all compounds of cesium are class D with an f_1 of 1.0[1].

Systemic Retention

The systemic retention function for cesium given in ICRP Publication 54[12] is

$$R_s(t) = 0.1e^{-3.466 \cdot 10^{-1}t} + 0.9e^{-6.301 \cdot 10^{-3}t},$$

where t is the time in days after uptake of the cesium. Cesium is assumed to be uniformly distributed throughout the body. The ICRP 54 retention function is used to calculate dose and evaluate *in vivo* bioassay data following uptakes of cesium. To customize coefficients and rate constants based on total body potassium consult Leggett[19].

Urinary Excretion

The derivative of the ICRP cesium retention function with respect to time multiplied by a urinary excretion fraction of 0.8 adequately describes the urinary excretion of cesium and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 2.773 \cdot 10^{-2}e^{-3.466 \cdot 10^{-1}t} + 4.537 \cdot 10^{-3}e^{-6.301 \cdot 10^{-3}t} \text{days}^{-1},$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the urine at time t after uptake.

Iodine

The respiratory and GI tract parameters for all compounds of iodine are class D with an f_1 of 1.0[1].

Thyroid Retention

ICRP 54[12] recommends Riggs mammillary model for iodine; it is used to calculate dose after uptakes of iodine. A retention function for iodine in the thyroid can be derived from the model (see Chapter 7, page 37 for details)

$$R_{thy}(t) = -3.311 \cdot 10^{-1} e^{-2.773t} + 1.796 \cdot 10^{-2} e^{-6.068 \cdot 10^{-2} t} + 3.131 \cdot 10^{-1} e^{-5.798 \cdot 10^{-3} t},$$

where t is the time after uptake in days. This function is used to evaluate thyroid count data following uptakes of iodine.

Systemic Retention

Systemic retention of iodine may be expressed as a sum of three exponentials

$$R_s(t) = 6.700 \cdot 10^{-1} e^{-2.773t} - 3.632 \cdot 10^{-2} e^{-6.068 \cdot 10^{-2} t} + 3.663 \cdot 10^{-1} e^{-5.798 \cdot 10^{-3} t},$$

where t is the time after uptake in days. This function is used to evaluate systemic whole body count data following uptakes of iodine.

Urinary Excretion

The urinary excretion rate of iodine predicted by Riggs model can be expressed as a sum of three exponentials

$$\dot{E}_u(t) = 1.858 e^{-2.773t} - 1.897 \cdot 10^{-3} e^{-6.068 \cdot 10^{-2} t} + 1.822 \cdot 10^{-3} e^{-5.798 \cdot 10^{-3} t} \text{ days}^{-1},$$

where t is the time in days after uptake. The pseudo-retention function derived from this excretion function is

$$R_p(t) = 7.042 \cdot 10^{-1} e^{-2.773t} - 3.450 \cdot 10^{-2} e^{-6.068 \cdot 10^{-2} t} + 3.303 \cdot 10^{-1} e^{-5.798 \cdot 10^{-3} t},$$

with an f_u of 0.95. These excretion and pseudo-retention functions are used to evaluate urine bioassay data following uptakes of iodine.

Antimony

The respiratory and GI tract parameters for oxides, hydroxides, halides, sulfides, sulfates, and nitrates are class W with an f_1 of 0.01; all other compounds are class D with an f_1 of 0.1[14].

Systemic Retention

The systemic retention function for antimony given in ICRP Publication 54[12] is

$$R_s(t) = 0.20e^{-2.773t} + 0.76e^{-1.386 \cdot 10^{-1}t} + 0.04e^{-6.931 \cdot 10^{-3}t},$$

where t is the time in days after uptake. Of the retention described by the second and third terms 25% is in bone, 12.5% in liver, and 62.5% in all other tissues. This function is used to evaluate *in vivo* bioassay data and calculate doses after uptakes of antimony.

Excretion

No excretion functions are available for antimony.

Ruthenium

The respiratory and GI tract parameters for oxides and hydroxides are class Y with an f_1 of 0.05; halides are class W with an f_1 of 0.05; all other compounds are class D with an f_1 of 0.05[15].

Systemic Retention

The systemic retention function for ruthenium given in ICRP Publication 54[12] is

$$R_s(t) = 0.15e^{-2.310t} + 0.35e^{-8.664\cdot10^{-2}t} + 0.30e^{-1.980\cdot10^{-2}t} + 0.20e^{-6.931\cdot10^{-4}t}$$

where t is the time in days after uptake. The last three terms describe the retention of ruthenium in all tissues of the body. This function is used to evaluate *in vivo* bioassay data and calculate doses following uptakes of ruthenium.

Urinary Excretion

The derivative of the ICRP ruthenium retention function with respect to time multiplied by a urinary excretion fraction of 0.8 adequately describes the urinary excretion of ruthenium and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 2.772\cdot10^{-1}e^{-2.310t} + 2.426\cdot10^{-2}e^{-8.664\cdot10^{-2}t} + 4.752\cdot10^{-3}e^{-1.980\cdot10^{-2}t} + 1.109\cdot10^{-4}e^{-6.931\cdot10^{-4}t} \text{ days}^{-1},$$

where t is the time in days after uptake. This function is used to evaluate urine bioassay data.

Zirconium

The respiratory and GI tract parameters for carbides are class Y; oxides, hydroxides, halides, and nitrates are class W; all other compounds are class D. All classes have an f_1 of 0.002[1].

Systemic Retention

The systemic retention function for zirconium given in ICRP Publication 54[14] is

$$R_s(t) = 0.5e^{-9.902 \cdot 10^{-2}t} + 0.5e^{-8.664 \cdot 10^{-5}t},$$

where t is the time in days after uptake. The second term describes the retention of zirconium in the bone whereas the first term describes the retention of zirconium in all other tissues of the body. This function is used to evaluate *in vivo* bioassay data and calculate doses following uptakes of zirconium.

Excretion

No excretion functions are available for zirconium.

Niobium

The respiratory and GI tract parameters for oxides and hydroxides are class Y; all other compounds are class W. All classes have an f_1 of 0.01[1]

Systemic Retention

The systemic retention function for niobium given in ICRP Publication 30[1] is

$$R_s(t) = 0.5e^{-1.155 \cdot 10^{-1}t} + 0.5e^{-3.466 \cdot 10^{-3}t},$$

where t is the time in days after uptake. Of the retention described by these two terms 71% is in bone, 1.8% in kidney, 1% in spleen, 0.2% in testis, and 26 % in all other tissues of the body. This function is used evaluate *in vivo* bioassay data and calculate doses following uptakes of niobium.

Excretion

No excretion functions are available for niobium.

Strontium

The respiratory and GI tract parameter for $SrTiO_3$ is class Y with an f_1 of 0.01; all other compounds are class D with and f_1 of 0.3[1].

Systemic Retention

The ICRP alkaline earth systemic retention function is presented in Publication 20[16]. Johnson and Meyers[17] converted this function to a mammillary model which is mathematically less complex yet predicts systemic retention to within +/- 10% of the ICRP function. The Johnson and Meyers model is used to calculate dose following uptakes of strontium. A retention function may be derived from the Johnson-Meyers model (see Chapter 7, page 43 for details)

$$R_s(t) = 1.29\cdot10^{-1}e^{-25.6t} + 6.99\cdot10^{-2}e^{-5.57\cdot10^{-1}t} + 5.89\cdot10^{-1}e^{-1.59\cdot10^{-1}t} + 2.71\cdot10^{-2}e^{-1.23\cdot10^{-2}t} + 6.65\cdot10^{-2}e^{-3.30\cdot10^{-3}t} + 5.80\cdot10^{-2}e^{-3.72\cdot10^{-4}t} + 6.03\cdot10^{-2}e^{-8.43\cdot10^{-5}t},$$

where t is the time in days after uptake of the strontium. The individual terms in this function do not correspond to any distinct organ or tissue. The Johnson-Meyers systemic retention function is used to evaluate *in vivo* bioassay data following uptakes of strontium.

Urinary Excretion

The derivative of the Johnson-Meyers systemic retention function with respect to time multiplied by a urinary excretion fraction of 0.85[18] adequately describes the urinary excretion of strontium and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 2.81e^{-25.6t} + 3.1\cdot10^{-2}e^{-5.57\cdot10^{-1}t} + 7.96\cdot10^{-2}e^{-1.59\cdot10^{-1}t} + 2.83\cdot10^{-4}e^{-1.23\cdot10^{-2}t} + 1.86\cdot10^{-4}e^{-3.30\cdot10^{-3}t} + 1.84\cdot10^{-5}e^{-3.72\cdot10^{-4}t} + 4.32\cdot10^{-6}e^{-8.43\cdot10^{-5}t} \text{ days}^{-1},$$

where $\dot{E}_u(t)$ is the instantaneous rate of excretion of stable material into the urine at time t after uptake.

Zinc

The respiratory and GI tract parameters for all compounds of zinc are classified as class Y with an f_1 of 0.5[15].

Systemic Retention

The systemic retention function for zinc[15] is

$$R_s(t) = 0.24e^{-3.466 \cdot 10^{-2}t} + 0.76e^{-1.733 \cdot 10^{-3}t} ,$$

where t is the time in days after uptake. The first term describes the retention in soft tissues. Of the retention described by the second term, 74% is in soft tissue and 26% is in bone. This function is used to evaluate *in vivo* bioassay data and to calculate doses following uptakes of zinc.

Urinary Excretion

The derivative of the systemic retention function with respect to time multiplied by a urinary excretion fraction of 0.25[20] adequately describes the urinary excretion of zinc and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 2.080 \cdot 10^{-3}e^{-3.466 \cdot 10^{-2}t} + 3.293 \cdot 10^{-4}e^{-1.733 \cdot 10^{-3}t} ,$$

where t is the time in days after uptake.

Cobalt

The respiratory and GI tract parameters for oxides, hydroxides, halides, and nitrates are class Y; all other compounds are class W. All classes have an f_1 of 0.05[1].

Systemic Retention

The systemic retention function for cobalt given in ICRP Publication 54[12] is

$$R_s(t) = 0.5e^{-1.386t} + 0.3e^{-1.155\cdot10^{-1}t} + 0.1e^{-1.155\cdot10^{-2}t} + 0.1e^{-8.664\cdot10^{-4}t},$$

where t is the time in days after uptake. Of the retention described by the last three terms 10% is in liver and 90% in all other tissues of the body. This function is used to evaluate *in vivo* bioassay data and to calculate doses following uptakes of cobalt.

Urinary Excretion

The derivative of the systemic retention function with respect to time multiplied by a urinary excretion fraction of 0.7 adequately describes the urinary excretion of cobalt and is used to evaluate urine bioassay data

$$\dot{E}_u(t) = 4.851\cdot10^{-1}e^{-1.386t} + 2.426\cdot10^{-2}e^{-1.155\cdot10^{-1}t} + 8.085\cdot10^{-4}e^{-1.155\cdot10^{-1}t} + 6.065\cdot10^{-5}e^{-8.664\cdot10^{-4}t} \text{ day}^{-1},$$

where t is the time in days after uptake.

Iron

The respiratory and GI tract parameters for oxides, hydroxides, and halides are class W; all other compounds are class D. All classes have an f_1 of 0.1[15].

Systemic Retention

The systemic retention function for iron given in ICRP Publication 54[12] is

$$R_s(t) = 1.0e^{-3.466 \cdot 10^{-4} t},$$

where t is the time in days after uptake. This function is used to evaluate *in vivo* bioassay data and to calculate doses following uptakes of iron.

Urinary Excretion

Johnson and Dunford[21] presented a mammillary model for iron. The urinary excretion predicted by this model for times less than 120 days after uptake was converted to a sum of exponentials (see Chapter 7, page 50 for details)

$$\dot{E}_u(t) = 1.368 \cdot 10^{-1} e^{-9.754t} + 1.600 \cdot 10^{-3} e^{-7.861 \cdot 10^{-1} t} + 1.648 \cdot 10^{-3} e^{-2.638 \cdot 10^{-1} t} + 8.192 \cdot 10^{-7} e^{-1.641 \cdot 10^{-3} t} \text{ day}^{-1},$$

where t is the time in days after uptake. The pseudo-retention function derived from this excretion function is

$$R_p(t) = 6.150 \cdot 10^{-1} e^{-1.386t} + 8.921 \cdot 10^{-2} e^{-7.861 \cdot 10^{-1} t} + 2.739 \cdot 10^{-1} e^{-2.638 \cdot 10^{-1} t} + 2.189 \cdot 10^{-2} e^{-1.641 \cdot 10^{-3} t} \text{ day}^{-1},$$

with an f_u of 0.022. These excretion and pseudo-retention functions are used to evaluate urine bioassay data at times less than 120 days following uptakes of iron. For times greater than 120 days the derivative of the ICRP retention function with respect to time times an f_u of 0.022 should be used.

Manganese

The respiratory and GI tract parameters for oxides, hydroxides, halides, and nitrates are class W; all other compounds are class D. All classes have an f_1 of 0.1[12].

Systemic Retention

The systemic retention function for manganese given in ICRP Publication 54[12] is

$$R_s(t) = 0.3e^{-1.733 \cdot 10^{-1}t} + 0.7e^{-1.824 \cdot 10^{-2}t},$$

where t is the time in days after uptake. Of the retention described by the first term 33% is in liver and 67% in in other tissues of the body. Of the retention described by the second term 21% is in liver, 50% is in bone, and 29% is in other tissues of the body. This function is used to evaluate *in vivo* bioassay data and to calculate doses following uptakes of cobalt.

Urinary Excretion

No excretion functions are available for manganese.

Tritiated Water

All inhaled tritiated water is completely and instantaneously absorbed into the bloodstream. The tritiated water in air is absorbed through the skin at a rate of 0.01C Bq per minute, where C is the concentration of tritiated water in $Bq \cdot m^{-3}$. Tritiated water has an f_1 of 1.0[1].

Systemic Retention

The default systemic retention function for tritiated water is

$$r_s(t) = 1.0e^{-7.158 \cdot 10^{-2}t},$$

where t is the time in days after uptake and the rate constant assumes a turnover rate of 3 liters per day in a 42 liter free-body water volume[22] and a physical half-life of 12.46 years. The tritiated water is assumed to be uniformly distributed throughout 63 kg of soft tissues in the body[1].

Biological removal half-lives in the range of 4 to 18 days have been observed at SRS[23]. The observed biological retention half-life for an individual should be used whenever available.

Free-body water mass and soft-tissue mass should be customized for an individual if annual effective dose equivalent from intakes of tritiated water exceed 1.0 rem. Equations in ICRP 23[22] give the volume V_{bw} of free-body water as a function of sex, age, and weight of an individual:

Male

$$V_{bw} = 0.7945W - 0.0024W^2 - 0.0015AW \text{ liters},$$

Female

$$V_{bw} = 0.6981W - 0.0026W^2 - 0.0012AW \text{ liters},$$

where

W = weight of individual in kg, and

A = age of individual in years.

The soft-tissue mass is assumed to be 90% of the total body mass.

Tritium bound to the organic component of tissues can add additional compartments with longer retention times. These compartments are assumed to contribute less than 10% of the total dose and are neglected[1,24].

The function given in this section is used to calculate doses following uptakes of tritiated water.

Urinary Excretion

The derivative of the stable element systemic retention function with respect to time multiplied by a urinary excretion fraction of 0.47 gives the default urinary excretion function for tritiated water and is used to evaluate urine bioassay data

$$\dot{e}_u(t) = 3.357 \cdot 10^{-2} e^{-7.158 \cdot 10^{-2} t} \text{ day}^{-1},$$

where t is the time in days after uptake. The concentration of tritium in the free body water is assumed to be the same as the concentration of tritium in the urine. The tritium urine concentration is thus the systemic retention divided by the 42 L free body water volume

$$c_u(t) = 2.381 \cdot 10^{-2} e^{-7.158 \cdot 10^{-2} t} \text{ L}^{-1}.$$

Any changes made in retention half-life or free-body water volume should be incorporated into the above functions.

Uptake of Tritiated (Liquid) Water Through the Skin

Pinson[28] reported that 0.040 mg of water is taken up through the skin per minute for every square cm of skin immersed.

Tritium Gas

If pure tritium gas is inhaled, approximately $5 \cdot 10^{-5}$ is converted to tritiated water. The tritiated water may be assessed with the models discussed in the previous section. The total effective dose equivalent from an intake of tritium gas is approximately twice the effective dose equivalent determined from the tritiated water. More details on biokinetic models for tritium gas are given in the next section.

Tritium Gas Biokinetic Model

Data from Human Exposure Studies

Peterman et al.[27] measured the peak concentration of tritium in the urine following acute exposures to known concentrations of tritium gas. The mean of the measurements is

$2.4 \cdot 10^{-8}$ Bq/L per Bq·min/m^3.

A peak concentration of tritium in the urine of 1 Bq/L corresponds to an exposure of

$$(1 \text{ Bq/L})/(2.4 \cdot 10^{-8} \text{ Bq/L per Bq} \cdot \text{min/m}^3) = 4.167 \cdot 10^7 \text{ Bq} \cdot \text{min/m}^3$$

of elemental tritium gas. Given that exposure to a tritium gas concentration of 1 Bq/m^3 will deliver $9.85 \cdot 10^{-15}$ Sv/h[15], this exposure will give a lung dose equivalent of

$$(9.85 \cdot 10^{-15} \text{ Sv} \cdot \text{m}^3/\text{h} \cdot \text{Bq})(4.167 \cdot 10^7 \text{ Bq} \cdot \text{min/m}^3)(\text{h}/60 \text{ min})$$

$$= 6.841 \cdot 10^{-9} \text{ Sv} \quad \text{Result 1}$$

Weighting the lung dose by a factor of 0.12 (ICRP 1979) gives the effective dose equivalent

$$8.209 \cdot 10^{-10} \text{ Sv} \quad \text{Result 2}$$

from tritium gas. This compares to an effective dose equivalent of

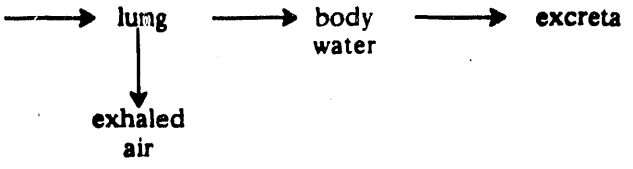
$$7.299 \cdot 10^{-10} \text{ Sv} \quad \text{Result 3}$$

from the tritiated water produced, assuming a 9.7 day halflife for the tritiated water in the body. The total effective dose equivalent is the sum of the weighted lung dose equivalent and the effective dose equivalent from tritiated water:

$$1.551 \cdot 10^{-9} \text{ Sv} \quad \text{Result 4}$$

Derivation of Tritium Gas Biokinetic Model

The following biokinetic model for an acute inhalation intake of pure tritium gas was devised so that it would produce values of lung and effective dose equivalent equal to those derived above:



λ = physical decay constant for tritium

$\lambda = \text{Ln}(2) / (12.28\ \text{y} \cdot 3.156 \cdot 10^{7}\ \text{s/y}) = 1.789 \cdot 10^{-9}\ \text{s}^{-1}$

where 12.28 years is the physical halflife of tritium.

k_1 = total removal rate constant for lung

$k_1 = (0.02\ \text{m}^3 \cdot \text{min}^{-1})/(3 \cdot 10^{-3}\ \text{m}^3)/(60\ \text{s} \cdot \text{min}^{-1}) + \lambda = 1.111 \cdot 10^{-1}\ \text{s}^{-1}$

where

$0.02\ \text{m}^3 \cdot \text{min}^{-1}$ is the mean breathing rate[1], and

$3 \cdot 10^{-3}\ \text{m}^3$ is the mean volume of the lung[1].

k_2 = total removal rate constant for body water

Given a biological removal rate constant of 3 L/(42 L/d) = 0.07143 d^{-1} for water in the body[1],

$k_2 = 0.07143\ \text{d}^{-1} / (86400\ \text{s/d}) + \lambda = 8.285 \cdot 10^{-7}\ \text{s}^{-1}$

k_{12} = transfer rate constant from lung to body water

$k_{12} = k_1\ \ 5.039 \cdot 10^{-5} = 5.599 \cdot 10^{-6}\ \text{s}^{-1}$

where $5.039 \cdot 10^{-5}$ is the fraction of tritium gas converted to tritiated water in the body.

I = intake of tritium gas

$I = 8.334 \cdot 10^{5}$ Bq

D_1 = number of transformations in lung

$$D_1 = I \cdot \int_0^{\infty} \exp(-k_1 \cdot t)\ dt$$

$D_1 = 7.501 \cdot 10^6$

D_2 = number of transformations in body water

$$D_2 = I \cdot k_{12}/(k_2 - k_1) \int_0^\infty \exp(-k_1 \cdot t) - \exp(-k_2 \cdot t)\, dt$$

$D_2 = 5.069 \cdot 10^7$

H_1 = dose equivalent to lung from tritium gas

$H_1 = D_1\ (5.7 \cdot 10^{-6}$ Mev/g/d)$(1.6 \cdot 10^{-10}$ Sv/MeV/g)

$H_1 = 6.841 \cdot 10^{-9}$ Sv

which is the same as Result 1

Multiplying H_1 by 0.12 gives

$0.12\ H_1 = 8.209 \cdot 10^{-10}$ Sv

which is the same as Result 2.

H_2 = effective dose equivalent from tritiated water in body

$H_2 = D_2\ (9.0 \cdot 10^{-8}$ MeV/g/d)$(1.6 \cdot 10^{-10}$ Sv/MeV/g)

$H_2 = 7.299 \cdot 10^{-10}$ Sv

which is the same as Result 3

H_T = effective dose equivalent from tritium gas and tritiated water

$H_T = 0.12H_1 + H_2$

$H_T = 1.551 \cdot 10^{-9}$ Sv

which is the same as Result 4

H_1 = dose equivalent to lung from tritium gas and tritiated water

$H'_1 = H_1 + H_2$

$H'_1 = 7.571 \cdot 10^{-9}$ Sv

Summary and Conclusions

The intake to dose conversion factors are:

Dose	Intake to Dose Factor (Sv per Bq intake)	Comment
lung dose equivalent	$8.208 \cdot 10^{-15}$	from tritium gas
lung dose equivalent	$9.084 \cdot 10^{-15}$	from tritium gas and tritiated water
effective dose equivalent	$8.758 \cdot 10^{-16}$	from tritiated water
effective dose equivalent	$1.861 \cdot 10^{-15}$	from tritiated water and tritium gas

The lung dose equivalent from tritium gas may be checked against the published value[15]. Exposure to 1 Bq/m^3 for 1 hour gives

$$(9.85 \cdot 10^{-15}\ Sv \cdot m^3/Bq \cdot h)(1\ Bq/m^3)(1h) = 9.85 \cdot 10^{-15}\ Sv$$

This exposure gives an intake of

$$(1\ Bq/m^3)(0.02\ m^3 \cdot min^{-1})(60\ min) = 1.2\ Bq$$

The intake to dose conversion factor is thus

$$(9.85 \cdot 10^{-15}\ Sv)\ /\ (1.2\ Bq) = 8.208 \cdot 10^{-15}\ Sv/Bq,$$

which agrees quite well with the value given in the table above.

A person exposed to pure elemental tritium gas will have a total effective dose equivalent approximately twice the effective dose equivalent calculated from tritiated water alone. For a specific exposure the total effective dose depends on the biological halflife of tritiated water in the body, the amount of tritiated water vapor in the tritium gas, and the fraction of the tritium gas converted to tritiated water in the body. The total effective dose equivalent could range from 1 to 5 or more times the effective dose equivalent calculated from tritiated water alone depending on the value of the parameters.

Mammillary Biokinetic Models

Mammillary biokinetic models incorporate explicit recycling of material in the system as opposed to cantenary models, which do not. Biokinetic functions suitable for use in computer codes such as CAIBEC may be derived for a mammillary model. Such models are given in this chapter for the following elements:

- iodine
- strontium
- barium
- iron
- americium

The derivation of retention and excretion functions from the mammillary models for these elements is documented in this appendix.

Review of Mathematical Methods: Iodine Mammillary Model

Solutions for the compartments in the ICRP[12] iodine mammillary model will be derived in this section to illustrate the mathematical methods used. Jacquez[25] may

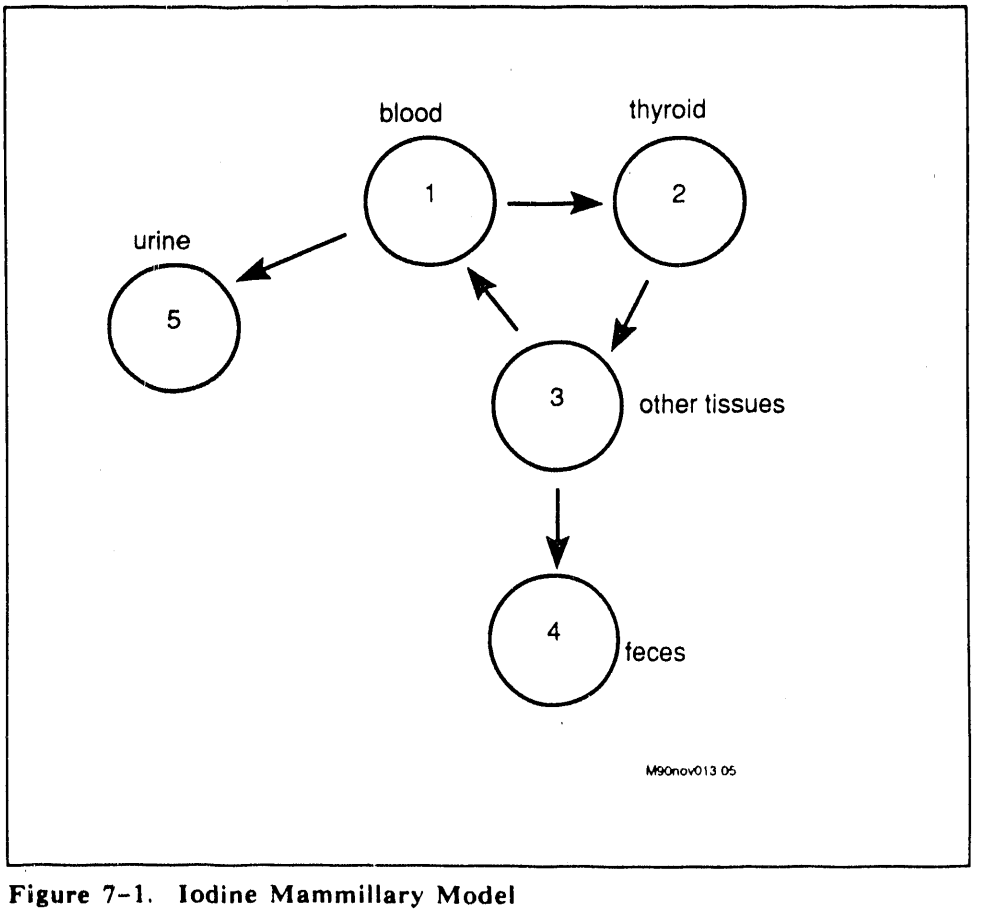


Figure 7-1. Iodine Mammillary Model

be consulted for more details on the mathematics used.The ICRP iodine model is shown in Figure 7-1.

A System of Differential Equations

The system of differential equations that describe this model is

$$\frac{dn_1}{dt} = k_{31}n_3 - k_1n_1$$

$$\frac{dn_2}{dt} = k_{12}n_1 - k_2n_2$$

$$\frac{dn_3}{dt} = k_{23}n_2 - k_3n_3$$

$$\frac{dn_4}{dt} = k_{34}n_3 - k_4n_4$$

$$\frac{dn_5}{dt} = k_{15}n_1 - k_5n_5$$

where

k_{12} = transfer rate constant from blood to thyroid

$k_{12} = 0.33\ \mathrm{Ln}(2.0) / 0.25\ \mathrm{day} = 9.150 \cdot 10^{-1}\ \mathrm{day}^{-1}$

k_{23} = transfer rate constant from thyroid to body tissue

$k_{23} = \mathrm{Ln}(2.0) / 80\ \mathrm{days} = 8.664 \cdot 10^{-3}\ \mathrm{day}^{-1}$

k_{31} = transfer rate constant from body tissue to blood

$k_{31} = 0.9\ \mathrm{Ln}(2.0) / 12\ \mathrm{days} = 5.199 \cdot 10^{-2}\ \mathrm{day}^{-1}$

k_{34} = transfer rate constant from body tissue to feces

$k_{34} = 0.1\ \mathrm{Ln}(2.0) / 12\ \mathrm{days} = 5.776 \cdot 10^{-3}\ \mathrm{day}^{-1}$

k_{15} = transfer rate constant from blood to urine

$k_{15} = 0.67\ \mathrm{Ln}(2.0) / 0.25\ \mathrm{day} = 1.858\ \mathrm{day}^{-1}$

k_{11} = total removal rate constant for blood

$k_{11} = k_{12} + k_{15} = 2.773\ day^{-1}$

k_{22} = total removal rate constant for thyroid

$k_{22} = k_{23}$

k_{33} = total removal rate constant for body tissue

$k_{33} = k_{31} + k_{34} = 5.776 \cdot 10^{-2}\ day^{-1}$

k_{44} = total removal rate constant for feces

$k_{44} = 0$

k_{55} = total removal rate constant for urine

$k_{55} = 0$

n_1 = number of atoms in blood

n_2 = number of atoms in thyroid

n_3 = number of atoms in other tissues

n_4 = number of atoms in feces

n_5 = number of atoms in urine

This system may be represented in matrix notation as follows:

$$K = \begin{bmatrix} -k_{11} & 0 & k_{31} & 0 & 0 \\ k_{12} & -k_{22} & 0 & 0 & 0 \\ 0 & k_{23} & -k_{33} & 0 & 0 \\ 0 & 0 & k_{34} & -k_{44} & 0 \\ k_{15} & 0 & 0 & 0 & -k_{55} \end{bmatrix}$$

$$N' = \begin{bmatrix} dn_1/dt \\ dn_2/dt \\ dn_3/dt \\ dn_4/dt \\ dn_5/dt \end{bmatrix}$$

$$N = \begin{bmatrix} n_1 \\ n_2 \\ n_3 \\ n_4 \\ n_5 \end{bmatrix}$$

Thus

$$N' = K \cdot N$$

The retention R_i of stable material in compartment i in the mammillary model is expressed as the sum of five exponentials (some of which may be zero)

$$R_i = \sum_{j=1}^{5} C_{ij} e^{-\lambda_i e}$$

The eigenvalues of matrix K are the rate constants λ. The coefficients are calculated from the eigenvectors v of matrix K. The matrix V of eigenvectors is

$$V = \left[v_1 v_2 v_3 v_4 v_5 \right]$$

where v_j is the 5 x 1 column eigenvector for j^{th} eigenvalue. Note that v_{ij} is the i^{th} entry for j^{th} eigenvector.

If N_0 is the column vector of the content of each compartment at time 0, and M is a column vector of multipliers that are to be determined, then

$$V \cdot M = N_0.$$

The initial content of all the compartments is zero except the blood, which is 1.0. This equation is solved for M to give the multipliers m which take into account the initial content of the compartments. The the j^{th} coefficient C_{ij} for compartment i is given by

$$C_{ij} = v_{ij} \cdot m_j$$

Solving the System of Differential Equations

The computer code Mathematica[26] was used to evaluate the equations given above and produce the following table:

	coefficients for				
Eigenvalue	thyroid	other tissue	blood	urine	feces
$2.773 \cdot 10^{0}$	$-3.311 \cdot 10^{-1}$	$1.057 \cdot 10^{-3}$	1.000	$-6.700 \cdot 10^{-1}$	$-2.201 \cdot 10^{-6}$
0	0	0	0	$9.531 \cdot 10^{-1}$	$4.694 \cdot 10^{-2}$
$6.068 \cdot 10^{-2}$	$1.796 \cdot 10^{-2}$	$-5.326 \cdot 10^{-2}$	$-1.021 \cdot 10^{-3}$	$3.125 \cdot 10^{-2}$	$5.070 \cdot 10^{-3}$
$5.798 \cdot 10^{-3}$	$3.131 \cdot 10^{-1}$	$5.220 \cdot 10^{-2}$	$9.809 \cdot 10^{-4}$	$-3.143 \cdot 10^{-1}$	$-5.201 \cdot 10^{-1}$
0	0	0	0	0	0

All results are given with four significant digits. In the previous sections of this chapter, sufficient significant digits are retained to permit the coefficients to sum to 1 or 0, whichever is appropriate. The thyroid retention function given previously Chapter 7, page 20 is

$$R_{thy}(t) = -3.311 \cdot 10^{-1} e^{-2.773t} + 1.796 \cdot 10^{-2} e^{-6.068 \cdot 10^{-2} t} + 3.131 \cdot 10^{-1} e^{-5.798 \cdot 10^{-3} t}.$$

Comparison of this function with the table above should make clear the origins of the rate constants and coefficients for the thyroid retention function.

The systemic retention function is given by the sum of the thyroid, blood, and other tissue retention functions:

$$R_s(t) = 6.700 \cdot 10^{-1} e^{-2.773t} - 3.632 \cdot 10^{-2} e^{-6.068 \cdot 10^{-2} t} + 3.663 \cdot 10^{-1} e^{-5.798 \cdot 10^{-3} t}.$$

The solution for the urine compartment (compartment 5 in the mammillary model) is the cumulative urinary excretion function:

$$E_u(t) = -6.700 \cdot 10^{-1} e^{-2.773t} + 9.531 \cdot 10^{-1} + 3.125 \cdot 10^{-2} e^{-6.068 \cdot 10^{-2} t} - 3.143 \cdot 10^{-1} e^{-5.798 \cdot 10^{-3} t}.$$

The derivative of the cumulative urinary excretion function with respect to time gives the iodine urinary excretion function shown in Chapter 7, page 20.

$$\dot{E}_u(t) = 1.858 e^{-2.773} - 1.897 \cdot 10^{-3} e^{-6.068 \cdot 10^{-2} t} + 1.822 \cdot 10^{-3} e^{-5.798 \cdot 10^{-3} t} \text{ day}^{-1}.$$

Mathematica Session

The Mathematica session used to solve the iodine model is given in this section. This is the transcript of the actual input and output. Comments were added for those unfamiliar with the syntax of Mathematica:

```
In[1]:  = k = Table[0, {5}, {5}];                    {set matrix k to zero}
In[2]:  = k[[1,2,]] = 0.33 Log[2.0]/0.25;            {assign rate constants}
In[3]:  = k[[1,5]] = 0.67 Log[2.0]/0.25;             {Log = Ln}
In[4]:  = k[[1,1]] = -(k[[1,2]] + k[[1,5]]);
In[5]:  = k[[2,3]] = Log[2.0]/80.0;
In[6]:  = k[[2,2]] = K[[2,3]];
In[7]:  = k[[3,1]] = 0.9 Log[2.0]/12.0;
In[8]:  = k[[3,4]] = 0.1 Log[2.0]/12.0;
In[9]:  = k[[3,3]] = -(k[[3,1]] + k[[3,4]]);
In[10]: = k = Transpose[k];                          {take transpose of k}
In[11]: = {a,v} = N[Eigensystem[k]];                 {solve for eigenvalues a}
In[12]: = v = Transpose[v];                          {and eigenvectors v}
In[13]: = m = [Linear Solve[v,{1,0,0,0,0}]];         {solve for m}
```

```
In[14]: = Do[
            Print[i,"   ",Scientific Form[N[a[[i]],16]]],
          {i,5}]
```

1	-2.772533797149725	{these are the eigenvalues} {the integer is the compartment} {number}
2	$9.8771359902825 \times 10^{-18}$	
3	$-6.068361002525243 \times 10^{-2}$	
4	$-5.797919868465631 \times 10^{-3}$	
5	$0. \times 10^{-1}$	

```
In[15]: = Do[
            Print[i,"   ",j, Scientific Form[N[v[[i,j]] m[[j]],16]]],
          {i,5}, {j,5}]
```

1	1	1.00004010608081	{these are the coefficients} {the first integer is the compartment} {number, the second is the coefficient} {number}
1	2	$0.\times 10^{-1}$	
1	3	$-1.020991855205588 \times 10^{-3}$	
1	4	$9.808857743956369 \times 10^{-4}$	
1	5	$0. \times 10^{-1}$	
2	1	$-3.310543380122235 \times 10^{-1}$	
2	2	$1.916083466709762 \times 10^{-17}$	
2	3	$1.795797713525147 \times 10^{-2}$	
2	4	$3.130963608769721 \times 10^{-1}$	
2	5	$0. \times 10^{-1}$	
3	1	$1.056577774095174 \times 10^{-3}$	
3	2	$0. \times 10^{-1}$	
3	3	$-5.326108911769997 \times 10^{-2}$	
3	4	$5.220451134360483 \times 10^{-2}$	
3	5	$0. \times 10^{-1}$	
4	1	$-2.201247302825625 \times 10^{-6}$	
4	2	$4.694167852062586 \times 10^{-2}$	
4	3	$5.069706869796066 \times 10^{-3}$	
4	4	$-5.200918414311911 \times 10^{-2}$	
4	5	$9.540979117872439 \times 10^{-18}$	
5	1	$-6.700401445953786 \times 10^{-1}$	
5	2	$9.530583214793735 \times 10^{-1}$	
5	3	$3.125439696785801 \times 10^{-2}$	

5 4 $-3.142725738518529\ 10^{-1}$

5 5 $0.\ 10^{-1}$

In[16]: = Quit

Evaluations for Other Materials

The evaluations of mammillary models in the following sections will give the diagram of the mammillary model, a transcript of the Mathematica session used to solve the model, and the retention function that appears earlier. Given this information one should be able to easily reproduce the calculations.

Strontium Mammillary Model

The Johnson and Meyers[17] mammillary model for strontium shown in Figure 7-2 is based on the ICRP 20 alkaline earth model.

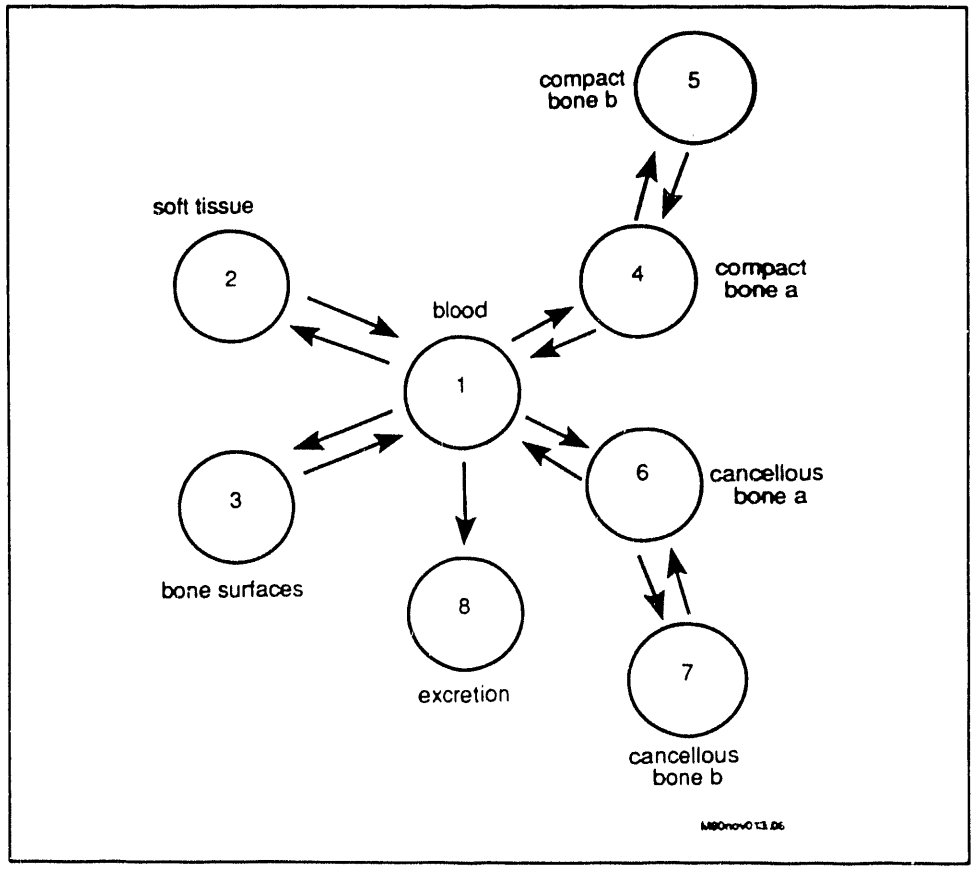


Figure 7-2. Johnson-Meyers Alkaline Earth Mammillary Model

Mathematica Session

The Mathematica section used to solve the strontium model is given in this section. Refer to the Mathematica session for the iodine mammillary model for comments on the syntax of Mathematica.

```
In[1]:  = k = table[0, {8}, {8}];
In[2]:  = k[[1,2]] = 17.6;
In[3]:  = k[[2,1]] = 1.29;
In[4]:  = k[[1,3]] = 2.66;
In[5]:  = k[[3,1]] = 0.403;
In[6]:  = k[[1,6]] = 0.378;
In[7]:  = k[[6,1]] = 0.00479;
In[8]:  = k[[6,7]] = 0.00725;
In[9]:  = k[[7,6]] = 0.00105;
In[9]:  = k[[7,6]] = 0.00105;
In[10]: = k[[1,4]] = 0.516;
In[11]: = k[[4,1]] = 0.00214;
In[12]: = k[[4,5]] = 0l00136;
In[13]: = k[[5,4]] = 0.000149;
In[14]: = k[[1,8]] = 3.44;
In[15]: = k[[1,1]] = -(k[[1,2]] + k[[1,3]] + k[[1,4]] + k[[1,6]] + k[[1,8]]);
In[16]: = k[[2,2]] = -k[[2,1]];
In[17]: = k[[3,3]] = k[[3,1]];
In[18]: = k[[4,4]] = -(k[[4,1]] + k[[4,5]]);
In[19]: = k[[5,5]] = -k[[5,4]];
In[20]: = k[[6,6]] = -(k[[6,1]] + k[[6,7]]);
In[21]: = k[[7,7]] = -k[[7,6]];
In[22]: = k = Transpose[k];
In[23]: = {a,v} = N[Eigensystem[k]];
In[24]: = v = Transpose[v];
In[25]: = m = N[Linear Solve[v,{1,0,0,0,0,0,0,0}]];
In[26]: = Do[
            Print[i,"    ",Scientific Form[N[a[[i]],16]]],
          {i,8}]

1       -2.557172976337359 10^1

2       -5.573297627903471 10^-1

3       -1.586416697213238 10^-1

4       -1.228587869015331 10^-2

5       -3.296010854401425 10^-3

6       -4.915104293200424 10^-18

7       -3.716212795042755 10^-4

8       -8.429329068235206 10^-5


In[27]: = Do[
            Print[i,"  ",j,"  ",Scientific Form[N[v[[i,j]] m[[j]],16]]],
          {i.8},{j,8}]

1    1    9.61349875328245 10^-1

1    2    1.132347983938305 10^-2

1    3    2.715848342056876 10^-2

1    4    9.670673451053028 10^-5

1    5    6.370782040117539 10^-5

1    6    1.14193730277659 10^-17
```

1	7	$6.270048200601657\ 10^{-6}$
1	8	$1.476808690771299\ 10^{-6}$
2	1	$-6.968102343062384\ 10^{-1}$
2	2	$2.720094730913905\ 10^{-1}$
2	3	$4.224915266980639\ 10^{-1}$
2	4	$1.332096514391259\ 10^{-3}$
2	5	$8.714184835979591\ 10^{-4}$
2	6	$2.569358931247327\ 10^{-17}$
2	7	$8.556949439968779\ 10^{-5}$
2	8	$2.015002439492658\ 10^{-5}$
3	1	$-1.016018961789023\ 10^{-1}$
3	2	$-1.951694593976458\ 10^{-1}$
3	3	$2.956378275147228\ 10^{-1}$
3	4	$6.583839686562378\ 10^{-4}$
3	5	$4.239707555317896\ 10^{-4}$
3	6	$-5.70968651388295\ 10^{-18}$
3	7	$4.142362807759445\ 10^{-5}$
3	8	$9.749709559694092\ 10^{-6}$
4	1	$-1.940128590887891\ 10^{-2}$
4	2	$-1.055002787246891\ 10^{-2}$
4	3	$-9.032965134837314\ 10^{-2}$
4	4	$-5.690459193751659\ 10^{-3}$
4	5	$1.224874633673091\ 10^{-1}$
4	6	$-3.711296234023917\ 10^{-16}$
4	7	$8.011007608501146\ 10^{-4}$
4	8	$2.682860195313741\ 10^{-3}$
5	1	$1.031838785417127\ 10^{-6}$
5	2	$2.575113655163381\ 10^{-5}$
5	3	$7.751041486637241\ 10^{-4}$
5	4	$6.376453700390822\ 10^{-4}$
5	5	$-5.293370690048834\ 10^{-2}$
5	6	$-3.094650090524559\ 10^{-15}$

5	7	$-4.893948310701491\ 10^{-3}$
5	8	$5.638812271715308\ 10^{-2}$
6	1	$-1.421731877306908\ 10^{-2}$
6	2	$-7.849739885841477\ 10^{-3}$
6	3	$-7.004893246759653\ 10^{-2}$
6	4	$8.468927788824883\ 10^{-2}$
6	5	$1.984743914876261\ 10^{-3}$
6	6	$-1.170485735346005\ 10^{-16}$
6	7	$5.304908109226325\ 10^{-3}$
6	8	$1.370612141558023\ 10^{-4}$
7	1	$4.031005904363648\ 10^{-6}$
7	2	$1.023057425042422\ 10^{-4}$
7	3	$3.22259901991099\ 10^{-3}$
7	4	$-5.464612796397381\ 10^{-2}$
7	5	$-6.406644631594075\ 10^{-3}$
7	6	$-8.678723501102084\ 10^{-16}$
7	7	$5.669485588195606\ 10^{-2}$
7	8	$1.028980945293107\ 10^{-3}$
8	1	$-1.293242030058461\ 10^{-1}$
8	2	$-6.989178265387333\ 10^{-2}$
8	3	$-5.889069569859605\ 10^{-1}$
8	4	$-2.707752331812048\ 10^{-2}$
8	5	$-6.649095280963398\ 10^{-2}$
8	6	1.000000000000009
8	7	$-5.804017961200965\ 10^{-2}$
8	8	$-6.026840161456465\ 10^{-2}$

In[28]: = Quit

Strontium Systemic Retention Function

The sum of the retention equations for all the systemic compartments gives the strontium systemic retention function:

$$R_s(t) = 1.292\cdot10^{-1}e^{-25.6t} + 5.89\cdot10^{-1}e^{-1.59\cdot10^{-1}t} + 6.99\cdot10^{-2}e^{-5.57\cdot10^{-1}t} + 6.65\cdot10^{-2}e^{-3.30\cdot10^{-3}t} + 2.71\cdot10^{-2}e^{-1.23\cdot10^{-2}t} + 6.03\cdot10^{-2}e^{-8.43\cdot10^{-5}t} + 5.80\cdot10^{-2}e^{-3.72\cdot10^{-4}t}$$

Barium Mammillary Model

The Johnson and Meyers[17] mammillary model for barium shown in Figure 7-2 is based on the ICRP 20 alkaline earth model[16].

Mathematica Session

The Mathematica Session used to solve to Barium model is given in this section. Refer to the Mathematica session for the iodine mammillary model for comments on the syntax of Mathematica.

```
In[1]:  = k = Table[0, {8}, {8}];
In[2]:  = k[[1,2]] = 11.1;
In[3]:  = k[[2,1]] = 1.54;
In[4]:  = k[[1,3]] = 10.05;
In[5]:  = k[[3,1]] = 0.883;
In[6]:  = k[[1,6]] = 0.426;
In[7]:  = k[[6,1]] = 0.00112;
In[8]:  = k[[6,7]] = 0.00605;
In[9]:  = k[[7,6]] = 0.000852
In[9]:  = k[[7,6]] = 0.00105;
In[10]: = k[[1,4]] = 0.543;
In[11]: = k[[4,1]] = 0.00439;
In[12]: = k[[4,5]] = 0.00366;
In[13]: = k[[5,4]] = 0.000189;
In[14]: = k[[1,8]] = 0.919;
In[15]: = k[[1,1]] = -(k[[1,2]] + k[[1,3]] + k[[1,4]] + k[[1,6]] + k[[1,8]]);
In[16]: = k[[2,2]] = -k[[2,1]];
In[17]: = k[[3,3]] = k[[3,1]];
In[18]: = k[[4,4]] = -(k[[4,1]] + k[[4,5]]);
In[19]: = k[[5,5]] = -k[[5,4]];
In[20]: = k[[6,6]] = -(k[[6,1]] + k[[6,7]]);
In[21]: = k[[7,7]] = -k[[7,6]];
In[22]: = k = Transpose[k];
In[23]: = {a,v} = N[Eigensystem[k]];
In[24]: = v = Transpose[v];
In[25]: = m = N[Linear Solve[v,{1,0,0,0,0,0,0,0}]];
In[26]: = Do[
              Print[i,"     ",Scientific Form[N[a[[i]],16]]],
          {i,8}]

1       -2.461938031470023 10^1

2       -1.205214188565213

3       -8.801477920039391 10^-2

4       -6.86077129845387 10^-3
```

5 -1.897170849807361 10^{-3}

6 -1.463672932855431 10^{-18}

7 -3.701352885112255 10^{-4}

8 -7.864009739525252 10^{-5}

```
In[27]:= Do[Print[i,"  ",j,"  ",ScientificForm[N[v[[i,j]] m[[j]],16]]],
         {i,8},{j,8}]
```

1 1 9.536956919361768 10^{-1}

1 2 4.118313308135649 10^{-3}

1 3 4.1296453779843 10^{-2}

1 4 6.722626131951742 10^{-4}

1 5 1.448526759800047 10^{-4}

1 6 -3.666337889847799 10^{-18}

1 7 5.177815304595677 10^{-5}

1 8 2.064753362330491 10^{-5}

2 1 -4.586787875647112 10^{-1}

2 2 1.365448479563485 10^{-1}

2 3 3.156992443103672 10^{-1}

2 4 4.867212883715896 10^{-3}

2 5 1.045355793452642 10^{-3}

2 6 3.666337889847799 10^{-18}

2 7 3.73295888257354 10^{-4}

2 8 1.488307320012701 10^{-4}

3 1 -4.218758139432147 10^{-1}

3 2 -1.342035554920094 10^{-1}

3 3 5.454350009830602 10^{-1}

3 4 8.056661780811382 10^{-3}

3 5 1.726192502703665 10^{-3}

3 6 2.933070311878239 10^{-17}

3 7 6.159667021467238 10^{-4}

3 8 2.455474665021401 10^{-4}

4 1 -2.104139657185631 10^{-2}

4	2	$-1.867951965125432\ 10^{-3}$
4	3	$-2.804507626241224\ 10^{-1}$
4	4	$2.823386860421597\ 10^{-1}$
4	5	$1.199413511868621\ 10^{-2}$
4	6	$-2.493109765096503\ 10^{-16}$
4	7	$2.445089084096036\ 10^{-3}$
4	8	$6.582200916162441\ 10^{-3}$
5	1	$3.128108899621337\ 10^{-6}$
5	2	$5.673494842459835\ 10^{-6}$
5	3	$1.168734055706146\ 10^{-2}$
5	4	$-1.548853437397917\ 10^{-1}$
5	5	$-2.569914744730727\ 10^{-2}$
5	6	$-4.216288573324969\ 10^{-15}$
5	7	$-4.940520492359425\ 10^{-2}$
5	8	$2.18293553949894\ 10^{-1}$
6	1	$-1.650337369325964\ 10^{-2}$
6	2	$-1.457763069123594\ 10^{-3}$
6	3	$-2.038885163638646\ 10^{-1}$
6	4	$-5.670982715842137\ 10^{-2}$
6	5	$1.922274001504736\ 10^{-1}$
6	6	$-5.792813865959523\ 10^{-16}$
6	7	$7.735525531736283\ 10^{-2}$
6	8	$8.976824816833319\ 10^{-3}$
7	1	$4.055701850569236\ 10^{-7}$
7	2	$7.32293545241956\ 10^{-7}$
7	3	$1.415197559459882\ 10^{-3}$
7	4	$5.709893708165123\ 10^{-3}$
7	5	$-1.112713554080386\ 10^{-1}$
7	6	$-5.132873045786919\ 10^{-16}$
7	7	$9.712254985929754\ 10^{-2}$
7	8	$7.022576417386179\ 10^{-3}$
8	1	$-3.55998538422196\ 10^{-2}$

```
8    2    -3.140296526613515 10^-3
8    3    -4.311939582018047 10^-1
8    4    -9.004954612983411 10^-2
8    5    -7.016743338595035 10^-2
8    6    1.00000000000001
8    7    -1.285587300811811 10^-1
8    8    -2.412901818324071 10^-1

In[28]: = Quit
```

Barium Systemic Retention Function

The sum of the retention equations for all the systemic compartments gives the barium systemic retention function:

$$R_s(t) = 5.623 \cdot 10^{-1} e^{-54.6t} + 2.62 \cdot 10^{-1} e^{-6.21 \cdot 10^{-1} t} + 1.15 \cdot 10^{-1} e^{-2.38 \cdot 10^{-1} t} + 2.10 \cdot 10^{-2} e^{-7.92 \cdot 10^{-3} t} + 1.02 \cdot 10^{-2} e^{-1.82 \cdot 10^{-3} t} + 1.35 \cdot 10^{-2} e^{-5.05 \cdot 10^{-4} t} + 1.60 \cdot 10^{-2} e^{-1.29 \cdot 10^{-4} t}$$

Iron Mammillary Model

Johnson and Dunford[21] presented a mammillary model for iron, which is shown in Figure 7-3. The functions derived below are applicable for the first 120 days following intake, i.e., it is good until the red blood cells begin to recycle.

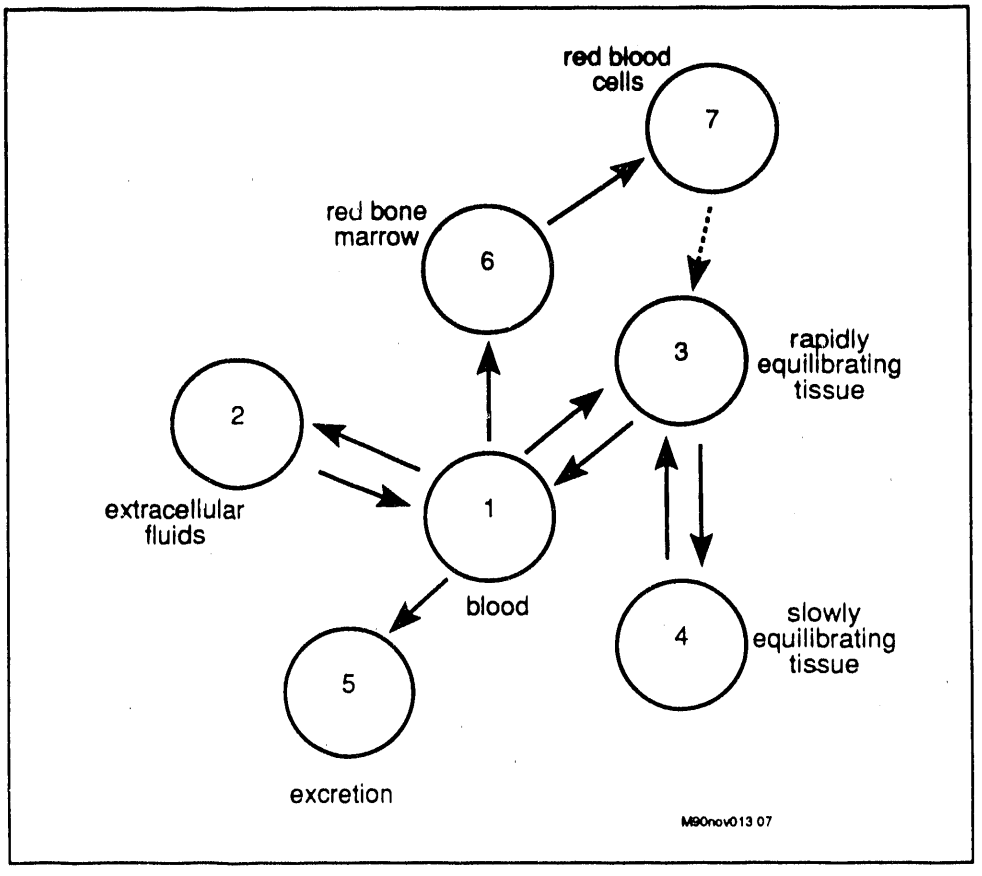


Figure 7-3. Johnson-Dunford Iron Mammillary Model

Mathematica Session

The Mathematica session for the iron model is given in this section. Refer to the Mathematica session for the iodine mammillary model for comments on the syntax of Mathematica.

```
In[1]:   = k = Table[0, {7}, {7}];
In[2]:   = k[[1,2]]  =  1.4;
In[3]:   = k[[2,1]]  =  1.91;
In[4][:  = k[[1,3]]  =  2.0;
In[5]:   = k[[3,1]]  =  0.33;
In[6]:   = k[[3,4]]  =  0.024;
In[7]:   = k[[4.3]]  =  0.0018;
In[8]:   = k[[1,6]]  =  6.0;
In[9]:   = k[[6,7]]  =  0.70
In[10]:  = k[[1,5]]  =  0.14;
In[11]:  = k[[1,1]]  =  -(k[[1,2]] + k[[1,3]] + k[[1,6]] + k[[1,8]]);
In[12]:  = k[[2,2]]  =  -k[[2,1]];
In[13]:  = k[[3,3]]  =  -k[[3,1]] + k[[3,4]];
In[14]:  = k[[4,4]]  =  -k[[4,3]];
In[15]:  = k[[6,6]]  =  -k[[6,7]];
In[16]:  = k Transpose[k];
In[17]:  = {a,v} = N[Eignesystem[k]];
In[18]:  = v = Transpose[v];
In[19]:  = m = N[Linear Solve[v,{1,0,0,0,0,0,0}]];
In[20]:  = Do[
```

```
                Print[i,"    ",Scientific Form[N[a[[i]],16]]],
            {i,7}]
```

```
1       -9.754259108374356
2       -7.860889516630082 10^-1
3       -7.000000000000002 10^-1
4       -2.638112104820467 10^-1
5       -2.927345865710862 10^-18
6       -1.640729480587243 10^-3
7       0. 10^-1
```

```
In[21]: = Do[
            Print[i,"    ",j,"  ",Scientific Form[N[v[[i,j]], m[[j]],16]]],
          {i.7},{j,7}]
```

```
1   1   9.767950182472379 10^-1
1   2   1.142705410692996 10^-2
1   3   -1.441569312207913 10^-16
1   4   1.177207616498809 10^-2
1   5   2.678966732075966 10^-17
1   6   5.851480844096667 10^-6
1   7   0. 10^-1
2   1   -1.546215470158803 10^-1
2   2   1.291077427268116 10^-1
2   3   -4.087962407301789 10^-16
2   4   2.550478575042723 10^-2
2   5   5.953259404613257 10^-17
2   6   9.018538641710429 10^-6
2   7   0. 10^-1
3   1   -2.078230966355052 10^-1
3   2   -5.289888086303156 10^-2
3   3   2.569040595414427 10^-16
3   4   2.605777149034104 10^-1
3   5   -5.953259404613258 10^-18
3   6   1.442625951260775 10^-4
```

3	7	$0.\ 10^{-1}$
4	1	$5.114355532102853\ 10^{-4}$
4	2	$1.618756885483023\ 10^{-3}$
4	3	$-6.270240504318971\ 10^{-18}$
4	4	$-2.386869304628616\ 10^{-2}$
4	5	$-4.390528810902277\ 10^{-17}$
4	6	$2.17385006075929\ 10^{-2}$
4	7	$0.\ 10^{-1}$
5	1	$-1.401965039427831\ 10^{-2}$
5	2	$-2.035122833854566\ 10^{-3}$
5	3	$1.048946775956164\ 10^{-17}$
5	4	$-6.247235134878751\ 10^{-3}$
5	5	$2.280130293159609\ 10^{-2}$
5	6	$-4.992945685844748\ 10^{-4}$
5	7	$-6.071532165918825\ 10^{-18}$
6	1	$-6.472942776800759\ 10^{-1}$
6	2	$-7.964125862510668\ 10^{-1}$
6	3	1.281725659060363
6	4	$1.619309314849355\ 10^{-1}$
6	5	$4.762607523690606\ 10^{-17}$
6	6	$5.027338584403829\ 10^{-5}$
6	7	$0.\ 10^{-1}$
7	1	$4.645211792529137\ 10^{-2}$
7	2	$7.091930362287282\ 10^{-1}$
7	3	-1.281725659060363
7	4	$-4.296695801225963\ 10^{-1}$
7	5	$9.77198697068404\ 10^{-1}$
7	6	$-2.144861203946439\ 10^{-2}$
7	7	$0.\ 10^{-1}$

In[22]: = Quit

Urinary Excretion Function

The solution for compartment 5 of the mammillary model is the cumulative urinary excretion function:

$$E_u(t) = -1.401 \cdot 10^{-2} e^{-9.754t} - 2.035 \cdot 10^{-3} e^{-7.861 \cdot 10^{-1}t}$$
$$- 6.247 \cdot 10^{-3} e^{-2.638 \cdot 10^{-1}t} + 2.280 \cdot 10^{-2}$$
$$- 4.993 \cdot 10^{-4} e^{-1.641 \cdot 10^{-3}t}$$

The derivative of the cumulative urinary excretion function with respect to time gives the urinary excretion function:

$$\dot{E}_u(t) = 1.368 \cdot 10^{-1} e^{-9.754t} + 1.600 \cdot 10^{-3} e^{-7.861 \cdot 10^{-1}t}$$
$$+ 1.648 \cdot 10^{-3} e^{-2.638 \cdot 10^{-1}t} + 8.192 \cdot 10^{-7} e^{-1.641 \cdot 10^{-3}t} \text{ day}^{-1}.$$

Americium Mammillary Model

Durbin and Schmidt[6] present a mammillary model for americium, which is shown in Figure 7-4.

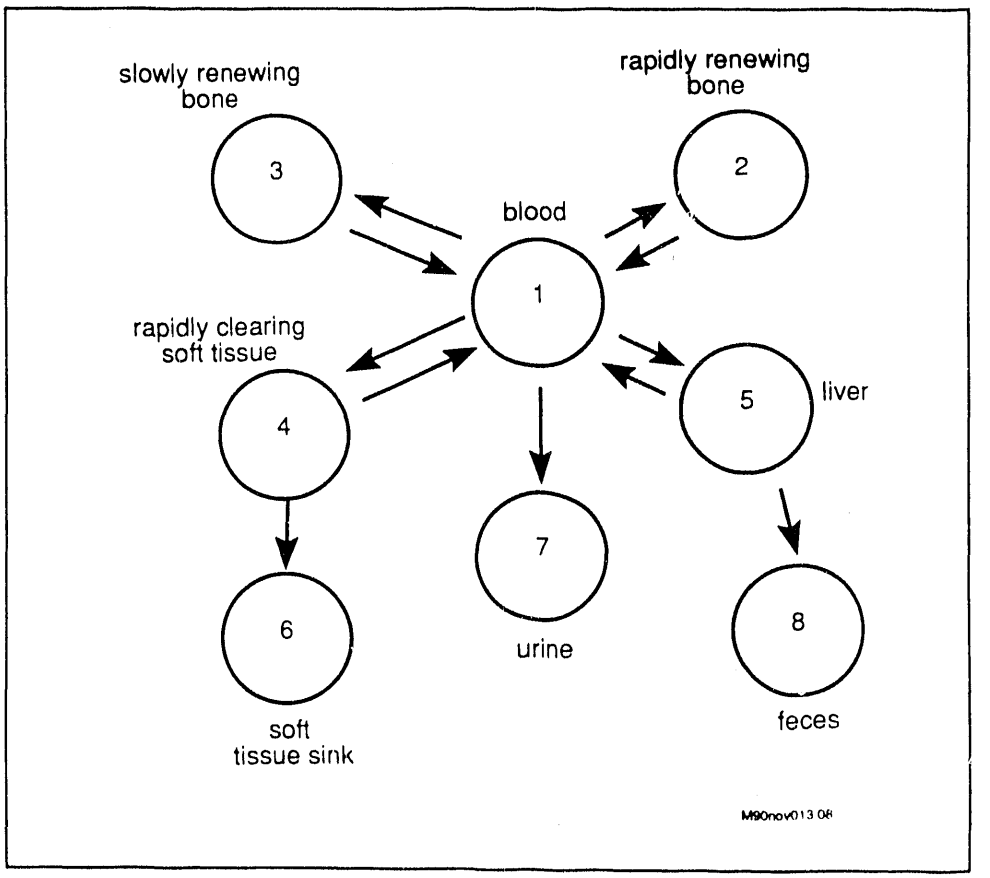


Figure 7-4. Durbin-Schmidt Americium Mammillary Model

Mathematica Session

The Mathematica session for Americium is given in this section. Refer to the Mathematica session for the Iodine Mammillary model for comments on the syntax of Mathematica.

```
In[1]:  = k = Table[0,{8},{8}];
In[2]:  = k[[1,2]]  =  0.13 Log[2.0] / 0.25;
In[3]:  = k[[2,1]]  =  0.112 / 365.25;
In[4]:  = k[[1,3]]  =  0.17 Log[2.0] / 0.25;
In[5]:  = k[[3,1]]  =  0.03 / 365.25;
In[6]:  = k[[1,4]]  =  0.12 Log[2.0] / 0.25;
In[7]:  = k[[4,1]]  =  2.0 / 365.25;
In[8]:  = k[[4,6]]  =  0.32/ 365.25;
In[9]:  = k[[1,5]]  =  0.53 Log[2.0] / 0.25;
In[10]: = k[[5,1]]  =  0.84 / 365.25;
In[11]: = k[[1,7]]  =  0.05 Log[2.0] / 0.25;
In[12]: = k[[5,8]]  =  0.019 / 365.25;
In[13]: = k[[1,1]]  =  -(k[[1,2]] + k[[1,3]] + k[[1,4]] + k[[1,5]] + k[[1,7]]);
In[14]: = k[[2,2]]  =  -k[[2,1]];
In[15]: = k[[3,3]]  =  -k[[3,1]];
In[16]: = k[[4,4]]  =  -(k[[4,1]] + k[[4,6]]);
In[17]: = k[[5,5]]  =  -(k[[5,1]] + k[[5,8]]);
In[18]: = k = transpose[k];
In[19]: = {a,v} = N[Eigensystem[k]];
In[20]: = v = transpose[v];
In[21]: = m = N[Linear Solve[v,{1,0,0,0,0,0,0,0}]];
In[20]: = Do[
              Print[i,"    ",Scientific Form[N[a[[i]],16]]],
          {i,8}]

1       -2.774519725500132

2       -5.866614344646408 10^-3

3       -1.058915284216627 10^-3

4       -2.13701839749644 10^-4

5       -2.216773510250578 10^-5

6       5.962379012454147 10^-21

7       0. 10^-1

8       0.10^-1


In[21]: = Do[
              Print[i,"    ",j,"  ",Scientific Form[N[v[[i,j]], m[[j]],16]]],
          {i,8},{j,8}]

1    1  9.993035890562133 10^-1

1    2  1.247201454082291 10^-4

1    3  4.303612387916671 10^-4

1    4  6.3206475554415 10^-5

1    5  7.812308403227863 10^-5
```

1	6	$1.929328324972102\ 10^{-18}$
1	7	$0.\ 10^{-1}$
1	8	$0.\ 10^{-1}$
2	1	$-1.298334016714778\ 10^{-1}$
2	2	$-8.085233559616434\ 10^{-3}$
2	3	$-2.061981427615035\ 10^{-1}$
2	4	$2.4513317887275834\ 10^{-1}$
2	5	$9.898498926501414\ 10^{-2}$
2	6	$6.559716304905148\ 10^{-17}$
2	7	$0.\ 10^{-1}$
2	8	$0.\ 10^{-1}$
3	1	$-1.697684021044624\ 10^{-1}$
3	2	$-1.016264480862075\ 10^{-2}$
3	3	$-2.076686169178496\ 10^{-1}$
3	4	$-2.264390027412326\ 10^{-1}$
3	5	$6.14038666572161\ 10^{-1}$
3	6	$8.10317896488283\ 10^{-17}$
3	7	$0.\ 10^{-1}$
3	8	$0.\ 10^{-1}$
4	1	$-1.201079392370603\ 10^{-1}$
4	2	$8.552301025364685\ 10^{-2}$
4	3	$2.70524297651374\ 10^{-2}$
4	4	$3.426048172788431\ 10^{-3}$
4	5	$4.106451045487636\ 10^{-3}$
4	6	$-9.646641624860511\ 10^{-19}$
4	7	$0.\ 10^{-1}$
4	8	$0.\ 10^{-1}$
5	1	$-5.297112993065769\ 10^{-1}$
5	2	$-5.214314820732513\ 10^{-2}$
5	3	$4.891364477466346\ 10^{-1}$
5	4	$4.344026322153541\ 10^{-2}$
5	5	$4.927773654573215\ 10^{-2}$

5	6	$2.701059654960943\ 10^{-17}$
5	7	$0.\ 10^{-1}$
5	8	$0.\ 10^{-1}$
6	1	$3.792657739888722\ 10^{-5}$
6	2	$-1.277189068503347\ 10^{-2}$
6	3	$-2.238230528323635\ 10^{-2}$
6	4	$-1.404575076768294\ 10^{-2}$
6	5	$-1.622949776362046\ 10^{-1}$
6	6	$2.114569977947585\ 10^{-1}$
6	7	$-2.862293735361732\ 10^{-17}$
6	8	$0.\ 10^{-1}$
7	1	$-.993040481288267\ 10^{-2}$
7	2	$-2.947165505352433\ 10^{-3}$
7	3	$-5.634136814096925\ 10^{-2}$
7	4	$-4.100235203387137\ 10^{-2}$
7	5	$-4.885550570070816\ 10^{-1}$
7	6	$6.387763475049997\ 10^{-1}$
7	7	$0.\ 10^{-1}$
7	8	$0.\ 10^{-1}$
8	1	$9.931498847952596\ 10^{-6}$
8	2	$4.623523668931287\ 10^{-4}$
8	3	$-2.402880564700496\ 10^{-2}$
8	4	$-1.057420104983325\ 10^{-2}$
8	5	$-1.156359318691445\ 10^{-1}$
8	6	$1.497666547002409\ 10^{-1}$
8	7	$0.\ 10^{-1}$
8	8	$-1.040834085586084\ 10^{-17}$

In[24]: = Quit

Urinary Excretion Function

The solution for compartment 7 is the cumulative urinary excretion function:

$$E_u = -4.993 \cdot 10^{-2} e^{-2.775t} - 2.947 \cdot 10^{-3} e^{-5.867 \cdot 10^{-3} t}$$
$$- 5.634 \cdot 10^{-2} e^{-1.059 \cdot 10^{-3} t} - 4.100 \cdot 10^{-2} e^{-2.137 \cdot 10^{-4} t}$$
$$- 4.886 \cdot 10^{-1} e^{-2.217 \cdot 10^{-5} t} + 6.388 \cdot 10^{-1}.$$

The derivative of this function with respect to time gives the instantaneous urinary excretion function:

$$E_u(t) = 1.385 \cdot 10^{-1} e^{-2.775t} + 1.729 \cdot 10^{-5} e^{-5.867 \cdot 10^{-3} t}$$
$$+ 5.966 \cdot 10^{-5} e^{-1.059 \cdot 10^{-3} t} + 8.762 \cdot 10^{-6} e^{-2.137 \cdot 10^{-4} t}$$
$$+ 1.083 \cdot 10^{-5} e^{-2.217 \cdot 10^{-5} t} \text{ day}^{-1}.$$

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Chapter 8

Bioassay Methods

Chapter 8 Preview

- Elements Found at SRS

8

Bioassay Methods

The radiobioassay methods used at SRS to detect and quantify radioactive material in the human body and excreta are reviewed in this chapter. Unless otherwise noted the radioactive decay information is taken from Kocher[1]. The minimum detectable amount (MDA) quoted for each radiobioassay method is that quantity of material that has a 5% chance of not being detected under the assay conditions indicated. The MDA is a representative quantity used to design internal dosimetry programs. The MDA should not be used to decide if a specific radiobioassay has detected activity or not.

Whole-body counts are performed with a Canberra FastScan which uses two large NaI detectors. The routine count time for the whole body count is 2 minutes.

Chest counts for low energy photon emitters are performed with a pair of thin NaI-CsI phoswich detectors or a planar germanium detector array. The routine count time for a chest count is 30 minutes.

Silicon surface barrier detectors are used for both gross alpha and alpha spectrometry measurements of alpha emitters in urine. The typical count time is 1000 minutes.

Tritium in urine is quantified by liquid scintillation counting of a 0.5 ml sample for 1 minute.

Fecal samples are processed by an off-site vendor. The MDA for alpha emitting isotopes of thorium, uranium, neptunium, plutonium, americium, and curium is 0.1 pCi per sample. The MDA for Sr-89 and Sr-90 in feces is 20 pCi per sample.

Californium-252

Cf-252 decays with a halflife of 2.6 years[2], 96.9% of the time by alpha emission to Cm-248 and 3.1% of the time by spontaneous fission. Because of the difference in halflife between Cf-252 and Cm-248 there is no significant in-growth of any radioactive daughters. The two primary alpha particles emitted are

6076 keV	15.2%
6118 keV	81.5%

The following low-energy photons[2] may be useful for radiobioassay:

14.93 keV	2.39%
19.22 keV	3.24%
23.0 keV	0.76%

Gross alpha counting is used to quantify Cf-252 in urine; the method has an MDA of 0.2 pCi/L. Because gross alpha counting is used and californium, curium, and americium have similar chemistry, these three radionuclides are not differentiated from one another; therefore, the results of urine bioassay should be thought of as Cf/Cm/Am even through they are reported as Am. Chest counting with germanium detectors has an MDA of 30 nCi based on the 19.2 keV photon and a chest-wall thickness of 26 mm. Low levels of short-lived fission products from the fission of Cf-252 may be detectable with the chest counter. Therefore, any unusual peaks in the germanium counter spectrum should be investigated.

Curium-242

Cm-242 has a halflife of 163 days[2] and decays to Pu-238 by alpha emission. Because of the difference in the halflives of Cm-242 and Pu-238, there is no significant in-growth of any radioactive daughters. The two alphas emitted are

6070 keV	25.9%
6113 keV	74.1%

Cm-242 emits the following low-energy photons[2], which are useful for radiobioassay:

12.1 keV	0.2%
14.3 keV	4.04%
16.3 keV	0.1%
18.1 keV	5.21%
21.6 keV	1.18%
44.1 keV	0.032%

Gross alpha counting is used to quantify Cm-242 in urine; the method has an MDA of 0.2 pCi/L. Because gross alpha counting is used and californium, curium, and americium have similar chemistry, these three radionuclides are not differentiated from one another, therefore, the results of urine bioassay should be thought of as Cf/Cm/Am even through they are reported as Am. Cm-242 is measured in-vivo by chest counting. Germanium detectors have an MDA of 27 nCi and the phoswich detectors have an MDA of 23 nCi. The MDA for the germanium detectors is based on the 21.6 keV photon whereas the MDA for the phoswich detectors is based on the 18.1 keV photon. The MDAs for both methods are based on a chest-wall thickness of 26 mm.

Curium-244

Cm-244 has a halflife of 18.1 years[2] and decays to Pu-240 by alpha emission. Because of the difference in halflife between Cm-244 and Pu-240 there is no significant in-growth of radioactive daughters. The two alphas emitted are

5763 keV	23.6%
5805 keV	76.4%

Cm-244 emits the following low-energy photons[2], which are useful for radiobioassay

12.1 keV	0.193%
14.3 keV	3.77%
16.3 keV	0.0976%
18.1 keV	4.83%
21.6 keV	1.09%
42.8 keV	0.0260%

Gross alpha counting is used to quantify Cm-244 in urine; the method has an MDA of 0.2 pCi/L. Because gross alpha counting is used and californium, curium, and americium have similar chemistry, these three radionuclides are not differentiated from one another; therefore the results of urine bioassay should be thought of as Cf/Cm/Am even through they are reported as Am. Cm-244 is measured in-vivo by chest counting. Germanium detectors have an MDA of 29 nCi and the phoswich detectors have an MDA of 25 nCi. The MDA for the germanium detectors is based on the 21.6 keV photon whereas the MDA for the phoswich detectors is based on the 18.1 keV photon. The MDAs for both methods are based on a chest-wall thickness of 26 mm.

Americium-241

Am-241 decays with a halflife of 432 years[2] to Np-237 by alpha emission. Because of the difference in halflife between Am-241 and Np-237 there is no significant in-growth of any radioactive daughters.The two primary alpha particles emitted are

5443 keV 12.8%
5486 keV 85.2%

Am-241 emits the following low-energy photons[3,4], which are useful for radiobioassay

11.9 keV 0.808%
13.9 keV 13.2%
15.9 keV 0.505%
17.7 keV 19.5%
20.9 keV 4.8%
26.4 keV 2.4%
59.5 keV 35.9%

Gross alpha counting is used to quantify Am-241 in urine; the method has an MDA of 0.2 pCi/L. Because gross alpha counting is used and californium, curium, and americium have similar chemistry, these three radionuclides are not differentiated from one another; therefore, the results of urine bioassay should be thought of as Cf/Cm/Am even through they are reported as Am. Am-241 is measured in-vivo by chest counting. Germanium detectors have an MDA of 0.13 nCi and the phoswich detectors have an MDA of 0.16 nCi, both of which are measure the 59 keV photon assuming a chest-wall thickness of 26 mm.

Plutonium-238

Pu-238 has a halflife of 87.7 years[2] and decays to U-234 by alpha emission. Because of the difference in halflife between Pu-238 and U-234 there is no significant in-growth of radioactive daughters. The two primary alphas emitted are

5456 keV	28.3%
5499 keV	71.6%

Pu-238 emits the following low-energy photons[2] which are useful for radiobioassay

11.6 keV	0.196%
13.6 keV	3.97%
15.4 keV	0.114%
17.1 keV	5.57%
20.3 keV	1.28%
43.5 keV	0.0389%

The electrodeposition method followed by alpha spectrometry is used to quantify Pu-238 in urine; the method has an MDA of 0.02 pCi/L. Pu-238 is measured in-vivo by chest counting. Germanium detectors have an MDA of 43 nCi and the phoswich detectors have an MDA of 20 nCi. The MDA for the germanium detectors is based on the 20.3 keV photon whereas the MDA for the phoswich detectors is based on the 17.1 keV photon. The MDAs for both methods are based on a chest-wall thickness of 26 mm.

Plutonium-239

Pu-239 has a halflife of 24065 years[2] and decays to U-235 by alpha emission. Because of the difference in halflife between Pu-239 and U-235, there is no significant in-growth of radioactive daughters. The three primary alphas emitted are

5105 keV	10.7%
5143 keV	15.2%
5156 keV	73.8%

Pu-239 emits the following low-energy photons[2] which are useful for radiobioassay

11.6 keV	0.073%
13.6 keV	1.48%
15.4 keV	0.042%
17.1 keV	2.09%
20.3 keV	0.486%
51.6 keV	0.0208%

The electrodeposition method followed by alpha spectrometry is used to quantify Pu-239 in urine. Alpha spectrometry can not differentiate between Pu-239 and Pu-240. The results of urine bioassay are reported as Pu-239, which includes any Pu-240 that may be present. The method has an MDA of 0.02 pCi/L. Pu-239 is measured in-vivo by chest counting. Germanium detectors have an MDA of 110 nCi and the phoswich detectors have an MDA of 40 nCi. The MDA for the germanium detectors is based on the 20.3 keV photon whereas the MDA for the phoswich detectors is based on the 17.1 keV photon. The MDAs for both methods are based on a chest- wall thickness of 26 mm.

There are two typical types of Pu-239 processed at SRS which may be described by their Pu-240 mass fraction: 6% Pu and 12% Pu. The MDA for chest counting of 6% Pu is 0.75 times the MDA for pure Pu-239 based on either the 20.3 keV or 17.1 keV x-rays. The MDA for chest counting of 12% Pu is 0.5 times the MDA for pure Pu-239 based on either the 20.3 keV or 17.1 keV x-ray.

Plutonium-240

Pu-240 has a halflife of 6537 years[2] and decays to U-236 by alpha emission. Because of the difference in halflife between Pu-240 and U-236, there is no significant in-growth of radioactive daughters. The two primary alphas emitted are

5124 keV	26.5%
5168 keV	73.4%

Pu-240 emits the following low-energy photons[2] which are useful for radiobioassay:

11.6 keV	0.186%
13.6 keV	3.77%
15.4 keV	0.109%
17.1 keV	5.32%
20.3 keV	1.22%
51.6 keV	0.045%

The electrodeposition method followed by alpha spectrometry is used to quantify Pu-240 in urine. Alpha spectrometry can not differentiate between Pu-239 and Pu-240; therefore, the results of urine bioassay are reported as Pu-239. The method has an MDA of 0.02 pCi/L. Pu-240 is measured in-vivo by chest counting. Germanium detectors have an MDA of 46 nCi and the phoswich detectors have an MDA of 22 nCi. The MDA for the germanium detectors is based on the 20.3 keV photon whereas the MDA for the phoswich detectors is based on the 17.1 keV photon. The MDAs for both methods are based on a chest-wall thickness of 26 mm.

Plutonium-241

Pu-241 decays with a halflife of 14.4 years[2] to Am-241 by negatron emission. No photons useful for radiobioassay are emitted. The beta particles emitted have a maximum energy of 21 keV. Currently no radiobioassay procedures are performed for Pu-241.

Neptunium-237

Np-237 has a halflife of $2.14 \cdot 10^6$ years[2] and decays to Pa-233 by alpha emission. Pa-233 has a halflife of 27 days and grows into equilibrium with Np-237. The two primary alphas emitted by Np-237 are

4772 keV	25.0%
4789 keV	47.1%

Np-237 emits the following low-energy photons[2] which are useful for radiobioassay:

11.4 keV	1.04%
13.3 keV	21.5%
16.6 keV	28.4%
19.7 keV	6.61%
29.4 keV	14.0%
86.5 keV	12.6%
92.3 keV	1.75%
95.9 keV	2.84%

In addition, Pa-233 emits the following photons[2]

13.6	17.3%
17.0	20.4%
20.4	5.07%
75.3	1.17%
86.6	1.76%
94.7	10.1%
98.4	16.3%
110.4	2.00%
111.3	3.80%
114.7	2.00%
300.1	6.19%
312.0	36.0%
340.5	4.21%
398.6	1.19%
415.8	1.51%

Gross alpha counting is used to quantify Np-237 in urine. The method has an MDA of 0.2 pCi/L. Np-237 is measured in-vivo by chest and whole body counting. For chest counting, the germanium detectors have an MDA of 0.35 nCi and the phoswich detectors have an MDA of 0.48 nCi, both of which are based on the 86.5 keV photon assuming a chest-wall thickness of 26 cm. The 312 keV photon from Pa-233 is used to measure Np-237 by whole body counting. The MDA for this procedure is 13 nCi of Pa-233, which in equilibrium represents 13 nCi of Np-237.

Uranium-234

U-234 decays with a halflife of $2.445 \cdot 10^5$ years[2] to Th-230 by alpha emission. Because of the halflife of Th-230 there is no significant in-growth of radioactive daughters. The two primary alphas emitted are

4721 keV	27.4%
4773 keV	72.3%

U-234 emits the following low-energy photons[2], which are useful for radiobioassay

11.1 keV	0.17%
13.0 keV	3.56%
14.5 keV	0.113%
16.1 keV	5.44%
19.1 keV	1.25%
53.2 keV	0.118%

Gross alpha counting is used to quantify U-234 in urine; the method has an MDA of 0.4 pCi/L. U-234 can be measured in-vivo by chest counting using the 53.2 keV photon, which gives an an MDA of 43 nCi. Lower detection levels are usually obtained by quantifying the U-234 indirectly based on the U-235 or U-238 present.

Uranium-235

U-235 decays with a halflife of $7.038 \cdot 10^8$ years[2] to Th-231 by alpha emission. Th-231 decays by negatron emission and with a halflife of 26 hours and grows rapidly into equilibrium with the U-235, but no other daughters grow in to a significant extent. The two primary alphas emitted by U-235 are

4366 keV	17.6%
4398 keV	56.0%

U-235 emits the following low-energy photons[2] which are useful for radiobioassay

144 keV	10.5%
186 keV	54.0%

The Th-231 does not emit any photons useful for radiobioassay.

Gross alpha counting is used to quantify U-235 in urine. The method has an MDA of 0.4 pCi/L. U-235 is measured in-vivo by chest and whole body counts. For chest counts, the germanium detectors have an MDA of 0.1 nCi and the thick phoswich detectors have an MDA of 0.2 nCi, both of which are based on the 186 keV photon. The 186 keV photon can also be detected by the whole body counter, which has an MDA of 13 nCi.

Uranium-236

U-236 decays with a halflife of $2.342 \cdot 10^7$ years[2] to Th-232 by alpha emission. Because of the halflife of Th-232 there is no significant in-growth of radioactive daughters. The two primary alphas emitted are

4470 keV	25.9%
4518 keV	73.8%

U-236 emits the following low-energy photons[2] which are useful for radiobioassay:

11.1 keV	0.16%
13.0 keV	3.36%
14.5 keV	0.105%
16.1 keV	5.08%
19.1 keV	1.16%

Gross alpha counting is used to quantify U-236 in urine; the method has an MDA of 0.4 pCi/L. U-236 can be measured in-vivo by chest counting using the 19.1 keV photon, which gives an an MDA of 91 nCi. Lower detection levels are usually obtained by quantifying the U-236 indirectly based on the U-235 or U-238 present.

Uranium-238

U-238 decays with a halflife of $4.468 \cdot 10^9$ years[2] to Th-234 by alpha emission. Th-234 decays by negatron emission and with a halflife of 24 days and grows into equilibrium with the U-238. The two primary alphas emitted by U-238 are

4149 keV	22.9%
4198 keV	76.8%

U-238 emits the following low-energy photons[2] which are useful for radiobioassay:

11.1 keV	0.141%
13.0 keV	2.96%
14.5 keV	0.0922%
16.1 keV	4.47%
19.1 keV	1.02%

In addition Th-234 emits the following photons:

11.4 keV	0.178%
13.3 keV	3.66%
16.5 keV	4.7%
19.8 keV	1.23%
63.3 keV	3.81%
92.3 keV	2.73%
92.8 keV	2.69%
113.0 keV	0.242%

Gross alpha counting is used to quantify U-238 in urine; the method has an MDA of 0.4 pCi/L. U-238 is measured in-vivo by chest counting. The germanium detectors have an MDA of 1.1 nCi and the phoswich detectors have an MDA of 2.0 nCi, both of which are based on the 63 keV photon of Th-234 assuming a 26 mm chest wall thickness. The MDA based on detection of the 19.1 keV photon of U-238 is 100 nCi.

Elemental Uranium

Elemental uranium is measured by kinetic phosphorimeter analysis with an MDA of 0.3 ug/L.

Thorium-228

Th-228 decays with a halflife of 1.91 years by alpha emission. A series of seven daughters, five alpha emitters and two beta emitters, grow into equilibrium with Th-228 within several weeks. The two primary alphas emitted by Th-228 (not including daughters) are

5341 keV 26.7%
5423 keV 72.7%

Th-228 emits the following low-energy photon which is useful for radiobioassay:

84.4 keV 1.21%

The most important daughter of Th-228 for in-vivo bioassay is Pb-212, which emits the following photon:

239 keV 45%

The Th-228 in urine is quantified by an off-site vendor. The MDA for the analysis is 0.1 pCi/L. Th-228 is measured in-vivo by chest counting. The germanium detectors have an MDA of 0.13 nCi for Pb-212 based on the 239 keV photon. Assuming secular equilibrium of Pb-212 with Th-228, the MDA for Th-228 would also be 0.13 nCi. The MDA based on the 84.4 keV photon of Th-228 and a 26 mm chest wall thickness is 3.4 nCi.

Thorium-232

Th-232 decays with a halflife of $1.405 \cdot 10^{10}$ years by alpha emission. A series of ten daughters, six alpha emitters and four beta emitters, will grow into equilibrium with Th-232 in two or three decades. The two primary alphas emitted by Th-232 (not including daughters) are

3953 keV 23.0%
4010 keV 77.0%

Th-232 emits the following low-energy photon which is useful for radiobioassay:

84.4 keV 1.21%

The most important daughter for in-vivo bioassay is Th-228, which was discussed in the previous section.

The Th-232 in urine is quantified by an off-site vendor. The MDA for the analysis is 0.1 pCi/L. Th-232 is measured in-vivo by chest counting. The MDA based on the 106 keV photon of Th-232 and a 26 mm chest wall thickness is 28 nCi. If the ratio of Th-228 to Th-232 is known, the Th-228 may be used to quantify the Th-232.

Europium-152

Eu-152 decays with a halflife of 13.6 years, 72% of the time by electron capture and positron emission and 28% of the time by negatron emission. Photons emitted by Eu-152 that are useful for radiobioassay are

1408 keV	20.7%
1112 keV	13.3%
1086 keV	10.0%
964 keV	14.4%
779 keV	12.7%
344 keV	26.5%
40.1 keV	37.7%
39.5 keV	20.8%

Eu-152 is measured in-vivo by chest and whole body counting. Whole body counting has an MDA of 16 nCi based on the 344 keV photon. The germanium chest counter has an MDA of 0.13 nCi based on the 40.1 keV photon.

Europium-154

Eu-154 decays with a halflife of 8.8 years by negatron emission. Photons emitted by Eu-154 that are useful for radiobioassay are

1274 keV	35.5%
1005 keV	17.9%
996 keV	10.3%
873 keV	11.5%
779 keV	12.7%
723 keV	19.7%

Eu-154 is measured in-vivo by whole body counting. The method has an MDA of 10 nCi based on the 1274 keV photon.

Cerium-141

Ce-141 decays with a halflife of 32.5 days by negatron emission. One photon emitted by Ce-141 that is useful for radiobioassay is

145 keV 48.4%

Ce-141 is measured in-vivo by chest and whole body counting. Whole body counting has an MDA of 14 nCi whereas chest counting has an MDA of 0.10 nCi.

Cerium-144

Ce-144 decays with a halflife of 284 days by negatron emission. One photon emitted by Ce-144 that is useful for radiobioassay is

134 keV 10.8%

Ce-144 is measured in-vivo by chest and whole body counting. Whole body counting has an MDA of 55 nCi whereas chest counting has an MDA of 0.43 nCi.

Barium-140

Ba-140 decays with a halflife of 12.8 days by negatron emission to the radioactive daughter La-140, which has a halflife of 1.7 days. One photon emitted by Ba-140 that is useful for radiobioassay is

537 keV 25.0%

La-140 also emits photons that may be useful for detecting and quantifying Ba-140.

Ba-140 is measured in-vivo by whole body counting. The method has an MDA of 17 nCi based on the 537 keV photon.

Lanthanum-140

La-140 decays with a halflife of 1.7 days by negatron emission. Photons emitted by La-140 that are useful for radiobioassay are

1596 keV	95.5%
815 keV	23.5%
487 keV	45.5%
329 keV	20.5%

La-140 is measured in-vivo by whole body counting. The method has an MDA of 3 nCi based on the 1596 keV photon.

Cesium-134

Cs-134 decays with a halflife of 2.1 years by negatron emission. Photons emitted by Cs-134 that are useful for radiobioassay are

796 keV	85.4%
605 keV	97.6%
569 keV	15.4%

Cs-134 is measured in-vivo by whole body counting. The method has an MDA of 4 nCi based on the 605 keV photon.

Cesium-137

Cs-137 decays with a halflife of 30 years by negatron emission with 94.6% of the decays producing the radioactive daughter Ba-137m, which has a halflife of 2.6 minutes. Cs-137 does not emit any photons useful for radiobioassay but Ba-137m emits one photon that may be used

662 keV 90.0%

Thus, Cs-137 appears to emit

662 keV 85.4%

taking into account the fraction of Cs-137 decaying to Ba-137m.

Cs-137 is measured in-vivo by whole body counting. The method has an MDA of 3 nCi based on the 662 keV photon.

Iodine-131

I-131 decays with a halflife of 8.04 days by negatron emission with 1.1% of the decays producing the radioactive daughter Xe-131, which has a halflife of 11.8 days. One photon emitted by I-131 that is useful for radiobioassay is

364 keV 81.2%

I-131 is measured in-vivo by whole body counting. The method has an MDA of 4 nCi based on the 364 keV photon.

Iodine-133

I-133 decays with a halflife of 21 hours by negatron emission to the radioactive daughter Xe-133, which has a halflife of 5.2 days. One photon emitted by I-133 that is useful for radiobioassay is

530 keV 86.3%

I-133 is measured in-vivo by whole body counting. The method has an MDA of 5 nCi based on the 530 keV photon.

Antimony-125

Sb-125 decays with a halflife of 2.8 years by negatron emission with 23.1% of the decays producing the radioactive daughter Te-125,which has a halflife of 58 days. Photons emitted by Sb-125 that are useful for radiobioassay are

636 keV	11.3%
601 keV	17.8%
463 keV	10.3%
428 keV	29.3%

Sb-125 is measured in-vivo by whole body counting. The method has an MDA of 13 nCi based on the 428 keV photon.

Ruthenium-103

Ru-103 decays with a halflife of 39.4 days by negatron emission with 99.7% of the decays producing the radioactive daughter Rh-103, which has a halflife of 56 minutes. One photon emitted by Ru-103 that is useful for radiobioassay is

497 keV 89.0%

Ru-103 is measured in-vivo by whole body counting. The method has an MDA of 5 nCi based on the 497 keV photon.

Ruthenium-106

Ru-106 decays with a halflife of 368 days by negatron emission to the radioactive daughter Rh-106, which has a halflife of 30 seconds. Ru-106 has no photons that are useful for radiobioassay, but Rh-106 emits

622 keV	9.9%
512 keV	20.6%

Ru-106 is measured in-vivo by whole body counting. The method has an MDA of 25 nCi based on the 622 keV photon.

Zirconium-95

Zr-95 decays with a halflife of 64 days by negatron emission to the radioactive daughter Nb-95, which has a halflife of 35 days. Photon emitted by Zr-95 that are useful for radiobioassay are

757 keV	55.3%
724 keV	43.7%

Zr-95 is measured in-vivo by whole body counting. The method has an MDA of 5 nCi based on the 757 keV photon.

Niobium-95

Nb-95 decays with a halflife of 35 days by negatron emission. One photon emitted by Nb-95 that is useful for radiobioassay is

766 keV 99.8%

Nb-95 is measured in-vivo by whole body counting. The method has an MDA of 2 nCi based on the 766 keV photon.

Strontium-89

Sr-89 decays with a halflife of 50.6 days by negatron emission. No photons useful for radiobioassay are emitted. The beta particles have a maximum energy of 1491 keV. Gas-flow proportional counting is used to quantify Sr-89 in urine. The method has an MDA of 6 pCi/L but does not differentiate between Sr-89 and Sr-90.

Strontium-90

Sr-90 decays with a halflife of 28.6 years by negatron emission to the radioactive daughter Y-90, which has a halflife of 64 hours. Neither nuclide emits photons that are useful for radiobioassay. Sr-90 emits beta particles with a maximum energy of 546 keV and Y-90 emits beta particles with a maximum energy of 2283 keV. Gas-flow proportional counting is used to quantify Sr-90 in urine. The method has an MDA of pCi/L but does not differentiate between Sr-89 and Sr-90.

Zinc-65

Zn-65 decays with a halflife of 244 days by electron capture. One photon emitted by Zn-65 that is useful for radiobioassay is

1116 keV 50.8%

Zn-65 is measured in-vivo by whole body counting. The method has an MDA of 4 nCi based on the 1116 keV photon.

Cobalt-58

Co-58 decays with a halflife of 71 days by electron capture. One photon emitted by Co-58 that is useful for radiobioassay is

811 keV 99.4%

Co-58 is measured in-vivo by whole body counting. The method has an MDA of 4 nCi based on the 811 keV photon.

Cobalt-60

Co-60 decays with a halflife of 5.3 years by negatron emission. Photons emitted by Co-60 that are useful for radiobioassay are

1173 keV	100.0%
1332 keV	100.0%

Co-60 is measured in-vivo by whole body counting. The method has an MDA of 4 nCi based on the 1332 keV photon.

Iron-59

Fe-59 decays with a halflife of 44.6 days by negatron emission. Photons emitted by Co-58 that are useful for radiobioassay are

1099 keV 56.5%
1292 keV 43.2%

Fe-59 is measured in-vivo by whole body counting. The method has an MDA of 6 nCi based on the 1099 keV photon.

Manganese-54

Mn-54 decays with a halflife of 313 days by electron capture. One photon emitted by Mn-54 that is useful for radiobioassay is

835 keV 100.0%

Mn-54 is measured in-vivo by whole body counting. The method has an MDA of 4 nCi based on the 835 keV photon.

Tritium

Tritium decays with a halflife of 12.3 years by negatron emission. No photons useful for radiobioassay are emitted. The beta particles emitted have a maximum energy of 18.6 keV. Liquid scintillation counting is used to quantify tritium in the urine. The method has an MDA of 0.02 uCi/L.

Radionuclide	MDA for Urine Bioassay (pCi)	MDA for Whole-body Count (nCi)	MDA for Chest Count (nCi)
Cf-252	0.2	NA	30
Cm-242	0.2	NA	2
Cm-244	0.2	NA	29
Am-241	0.2	NA	0.13
Pu-238	0.02	NA	43
Pu-239	0.02	NA	110
Np-237	0.2	13	0.35
U-234	0.4	NA	43
U-235	0.4	13	0.10
U-236	0.4	NA	91
U-238	0.4	NA	1.1
Th-228	0.1	NA	3.4
Th-232	0.1	NA	28
Eu-15	NA	16	0.13
Eu-154	NA	10	NA
Ce-141	NA	14	0.10
Ce-144	NA	55	0.43
Ba-140	NA	17	NA
La-140	NA	3	NA
Cs-134	NA	4	NA
Cs-137	NA	3	NA
I-131	NA	4	NA
I-133	NA	5	NA
Sb-125	NA	13	NA
Ru-103	NA	5	NA
Ru-106	NA	25	NA
Zr-95	NA	5	NA
Nb-95	NA	2	NA
Sr-89	6	NA	NA
Sr-90	6	NA	NA
Zn-65	NA	4	NA
Co-58	NA	4	NA
Co-60	NA	4	NA
Fe-59	NA	6	NA
Mn-54	NA	4	NA
H-3	2000	NA	NA

NA = not applicable, method not used.

References

(1) *Radioactive Decay Data Tables.* DOE/TIC-11026. 1981.

(2) *Radionuclide Transformations.* ICRP Publication 38. 1983.

(3) Vaninbroukx, R., Denecke, B. "Determination of Gamma-Ray Emission Probabilities in The Decay of U-235 and Th-231". *Nuclear Instruments and Methods* (193) pp. 191-196. 1982.

(4) Dias, M.S., Renner,C., "Si(Li) Efficiency Curve for X-Ray Parallel Beam". *Nuclear Instruments and Methods* (193) pp. 91-93. 1982.

Chapter 9

Missed Dose Tables

Chapter 9 Preview

- Intake to Dose Conversion Factors
- Intake Retention Fraction Tables
- Missed Dose Tables
- Missed Dose for Intakes of Tritiated Water
- Chemical Toxicity of Uranium
- Missed Dose for Lapel Air Samplers
- Missed Dose for Chest Counts of Transuranics

9

Missed Dose Tables

For a given intake scenario and bioassay program, the missed dose is the maximum effective dose equivalent associated with a less-than-MDA bioassay result. These tables are used to design routine bioassay programs needed to comply with the design goals given in Chapter 2 and should not be used for any other purpose. The tables were calculated with the following information:

- the Minimum detectable amounts (MDA) for all bioassay techniques used (Chapter 8)
- intake to dose conversion factors (DCF)
- intake retention fraction (IRF) tables

Intake to Dose Conversion Factors

Intake to effective dose equivalent conversion factors are presented in Table 9-1 and 9-2 for two time intervals:

- 50 years following the intake, that is, the committed effective dose equivalent
- 12 months following the intake, that is, the 12-month effective dose equivalent.

Intake Retention Fraction Tables

The intake retention fraction (IRF) is the fraction of an acute inhalation intake I expected to be present in bioassay compartment B at time T following the intake:

$$IRF = q/I,$$

where q is the quantity of material in B at time T. The IRF tables given in Tables 9-3 through 9-7 were generated with INDOS[4] for stable elements (radioactive decay was accounted for in the calculation of the missed dose tables).

Missed Dose Tables

Given

- the intake retention fraction IRF for time T and bioassay type B,
- the minimum detectable amount MDA for B in nCi, and
- the dose conversion factor DCF for the intake I in mrem/nCi,

the missed dose D in mrem is given by

$$D = DCF \cdot MDA / IRF \quad \text{mrem.}$$

The values for missed committed and 12-month effective dose equivalent are given in Tables 9-8 through 9-15 for pure radionuclides for the indicated bioassay period.

The missed dose for a bioassay period other than 365 days is determined by dividing the missed dose by the fraction of a year the bioassay period represents. If this dose is lower than the 365 day missed dose then it is used; if it is not, then the 365 day missed dose is used. For example, referring to Table 9-10, the missed committed dose for a 180 day whole-body count for Eu-154 is 47 mrem. The missed dose for the year is thus 47 mrem/0.5 = 94 mrem. The missed dose for an annual whole-body count is 55 mrem, and because it is less than 94 mrem it is used as the missed dose for biannual whole-body counts.

The missed dose for a mixture of radionuclides may be calculated from the tables in the following manner. Assume, for example, that the missed committed effective dose equivalent is needed for annual chest counts for commercial-grade class Y enriched uranium, which has the following composition:

Isotope	Mass Fraction	Activity Fraction	MDA (nCi)	Missed Dose (mrem)
U-234	0.0003	0.82774	43	54000
U-235	0.0296	0.02825	0.10	120
U-238	0.9701	0.14401	1.1	1300

The radionuclide that has the most activity relative to its MDA will be detected at the lowest level. Assuming negligible radioactive decay and identical biological elimination rates:

Isotope	Activity Fraction / MDA
U-234	C.82774 / 43 = 0.019
U-235	0.02825 / 0.10 = 0.28
U-238	0.14401 / 1.1 = 0.13

Thus, U-235 will be detected in this mixture before any of the other uranium isotopes. The intake I of U-235 that would produce a lung content of 0.1 nCi (the MDA) one year after the intake is given by the ratio of the missed dose D and the dose conversion factor DCF

$$I = D / DCF = 120 \text{ mrem} / 120 \text{ mrem/nCi} = 1 \text{ nCi.}$$

The missed dose D_j for component j with activity fraction f_j is given by

$$D_j = (1 \text{ nCi})(f_j/f_{U-235})\ DCF_j \text{ mrem.}$$

Thus, the missed dose D_{U-234} for U-234 is

$$(1 \text{ nCi})(0.82774/0.02825)(130 \text{ mrem/nCi}) = 3800 \text{ mrem.}$$

The missed dose D_{U-238} for U-238 is

$$(1 \text{ nCi})(0.14401/0.02825)(120 \text{ mrem/nCi}) = 610 \text{ mrem.}$$

The missed dose for the uranium mixture is the sum of the missed dose for its components, 4530 mrem.

Missed Dose For Intakes of Tritiated Water

Workers frequently have chronic or repeated intakes of tritiated water. Missed dose tables for such intakes are difficult to generate and read. Therefore, the missed dose for tritium is presented as the concentration of tritium in urine at various times following intakes in a year that will deliver a committed effective dose equivalent in of 10 mrem (approximately 170 μCi of tritiated water). Four intake patterns are presented:

- an intake of 170/365 μCi at the beginning of every day for a year (Figure 9-1)
- an intake of 170/52 μCi at the beginning of every week for a year (Figure 9-2)
- an intake of 170/12 μCi at the beginning of every month for a year (Figure 9-3)
- an intake of 170/4 μCi at the beginning of every three months for a year (Figure 9-4)

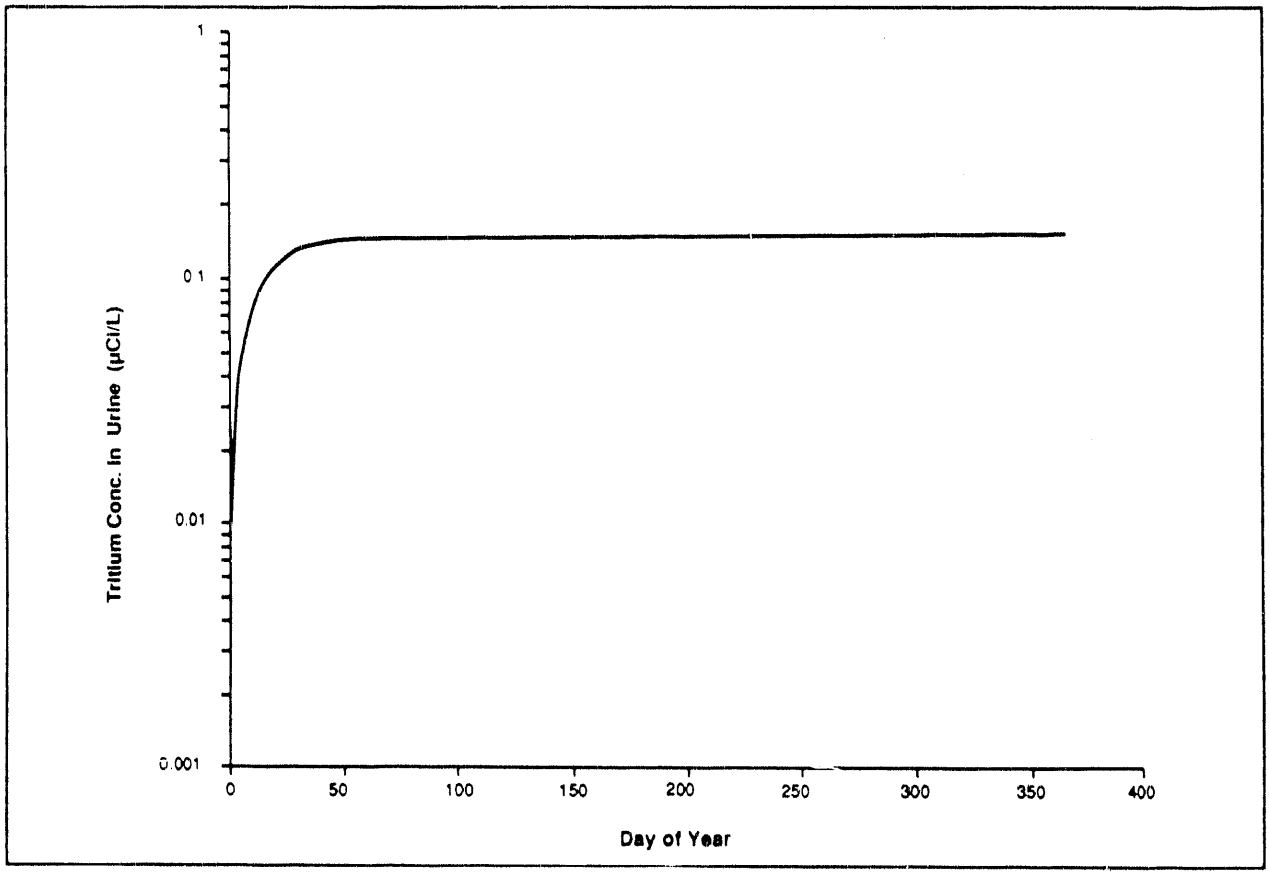


Figure 9-1. Intake of 170/365 μCi Tritiated Water at the Beginning of Every Day for a Year

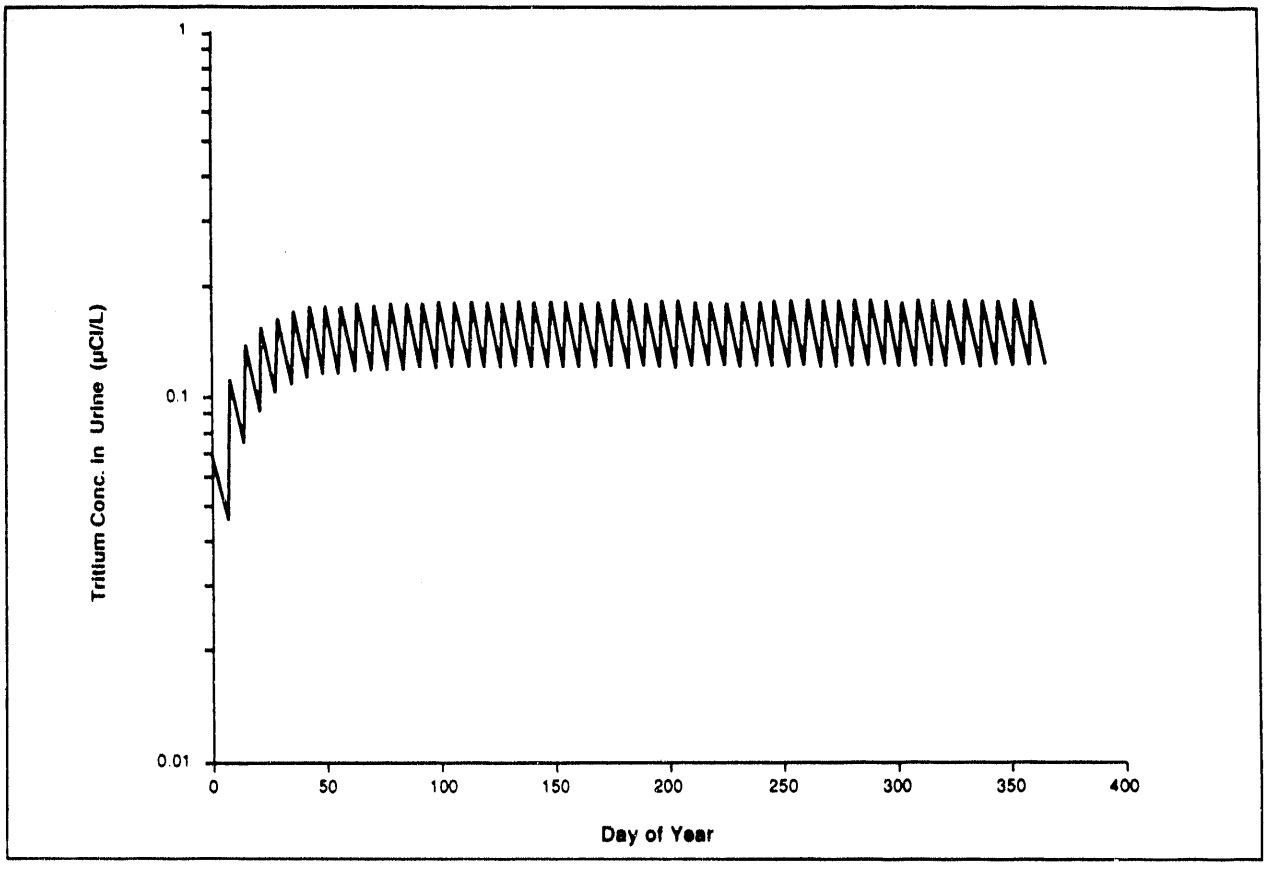


Figure 9-2. Intake of 170/52 µCi Tritiated Water at the Beginning of Every Week for a Year

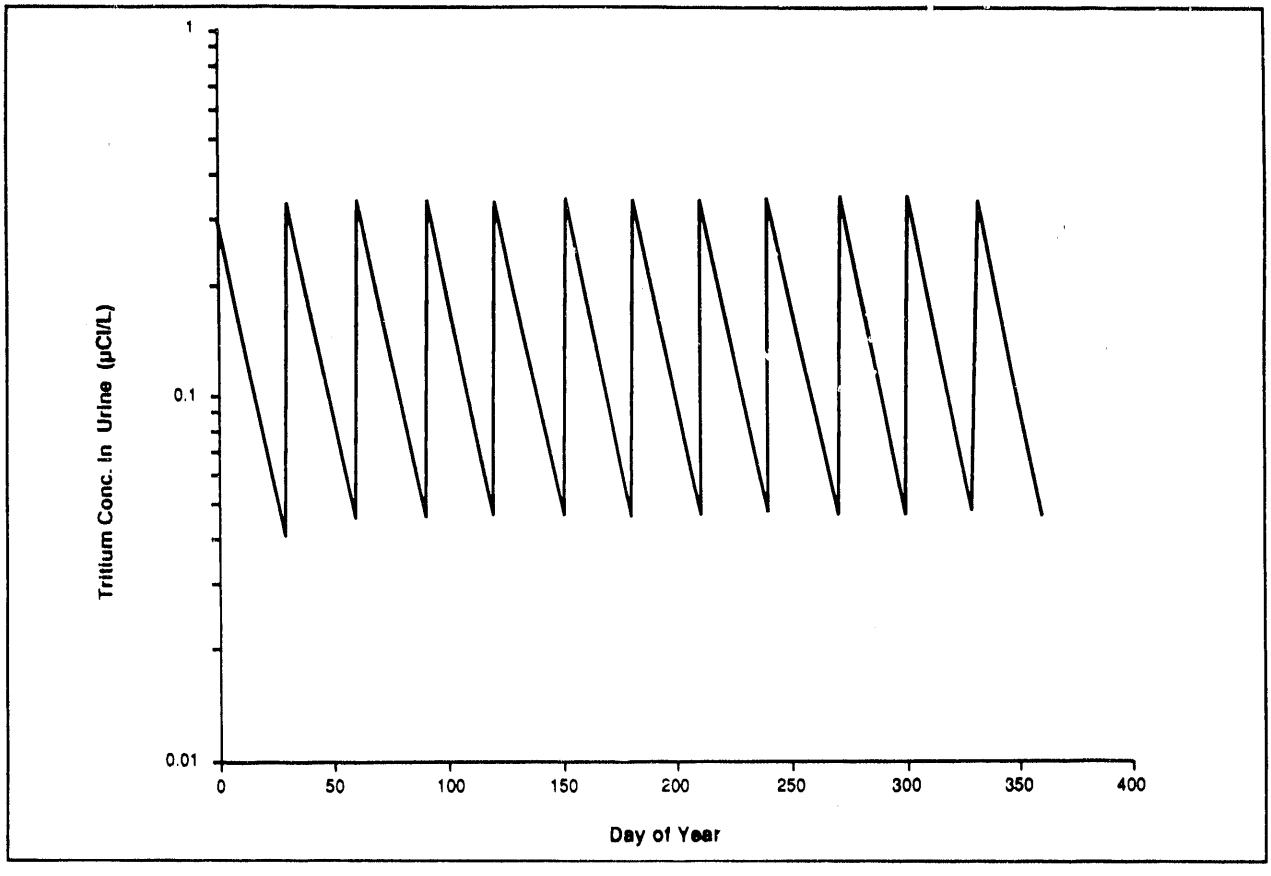


Figure 9-3. Intake of 170/12 μCi Tritiated Water at the Beginning of Every Month for a Year

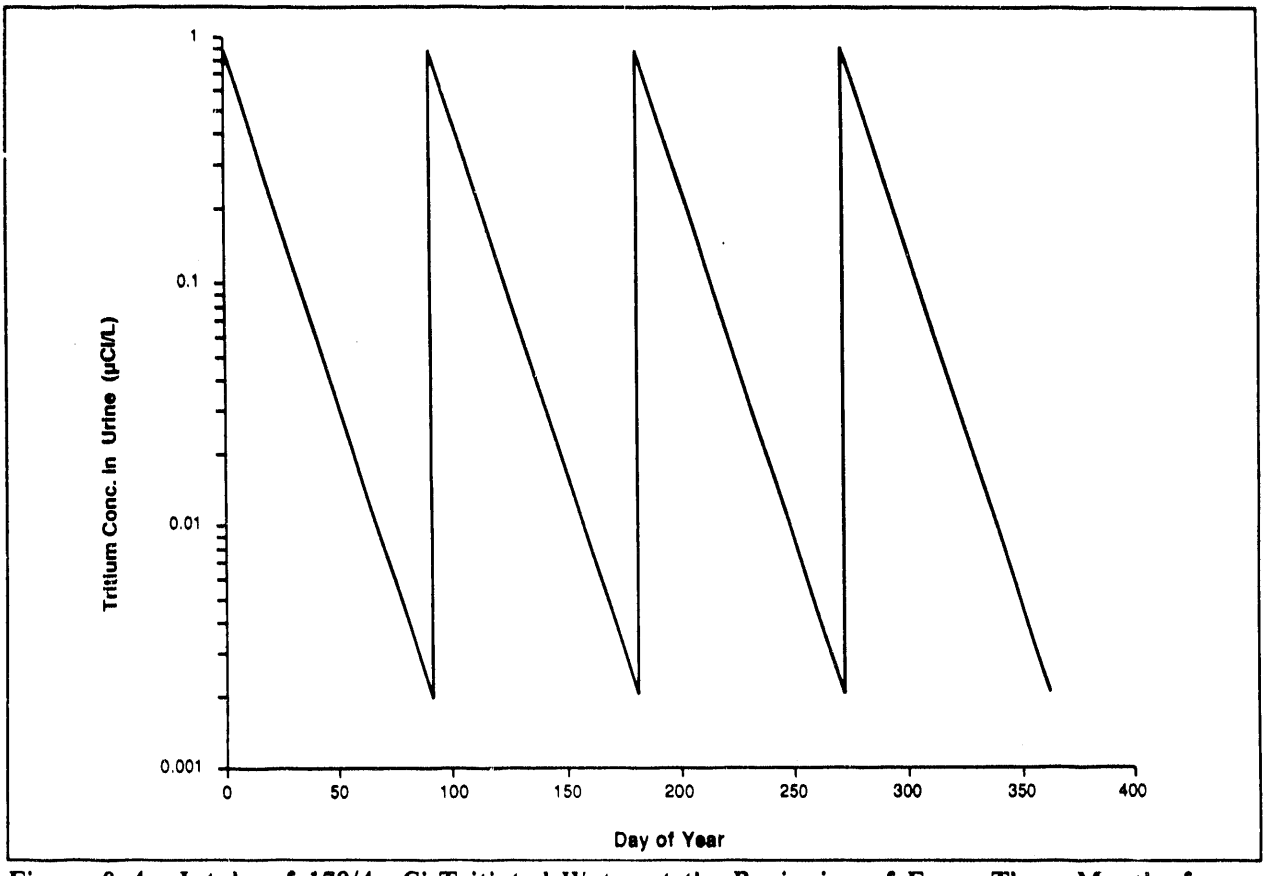


Figure 9-4. Intake of 170/4 µCi Tritiated Water at the Beginning of Every Three Months for a Year

Chemical Toxicity of Uranium

Chemical toxicity is a concern for uranium because of the large quantity of low-specific activity material processed at SRS. Assuming standard ICRP 30 biokinetic models, the intakes of uranium that would cause an individual to exceed the 900 μg nephrotoxic limit[5] are

Class	Intake (mg)
D	20
W	99
Y	1900

The following table gives ICRP 30 stocastic ALIs for uranium isotopes in terms of activity and mass:

	class D		class W		class Y	
Isotope	(nCi)	(mg)	(nCi)	(mg)	(nCi)	(mg)
U-234	1900	0.30	700	0.11	38	6.1E-3
U-235	2000	940	760	350	41	19
U-236	2000	30	760	12	41	0.63
U-238	2000	6300	7800	2300	43	130

The following table gives the concentration of uranium in the urine at various times following a 20 mg intake of class D uranium and a 99 mg intake of class W uranium.

Time After Intake	Uranium Excreted in Urine After a	
	Class D Intake	Class W Intake
(days)	(μg/day)	(μg/day)
1.00	3900	4200
7.00	190	210
15.00	94	130
30.0	34	72
60.00	9.0	41
90.00	3.1	27
180.00	0.17	9.4
365.00	0.033	1.1

Missed Dose for Lapel Air Samplers

Lapel air samplers may be used for two purposes.

- The filter paper is counted at the end of each shift to determine if the individual was exposed to relatively high levels of radioactive material.
- The filter paper is held for two weeks (to allow the radon daughters to decay) and counted to quantify exposures to low levels of radioactive material.

The missed dose for the second case above is calculated assuming

- an MDA of 0.6 pCi for an alpha emitter on a filter paper (20 minute count on a gas-flow proportional counter)
- a sampling rate of 2 liters per minute
- a sampling period of 8 hours per day
- a breathing rate of 20 liters per minute
- a dose conversion factor (DCF) in mrem per nCi intake

The missed effective dose equivalent for a worker would be

$$(0.6 \text{ pCi})(20 \text{ l/m})/(2 \text{ l/m})(10^{-3} \text{nCi/pCi})\ \text{DCF},$$

which equals $6 \cdot 10^{-3}$ DCF mrem per day or $1.5 \cdot$DCF mrem per year, assuming 250 work days per year. Because the missed dose for a particular alpha emitter is simply $1.5 \cdot$DCF, tables of missed dose for lapel air samplers will not be presented. Lower missed doses may be achieved by combining filter papers and analyzing them by radiochemical methods.

Missed Dose for Chest Counts of Transuranics

Chest counts for transuranics have very high missed doses, so high in fact that they may seem to be of little use in a routine bioassay program. To a certain extent this is true, chest counting has its greatest usefulness in special bioassay. Routine chest counts are still performed for the following reasons:

- they monitor the buildup of long-lived radioactive materials in the chest
- they help ensure that the chest counting equipment is operating properly for when it is really needed, for example, for special chest counts
- they provide baseline information that can be useful for evaluating subsequent chest counts
- the missed dose tables may not accurately describe the ability of a chest count to detect radioactive material in the chest for a specific application. For example, the tables do not address the use of Am-241 as a tracer for Pu-239 or the possibility of a person having a very thin chest wall, which decreases the missed dose

The objective of this section is to point out that routine chest counts are useful but that the frequency at which they should be performed is largely a matter of professional judgement.

References

(1) *RADOSE Documentation* ESH-TBD-91-0004.

(2) Dunning, D.E., *Estimates of Internal Dose Equivalents from Inhalation and Ingestion of Selected Radionuclides*. WIPP-DOE-176, Rev. 1, 1985.

(3) Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. EPA Federal Guidance Report Number 11, EPA-520/1-88-020, 1988.

(4) INDOS Documentation ESH-TBD-91-0001.

(5) N.L. Spoor and J.B. Hursh, *Uranium, Plutonium, Transplutonic Elements Chapters*. H.C. Hodge, J.N. Stannard, and J.B. Hursh, Editors Springer-Verlag, New York. 1973.

Table 9-1 Intake to dose conversion factors for thorium, uranium and transuranics. Please note that this table has the units of mrem/nCi.

Radionuclide		Committed mrem/nCi	12-month mrem/nCi	Reference
Cf-252	W	140	43	1
Cf-252	Y	160	62	1
Am-241	W	450	20	1
Cm-242	W	17	13	1
Cm-244	W	250	21	1
Pu-238	W	390	20	1
Pu-238	Y	290	32	1
Pu-239	W	420	18	1
Pu-239	Y	300	29	1
Pu-241	W	10	0.014	2
Pu-241	Y	5.8	0.026	2
Np-237	W	370	16	1
U-234	D	2.7	0.85	1
U-234	W	7.9	7.3	1
U-234	Y	130	27	1
U-235	D	2.5	0.79	1
U-235	W	7.3	6.7	1
U-235	Y	120	25	1
U-236	D	2.5	0.82	2
U-236	W	7.4	6.8	2
U-236	Y	130	26	2
U-238	D	2.4	0.76	1
U-238	W	7.0	6.4	1
U-238	Y	120	24	1
Th-228	W	250	100	2
Th-228	Y	340	160	2
Th-232	W	1600	16	2
Th-232	Y	1200	25	2

Table 9-2 Intake to dose conversion factors for fission and activation products. Please note that this table has the units of mrem/μCi.

Radionuclide		Committed mrem/μCi	12-month mrem/μCi	Reference
Eu-152	W	220	---	3
Eu-154	W	290	---	3
Ce-141	W	8.3	8.3	2
Ce-141	Y	9.0	8.9	2
Ce-144	W	220	160	2
Ce-144	Y	380	260	2
Ba-140	D	3.7	3.7	2
La-140	D	3.4	3.4	2
La-140	W	4.8	4.8	2
Cs-134	D	46	43	2
Cs-137	D	32	29	2
I-131	D	33	33	2
I-133	D	5.9	5.9	2
Sb-125	D	2.1	---	3
Sb-125	W	12	---	3
Ru-103	D	3.1	3.1	2
Ru-103	W	6.5	6.5	2
Ru-103	Y	9.0	9.0	2
Ru-106	D	57	40	2
Ru-106	W	120	110	2
Ru-106	Y	480	310	2
Zr-95	D	24	23	2
Zr-95	W	16	16	2
Zr-95	Y	23	23	2
Nb-95	W	4.8	4.8	2
Nb-95	Y	5.8	5.8	2
Sr-89	D	6.6	6.5	2
Sr-89	Y	41	41	2
Sr-90	D	220	35	2
Sr-90	Y	1300	320	2
Zn-65	Y	20	16	2
Co-58	W	6.4	6.3	2
Co-58	Y	11	11	2
Co-60	W	33	27	2
Co-60	Y	220	79	2
Fe-59	D	15	15	2
Fe-59	W	12	12	2
Mn-54	D	5.3	5.2	2
Mn-54	W	6.7	6.6	2
H-3	-	0.060	0.060	2

Table 9-3 Intake Retention Fractions.

Nuclide, class, and Bioassay Type		Intake Retention Fraction at Indicated Day after Intake							
		1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
BAD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BAD	WB	3.100E-01	4.431E-02	3.033E-02	2.734E-02	2.518E-02	2.345E-02	1.985E-02	1.618E-02
BAD	U	2.806E-02	5.637E-04	5.869E-05	9.155E-06	6.366E-06	5.234E-06	3.043E-06	1.303E-06
BAD	F	3.942E-02	5.927E-04	2.000E-07	6.117E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CEW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.768E-02	1.498E-02	1.356E-03
CEW	WB	5.889E-01	2.202E-01	2.035E-01	1.876E-01	1.639E-01	1.481E-01	1.252E-01	1.133E-01
CEW	U	1.372E-06	1.452E-06	1.524E-06	1.645E-06	1.836E-06	1.974E-06	2.178E-06	2.216E-06
CEW	F	4.112E-02	8.711E-03	1.190E-03	9.472E-04	6.249E-04	4.123E-04	1.184E-04	9.111E-06
CEU	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
CEY	WB	5.791E-01	1.604E-01	1.522E-01	1.501E-01	1.462E-01	1.424E-01	1.319E-01	1.139E-01
CEY	U	7.530E-08	8.021E-08	8.286E-08	8.782E-08	9.779E-08	1.078E-07	1.381E-07	1.998E-07
CEY	F	5.095E-02	8.047E-03	1.601E-04	1.334E-04	1.280E-04	1.228E-04	1.084E-04	8.385E-05
CFW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
CFW	WB	5.861E-01	2.134E-01	1.955E-01	1.780E-01	1.527E-01	1.358E-01	1.113E-01	9.987E-02
CFW	U	2.560E-03	1.431E-04	9.804E-05	6.525E-05	4.376E-05	3.481E-05	1.997E-05	9.468E-06
CFW	F	4.131E-02	8.805E-03	1.236E-03	9.769E-04	6.444E-04	4.278E-04	1.274E-04	1.338E-05
CFY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
CFY	WB	5.789E-01	1.603E-01	1.520E-01	1.499E-01	1.459E-01	1.420E-01	1.313E-01	1.130E-01
CFY	U	1.453E-04	8.094E-06	5.150E-06	3.338E-06	2.524E-06	2.433E-06	2.464E-06	2.613E-06
CFY	F	5.093E-02	8.047E-03	1.624E-04	1.349E-04	1.290E-04	1.238E-04	1.094E-04	8.496E-05
COW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
COW	WB	5.584E-01	1.779E-01	1.506E-01	1.245E-01	8.963E-02	6.599E-02	3.011E-02	1.274E-02
COW	U	1.595E-02	1.036E-03	5.503E-04	2.821E-04	1.773E-04	1.324E-04	5.574E-05	1.146E-05
COW	F	3.974E-02	8.229E-03	1.130E-03	9.000E-04	5.938E-04	3.918E-04	1.125E-04	8.657E-06
COY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
COY	WB	5.751E-01	1.655E-01	1.548E-01	1.508E-01	1.456E-01	1.409E-01	1.283E-01	1.069E-01
COY	U	2.832E-03	3.055E-04	1.236E-04	4.016E-05	2.011E-05	1.736E-05	1.378E-05	1.169E-05
COY	F	4.924E-02	7.594E-03	1.520E-04	1.268E-04	1.216E-04	1.167E-04	1.030E-04	7.968E-05
CSD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CSD	WB	6.135E-01	5.503E-01	5.175E-01	4.704E-01	3.894E-01	3.223E-01	1.828E-01	5.699E-02

U = urine bioassay C = chest count WB = whole body count F = feces bioassay

Note: CFW (californium class W) tables apply to class W americium and curium.

Table 9-4 Intake Retention Fractions

Nuclide, class, and Bioassay Type		Intake Retention Fraction at Indicated Day after Intake							
		1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CSD	U	1.322E-02	4.914E-03	2.750E-03	2.380E-03	1.969E-03	1.630E-03	9.245E-04	2.882E-04
CSD	F	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EUW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
EUW	WB	5.794E-01	2.085E-01	1.901E-01	1.720E-01	1.460E-01	1.287E-01	1.035E-01	9.116E-02
EUW	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
EUW	F	4.110E-02	8.704E-03	1.189E-03	9.466E-04	6.245E-04	4.120E-04	1.183E-04	9.104E-06
FED	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FED	WB	5.905E-01	4.906E-01	4.889E-01	4.864E-01	4.814E-01	4.764E-01	4.618E-01	4.331E-01
FED	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FED	F	3.942E-02	5.927E-04	2.000E-07	6.117E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FEW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
FEW	WB	5.917E-01	2.608E-01	2.456E-01	2.308E-01	2.087E-01	1.935E-01	1.698E-01	1.529E-01
FEW	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
FEW	F	3.831E-02	7.749E-03	1.071E-03	8.526E-04	5.625E-04	3.711E-04	1.066E-04	8.200E-06
ID	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ID	WB	3.096E-01	2.068E-01	2.026E-01	1.905E-01	1.627E-01	1.371E-01	8.144E-02	2.786E-02
ID	U	3.203E-01	4.789E-04	6.433E-04	8.928E-04	9.112E-04	7.918E-04	4.735E-04	1.620E-04
ID	F	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LAD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LAD	WB	5.873E-01	4.759E-01	4.748E-01	4.734E-01	4.706E-01	4.678E-01	4.595E-01	4.430E-01
LAD	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LAD	F	4.263E-02	6.709E-04	2.264E-07	6.924E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LAW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
LAW	WB	5.889E-01	2.205E-01	2.038E-01	1.879E-01	1.643E-01	1.485E-01	1.255E-01	1.136E-01
LAW	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LAW	F	4.110E-02	8.704E-03	1.189E-03	9.466E-04	6.245E-04	4.120E-04	1.183E-04	9.104E-06
MND	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MND	WB	4.972E-01	3.057E-01	2.638E-01	2.007E-01	1.161E-01	6.718E-02	1.301E-02	4.455E-04
MND	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00

U = urine bioassay C = chest count WB = whole body count F = feces bioassay

Table 9-5 Intake Retention Fractions

Nuclide, class, and Bioassay Type		Intake Retention Fraction at Indicated Day after Intake							
		1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
MND	F	3.942E-02	5.927E-04	2.000E-07	6.117E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MNW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
MNW	WB	5.690E-01	2.176E-01	1.913E-01	1.585E-01	1.090E-01	7.509E-02	2.461E-02	2.451E-03
MNW	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
MNW	F	3.831E-02	7.749E-03	1.071E-03	8.526E-04	5.625E-04	3.711E-04	1.066E-04	8.200E-06
NBW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
NBW	WB	5.852E-01	2.024E-01	1.735E-01	1.473E-01	1.134E-01	8.991E-02	5.038E-02	2.175E-02
NBW	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NBW	F	4.085E-02	8.616E-03	1.178E-03	9.380E-04	6.189E-04	4.083E-04	1.173E-04	9.022E-06
NBY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
NBY	WB	5.791E-01	1.626E-01	1.531E-01	1.500E-01	1.453E-01	1.409E-01	1.286E-01	1.071E-01
NBY	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
NBY	F	5.062E-02	7.958E-03	1.585E-04	1.321E-04	1.267E-04	1.216E-04	1.073E-04	8.304E-05
NPW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
NPW	WB	5.626E-01	1.909E-01	1.727E-01	1.545E-01	1.272E-01	1.089E-01	8.270E-02	7.186E-02
NPW	U	1.316E-02	9.930E-05	9.217E-05	7.997E-05	5.966E-05	4.411E-05	1.746E-05	3.459E-06
NPW	F	4.112E-02	8.713E-03	1.190E-03	9.474E-04	6.251E-04	4.124E-04	1.184E-04	9.113E-06
PUW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
PUW	WB	5.886E-01	2.189E-01	2.019E-01	1.856E-01	1.617E-01	1.458E-01	1.231E-01	1.131E-01
PUW	U	2.786E-04	3.384E-05	2.161E-05	1.743E-05	1.347E-05	1.156E-05	8.363E-06	4.876E-06
PUW	F	4.117E-02	8.757E-03	1.209E-03	9.625E-04	6.367E-04	4.223E-04	1.256E-04	1.328E-05
PUY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
PUY	WB	5.790E-01	1.602E-01	1.520E-01	1.499E-01	1.459E-01	1.421E-01	1.316E-01	1.137E-01
PUY	U	1.510E-05	1.738E-06	1.072E-06	8.653E-07	7.186E-07	6.920E-07	7.238E-07	8.021E-07
PUY	F	5.096E-02	8.052E-03	1.611E-04	1.342E-04	1.286E-04	1.234E-04	1.090E-04	8.456E-05
RUD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RUD	WB	5.286E-01	3.204E-01	2.527E-01	1.887E-01	1.384E-01	1.156E-01	8.956E-02	7.525E-02
RUD	U	4.826E-02	9.091E-03	5.279E-03	2.296E-03	8.362E-04	4.499E-04	1.137E-04	4.338E-05
RUD	F	4.106E-02	6.318E-04	2.132E-07	6.521E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RUW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
RUW	WB	5.769E-01	2.086E-01	1.786E-01	1.477E-01	1.079E-01	8.199E-02	4.346E-02	2.476E-02
RUW	U	1.071E-02	1.828E-03	1.177E-03	6.577E-04	3.678E-04	2.573E-04	1.004E-04	2.257E-05

U = urine bioassay C = chest count WB = whole body count F = feces bioassay

Table 9-6 Intake Retention Fractions.

Nuclide, class, and Bioassay Type		Intake Retention Fraction at Indicated Day After Intake							
		1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
RUW	F	3.974E-02	8.229E-03	1.130E-03	9.000E-04	5.938E-04	3.918E-04	1.125E-04	8.657E-06
RUY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
RUY	WB	5.783E-01	1.746E-01	1.628E-01	1.569E-01	1.498E-01	1.444E-01	1.313E-01	1.101E-01
RUY	U	1.979E-03	5.460E-04	3.184E-04	1.454E-04	6.199E-05	4.055E-05	2.206E-05	1.797E-05
RUY	F	4.924E-02	7.594E-03	1.520E-04	1.268E-04	1.216E-04	1.167E-04	1.030E-04	7.968E-05
SBD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SBD	WB	4.874E-01	1.707E-01	6.779E-02	2.227E-02	1.311E-02	1.057E-02	5.663E-03	1.571E-03
SBD	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SBD	F	3.942E-02	5.927E-04	2.000E-07	6.117E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SBW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
SBW	WB	5.687E-01	1.729E-01	1.387E-01	1.104E-01	7.558E-02	5.216E-02	1.735E-02	2.002E-03
SBW	U	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SBW	F	4.085E-02	8.616E-03	1.178E-03	9.380E-04	6.189E-04	4.083E-04	1.173E-04	9.022E-06
SRD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SRD	WB	4.983E-01	2.192E-01	1.371E-01	1.055E-01	9.631E-02	9.109E-02	8.010E-02	6.757E-02
SRD	U	8.451E-02	1.690E-02	4.717E-03	6.162E-04	1.668E-04	1.336E-04	8.148E-05	4.153E-05
SRD	F	3.230E-02	4.437E-04	1.497E-07	4.578E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SRY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
SRY	WB	5.778E-01	1.600E-01	1.504E-01	1.476E-01	1.431E-01	1.388E-01	1.269E-01	1.061E-01
SRY	U	1.305E-03	2.930E-04	9.080E-05	2.305E-05	1.592E-05	1.563E-05	1.530E-05	1.505E-05
SRY	F	5.062E-02	7.958E-03	1.585E-04	1.321E-04	1.267E-04	1.216E-04	1.073E-04	8.304E-05
THW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
THW	WB	5.836E-01	2.128E-01	1.957E-01	1.790E-01	1.543E-01	1.375E-01	1.130E-01	9.985E-02
THW	U	5.258E-03	6.953E-05	6.545E-05	6.069E-05	5.254E-05	4.607E-05	3.402E-05	2.503E-05
THW	F	4.112E-02	8.712E-03	1.190E-03	9.473E-04	6.250E-04	4.123E-04	1.184E-04	9.112E-06
THY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
THY	WB	5.788E-01	1.599E-01	1.517E-01	1.496E-01	1.456E-01	1.418E-01	1.311E-01	1.127E-01
THY	U	2.878E-04	2.895E-06	2.808E-06	2.875E-06	3.006E-06	3.133E-06	3.493E-06	4.130E-06
THY	F	5.095E-02	8.048E-03	1.601E-04	1.334E-04	1.280E-04	1.228E-04	1.084E-04	8.386E-05
UD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
UD	WB	3.944E-01	1.439E-01	9.178E-02	5.018E-02	2.393E-02	1.582E-02	1.151E-02	1.100E-02
UD	U	1.946E-01	9.538E-03	4.723E-03	1.698E-03	4.499E-04	1.554E-04	8.490E-06	1.674E-06

U = urine bioassay C = chest count WB = whole body count F = feces bioassay

Table 9-7 Intake Retention Fractions

Nuclide, class, and Bioassay Type		Intake Retention Fraction at Indicated Day After Intake							
		1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
UD	F	4.106E-02	6.318E-04	2.132E-07	6.521E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
UW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
UW	WB	5.476E-01	1.743E-01	1.457E-01	1.168E-01	7.892E-02	5.441E-02	1.937E-02	4.785E-03
UW	U	4.265E-02	2.158E-03	1.291E-03	7.239E-04	4.110E-04	2.751E-04	9.471E-05	1.093E-05
UW	F	3.974E-02	8.229E-03	1.130E-03	9.000E-04	5.938E-04	3.918E-04	1.125E-04	8.657E-06
UY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
UY	WB	5.768E-1	1.577E-01	1.490E-01	1.463E-01	1.416E-01	1.373E-01	1.253E-01	1.042E-01
UY	U	2.281E-03	1.043E-04	5.980E-05	3.239E-05	2.147E-05	1.908E-05	1.817E-05	1.833E-05
UY	F	5.089E-02	8.032E-03	1.598E-04	1.332E-04	1.278E-04	1.226E-04	1.082E-04	8.371E-05
ZNY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
ZNY	WB	5.987E-01	3.819E-01	3.641E-01	3.438E-01	3.187E-01	3.028E-01	2.695E-01	2.166E-01
ZNY	U	2.317E-04	4.924E-04	3.942E-04	2.665E-04	1.439E-04	9.855E-05	6.794E-05	5.249E-05
ZNY	F	3.035E-02	3.758E-03	7.932E-05	6.672E-05	6.400E-05	6.139E-05	5.419E-05	4.193E-05
ZRD	C	7.995E-02	4.487E-05	1.201E-09	2.020E-18	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZRD	WB	5.721E-01	3.634E-01	2.945E-01	2.504E-01	2.376E-01	2.363E-01	2.345E-01	2.307E-01
ZRD	U	5.515E-04	4.690E-04	2.129E-04	4.877E-05	3.202E-06	8.634E-07	7.314E-07	7.198E-07
ZRD	F	4.260E-02	6.700E-04	2.261E-07	6.916E-14	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ZRW	C	2.115E-01	1.386E-01	1.243E-01	1.027E-01	6.999E-02	4.767E-02	1.498E-02	1.356E-03
ZRW	WB	5.856E-01	2.028E-01	1.752E-01	1.503E-01	1.199E-01	1.003E-01	7.202E-02	5.988E-02
ZRW	U	1.182E-04	7.095E-05	3.685E-05	1.426E-05	6.315E-06	4.468E-06	1.785E-06	3.656E-07
ZRW	F	4.107E-02	8.694E-03	1.188E-03	9.456E-04	6.239E-04	4.116E-04	1.182E-04	9.095E-06
ZRY	C	2.141E-01	1.499E-01	1.477E-01	1.454E-01	1.410E-01	1.367E-01	1.247E-01	1.037E-01
ZRY	WB	5.789E-01	1.600E-01	1.511E-01	1.485E-01	1.442E-01	1.402E-01	1.290E-01	1.095E-01
ZRY	U	6.941E-06	4.704E-06	2.312E-06	7.777E-07	3.554E-07	3.376E-07	3.459E-07	3.549E-07
ZRY	F	5.089E-02	8.032E-03	1.598E-04	1.332E-04	1.278E-04	1.226E-04	1.082E-04	8.371E-05

U = urine bioassay C = chest count WB = whole body count F = feces bioassay

Table 9-8 Summary of missed committed dose for urine bioassay

Nuclide	Missed Committed Effective Dose Equivalent in mrem at Indicated Day After Intake							
	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W AM-241	35	630	920	1400	2100	2600	4500	9500
CLASS Y CF-252	220	4000	6300	9800	13000	14000	15000	16000
CLASS W CF-252	11	200	290	440	670	860	1600	3800
CLASS W CM-242	< 1	24	37	59	100	140	370	1700
CLASS W CM-244	20	350	510	770	1100	1400	2600	5500
CLASS W NP-237	6	750	800	930	1200	1700	4200	21000
CLASS W PU-238	28	230	360	450	580	680	940	1600
CLASS Y PU-238	380	3300	5400	6700	8100	8400	8000	7300
CLASS Y PU-239	400	3500	5600	6900	8300	8700	8300	7500
CLASS W PU-239	30	250	390	480	620	730	1000	1700
CLASS D SR-89	< 1	< 1	< 1	< 1	< 1	< 1	6	140
CLASS Y SR-89	< 1	< 1	3	16	35	54	190	2400
CLASS D SR-90	< 1	< 1	< 1	2	8	10	16	33
CLASS Y SR-90	6	27	86	340	490	500	520	530
CLASS Y TH-228	120	12000	12000	12000	12000	12000	12000	12000
CLASS W TH-228	5	360	390	420	510	590	880	1400
CLASS W TH-232	30	2300	2400	2600	3000	3500	4700	6400
CLASS Y TH-232	420	41000	43000	42000	40000	38000	34000	29000
CLASS D U-234	< 1	< 1	< 1	< 1	2	7	130	650
CLASS W U-234	< 1	< 1	2	4	8	11	33	290
CLASS Y U-234	23	500	870	1600	2400	2700	2900	2800
CLASS W U-235	< 1	< 1	2	4	7	11	31	270
CLASS Y U-235	21	460	800	1500	2200	2500	2600	2600
CLASS D U-235	< 1	< 1	< 1	< 1	2	6	120	600
CLASS W U-236	< 1	< 1	2	4	7	11	31	270
CLASS Y U-236	23	500	870	1600	2400	2700	2900	2800
CLASS D U-236	< 1	< 1	< 1	< 1	2	6	120	600
CLASS Y U-238	21	460	800	1500	2200	2500	2600	2600
CLASS W U-238	< 1	< 1	2	4	7	10	30	260
CLASS D U-238	< 1	< 1	< 1	< 1	2	6	110	570

Table 9-9 Summary of missed committed dose for chest counting

Nuclide	Missed Committed Effective Dose Equivalent in mrem at Indicated Day After Intake							
	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W AM-241	280	420	470	570	840	1200	3900	43000
CLASS W CE-141	< 1	< 1	< 1	< 1	< 1	< 1	3	1500
CLASS Y CE-141	< 1	< 1	< 1	< 1	< 1	< 1	< 1	21
CLASS W CE-144	< 1	< 1	< 1	< 1	2	2	10	170
CLASS Y CE-144	< 1	< 1	< 1	< 1	< 1	< 1	2	4
CLASS W CF-252	20000	30000	34000	42000	63000	94000	> 250000	> 250000
CLASS Y CF-252	22000	32000	33000	34000	36000	37000	44000	60000
CLASS W CM-242	2200	3400	3900	5100	8500	14000	66000	> 250000
CLASS W CM-244	34000	52000	58000	71000	100000	150000	> 250000	> 250000
CLASS W EU-152	< 1	< 1	< 1	< 1	< 1	< 1	2	22
CLASS W NP-237	610	930	1000	1300	1900	2700	8600	96000
CLASS Y PU-238	58000	83000	84000	86000	89000	91000	100000	120000
CLASS W PU-238	79000	120000	130000	160000	240000	> 250000	> 250000	> 250000
CLASS Y PU-239	150000	220000	220000	230000	230000	240000	> 250000	> 250000
CLASS W PU-239	220000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS W TH-228	4000	6200	6900	8500	13000	19000	68000	> 250000
CLASS Y TH-228	5400	7800	7900	8200	8700	9200	11000	16000
CLASS W TH-232	210000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS Y TH-232	73000	100000	110000	110000	110000	110000	130000	150000
CLASS D U-234	1500	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS Y U-234	26000	37000	38000	38000	40000	41000	45000	54000
CLASS W U-234	1600	2500	2700	3300	4900	7100	23000	> 250000
CLASS Y U-235	56	80	81	83	85	88	96	120
CLASS W U-235	3	5	6	7	10	15	49	540
CLASS D U-235	3	5600	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS Y U-236	55000	79000	80000	81000	84000	87000	95000	110000
CLASS W U-236	3200	4900	5400	6600	9600	14000	45000	> 250000
CLASS D U-236	2800	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS W U-238	36	56	62	75	110	160	510	5700
CLASS Y U-238	620	880	890	910	940	970	1100	1300
CLASS D U-238	33	59000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000

Table 9-10 Summary of missed committed dose for whole body counting

Nuclide	Missed Committed Effective Dose Equivalent in mrem at Indicated Day After Intake							
	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS D BA-140	< 1	2	5	12	64	350	54000	> 250000
CLASS Y CE-141	< 1	< 1	< 1	2	3	6	44	2700
CLASS W CE-141	< 1	< 1	< 1	< 1	3	5	43	2500
CLASS Y CE-144	36	130	140	150	170	180	250	450
CLASS W CE-144	21	56	62	69	85	100	150	260
CLASS W CO-58	< 1	< 1	< 1	< 1	< 1	< 1	5	72
CLASS Y CO-58	< 1	< 1	< 1	< 1	< 1	< 1	2	15
CLASS W CO-60	< 1	< 1	< 1	< 1	2	2	5	12
CLASS Y CO-60	2	5	6	6	6	6	7	9
CLASS D CS-134	< 1	< 1	< 1	< 1	< 1	< 1	< 1	5
CLASS D CS-137	< 1	< 1	< 1	< 1	< 1	< 1	< 1	2
CLASS W EU-152	6	17	19	21	24	28	35	41
CLASS W EU-154	8	22	24	27	32	37	47	55
CLASS D FE-59	< 1	< 1	< 1	< 1	< 1	< 1	3	60
CLASS W FE-59	< 1	< 1	< 1	< 1	< 1	2	7	140
CLASS D I-131	< 1	< 1	2	9	140	2300	> 250000	> 250000
CLASS D I-133	< 1	38	23000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS D LA-140	< 1	< 1	10	4400	> 250000	> 250000	> 250000	> 250000
CLASS W LA-140	< 1	< 1	32	16000	> 250000	> 250000	> 250000	> 250000
CLASS W MN-54	< 1	< 1	< 1	< 1	< 1	< 1	2	25
CLASS D MN-54	< 1	< 1	< 1	< 1	< 1	< 1	2	110
CLASS W NB-95	< 1	< 1	< 1	< 1	< 1	< 1	7	610
CLASS Y NB-95	< 1	< 1	< 1	< 1	< 1	< 1	3	150
CLASS W NP-237	8500	25000	28000	31000	38000	44000	58000	67000
CLASS Y RU-103	< 1	< 1	< 1	< 1	< 1	2	8	250
CLASS W RU-103	< 1	< 1	< 1	< 1	< 1	2	18	810
CLASS D RU-103	< 1	< 1	< 1	< 1	< 1	< 1	4	130
CLASS D RU-106	3	5	6	8	12	15	22	38
CLASS Y RU-106	21	70	76	81	90	98	130	220
CLASS W RU-106	5	15	17	21	31	43	97	240
CLASS W SB-125	< 1	< 1	< 1	< 1	2	3	10	100
CLASS D SB-125	< 1	< 1	< 1	< 1	2	3	5	22

Table 9-10 Summary of missed committed dose for whole body counting (contd)

	Missed Committed Effective Dose Equivalent in mrem at Indicated Day After Intake							
Nuclide	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W U-235	170	540	650	810	1200	1700	4900	20000
CLASS D U-235	82	230	350	650	1400	2100	2800	3000
CLASS Y U-235	2700	9900	10000	11000	11000	11000	12000	15000
CLASS Y ZN-65	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1
CLASS Y ZR-95	< 1	< 1	< 1	< 1	2	2	6	55
CLASS D ZR-95	< 1	< 1	< 1	< 1	< 1	< 1	4	27
CLASS W ZR-95	< 1	< 1	< 1	< 1	< 1	2	8	70

Table 9-11 Summary of missed committed dose for 1-day fecal sample

	Missed Committed Effective Dose Equivalent in mrem at Indicated Day After Intake							
Nuclide	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W AM-241	< 1	5	36	46	70	110	350	3400
CLASS W CF-252	< 1	2	11	15	23	35	130	1400
CLASS Y CF-252	< 1	2	100	120	130	140	170	240
CLASS W CM-242	< 1	< 1	< 1	2	3	6	29	600
CLASS W CM-244	< 1	3	20	26	39	59	200	1900
CLASS W NP-237	< 1	4	31	39	59	90	310	4100
CLASS Y PU-238	< 1	4	180	220	230	240	270	350
CLASS W PU-238	< 1	4	32	41	61	93	310	3000
CLASS W PU-239	< 1	5	35	44	66	99	330	3200
CLASS Y PU-239	< 1	4	190	220	230	240	280	350
CLASS Y SR-89	< 1	< 1	6	9	15	23	90	1500
CLASS Y SR-90	< 1	3	160	200	210	220	250	320
CLASS W TH-228	< 1	3	21	27	42	66	250	3900
CLASS Y TH-228	< 1	4	220	260	280	300	380	580
CLASS Y TH-232	2	15	750	900	940	980	1100	1400
CLASS W TH-232	4	18	130	170	260	390	1400	18000
CLASS W U-234	< 1	< 1	< 1	< 1	< 1	2	7	91
CLASS Y U-234	< 1	2	81	98	100	110	120	160
CLASS W U-235	< 1	< 1	< 1	< 1	< 1	2	6	84
CLASS Y U-235	< 1	< 1	75	90	94	98	110	140
CLASS W U-236	< 1	< 1	< 1	< 1	< 1	2	7	85
CLASS Y U-236	< 1	2	81	98	100	110	120	160
CLASS Y U-238	< 1	< 1	75	90	94	98	110	140
CLASS W U-238	< 1	< 1	< 1	< 1	< 1	2	6	81

9-12 Summary of missed 12-month dose for urine bioassay

	Missed 12-Month Effective Dose Equivalent in mrem at Indicated Day After Intake							
Nuclide	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W AM-241	2	28	41	61	91	110	200	420
CLASS Y CF-252	85	1500	2400	3800	5100	5400	5700	6200
CLASS W CF-252	3	60	89	130	210	260	490	1200
CLASS W CM-242	< 1	19	28	45	77	110	280	1300
CLASS W CM-244	2	29	43	65	97	120	210	460
CLASS W NP-237	< 1	32	35	40	54	73	180	930
CLASS W PU-238	< 1	12	19	23	30	35	48	83
CLASS Y PU-238	42	370	600	740	890	930	890	800
CLASS Y PU-239	38	330	540	670	810	840	800	720
CLASS W PU-239	< 1	11	17	21	27	31	43	74
CLASS D SR-89	< 1	< 1	< 1	< 1	< 1	< 1	6	140
CLASS Y SR-89	< 1	< 1	3	16	35	54	190	2400
CLASS D SR-90	< 1	< 1	< 1	< 1	< 1	2	3	5
CLASS Y SR-90	< 1	7	21	83	120	120	130	130
CLASS W TH-228	2	140	160	170	200	240	350	570
CLASS Y TH-228	56	5600	5800	5700	5600	5600	5500	5600
CLASS W TH-232	< 1	23	24	26	30	35	47	64
CLASS Y TH-232	9	860	890	870	830	800	720	610
CLASS D U-234	< 1	< 1	< 1	< 1	< 1	2	40	200
CLASS W U-234	< 1	< 1	2	4	7	11	31	270
CLASS Y U-234	5	100	180	330	500	570	590	590
CLASS W U-235	< 1	< 1	2	4	7	10	28	250
CLASS Y U-235	4	96	170	310	470	520	550	550
CLASS D U-235	< 1	< 1	< 1	< 1	< 1	2	37	190
CLASS W U-236	< 1	< 1	2	4	7	10	29	250
CLASS Y U-236	5	100	170	320	480	550	570	570
CLASS D U-236	< 1	< 1	< 1	< 1	< 1	2	39	200
CLASS Y U-238	4	92	160	300	450	500	530	520
CLASS W U-238	< 1	< 1	2	4	6	9	27	230
CLASS D U-238	< 1	< 1	< 1	< 1	< 1	2	36	180

9-13 Summary of missed 12-month dose for chest count

Nuclide	Missed 12-Month Effective Dose Equivalent in mrem at Indicated Day After Intake							
	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W AM-241	12	19	21	25	37	55	170	1900
CLASS W CE-141	< 1	< 1	< 1	< 1	< 1	< 1	3	1500
CLASS Y CE-141	< 1	< 1	< 1	< 1	< 1	< 1	< 1	21
CLASS W CE-144	< 1	< 1	< 1	< 1	< 1	2	7	120
CLASS Y CE-144	< 1	< 1	< 1	< 1	< 1	< 1	< 1	3
CLASS W CF-252	6100	9400	10000	13000	19000	29000	98000	> 250000
CLASS Y CF-252	8700	12000	13000	13000	14000	15000	17000	23000
CLASS W CM-242	1700	2600	3000	3900	6500	11000	50000	> 250000
CLASS W CM-244	2900	4400	4900	5900	8800	13000	41000	> 250000
CLASS W NP-237	26	40	45	55	80	120	370	4100
CLASS Y PU-238	6400	9200	9300	9500	9800	10000	11000	13000
CLASS W PU-238	4100	6200	6900	8400	12000	18000	58000	> 250000
CLASS Y PU-239	15000	21000	22000	22000	23000	23000	26000	31000
CLASS W PU-239	9400	14000	16000	19000	28000	42000	130000	> 250000
CLASS W TH-228	1600	2500	2800	3400	5200	7800	27000	> 250000
CLASS Y TH-228	2500	3700	3700	3900	4100	4400	5200	7500
CLASS W TH-232	2100	3200	3600	4400	6400	9400	30000	> 250000
CLASS Y TH-232	1500	2200	2200	2200	2300	2400	2600	3100
CLASS D U-234	460	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS Y U-234	5400	7700	7900	8000	8200	8500	9300	11000
CLASS W U-234	1500	2300	2500	3100	4500	6600	21000	230000
CLASS Y U-235	12	17	17	17	18	18	20	24
CLASS W U-235	3	5	5	7	10	14	45	490
CLASS D U-235	< 1	1800	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS Y U-236	11000	16000	16000	16000	17000	17000	19000	23000
CLASS W U-236	2900	4500	5000	6000	8800	13000	41000	> 250000
CLASS D U-236	930	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS W U-238	33	51	57	69	100	150	470	5200
CLASS Y U-238	120	180	180	180	190	190	210	250
CLASS D U-238	10	19000	> 250000	> 250000	> 250000	> 250000	> 250000	> 250000

9-14 Summary of missed 12-month dose for whole body count

Nuclide	Missed 12-Month Effective Dose Equivalent in mrem at Indicated Day After Intake							
	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS D BA-140	< 1	2	5	12	64	350	54000	> 250000
CLASS Y CE-141	< 1	< 1	< 1	2	3	6	44	2700
CLASS W CE-141	< 1	< 1	< 1	< 1	3	5	43	2500
CLASS Y CE-144	25	91	97	100	110	130	170	310
CLASS W CE-144	15	41	45	50	62	74	110	190
CLASS W CO-58	< 1	< 1	< 1	< 1	< 1	< 1	5	70
CLASS Y CO-58	< 1	< 1	< 1	< 1	< 1	< 1	2	15
CLASS W CO-60	< 1	< 1	< 1	< 1	< 1	2	4	10
CLASS Y CO-60	< 1	2	2	2	2	2	3	3
CLASS D CS-134	< 1	< 1	< 1	< 1	< 1	< 1	< 1	4
CLASS D CS-137	< 1	< 1	< 1	< 1	< 1	< 1	< 1	2
CLASS D FE-59	< 1	< 1	< 1	< 1	< 1	< 1	3	60
CLASS W FE-59	< 1	< 1	< 1	< 1	< 1	2	7	140
CLASS D I-131	< 1	< 1	2	9	140	2300	> 250000	> 250000
CLASS D I-133	< 1	38	23000	> 250000	> 250000	> 250000	> 250000	> 250000
CLASS D LA-140	< 1	< 1	10	4400	> 250000	> 250000	> 250000	> 250000
CLASS W LA-140	< 1	< 1	32	16000	> 250000	> 250000	> 250000	> 250000
CLASS W MN-54	< 1	< 1	< 1	< 1	< 1	< 1	2	24
CLASS D MN-54	< 1	< 1	< 1	< 1	< 1	< 1	2	100
CLASS W NB-95	< 1	< 1	< 1	< 1	< 1	< 1	7	610
CLASS Y NB-95	< 1	< 1	< 1	< 1	< 1	< 1	3	150
CLASS W NP-237	370	1100	1200	1300	1600	1900	2500	2900
CLASS Y RU-103	< 1	< 1	< 1	< 1	< 1	2	8	250
CLASS W RU-103	< 1	< 1	< 1	< 1	< 1	2	18	810
CLASS D RU-103	< 1	< 1	< 1	< 1	< 1	< 1	4	130
CLASS D RU-106	2	3	4	6	8	10	16	26
CLASS Y RU-106	13	45	49	52	58	64	83	140
CLASS W RU-106	5	13	16	20	29	40	89	220
CLASS W U-235	160	500	600	750	1100	1600	4500	18000
CLASS D U-235	26	71	110	200	430	650	890	930
CLASS Y U-235	560	2100	2200	2200	2300	2400	2600	3100
CLASS Y ZN-65	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1
CLASS Y ZR-95	< 1	< 1	< 1	< 1	2	2	6	55
CLASS D ZR-95	< 1	< 1	< 1	< 1	< 1	< 1	3	26
CLASS W ZR-95	< 1	< 1	< 1	< 1	< 1	2	8	70

9-15 Summary of missed 12-month dose 1-day fecal samples

Nuclide	Missed 12-Month Effective Dose Equivalent in mrem at Indicated Day After Intake							
	1 day	7 days	15 days	30 days	60 days	90 days	180 days	365 days
CLASS W AM-241	< 1	< 1	2	2	3	5	16	150
CLASS W CF-252	< 1	< 1	4	4	7	11	38	420
CLASS Y CF-252	< 1	< 1	39	47	50	53	65	95
CLASS W CM-242	< 1	< 1	< 1	2	3	4	22	460
CLASS W CM-244	< 1	< 1	2	2	3	5	17	160
CLASS W NP-237	< 1	< 1	< 1	2	3	4	14	180
CLASS W PU-238	< 1	< 1	2	2	3	5	16	150
CLASS Y PU-238	< 1	< 1	20	24	25	26	29	38
CLASS W PU-239	< 1	< 1	< 1	2	3	4	14	140
CLASS Y PU-239	< 1	< 1	18	22	23	24	27	34
CLASS Y SR-89	< 1	< 1	6	9	15	23	90	1500
CLASS Y SR-90	< 1	< 1	40	49	51	53	60	79
CLASS W TH-228	< 1	< 1	9	11	17	27	100	1600
CLASS Y TH-228	< 1	2	100	120	130	140	180	270
CLASS W TH-232	< 1	< 1	< 1	2	3	4	14	180
CLASS Y TH-232	< 1	< 1	16	19	20	20	23	30
CLASS Y U-234	< 1	< 1	17	20	21	22	25	32
CLASS W U-234	< 1	< 1	< 1	< 1	< 1	2	6	84
CLASS W U-235	< 1	< 1	< 1	< 1	< 1	2	6	77
CLASS Y U-235	< 1	< 1	16	19	20	20	23	30
CLASS Y U-236	< 1	< 1	16	20	20	21	24	31
CLASS W U-236	< 1	< 1	< 1	< 1	< 1	2	6	79
CLASS Y U-238	< 1	< 1	15	18	19	20	22	29
CLASS W U-238	< 1	< 1	< 1	< 1	< 1	2	6	74

Chapter 10

Routine Monitoring Programs

Chapter 10 Preview

- Worker and Workgroup Monitoring Programs
- Monitoring Program for Plutonium
- Monitoring Program for Transuranics Other Than Plutonium
- Monitoring Program for Uranium
- Monitoring Program for Thorium
- Monitoring Program for Strontium
- Gamma-Emitting Fission and Activation Products
- Summary of Worker Monitoring Programs
- Summary of Workgroup Monitoring Programs

10

Routine Monitoring Programs

Minimum routine monitoring programs based on the guidelines in Chapters 2 and 3 and the missed dose tables are given in this section for workers exposed to the following radioactive materials:

- plutonium
- transuranics other than plutonium
- uranium
- thorium
- strontium
- gamma-emitting fission and activation products
- tritium

Additional monitoring may be recommended for a specific facility in Chapters 12 through 17.

Worker and Workgroup Monitoring Programs

As discussed in Chapter 2, workers who are

- likely to be exposed in excess of 40 DAC-h of a radioactive material in a calendar year
- use any form of respiratory protection
- enter Airborne Radioactivity Areas

- enter a Radiation Control Area (RCA)
- have radioactive material in their body that interferes with detecting assessing additional intakes

are required to be on a worker monitoring program.

Workers who have a reasonable potential for exposure to radioactive material but do not meet any of the criteria given above shall be on a workgroup monitoring program. As a minimum, workers who enter a RCA where protective clothing is required are considered to have a reasonable potential for exposure to radioactive materials. Workgroup programs may be also be appropriate for some areas where protective clothing is not required, the Waste Tank Areas for example.

Monitoring Program for Plutonium

The worker monitoring program for plutonium shall consist of

- annual urine bioassay
- annual chest count
- annual fecal bioassay
- personal air sampler (PAS)

If respiratory protection is not used then Health Protection Operations may use either fecal bioassay or PAS or both. If respiratory protection is used then fecal bioassay is required and PAS is optional. The missed 12-month effective dose equivalent in mrem for each monitoring method is given

	Annual Urine Bioassay	Annual Feces Bioassay	Annual Chest Count	PAS
Class W Pu-238	83	150	>250000	<100
Class W Pu-239	74	140	>250000	<100
Class Y Pu-238	800	38	13000	<100
Class Y Pu-239	720	34	31000	<100

A missed dose of less than 100 mrem for the PAS may be achieved by analysis of composited daily filters. Chest counting is recommended to monitor the ingrowth of Am-241 from any Pu-241 that may be retained in the lungs.

The workgroup monitoring program for plutonium shall consist of

- annual urine bioassay
- annual chest count

for each worker in the workgroup. This individual frequency is assumed to result in at least a monthly frequency for monitoring the workgroup.

The missed committed effective dose equivalent in mrem for each monitoring method is given below, assuming a monthly frequency for the workgroup:

	Monthly Urine Bioassay	Monthly Chest Count
Class W Pu-238	1600	>250000
Class W Pu-239	1700	>250000
Class Y Pu-238	7300	120000
Class Y Pu-239	7500	>250000

Chest counting is recommended to monitor the ingrowth of Am-241 from any Pu-241 that may be retained in the lungs. A special bioassay program is required for all members of a workgroup if any member of the workgroup has a confirmed intake of radioactive material.

Monitoring Program for Transuranics other then Plutonium

The worker monitoring program for neptunium, americium, curium, and californium shall consist of

- quarterly urine bioassay
- annual chest count
- semi-annual fecal bioassay
- personal air sampler (PAS)

If respiratory protection is not used then Health Protection Operations may use either fecal bioassay or PAS or both. If respiratory protection is used then fecal bioassay is required and PAS is optional. The missed 12-month effective dose equivalent in mrem for each monitoring method is given below:

	Quarterly Urine Bioassay	Semi-annual Feces Bioassay	Annual Chest Count	PAS
Class W Np-237	290	28	4100	<100
Class W Am-241	420	32	1900	<100
Class W Cm-242	440	44	>250000	<100
Class W Cm-244	460	34	>250000	<100
Class W Cf-252	1000	76	23000	<100

The relatively high missed dose for urine bioassay is the result of the high MDA for these radionuclides (0.2 pCi/L). Work is currently being done to lower the MDA by a factor of 5 to 10, which will enable annual urine bioassay to meet the design objective. Until this work is completed semi-annual feces bioassay is required. A missed dose of less than 100 mrem for the PAS may be achieved by analysis of composited daily filters.

The workgroup monitoring program for neptunium, americium, curium, and californium shall consist of

- annual urine bioassay
- annual chest count

for each worker in the workgroup. This individual frequency is assumed to result in at least a monthly frequency for monitoring the workgroup.

The missed committed effective dose equivalent in mrem for each monitoring method is given below, assuming a monthly frequency for the workgroup:

	Monthly Urine Bioassay	Monthly Chest Count
Class W Np-237	11000	16000
Class W Am-241	9500	6800
Class W Cm-242	710	61000
Class W Cm-244	5500	>250000
Class W Cf-252	3800	>250000

A special bioassay program is required for all members of a workgroup if any member of the workgroup has a confirmed intake of radioactive material.

Monitoring Program for Uranium

The worker monitoring program for uranium shall consist of

- semi-annual urine bioassay
- annual chest count
- annual fecal bioassay
- personal air sampler (PAS), if class Y uranium is present

If class Y uranium is present and respiratory protection is not used, then Health Physics Operations may use either fecal bioassay or PAS or both. If class Y uranium is present and respiratory protection is used, then fecal bioassay is required and PAS is optional.

The missed 12-month effective dose equivalent in mrem for each monitoring method is given below:

	Semi-annual Urine Bioassay	Annual Feces Bioassay	Annual Chest Count	PAS
Class D U-234	80	--	>250000	<100
Class W U-234	62	32	230000	<100
Class Y U-234	590	84	11000	<100

U-234, U-235, U-236, and U-238 have approximately equal missed doses for urine and feces bioassay; therefore, only U-234, which has the highest missed dose, is listed. Lower missed doses for chest counting are possible if significant quantities of U-235 or U-238 are present. Missed dose tables for specific uranium mixtures are presented in Chapters 12-17. A missed dose of less than 100 mrem for the PAS may be achieved by analysis of composited daily filters.

This monitoring program is based on radiological toxicity alone. Depending on the specific activity of the uranium, monthly urine bioassay is required to detect intakes of uranium that exceed the chemical toxicity limit (Chapter 9). An inhalation intake of 120 nCi class D U-234, U-235, U-236, or U-238 will deliver an effective dose equivalent of 100 mrem in the 12 months following the intake. The chemically toxic intake of class D uranium is 20 mg. Assuming intakes of uranium that deliver 100 mrem are detected and assessed, workers exposed to class D uranium with a specific activity lower than

$$120 \text{ nCi}/20 \text{ mg} = 6 \text{ nCi/mg}$$

require a monthly urine bioassay for elemental uranium in addition to the semi-annual isotopic uranium urine bioassay. A similar analysis shows that there are no uranium isotopes with a specific activity low enough to warrant monthly urine bioassay for class W and class Y uranium. This assumes that the chemical hazard is assessed for all intakes of uranium detected by isotopic uranium urine bioassay.

The workgroup monitoring program for uranium shall consist of

- annual urine bioassay
- annual chest count

for each worker in the workgroup. This individual frequency is assumed to result in at least a monthly frequency for monitoring the workgroup.

The missed committed effective dose equivalent in mrem for each monitoring method is given below, assuming a monthly frequency for the workgroup:

	Monthly Urine Bioassay	Monthly Chest Count
Class D U-234	<12	>250000
Class W U-234	48	40000
Class Y U-234	2800	54000

A special bioassay program is required for all members of a workgroup if any member of the workgroup has a confirmed intake of radioactive material.

Monitoring Program for Thorium

The worker monitoring program for thorium shall consist of

- annual urine bioassay
- annual chest count
- semi-annual fecal bioassay
- personal air sampler (PAS)

If respiratory protection is not used Health Protection Operations may use either fecal bioassay or PAS or both. If respiratory protection is used then fecal bioassay is required and PAS is optional.

The missed 12-month effective dose equivalent in mrem for each monitoring method is given below:

	Annual Urine Bioassay	Semi-annual Feces Bioassay	Annual Chest Count	PAS
Class W Th-228	570	200	>250000	<100
Class W Th-232	64	28	>250000	<100
Class Y Th-228	5600	270	7500	<100
Class Y Th-232	610	30	3100	<100

Unsupported Th-228 (i.e., no Th-232) is not anticipated at SRS facilities. The relatively high missed dose for Th-228 is therefore not viewed as a problem.

The missed dose for chest counting is based on detection of photons emitted by the Th-232 or Th-228. Lower missed doses may be achieved by counting the photons emitted by the daughters. Missed doses for thorium in specific facilities are given in the chapters on those facilities.

A missed dose of less than 100 mrem for the PAS may be achieved by analysis of composited daily filters.

The workgroup monitoring program for thorium shall consist of

- annual urine bioassay
- annual chest count

for each worker in the workgroup. This individual frequency is assumed to result in at least a monthly frequency for monitoring the workgroup.

The missed committed effective dose equivalent in mrem for each monitoring method is given below, assuming a monthly frequency for the workgroup

	Monthly Urine Bioassay	Monthly Chest Count
Class W Th-228	1400	100000
Class W Th-232	6400	>250000
Class Y Th-228	12000	16000
Class Y Th-232	29000	150000

A special bioassay program is required for all members of a workgroup if any member of the workgroup has a confirmed intake of radioactive material.

Monitoring Program for Strontium

The worker monitoring program for strontium shall consist of semi-annual urine bioassay.

The missed 12-month effective dose equivalent in mrem for this monitoring method is given below:

	Semi-monthly Urine Bioassay
Class D Sr-89	12
Class D Sr-90	5
Class Y Sr-89	380
Class Y Sr-90	130

The relatively high missed dose for class Y Sr-89 is caused by its short physical halflife. Significant exposures to pure Sr-89 are not anticipated and this is not considered a problem.

The workgroup monitoring program for strontium shall consist of semi-annual urine bioassay for each member of the workgroup. This individual frequency is assumed to result in at least a semi-monthly frequency for monitoring the workgroup.

The missed committed effective dose equivalent in mrem for each monitoring method is given below, assuming a semi-monthly frequency (every 15 days) for the workgroup:

	Semi-monthly Urine Bioassay
Class D Sr-89	<12
Class D Sr-90	<12
Class Y Sr-89	72
Class Y Sr-90	530

Gamma-Emitting Fission and Activation Products

The worker and workgroup monitoring programs for gamma-emitting fission and activation products shall consist of semi-annual whole-body counts. The missed doses for various pure gamma-emitting radionuclides on a semi-annual frequency are given in Tables 9-10 and 9-14. With the exception of Ce-144, Ru-106, and the short-lived radionuclides listed below, all the missed doses for individual radionuclides are less than 100 mrem:

Ba-140
La-140
I-131
I-133

These short-lived radionuclides can not be reliably detected by whole- body counts on a semi-annual frequency. Special efforts should be taken to detect them by air and surface monitoring. The missed 12- month EDE class Y Ru-106 is within the 200 mrem objective given in Chapter 2, and is considered acceptable. Class Y Ce-144 can not be reliably detected by the standard whole-body counting methods. A special 5 minute chest count once a year is required to detect this material. Mixtures of gamma-emitters are likely to produce larger missed doses than any one gamma-emitter. This possibility should be explored if the composition of the mixture is known.

Tritium

Tritium is unique at SRS in that urine bioassay is the primary method used to monitor and control exposure of workers to tritiated water. For all other radionuclides workplace monitoring is the primary method used to monitor and control exposure of workers. For this reason a missed dose analysis alone can not be used to select the appropriate bioassay frequency, but it can be used to select a minimum frequency. With this in mind, the worker and workgroup monitoring programs for tritiated water shall consist of monthly urine bioassay. The missed dose for this frequency is less than 100 mrem as shown in Figure 9-3. The bioassay frequencies for specific facilities are given in the appropriate chapters on those facilities.

Summary of Worker Monitoring Programs

The following table indicates the types and frequency of monitoring that should be performed for each category of radionuclide. The frequency is given as the number of times the monitoring should be performed for each worker in a calendar year.

	Urine	Feces	Chest	Whole-Body	PAS
Plutonium	1	1	1	-	+
Other TRU	4	2	1	-	+
Uranium	2**	1*	1	-	+*
Thorium	1	2	1	-	+
Strontium	2	-	-	-	-
FP and AP	-	-	1***	2	-
Tritium	12	-	-	-	-

\+ should be performed

\- should not be performed

* only if class Y uranium is present

** monthly urine bioassay for elemental uranium is also required for exposures to class D uranium with a specific activity of less than 6 nCi/mg

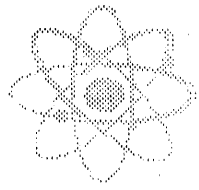
***a 5 minute count on the germanium chest counter every year is required for workers exposed to class Y Ce-144

Summary of Workgroup Monitoring Programs

The following table indicates the types and frequency of monitoring that should be performed for each category of radionuclide. The frequency is given as the number of times the monitoring should be performed for each person in the workgroup in a calendar year.

	Urine	Feces	Chest	Whole-Body	PAS
Plutonium	1	-	1	-	-
Other TRU	1	-	1	-	-
Uranium	1	-	1	-	-
Thorium	1	-	1	-	-
Strontium	2	-	-	-	-
FP and AP	-	-	-	2	-
Tritium	12	-	-	-	-

\+ should be performed
\- should not be performed



Part II

Part II - Facility Specific Routine Bioassay Programs

Chapter 11

Introduction

Chapter 11 Preview

- Overview of Part II

11

Introduction

One of the objectives of the Internal Dosimetry Technical Basis Manual is to derive routine monitoring programs for the various facilities at SRS. In Part II the facilities and the radionuclides of concern for each facility are identified and the minimal routine monitoring programs which are applicable to each facility are presented.

The facilities that are included in this section are those selected by HPO and HPT personnel to be of radiological concern. It is understood that other facilities will be added as needed.

Many radionuclides may be present in a facility. Waste streams such as those at DWPF or other waste treatment facilities may contain on the order of 60 radionuclides. Since it is not practical to design a program for each radionuclide, the radionuclides of concern are determined as follows. All radionuclides in a facility are identified from safety analysis reports (SAR), personal interviews, the open literature, etc. The radionuclides whose radiotoxicity and exposure potential combine to deliver 90% of the dose are considered to be the radionuclides of concern.

The recommended routine monitoring programs were selected from those presented in Chapter 10 of Part I. For example, if information in a facility's SAR and subsequent dose calculations indicate that Pu-238 and Am-241 deliver over 90% of the dose, then the monitoring program for plutonium and the monitoring program for americium (listed in Chapter 10) are the minimum programs required for that facility.

The routine monitoring programs presented are minimum programs and are based solely upon missed dose analyses. Additional sampling in a facility may be required. For example, if tritiated water is the only radionuclide of concern, then a monthly urine bioassay is an adequate routine worker monitoring program. However, if urine bioassay is used to detect exposures of workers to tritium and control their exposures, then a higher frequency may be necessary.

Chapter 12

100 Areas

Chapter 12 Preview

- Internal Dosimetry Surveillance Program for P, L, and K Reactors

12

100 Areas

Internal Dosimetry Surveillance Program for P, L, and K Reactors

Five heavy-water reactors designated R, P, L, K, and C were constructed at SRS in the early 1950's. Three of the reactors, P, L, and K, are currently operational. Each of the reactors is located in its own area which is designated with the same letter.

Description of Process

Targets and highly enriched fuel for the reactors are fabricated in M Area. Targets that are irradiated in the reactors include:

Target	Desired Product
depleted uranium	Pu-239
Np-237	Pu-238
lithium	tritium

In the past, targets have been irradiated to produce transplutonic elements such as curium and californium. The targets are irradiated and transferred to the Triltium Facilities (tritium), H Area Separations (Pu-238), and F Area Separations (Pu-239) for processing. Spent reactor fuel is sent to H Area Separations for processing.

Radionuclides of Concern

The composition of radionuclides that workers have been exposed to in the reactors is not documented. Discussions with Health Physics Operations personnel and ANSI 343[1] indicate that the following radionuclides may be of concern:

Pu-239	Cs-134,137
Am-241	Zn-65
U-234	Fe-59
Ce-144	Co-58,60
Ru-103,106	Mn-54
Eu-152,154	La-140
Sr-90	Ba-140
I-131,133	Nb-95
Zr-95	H-3

No information is available on the relative abundances of each radionuclide in the workplace, but tritium is assumed to produce most of the personnel exposures. This is because of the large quantities that are present, and tritium is difficult to completely contain.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the P, L, and K Reactor facilities.

Worker Monitoring Program

The tritium worker monitoring program as given in Chapter 10 should be instituted. This program is based solely on missed dose and requires monthly urine bioassay for tritium. If urine bioassay is used to detect and control workers' exposures to tritium, then a higher frequency may be necessary.

The radionuclides of concern, other than tritium, are considered to be a source independent of tritium. Fission product, strontium, and plutonium worker monitoring programs as given in Chapter 10 should be instituted. These programs require the following:

- urine - one sample every six months for strontium and one sample every year for plutonium
- feces - one sample every year for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count every year on the germanium chest counter
- whole-body count - one count every six months

Workgroup Monitoring Program

The tritium workgroup monitoring program as given in Chapter 10 should be instituted. This program requires monthly urine bioassay for tritium.

The radionuclides of concern, other than tritium, are considered to be a source independent of tritium. Fission product, strontium, and plutonium worker monitoring programs as given in Chapter 10 should be instituted. These programs require the following:

- urine - one sample every six months for strontium and one sample every year for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

References

(1) *American National Standard for Internal Dosimetry for Mixed Fission and Activation Products*. ANSI N343-1978

Chapter 13

F and G Areas

Chapter 13 Preview

- 221-F A-Line
- 221-F B-Line
- 221-F Canyon Facility
- 221-F New Special Recovery Facility
- 221-F Outside Facilities
- 221-F Pu Fuel Form Facility (PuFF) and Pu Experimental Facility (PEF)
- 235-F Vaults
- 772-F and 772-1F Production Control Laboratories
- F-Area Burial Grounds
- F-Area Plutonium Storage Facility (PSF)
- F- and H-Area Effluent Treatment Facility
- F- and H-Area Seepage and Retention Basins
- F- and H-Area Tank Farms
- G-Area Waste Certification Facility

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F and G Areas

Internal Dosimetry Surveillance Program for F Area A-Line

The A-Line facility accepts dilute, depleted uranyl nitrate solution from the 221-F Facility and converts it into uranium trioxide powder[1]. Dilute uranyl nitrate solution is sent to A-Line from the solvent extraction process in the 221-F Facility where it is initially decanted to remove TBP (tributyl phosphate n-paraffin) solvent; it is then concentrated by evaporation. A second evaporation yields a hydrate of uranyl-nitrate that is heated in a denitrator to yield the uranium trioxide product.

The A-Line facility also has a dissolver to prepare a uranyl nitrate solution from the uranium trioxide product and nitric acid when it is required as a process feed stream back into the solvent extraction process in the 221-F Facility. The dissolver is also used to recycle offstandard A-Line product. The uranium trioxide product is placed into drums for storage onsite or into packages for offsite shipment.

Description of Process

The A-Line Facility can be broken down into unit operations that perform specific functions in the creation of the final product.

Evaporation

The evaporator, which is located outside on the south end of the A-Line area, accepts decanted uranyl nitrate solution from the 221-F Warm Canyon and concentrates it by using steam heating.

Purification

Once the uranyl nitrate solution is concentrated, it can be sent to the silica gel purification system. This system is an optional step in the process and is

occasionally used when needed to ensure the uranium trioxide product will meet certain specifications. The silica gel beds will absorb residual zirconium and niobium out of the uranyl nitrate solution. Once the bed is used, regeneration of the silica can be accomplished by using hot oxalic acid. The acid will complex approximately 90% of the radioactivity and remove it from the silica. The waste from this process is either neutralized and sent to underground storage or collected in a low level oxalic acid waste tank and transferred to the seepage basin.

Hydrate Evaporation

Uranyl nitrate received from either the evaporator or the silica gel beds is fed to the hydrate evaporators where it is again concentrated. Condensate from the evaporation is sent to the seepage basin.

Denitration

During denitration the material received from the hydrate evaporators is converted from uranyl nitrate hexahydrate to uranium trioxide. There are two types of denitrator vessels, but the intermediate steps and the end product remain the same.

Material Handling

The uranium trioxide powder is removed from the denitrators using suction wands, sending the product to a dust collector where it is placed into storage bins from which it is mechanically loaded into containers. All process equipment in A-Line that handles uranium trioxide powder is vented to the oxide fines recovery system. The purpose of this system is to minimize the amount of material that can reach

the atmosphere; the system consists of a General Dust Collector (GDC) that is filtered. Collected particles from the filters are sent to a 55 gallon drum for storage.

Fume Recovery System

The Fume Recovery System converts oxides of nitrogen into nitric acid for reuse. The system consists of exhausters, a venturi scrubber, off-gas coolers, an absorption column, and multiple pumps and tanks. The venturi scrubber removes uranium trioxide particles from the denitrator off-gas and is the only piece of the system that is of a radiological concern.

Oxide Dissolution

Occasionally, 221-F requires a uranyl nitrate-nitric acid solution for use as a process stream. The dissolver in A-Line prepares this solution by dissolving uranium trioxide in nitric acid.

Radionuclides of Concern

The radionuclides of concern for A-Line feed material are

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-239	W	$5.599 \cdot 10^{-03}$	$2.061 \cdot 10^{-02}$	$3.097 \cdot 10^{-01}$
U235	W	$8.022 \cdot 10^{-04}$	$1.099 \cdot 10^{-03}$	$7.712 \cdot 10^{-04}$
U236	W	$8.179 \cdot 10^{-04}$	$1.137 \cdot 10^{-03}$	$7.971 \cdot 10^{-04}$
U238	W	$7.440 \cdot 10^{-01}$	$9.736 \cdot 10^{-01}$	$6.859 \cdot 10^{-01}$
Ce141	W	$1.305 \cdot 10^{-02}$	$2.216 \cdot 10^{-05}$	$1.427 \cdot 10^{-05}$
Ce144	W	$6.079 \cdot 10^{-02}$	$1.989 \cdot 10^{-03}$	$1.761 \cdot 10^{-03}$
Ru103	W	$5.969 \cdot 10^{-02}$	$7.934 \cdot 10^{-05}$	$5.110 \cdot 10^{-05}$
Ru106	W	$6.111 \cdot 10^{-02}$	$1.374 \cdot 10^{-03}$	$9.657 \cdot 10^{-04}$
Nb95	W	$5.418 \cdot 10^{-02}$	$5.318 \cdot 10^{-05}$	$3.425 \cdot 10^{-05}$

These were obtained from DPSTSA-100-10[1].

The activity fraction is the fraction of the total activity represented by a particular radionuclide. The annual dose fraction is the fraction of the annual dose conversion factor (DCF) represented by a particular radionuclide. The committed dose fraction is the fraction of the committed DCF represented by a particular radionuclide. The annual DCF for this mixture is 4.890 mrem/nCi. The committed DCF for this mixture is 7.593 mrem/nCi.

Dosimetrically, the feed material for A-Line is U-238 and Pu-239. The specific activity of the uranium is approximately 0.4 nCi/mg. The chemical forms of uranium that are produced in A-Line are uranyl nitrate and uranium trioxide; these materials are class D and W, respectively.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the F-Area A-Line facility.

Worker Monitoring Program

Uranium and plutonium worker monitoring programs as given in Chapter 10 should be instituted. These programs require the following:

- urine - one sample every month for elemental uranium and one sample every six months for plutonium and isotopic uranium
- feces - one sample every year to be analyzed for plutonium and isotopic uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

There are insufficient fission products present in the feed material to be useful as a tracer. If significant quantities of Zr-95 and Nb-95 are separated from the feed material in the purification process, then a fission product program will be required:

- whole-body count - one count every six months

Workgroup Monitoring Program

Uranium and plutonium workgroup monitoring programs as given in Chapter 10 should be instituted. These programs require the following:

- urine - one sample every year for plutonium, isotopic uranium, and elemental uranium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Surveillance Program 221-F B-Line

The FB-Line facility is located in the 200-F Separations Area and is used to convert plutonium nitrate into plutonium metal or plutonium oxide, depending upon the need[2]; it also has the capability to recover plutonium from onsite and offsite scrap. Initially, the plutonium produced in the facility is primarily Pu-239 in a dilute nitric acid solution. From there it undergoes many chemical processes that produce an end product.

Description of Process

The general location of the FB-Line facilities is in the mid-northern section of the 200-F Separations Area, west of the 704 Administration Building.

Cation Exchange

The dilute plutonium product solution is transferred from the canyon storage tanks to the FB-Line receiving tanks. These tanks are geometrically unfavorable from a nuclear criticality standpoint; therefore, other measures are taken to ensure that a criticality does not occur. At this point the product is cooled and filtered removing mainly dibutyl-phosphate and plutonium. Next, the material is sent through a cation exchange column that removes the plutonium along with some other impurities.

Precipitation and Filtration

The product solution from the cation exchange system is initially precipitated using hydrofluoric acid. This process creates plutonium trifloride crystals which can be filtered from the solution. The Plutonium trifloride that remains attached to the precipitator tanks can oxidize producing plutonium tetrafloride. This product is undesirable for a number of reasons, and the process is therefore carefully controlled to minimize its formation. The solution that is left from this process is tested to ensure that it is basic and transferred directly, without evaporation, to the F-Area tank farm.

Drying and Conversion

The plutonium trifloride taken from the filtration process is dried to a point where the moisture content is approximately 2-6%. Next, the dried material is roasted to produce plutonium dioxide and plutonium tetrafloride by the hydrolysis of plutonium trifloride at temperatures exceeding 500°C.

Reduction

The plutonium dioxide-plutonium tetrafloride mixture that was previously produced is weighed then mixed with metallic calcium and placed into a magnesium crucible whose inside is filled with magnesium oxide sand. The sand prevents any intermetallic reactions and subsequent contamination of the molten plutonium from contact with the metal pressure vessel. Finally, the chamber is sealed, lifted up into the furnace, pressurized with argon, checked for leaks, and heated.

The exothermic reactions that follow rapidly raise the temperature to about 1500°C, which produces a molten plutonium metal. The plutonium metal, being more dense than the calcium produc., separates and flows to the bottom of the

crucible and forms a button shaped solid. The button is cooled, separated from the calcium slag, and die stamped for identification.

Button Finishing

Once the plutonium metal button is stamped, it is placed in a basket and submerged into a nitric acid solution to remove particles of sand, slag, and calcium metal, and then it is rinsed in water and air dried. The next step involves removing one or two grams of the plutonium button using a drill; these drill turnings are sealed and sent to the 772-F Laboratory for analysis. The button is then weighed and placed inside a can, crimp sealed placed in a plastic bag, placed in a second can, and crimp sealed; the sealed unit is placed in storage. Upon successful completion of the laboratory analysis and seal tests, the can will be sent offsite.

Recovery

The recovery process takes solid scrap from onsite and offsite sources, miscellaneous solutions from FB-Line, and extracts the plutonium. The presence of other cation impurities, such as calcium magnesium and aluminum, causes the process of extracting plutonium from scrap to be accomplished by using an anion exchange resin in contrast to a cation exchange resin. The scrap is initially dissolved using a mixture of aluminum nitrate and nitric acid. Once in the solution, the plutonium scrap is sorbed on the anionic resin and then sent back to the canyon for processing.

Special Recoveries

Certain plutonium forms are received from both onsite and offsite that are not suitable for processing in the slag and crucible dissolver process discussed above. The special recovery process is also used to process certain plutonium containing solutions that cannot be prepared for anion exchange.

Storage and Vaults

All plutonium bearing material, both final products and recovery feed materials are stored in the FB-Line storage vaults. Materials that would be found in these vaults include:

- standard and off-standard plutonium buttons
- plutonium metal scrap
- slag and crucibles
- cabinet sweepings
- plutonium lab solutions
- plutonium alloy buttons
- plutonium oxide

Radionuclides of Concern

The radionuclide of concern in 221-F B-Line is plutonium. Many different mixtures of plutonium isotopes are possible, but we will assume either 6% Pu or 12% Pu (the percent refers to the mass percent of Pu-240).

The activity fractions of alpha-emitting plutonium (α-Pu) isotopes found in a typical 6% plutonium button from 221-F B- line[3] are

Isotope	Activity Fraction
Pu-238	0.048
Pu-239	0.776
Pu-240	0.176

The ratio of Pu-241 to α-Pu is 8.0 to 1.

Listed below are the activity fractions of alpha-emitting plutonium (α-Pu) isotopes found in a typical 12% plutonium button from 221-F B- line[3] are

Isotope	Activity Fraction
Pu-238	0.266
Pu-239	0.466
Pu-240	0.268

The ratio of Pu-241 to α-Pu is 23 to 1.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the 221-F B-Line facility.

Worker Monitoring Program

A plutonium worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every year for plutonium
- feces - one sample every year for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

A plutonium workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every year for plutonium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Surveillance Program for 221-F Canyon Facility

The 221-F Canyon is a long rectangular building located in the northwest section of F-Area Separations[1]. The canyon building consists of two parallel canyons that constitute the process area. These canyons contain the high activity and low activity materials and are referred to as the hot and warm canyons, respectively.

The F-Canyon facility uses the Purex process to recover plutonium-239, neptunium-237, and uranium- 238 from reactor-irradiated uranium targets. The plutonium is transferred to the B-Line and the uranium is transferred to the A-Line facility where they are processed into solid form.

Description of Process

This section gives an overview of the processes that must be monitored at the 221-F Canyon Facility.

Head End

Irradiated targets are received from the reactor areas. The aluminum cladding is dissolved in hydroxide-sodium nitrate solution, and the uranium metal is then dissolved in nitric acid. Noble gases and iodines are evolved at this point and are routed through silver nitrate-coated sorbent materials to reduce iodine emissions. Feed solution from the FB- Line special recovery process is also fed into the head end process (see Figure 13-1).

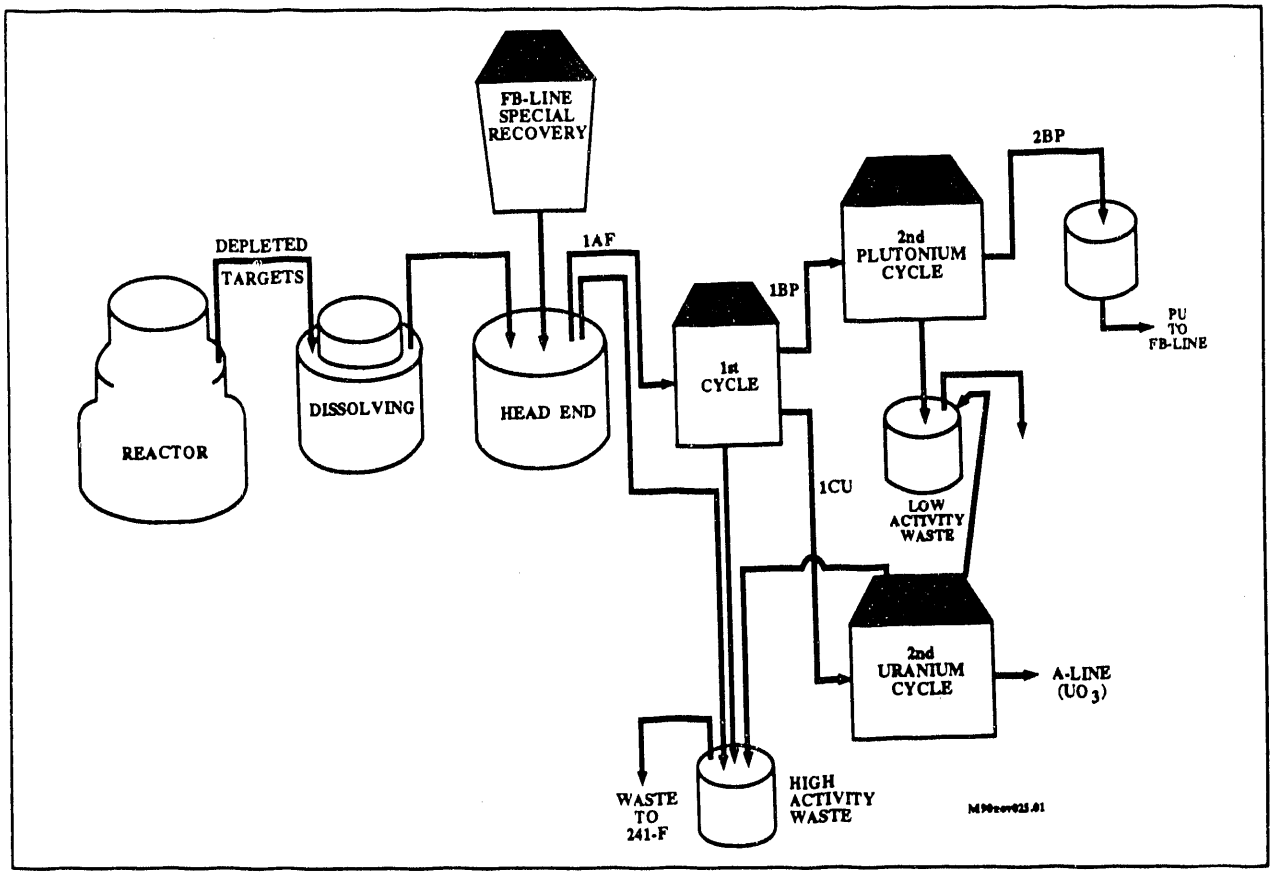

Figure 13-1. F Canyon Facility

Solvent Extraction

The first solvent extraction process (or cycle) separates the plutonium and the uranium from the fission products using tributyl phosphate in a hydrocarbon dilutant. In the second cycles, plutonium and uranium are separated and purified.

Anion Exchange

The aqueous waste produced in the first solvent extraction cycle and the second uranium cycle contain neptunium and plutonium in a nitric acid solution; these nuclides are recovered using a anion resin exchange column. After the solution is sent through the column, the diluted actinides are sent to the first solvent extraction cycle or warm canyon for further processing and separation. If the concentration of the actinides is such that the solution is sent to the warm canyon, a further series of columns are used to purify and separate the plutonium from the neptunium. At this stage, Th-232 and Th-234 are also removed.

Waste Evaporators

Aqueous waste from the first extraction cycle, the second uranium cycle, acid strip condensate, ventilation systems sumps and concentrate from the second stage high activity waste evaporator are all combined and evaporated in the first stage continuous evaporator. Condensate from the evaporator flows to the second stage continuous evaporator where it is once again evaporated. The bottoms (a term used to describe what's left after evaporation) from the first stage evaporators are sent continuously to a bottoms tank and are then transferred back to the primary recovery column. Concentrate from the second stage evaporator is periodically recycled to the high activity waste evaporator. This entire process reduces the volume of waste going to high level waste storage tanks. Continuous low activity waste evaporators are also used for volume reduction and acid recovery.

Locations of Increased Potential

The hot and warm crane cabs in 221-F Canyons have increased potential for exposure to radioactive materials. These areas are the major concern because the majority of the process is isolated and run remotely because of the high radiation fields. The areas where the cranes are decontaminated are also a location that should be closely monitored.

Radionuclides of Concern

The function of the 221-F Separations Facility is to isolate and purify plutonium, uranium, and neptunium from irradiated uranium. Fission products will also be present, and depending upon the age of the material (i.e. length of time out of the reactor) the isotopes of concern and the fraction by activity of these materials will change. Table 13-1 lists the mix of radionuclides at various stages in the separations cycles along with their fractions by activity. Three of these steams have been deemed to either deliver the majority of the dose or contain useful tracers. Lung solubility classes were chosen to maximize the missed committed dose.

The majority of the annual and committed dose from the Head End Stream (stream 1) comes from plutonium, cerium, and ruthenium as shown in the following table:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-239	Y	$1.191 \cdot 10^{-03}$	$1.981 \cdot 10^{-01}$	$6.247 \cdot 10^{-01}$
Ce144	W	$4.765 \cdot 10^{-01}$	$4.373 \cdot 10^{-01}$	$1.833 \cdot 10^{-01}$
Cs137	D	$2.621 \cdot 10^{-02}$	$4.359 \cdot 10^{-03}$	$1.466 \cdot 10^{-03}$
Ru106	Y	$1.588 \cdot 10^{-01}$	$2.824 \cdot 10^{-01}$	$1.333 \cdot 10^{-01}$
Zr95	Y	$3.177 \cdot 10^{-01}$	$4.190 \cdot 10^{-02}$	$1.277 \cdot 10^{-02}$
Sr90	D	$1.959 \cdot 10^{-02}$	$4.404 \cdot 10^{-03}$	$6.211 \cdot 10^{-03}$

The activity fraction is the fraction of the total activity represented by a particular radionuclide. The annual dose fraction is the fraction of the annual dose conversion factor (DCF) represented by a particular radionuclide. The committed dose fraction is the fraction of the committed DCF represented by a particular radionuclide. The annual DCF for this mixture is $1.557 \cdot 10^{-01}$ mrem/nCi. The committed DCF for this mixture is $6.939 \cdot 10^{-01}$ mrem/nCi.

The next stream to be considered is the second uranium cycle stream (stream 2). In this stream uranium 238 and plutonium 239 make up the majority of the dose, both annual and committed.

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-239	W	$4.226 \cdot 10^{-03}$	$1.677 \cdot 10^{-02}$	$2.636 \cdot 10^{-01}$
U235	Y	$7.433 \cdot 10^{-04}$	$4.095 \cdot 10^{-03}$	$1.324 \cdot 10^{-02}$
U238	W	$6.893 \cdot 10^{-01}$	$9.723 \cdot 10^{-01}$	$7.166 \cdot 10^{-01}$
Ce144	W	$5.633 \cdot 10^{-02}$	$1.986 \cdot 10^{-03}$	$1.840 \cdot 10^{-03}$
Cs137	D	$3.133 \cdot 10^{-03}$	$2.003 \cdot 10^{-05}$	$1.489 \cdot 10^{-05}$
Ru106	Y	$5.662 \cdot 10^{-02}$	$3.826 \cdot 10^{-03}$	$4.036 \cdot 10^{-03}$
Zr95	Y	$1.873 \cdot 10^{-01}$	$9.493 \cdot 10^{-04}$	$6.396 \cdot 10^{-04}$
Sr90	Y	$2.332 \cdot 10^{-03}$	$1.799 \cdot 10^{-05}$	$7.618 \cdot 10^{-05}$

The annual DCF for this mixture is 4.538 mrem/nCi. The committed DCF is 6.734 mrem/nCi. Based on the dose fractions, it can be assumed that this stream consists of depleted uranium with a specific activity of 0.4 nCi/mg and Pu-239.

The third stream is known as the second cycle plutonium stream (stream 3). As its name implies, there is much more plutonium present here thus making detection much more difficult.

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238	W	$2.045 \cdot 10^{-04}$	$3.680 \cdot 10^{-03}$	$2.293 \cdot 10^{-03}$
Pu-239	W	$6.075 \cdot 10^{-02}$	$9.838 \cdot 10^{-01}$	$7.334 \cdot 10^{-01}$
Pu-241	W	$9.194 \cdot 10^{-01}$	$1.158 \cdot 10^{-02}$	$2.643 \cdot 10^{-01}$
Ce144	W	$2.424 \cdot 10^{-03}$	$3.489 \cdot 10^{-04}$	$1.533 \cdot 10^{-05}$
Ru106	Y	$9.204 \cdot 10^{-04}$	$2.567 \cdot 10^{-04}$	$1.270 \cdot 10^{-05}$
Zr95	Y	$1.616 \cdot 10^{-02}$	$3.344 \cdot 10^{-04}$	$1.068 \cdot 10^{-05}$

The annual DCF for this stream is 1.111 mrem/nCi. The committed DCF for this stream is $3.479 \cdot 10^{+01}$ mrem/nCi.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the 221-F Canyon facility.

Worker Monitoring Program

Uranium, plutonium, and fission product worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every month for elemental uranium and one sample every six months for plutonium and isotopic uranium
- feces - one sample every year for plutonium and isotopic uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Ce-144 and Cs-137 are present in sufficient quantities in Stream 1 to make them useful as tracers for whole-body and chest counts. Annual Whole Body Counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Cs-137	2	2
Pure Ce-144	190	260
Stream 1 Cs-137 tracer	330	1500
Stream 1 Ce-144 tracer	400	1800
Stream 2 Cs-137 tracer	>250000	>250000
Stream 2 Ce-144 tracer	>250000	>250000
Stream 3 Cs-137 tracer	>250000	>250000
Stream 3 Ce-144 tracer	>250000	>250000

The table below gives the following missed doses for annual chest counts in mrem:

Nuclide	Annual	Committed
Pure Pu-239	>250000	>250000
Pure Pu-238	>250000	>250000
Pure Ce-144	120	170
Stream 1 Ce-144 tracer	260	1200
Stream 2 Ce-144 tracer	230000	>250000
Stream 3 Ce-144 tracer	>250000	>250000

Workgroup Monitoring Program

Uranium, plutonium, and fission product workgroup monitoring programs as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every year for plutonium, isotopic uranium, and elemental uranium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Ce-144 and Cs-137 may be useful as tracers (see Chapter 10).

Other Considerations

When calculating the missed dose for uranium, it is necessary to determine the ratio of uranium isotopes. In the 221-F Canyon the processes have different uranium ratios, and, since personnel are exposed to all the processes, it is impossible to determine the uranium ratios. It was noted that in every process stream in this facility that the main isotope was U-238 (Table 13-1). This is consistent with what is fed to the F-Area A-Line Facility; therefore, the missed dose was figured specifically for U-238. The missed dose is based entirely on this single isotope. It should also be noted that since the processes contain depleted uranium, that the possibility exists for a chemical toxicity problem.

Table 13-1. Mix of Radionuclides for Various Process Solutions

Table 13-1. Mix of Radionuclides for Various Process Solutions

Nuclide	Isotopic Curie Fraction					
	Head End	First Cycle	Second U Cycle	Second Pu Cycle	Waste	Ion Exchange
Sr-89	$0.7800 \cdot 10^{-01}$	$0.777 \cdot 10^{-01}$	$0.335 \cdot 10^{-01}$	$0.103 \cdot 10^{-02}$	$0.784 \cdot 10^{-01}$	$0.784 \cdot 10^{-01}$
Sr-90	$0.7400 \cdot 10^{-02}$	$0.744 \cdot 10^{-02}$	$0.321 \cdot 10^{-02}$	$0.985 \cdot 10^{-04}$	$0.751 \cdot 10^{-02}$	$0.751 \cdot 10^{-02}$
Y-90	$0.7400 \cdot 10^{-02}$	$0.744 \cdot 10^{-02}$	$0.321 \cdot 10^{-02}$	$0.985 \cdot 10^{-04}$	$0.751 \cdot 10^{-02}$	$0.751 \cdot 10^{-02}$
Y-91	$0.1200 \cdot 10^{+00}$	$0.119 \cdot 10^{-00}$	$0.516 \cdot 10^{-01}$	$0.158 \cdot 10^{-02}$	$0.121 \cdot 10^{-00}$	$0.121 \cdot 10^{-00}$
Zr-95	$0.1200 \cdot 10^{+00}$	$0.119 \cdot 10^{-00}$	$0.257 \cdot 10^{-00}$	$0.158 \cdot 10^{-01}$	$0.120 \cdot 10^{-00}$	$0.120 \cdot 10^{-00}$
Nb-95	$0.3200 \cdot 10^{-01}$	$0.319 \cdot 10^{-01}$	$0.689 \cdot 10^{-01}$	$0.105 \cdot 10^{-02}$	$0.322 \cdot 10^{-01}$	$0.322 \cdot 10^{-01}$
Ru-103	$0.5900 \cdot 10^{-01}$	$0.586 \cdot 10^{-01}$	$0.759 \cdot 10^{-01}$	$0.856 \cdot 10^{-03}$	$0.592 \cdot 10^{-01}$	$0.592 \cdot 10^{-01}$
Ru-106	$0.6000 \cdot 10^{-01}$	$0.600 \cdot 10^{-01}$	$0.777 \cdot 10^{-01}$	$0.868 \cdot 10^{-03}$	$0.606 \cdot 10^{-01}$	$0.606 \cdot 10^{-01}$
Rh-106	$0.6000 \cdot 10^{-01}$	$0.600 \cdot 10^{-01}$	$0.259 \cdot 10^{-01}$	$0.794 \cdot 10^{-03}$	$0.606 \cdot 10^{-01}$	$0.606 \cdot 10^{-01}$
Ag-110	$0.7600 \cdot 10^{-03}$	$0.758 \cdot 10^{-03}$	$0.327 \cdot 10^{-03}$	$0.100 \cdot 10^{-04}$	$0.765 \cdot 10^{-03}$	$0.765 \cdot 10^{-03}$
Sn-123	$0.1000 \cdot 10^{-02}$	$0.102 \cdot 10^{-02}$	$0.442 \cdot 10^{-03}$	$0.135 \cdot 10^{-04}$	$0.103 \cdot 10^{-02}$	$0.103 \cdot 10^{-02}$
Sb-125	$0.1300 \cdot 10^{-02}$	$0.128 \cdot 10^{-02}$	$0.554 \cdot 10^{-03}$	$0.170 \cdot 10^{-04}$	$0.130 \cdot 10^{-02}$	$0.130 \cdot 10^{-02}$
Te-127	$0.1900 \cdot 10^{-02}$	$0.194 \cdot 10^{-02}$	$0.839 \cdot 10^{-03}$	$0.257 \cdot 10^{-04}$	$0.196 \cdot 10^{-02}$	$0.196 \cdot 10^{-02}$
Te-129	$0.1200 \cdot 10^{-02}$	$0.122 \cdot 10^{-02}$	$0.526 \cdot 10^{-03}$	$0.161 \cdot 10^{-04}$	$0.123 \cdot 10^{-02}$	$0.123 \cdot 10^{-02}$
Cs-134	$0.1300 \cdot 10^{-02}$	$0.130 \cdot 10^{-02}$	$0.560 \cdot 10^{-03}$	$0.172 \cdot 10^{-04}$	$0.131 \cdot 10^{-02}$	$0.131 \cdot 10^{-02}$
Cs-137	$0.9900 \cdot 10^{-02}$	$0.995 \cdot 10^{-02}$	$0.430 \cdot 10^{-02}$	$0.132 \cdot 10^{-03}$	$0.100 \cdot 10^{-01}$	$0.100 \cdot 10^{-01}$
Ce-141	$0.3800 \cdot 10^{-01}$	$0.385 \cdot 10^{-01}$	$0.166 \cdot 10^{-01}$	$0.509 \cdot 10^{-03}$	$0.388 \cdot 10^{-01}$	$0.388 \cdot 10^{-01}$
Ce-144	$0.1800 \cdot 10^{+00}$	$0.179 \cdot 10^{-00}$	$0.773 \cdot 10^{-01}$	$0.237 \cdot 10^{-02}$	$0.181 \cdot 10^{-00}$	$0.181 \cdot 10^{-00}$
Pr-144	$0.1800 \cdot 10^{+00}$	$0.179 \cdot 10^{-00}$	$0.773 \cdot 10^{-01}$	$0.237 \cdot 10^{-02}$	$0.181 \cdot 10^{-00}$	$0.181 \cdot 10^{-00}$
Pm-147	$0.3200 \cdot 10^{-01}$	$0.321 \cdot 10^{-01}$	$0.124 \cdot 10^{-01}$	$0.425 \cdot 10^{-03}$	$0.324 \cdot 10^{-01}$	$0.324 \cdot 10^{-01}$
Pm-148	$0.3700 \cdot 10^{-03}$	$0.374 \cdot 10^{-03}$	$0.162 \cdot 10^{-03}$	$0.542 \cdot 10^{-05}$	$0.378 \cdot 10^{-03}$	$0.378 \cdot 10^{-03}$
Eu-155	$0.5600 \cdot 10^{-03}$	$0.563 \cdot 10^{-03}$	$0.243 \cdot 10^{-03}$	$0.745 \cdot 10^{-05}$	$0.568 \cdot 10^{-03}$	$0.568 \cdot 10^{-03}$
U-234	$0.1500 \cdot 10^{-08}$	$0.148 \cdot 10^{-08}$	$0.639 \cdot 10^{-05}$	$0.196 \cdot 10^{-11}$	$0.141 \cdot 10^{-12}$	$0.141 \cdot 10^{-12}$
U-235	$0.2400 \cdot 10^{-07}$	$0.237 \cdot 10^{-07}$	$0.102 \cdot 10^{-02}$	$0.314 \cdot 10^{-10}$	$0.235 \cdot 10^{-11}$	$0.235 \cdot 10^{-11}$
U-236	$0.2100 \cdot 10^{-07}$	$0.214 \cdot 10^{-07}$	$0.104 \cdot 10^{-02}$	$0.314 \cdot 10^{-10}$	$0.235 \cdot 10^{-11}$	$0.235 \cdot 10^{-11}$
U-238	$0.2200 \cdot 10^{-05}$	$0.219 \cdot 10^{-05}$	$0.946 \cdot 10^{-00}$	$0.290 \cdot 10^{-08}$	$0.235 \cdot 10^{-09}$	$0.235 \cdot 10^{-09}$
Np-237	$0.3100 \cdot 10^{-08}$	$0.315 \cdot 10^{-08}$	$0.136 \cdot 10^{-08}$	$0.417 \cdot 10^{-10}$	$0.640 \cdot 10^{-10}$	$0.318 \cdot 10^{-08}$
Pu-238	$0.1500 \cdot 10^{-05}$	$0.149 \cdot 10^{-05}$	$0.193 \cdot 10^{-04}$	$0.197 \cdot 10^{-03}$	$0.940 \cdot 10^{-10}$	$0.415 \cdot 10^{-08}$
Pu-239	$0.4500 \cdot 10^{-03}$	$0.448 \cdot 10^{-03}$	$0.580 \cdot 10^{-02}$	$0.594 \cdot 10^{-01}$	$0.270 \cdot 10^{-07}$	$0.136 \cdot 10^{-05}$
Pu-240	$0.1000 \cdot 10^{-03}$	$0.102 \cdot 10^{-03}$	$0.132 \cdot 10^{-02}$	$0.135 \cdot 10^{-01}$	$0.620 \cdot 10^{-08}$	$0.308 \cdot 10^{-06}$
Pu-241	$0.6800 \cdot 10^{-02}$	$0.679 \cdot 10^{-02}$	$0.880 \cdot 10^{-01}$	$0.899 \cdot 10^{-00}$	$0.410 \cdot 10^{-06}$	$0.206 \cdot 10^{-04}$
Pu-242	$0.9900 \cdot 10^{-08}$	$0.991 \cdot 10^{-08}$	$0.128 \cdot 10^{-06}$	$0.131 \cdot 10^{-05}$	$0.600 \cdot 10^{-12}$	$0.300 \cdot 10^{-10}$

Internal Dosimetry Program for the New Special Recovery Facility

Scrap plutonium from offsite sources and from FB-Line facility is processed through the New Special Recovery (NSR)[4] Facility. The processed plutonium is conditioned so that it can be blended back into the plutonium nitrate feed stream to the FB-Line process.

Description of Process

The following outlines the Processing of Radioactive Material on the New Special Recovery Facility.

Material Receiving and Assaying

Material for processing is brought from existing storage areas in FB-Line to the feed assay room (FAR). The feed material includes slightly impure plutonium oxide, plutonium metal, unirradiated reactor cores, scrap magnesium oxide crucibles, mixed oxides and metals, and plutonium and uranium oxide plates. The feed material is first assayed to determine its plutonium content before any processing begins.

Feed Preparation

After the feed has been analyzed in the FAR, it is introduced into the feed preparation glove box. The sequence of operations for preparation of the feed material is dependent on the type of scrap. One or more of the following operations may be required: oxidation, sawing, screening, or grinding.

Dissolution and Filtration

The dissolution and filtration steps convert the solid feed materials into a solution that is suitable for further purification by either anion exchange in the NSR facility or by Purex solvent extraction in the 221-F Canyon.

Anion Exchange Purification

Impurities such as fluoride, aluminum, americium, sulfate, and uranium are removed from the dissolver solutions by anion exchange.

Waste Handling

Solid waste generated in the facility glove boxes is transferred in cans to the waste handling glove box. This waste comes from sludge, residues, slurries, and from solid materials such as tools, filters, gloves, etc. The waste is cleaned, to a practical extent, and packaged for disposal in the burial ground.

Analytical and Assay Operations

Samples are taken from the process vessels and transported by the conveyor system to the sample assay room. Most samples are analyzed in the sample assay room and are then returned to the process without leaving the glove box confinement.

Radionuclides of Concern

The function of the NSR facility is to recover plutonium from a variety of materials. These materials also contain impurities such as americium and uranium. The isotopic content is not constant in the material or between materials (Table 13-2). For example, the unirradiated reactor cores percent weight range from 6% to 12% for Pu-240, 0% to 10% for Am-241, and 60% to 80% for uranium. The isotopes that deliver a majority of the dose are Pu-239, Pu-240, Am-241, U-235, and U-236 (assuming enriched uranium).

Recommended Bioassay Programs

The NSR facility is located on the top floors of Building 221-F, adjacent to FB-Line. The following programs were designed for and apply to all personnel assigned to the New Special Recovery facility.

Worker Monitoring Program

Plutonium, americium, and uranium worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every three months for americium; one sample every six months for isotopic uranium; one sample every 12 months for plutonium
- feces - one sample every six months for americium; one sample every 12 months for plutonium and isotopic uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

Plutonium, americium, and uranium workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every year for plutonium, americium, and isotopic uranium
- chest count - one count every year on the germanium chest counter

Other Considerations

It is assumed that the uranium material is principally enriched uranium. If this is not true, the uranium bioassay program will need to be modified for the appropriate uranium mixture.

Table 13-2 Examples of Feed Material for NSR Facility

	Percent of Isotopic Content			
Material	Pu-239	Pu-240	Am-241	Other
Impure Plutonium Oxide	87 to 94	6 to 12	0 to 1	Fe, Ni, C, U
Plutonium Metal	88 to 94	6 to 12	-	-
Unirradiated Reactor Cores	-	4 to 42	0 to 10	60 to 80 U
Scrap MgO Crucibles	88 to 94	6 to 12	0 to 1	<90 MgO
Mixed Oxides and Metals	*	*	-	3 Mo
ZPPR Pu and U Oxide Plates	2 to 88	6 to 18	-	78 U, 2 Mo

* Trace plutonium oxide in uranium oxides

Internal Dosimetry Program for F-Area Outside Facilities

The F-Area Outside Facilities provide general support, principally to the processing of irradiated fuels and targets in Building 221-F[5]. The term "Outside Facilities" is used to describe a wide variety of processes, utilities, and services that are ancillary to the primary 200-F Area operations. The main processes and services of radiological concern are Water Handling Facilities, Acid Recovery Unit (ACU), General Purpose Evaporators, General Purpose Waste Tankage, Segregated Solvent Facilities, Transfer Tanks, Sump Collection Tanks, Recycle Sumps, and Auxiliary Systems and Support Facilities.

Many of these processes and services do not have any connection to each other in terms of function in the Outside Facilities. Therefore it may, in some cases, be necessary to treat each process separately in terms of radiological protection.

Description of Process

The following outlines the processing of radioactive material at the F-Area Outside Facilities.

Water Handling Facilities

The water handling facilities are located in the Building 221-F complex between the chemical storage area and the Acid Recovery Unit (ACU). The primary equipment for water handling consists of tanks, skimmers, and coolers, all of which can contain radioactive materials. The facility is used to provide process water and acidified water streams for the canyons, retain discard water in hold tanks for analysis, and decant spent solvent from waste water.

Acid Recovery Unit (ARU)

The ARU concentrates nitric acid for reuse and is located in the Building 211-F complex between the general purpose evaporators and the water handling facilities. All material in this location can contain radioactive material.

General Purpose Evaporator

The General Purpose (GP) evaporators concentrate aqueous waste that have activity that is in excess of disposal limits yet is low enough to be evaporated in unshielded equipment. The GP evaporator is located in the building 211-F complex between the ARU and the general purpose waste tanks.

General Purpose Waste Tanks

The tanks in this facility are used to collect GP evaporator feed and to receive and dispense evaporator overheads for recycle to the process. The waste tanks receive a variety of low-level aqueous wastes from the canyon, A-Line, or building 211-F complex that do not exceed the specifications for feed to GP evaporators.

Waste Handling Facilities

The F-Area waste handling facilities are used for storage and transfer of high and low activity wastes that are primarily from the Metallurgical Building 235-F, SRL Waste Concentration Building 776-A, and the Production Control Facilities -

Building 772-F and 100-Area sources. Cold or very low-level wastes are transferred to the seepage basins by way of the general purpose waste tankage. Low-level activity wastes are sent to the GP evaporators for concentration. High-level wastes are pumped to the laboratory waste evaporator feed tank in the canyon.

Segregated Solvent Facilities

Segregated solvent facilities provide solvent purification and tank storage before the solvent is returned to F-Canyon for reuse. One of the purposes of the segregated solvent facilities is to remove the degradation end products and radioactive contaminants from the solvent.

Transfer Tanks

Eight transfer tanks in F-Area are located in four open basins along the east side of Building 221-F. These tanks provide intermediate pumping stations for those solutions transferring from building 221-F, where there is not enough power to transfer the material in a single operation.

Sump Collection Tanks

Two sump collection tanks are located in separate underground concrete vaults. One tank receives radioactive waste condensate, by steam jet, from sumps or tanks related to the canyon air exhaust system and transfers the waste to the hot or warm canyon waste evaporator feed tanks. Discharges to hot canyon cell sumps are first transferred to the second sump collection tank and then to the canyon for analysis and disposition.

Recycle Sump

The recycle sump is an underground stainless steel-lined concrete sump that is located between the ARU and the GP evaporators. The recycle sump collects drainage and overflow from all canyon auxiliary tanks that hold contaminated or recycled liquids.

Auxiliary Systems and Support Facilities

The only auxiliary system and support facility of concern is the plant laundry located in Building 723-F. All plant areas send soiled laundry to Building 723-F. Soiled clothing enters the laundry through the north receiving door and follows a relatively straight-line process until the clothing is loaded onto the trucks for return to the using departments. The sequence of laundering steps includes: soiled laundry storage, washing, spinning damp dry, drying, inspecting and monitoring, sorting, mending, and clean laundry storage.

Laundered items include coats, coveralls, cloth shoe covers, caps, hoods, rubber overshoes, rubber gloves and surgeons gloves. Laundry waste water is filtered and collected in waste tanks located at the laundry. These tanks are normally connected to process water lines that discharge to Four Mile Creek; however, laundry waste water can also be sent to the GP evaporators.

Radionuclides of Concern

The Safety Analysis Report for F-Area Outside Facility Operations considers six areas of radiological concern: the ARU, the GP evaporators, the segregated solvent (plutonium and uranium), laboratory wastes, water handling, and the basin transfer tanks.

The radionuclides of concern for the ARU, water handling, and the basin transfer tanks are Zr-95 and Ru-106 as shown in the following tables. The radionuclide distribution in the basin transfer tanks is assumed to be similar to the water handling distribution.

The ARU Radionuclide Distribution is as follows:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$7.040 \cdot 10^{-11}$	$3.980 \cdot 10^{-08}$	$1.405 \cdot 10^{-07}$
U235	Y	$1.012 \cdot 10^{-09}$	$5.297 \cdot 10^{-07}$	$1.864 \cdot 10^{-06}$
U236	Y	$1.008 \cdot 10^{-09}$	$5.489 \cdot 10^{-07}$	$2.012 \cdot 10^{-06}$
U238	Y	$1.008 \cdot 10^{-07}$	$5.067 \cdot 10^{-05}$	$1.857 \cdot 10^{-04}$
Ru103	Y	$6.950 \cdot 10^{-02}$	$1.310 \cdot 10^{-02}$	$9.600 \cdot 10^{-03}$
Ru106	Y	$1.023 \cdot 10^{-01}$	$6.638 \cdot 10^{-01}$	$7.534 \cdot 10^{-01}$
Zr95	Y	$6.176 \cdot 10^{-01}$	$2.974 \cdot 10^{-01}$	$2.180 \cdot 10^{-01}$
Nb95	Y	$2.107 \cdot 10^{-01}$	$2.558 \cdot 10^{-02}$	$1.875 \cdot 10^{-02}$

The annual dose conversion factor (DCF) for the mixture is $4.776 \cdot 10^{-02}$ mrem/nCi. The committed DCF for the mixture is $6.515 \cdot 10^{-02}$ mrem/nCi.

The water handling system distribution is as follows:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Ru103	Y	$1.417 \cdot 10^{-01}$	$8.229 \cdot 10^{-03}$	$5.385 \cdot 10^{-03}$
Ru106	Y	$4.815 \cdot 10^{-01}$	$9.633 \cdot 10^{-01}$	$9.760 \cdot 10^{-01}$
Zr95	Y	$1.298 \cdot 10^{-01}$	$1.926 \cdot 10^{-02}$	$1.260 \cdot 10^{-02}$
Nb95	Y	$2.470 \cdot 10^{-01}$	$9.246 \cdot 10^{-03}$	$6.050 \cdot 10^{-03}$

The annual dose conversion factor (DCF) for the mixture is $1.550 \cdot 10^{-01}$ mrem/nCi. The committed DCF for the mixture is $2.368 \cdot 10^{-01}$ mrem/nCi.

The radionuclide of concern for the GP evaporators, Segregated Plutonium Solvent, and Laboratory Wastes is plutonium as shown in the following tables.

The GP Evaporation distribution:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$9.626 \cdot 10^{-05}$	$3.195 \cdot 10^{-03}$	$2.497 \cdot 10^{-03}$
Pu-239 Y	$2.903 \cdot 10^{-02}$	$8.734 \cdot 10^{-01}$	$7.792 \cdot 10^{-01}$
Pu-241 Y	$3.913 \cdot 10^{-01}$	$1.055 \cdot 10^{-02}$	$2.030 \cdot 10^{-01}$
U234 Y	$2.911 \cdot 10^{-08}$	$8.154 \cdot 10^{-07}$	$3.386 \cdot 10^{-07}$
U235 Y	$4.813 \cdot 10^{-07}$	$1.248 \cdot 10^{-05}$	$5.167 \cdot 10^{-06}$
U236 Y	$4.343 \cdot 10^{-07}$	$1.171 \cdot 10^{-05}$	$5.051 \cdot 10^{-06}$
U238 Y	$4.343 \cdot 10^{-05}$	$1.081 \cdot 10^{-03}$	$4.663 \cdot 10^{-04}$
Ru103 Y	$4.852 \cdot 10^{-02}$	$4.530 \cdot 10^{-04}$	$3.907 \cdot 10^{-05}$
Ru106 Y	$3.369 \cdot 10^{-01}$	$1.083 \cdot 10^{-01}$	$1.447 \cdot 10^{-02}$
Zr95 Y	$9.743 \cdot 10^{-02}$	$2.325 \cdot 10^{-03}$	$2.005 \cdot 10^{-04}$
Nb95 Y	$9.665 \cdot 10^{-02}$	$5.815 \cdot 10^{-04}$	$5.015 \cdot 10^{-05}$

The annual dose conversion factor (DCF) for the mixture is $9.640 \cdot 10^{-01}$ mrem/nCi. The committed DCF for the mixture is $1.118 \cdot 10^{+01}$ mrem/nCi.

The Segregated Plutonium solvent distribution:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$2.068 \cdot 10^{-04}$	$3.738 \cdot 10^{-03}$	$2.613 \cdot 10^{-03}$
Pu-239 Y	$5.941 \cdot 10^{-02}$	$9.730 \cdot 10^{-01}$	$7.763 \cdot 10^{-01}$
Pu-241 Y	$8.701 \cdot 10^{-01}$	$1.278 \cdot 10^{-02}$	$2.198 \cdot 10^{-01}$
Ru103 Y	$2.418 \cdot 10^{-03}$	$1.229 \cdot 10^{-05}$	$9.478 \cdot 10^{-07}$
Ru106 Y	$5.941 \cdot 10^{-02}$	$1.040 \cdot 10^{-02}$	$1.242 \cdot 10^{-03}$
Zr95 Y	$5.941 \cdot 10^{-03}$	$7.717 \cdot 10^{-05}$	$5.952 \cdot 10^{-06}$
Nb95 Y	$2.532 \cdot 10^{-03}$	$8.293 \cdot 10^{-06}$	$6.396 \cdot 10^{-07}$

The annual dose conversion factor (DCF) for the mixture is 1.771 mrem/nCi. The committed DCF for the mixture is $2.296 \cdot 10^{+01}$ mrem/nCi.

The Laboratory waste distribution:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238	Y	$6.150 \cdot 10^{-01}$	$9.719 \cdot 10^{-01}$	$9.582 \cdot 10^{-01}$
Pu-239	Y	$1.921 \cdot 10^{-02}$	$2.751 \cdot 10^{-02}$	$3.097 \cdot 10^{-02}$
Pu-241	Y	$3.456 \cdot 10^{-01}$	$4.437 \cdot 10^{-04}$	$1.077 \cdot 10^{-02}$
U234	Y	$4.803 \cdot 10^{-08}$	$6.404 \cdot 10^{-08}$	$3.355 \cdot 10^{-08}$
U235	Y	$7.497 \cdot 10^{-07}$	$9.256 \cdot 10^{-07}$	$4.833 \cdot 10^{-07}$
U236	Y	$7.209 \cdot 10^{-07}$	$9.257 \cdot 10^{-07}$	$5.036 \cdot 10^{-07}$
U238	Y	$7.497 \cdot 10^{-05}$	$8.885 \cdot 10^{-05}$	$4.833 \cdot 10^{-05}$
Ce141	W	$1.369 \cdot 10^{-04}$	$5.612 \cdot 10^{-08}$	$6.105 \cdot 10^{-09}$
Ce144	W	$9.892 \cdot 10^{-03}$	$7.817 \cdot 10^{-05}$	$1.169 \cdot 10^{-05}$
Cs134	D	$1.822 \cdot 10^{-04}$	$3.869 \cdot 10^{-07}$	$4.502 \cdot 10^{-08}$
Cs137	D	$1.590 \cdot 10^{-03}$	$2.277 \cdot 10^{-06}$	$2.733 \cdot 10^{-07}$
Ru103	Y	$1.369 \cdot 10^{-04}$	$6.085 \cdot 10^{-08}$	$6.620 \cdot 10^{-09}$
Ru106	Y	$8.645 \cdot 10^{-04}$	$1.323 \cdot 10^{-05}$	$2.229 \cdot 10^{-06}$
Zr95	Y	$6.150 \cdot 10^{-03}$	$6.985 \cdot 10^{-06}$	$7.600 \cdot 10^{-07}$
Nb95	Y	$1.237 \cdot 10^{-03}$	$3.542 \cdot 10^{-07}$	$3.853 \cdot 10^{-08}$

The annual dose conversion factor (DCF) for the mixture is $2.025 \cdot 10^{+01}$ mrem/nCi. The committed DCF for the mixture is $1.861 \cdot 10^{+02}$ mrem/nCi.

The radionuclides of concern for the Segregated Uranium Solvent are Ru-106 and uranium as shown in the following table.

Segregated Uranium solvent distribution is:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$3.371 \cdot 10^{-07}$	$4.171 \cdot 10^{-05}$	$1.141 \cdot 10^{-04}$
U235	Y	$5.941 \cdot 10^{-06}$	$6.807 \cdot 10^{-04}$	$1.857 \cdot 10^{-03}$
U236	Y	$5.775 \cdot 10^{-06}$	$6.880 \cdot 10^{-04}$	$1.955 \cdot 10^{-03}$
U238	Y	$5.775 \cdot 10^{-04}$	$6.351 \cdot 10^{-02}$	$1.805 \cdot 10^{-01}$
Ru103	Y	$1.491 \cdot 10^{-01}$	$6.150 \cdot 10^{-03}$	$3.496 \cdot 10^{-03}$
Ru106	Y	$6.417 \cdot 10^{-01}$	$9.116 \cdot 10^{-01}$	$8.023 \cdot 10^{-01}$
Zr95	Y	$1.491 \cdot 10^{-01}$	$1.572 \cdot 10^{-02}$	$8.933 \cdot 10^{-03}$
Nb95	Y	$5.941 \cdot 10^{-02}$	$1.579 \cdot 10^{-03}$	$8.975 \cdot 10^{-04}$

The annual dose conversion factor (DCF) for the mixture is $2.182 \cdot 10^{-01}$ mrem/nCi. The committed DCF for the mixture is $3.840 \cdot 10^{-01}$ mrem/nCi.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the F-Area Outside facilities.

Worker Monitoring Program

A fission product worker monitoring program as given in Chapter 10 should be instituted for ARU, water handling, and the basin transfer tank facilities. This program requires the following:

- whole-body count - one count every six months

In addition the Segregated Uranium Solvent facility should institute a uranium worker monitoring program. This program requires the following:

- urine - one sample every six months for elemental uranium
- feces - one sample every 12 months for uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

A plutonium worker monitoring program, as given in Chapter 10, should be instituted for the GP evaporators, Segregated Plutonium Solvent, and Laboratory Wastes facilities. This program requires the following:

- urine - one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

A fission product workgroup monitoring program, as given in Chapter 10, should be instituted for ARU, water handling, and the basin transfer tank facilities. This program requires the following:

- whole-body count - one count every six months

In addition the Segregated Uranium Solvent facility should institute a uranium workgroup monitoring program. This program requires the following:

- urine - one sample every six months for elemental uranium
- chest count - one count every year on the germanium chest counter

A plutonium workgroup monitoring program, as given in Chapter 10, should be instituted for the GP evaporators, Segregated Plutonium Solvent, and Laboratory Wastes facilities. This program requires the following:

- urine - one sample every 12 months for plutonium
- chest count - one count every year on the germanium chest counter

Other Considerations

Specific information on the plant laundry, located in Building 723-F, is not available at this time. F-Area Health Protection should establish worker and workgroup monitoring programs based on their knowledge of the facility and on the bioassay monitoring programs given in Chapter 10.

Internal Dosimetry Program for the $^{238}PuO_2$ Fuel Form Facility (PuFF) and the $^{238}PuO_2$ Experimental Facility (PEF)

The Plutonium Fuel Form facility (PuFF) and the Plutonium Experimental Facility (PEF) are located on the first and second floors of Building 235-F[6,7]. The primary function of the PuFF facility is to produce encapsulated plutonium oxide fuel forms. This is accomplished by converting Pu-238 oxide powder into fuel forms for heat sources by powder ceramic and metallurgical process. The final products are compacted PuO_2 fuel shapes. The PEF provides space for the testing of processes to be used in the PuFF facility.

Description of Process

The following outlines the processing of radioactive material at the PuFF and PEF.

PuFF Process Description

The PuO_2 is fabricated and encapsulated in six manipulator cells filled with inert gas, three air-filled manipulator cells, five wing cabinets filled with inert gas, and one hood. The process enclosures are located on the first floor of the PuFF facility. The manipulator cells are divided into two parallel and facing lines with an operating area between and separate maintenance area behind the lines; they are separately vented, and interconnected by transfer locks to protect the purity of cell gas during intercell transfer. An enclosed conveyor in the transfer tunnel beneath the floor connects the two cell lines.

The fuel pellets are made by hot pressing a blended PuO_2 shard mixture prepared from calcined Pu oxalate powder received from HB-Line. Feed processing includes oxygen-16 enrichment, ball milling, compaction, granulation, and sintering. After final heat treatment, the fuel pellets are encapsulated in iridium-clad vent sets by tungsten inert gas welding. After decontamination of the exterior surface of the iridium and welding the primary shipping container, the encapsulated fuel pellets are shipped offsite.

The second floor area immediately above the PuFF operating and maintenance area is occupied by auxiliary equipment which includes ventilation system components, such as HEPA filters, blowers, and ducts; gas recirculators for the inert gas cells; a metallographic laboratory; electrical controls; equipment associated with the process and chilled water cooling loops; vacuum systems and hydraulic systems; and a motor-generator for the hot press.

PEF Process Description

The PEF processes involve converting plutonium oxide powder into fuel forms for heat sources by using powder ceramic and metallurgical processes. The final products are dense PuO_2 fuel forms.

The test fuel (Pu-239 and Pu-238) is fabricated in thirteen glove boxes of which eight are supplied with recirculated inert gas. The interconnected glove boxes are located on the first floor of Building 235-F.

The second floor which is immediately above the PEF Operating and Maintenance areas is occupied by auxiliary equipment, which includes ducts to the HEPA

exhaust filters, electrical distribution panels, equipment associated with the process and chilled water cooling loops, hydraulic systems, and a motor generator for the hot press.

Radionuclide Content of Material

The plutonium oxide blend, used in the PuFF and PEF facilities are very similar and can be represented by the distribution shown in the following table:

Nuclide	Specific Activity (Ci/g)	Activity Fraction
Pu-238	$1.66 \cdot 10^{+1}$	0.8350
Pu-239	$6.13 \cdot 10^{-2}$	0.1380
Pu-240	$2.27 \cdot 10^{-1}$	0.0200
Pu-241	$1.12 \cdot 10^{+2}$	0.0041
Am-241	$3.21 \cdot 10^{+0}$	0.0003
Pu-242	$3.85 \cdot 10^{-3}$	0.0016
Np-237	$6.87 \cdot 10^{-4}$	0.0005
Th-232	$1.11 \cdot 10^{-7}$	0.0005

The particle size distributions of ball-milled and feed powders, as measured with a Coulter Counter, showed the mass median diameters to be 2μm and 5μm respectively.

Radionuclides of Concern

The majority of the annual and committed dose from the PuFF and PEF material comes from plutonium, as expected. This is shown in the following table:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Am241 W	$3.067 \cdot 10^{-04}$	$1.952 \cdot 10^{-04}$	$4.742 \cdot 10^{-04}$
Pu-238 Y	$8.535 \cdot 10^{-01}$	$8.691 \cdot 10^{-01}$	$8.506 \cdot 10^{-01}$
Pu-239 Y	$1.411 \cdot 10^{-01}$	$1.302 \cdot 10^{-01}$	$1.454 \cdot 10^{-01}$
Pu-241 Y	$4.089 \cdot 10^{-03}$	$3.383 \cdot 10^{-06}$	$8.149 \cdot 10^{-05}$
Np237 W	$5.111 \cdot 10^{-04}$	$2.602 \cdot 10^{-04}$	$6.498 \cdot 10^{-04}$
Th232 W	$5.111 \cdot 10^{-04}$	$2.602 \cdot 10^{-04}$	$2.810 \cdot 10^{-03}$

Recommended Bioassay Programs

The PuFF and PEF facilities are located on the first and second floors of Building 235-F. The following programs apply to all personnel assigned to the PuFF and PEF facilities.

Worker Monitoring Program

The plutonium worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium

- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

The plutonium workgroup monitoring program, as given in Chapter 10, should be instituted. This program would require the following:

- urine - one sample every 12 months for plutonium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Surveillance Program for 235-F Vaults

The 235-F facility receives plutonium or plutonium dioxide and fabricates it into a variety of items used in the nuclear program[3]. Vaults were built to store plutonium at different steps prior to completion. These vaults were not only created for safety reasons, but strict safeguards were set up to for security purposes. There have also been times when neptunium, thorium, and uranium have been stored there.

Description of Process

There are three vaults located in the 235-F facility: Plutonium Oxide Vault, Finished Product Vault, and Scrap Vault.

Plutonium Oxide Vault

The Plutonium Oxide Vault is used for storage of plutonium oxide powder from the Plutonium Fuel Form (PuFF) facility, Plutonium Experimental facility (PEF), and the Actinide Billet Fabrication facility. Np-237 powder and billets is also stored here. The materials are stored in approved storage containers that are surveyed for contamination prior to being placed in the vault. Because of the surveys of containers prior to vault entry, this vault is considered and maintained as a contamination free area.

Finished Product Vault

The Finished Product Vault is used for storage of uranium and transuranics in powder form like the Plutonium Oxide Vault. In addition, this vault contains a water cooled, shielded storage area for the storage of finished Pu-238 products. These products are in the form of canned encapsulated pellets/spheres welded into cans for shipping and storage and produced in PuFF or PEF.

Scrap Vault

Metal and oxide scraps of uranium, thorium, and plutonium that are in the form of chips, sweepings, powder, blend samples, or metal turnings are stored in the Scrap vault. This vault is maintained as a regulated but contamination free area.

Radionuclides of Concern

Radioisotopes that are stored and found in the 235-F vaults are Pu-238, Pu-239, U-235, and Np-237. More detailed information on the composition of these materials is not available and they will, therefore, be considered pure.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the 235-F Vault facility.

Worker Monitoring Program

Uranium, plutonium, and neptunium worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every quarter for neptunium and one sample every six months for isotopic uranium and plutonium

- feces - one sample every six months for neptunium and one sample every year for uranium and plutonium
- PAS - in lieu of or in addition to feces bioassay
- hest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

Uranium, plutonium, and neptunium workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every year for plutonium, isotopic uranium, and neptunium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Surveillance Program for 772-F and 772-1F Production Control Laboratories

The production control laboratories 772-F and 772-1F are located near the 221-F Canyon facility. The new facility, 772-1F, was built to accommodate work transferred from 772-F while it is refurbished[8,9].

Description of Process

The production control laboratories provide analytical support for 200-F and 200-H separations processes. Support is also provided to Waste Management, Reactors, and Raw Materials.

Radionuclides of Concern

The radionuclides of concern at the production control laboratories are plutonium, uranium, and fission products.

Recommended Bioassay Programs

The following programs apply to all personnel assigned to the 772-F and 772-1F Production control laboratories.

Worker Monitoring Program

Uranium, plutonium, and fission product worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every month for elemental uranium and one sample every six months for plutonium and isotopic uranium
- feces - one sample every year for plutonium and isotopic uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Workgroup Monitoring Program

Uranium, plutonium, and fission product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every year for plutonium, isotopic uranium, and elemental uranium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Internal Dosimetry Program for 200-Area Burial Grounds

Solid and liquid low-level radioactive wastes generated at SRS and SRL and some wastes from offsite are stored at the SRS Burial Ground[10]. Solid waste is packaged in 55-gal drums, carbon steel engineered boxes, shipping containers, or not at all depending on the radiation level and potential for reuse (such as large equipment).

Solid wastes are located above and below ground. Most of the solid waste above ground is TRU waste, mainly Pu-238 and Pu-239 contaminated waste, and is packaged in galvanized drums, carbon steel boxes or in concrete containers. Contaminated process equipment contained in steel boxes is stored above ground until its future reuse. The solid non-TRU waste located below ground is contained in carbon steel boxes, drums, concrete or uncontained depending on its type and form.

Liquid wastes are stored at the Burial Grounds in below ground tanks. Thirty-two tanks have been used to store liquids. As of 1988, only three tanks still contain liquid wastes.

Description of Process

The liquid and solid wastes received at the Burial Grounds comes from a variety of both onsite and offsite sources. The waste is classified according to its type and form.

Solid Waste

Solid radioactive waste is classified as low-level alpha and beta-gamma waste (LLBG), intermediate-level beta-gamma waste (ILBG), low-level alpha wastes (LLA), intermediate-activity transuranic (TRU) waste, high-activity TRU waste and temporarily stored contaminated equipment.

Liquid Waste

Liquid radioactive waste is in the form of degraded organic process solvents, waste oil, scintillation solutions and contaminated hazardous waste chemicals. Tanks S-23, S-27 and S-30 contain the majority of the liquid waste (spent solvent).

Radionuclides of Concern

Due to the diverse nature of the radioactive waste, it would be inappropriate to try to define the radionuclide contents of each type of waste. Small quantities of certain radionuclides may be spread over large areas in the burial grounds while others may be concentrated in smaller locations. After reviewing the Safety Analysis Report for the Burial Grounds, it was determined that the radionuclides with the highest concentration (Ci/vol) are tritium, cobalt, plutonium, and curium.

Recommended Bioassay Programs

The Burial Ground is located in the areas labeled 643-G and 643-7G between the F-Area and H-Area separations facilities. The old Burial Ground is designated as 643-G and ceased operations in 1972. Burial operations were phased in during 1969 to 1972 in the present Burial Ground designated as 643-7G.

The following programs apply to all personnel assigned to 643-G and 643-7G.

Worker Monitoring Program

Tritium, curium, plutonium, and activation products worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine – one sample every month for tritium; one sample every 3 months for curium; one sample every 12 months for plutonium
- feces – one sample every six months for curium; one sample every 12 months for plutonium
- PAS – in lieu of or in addition to fecal bioassay
- chest count – one count every year on the germanium chest counter
- whole-body count – one count every six months

Workgroup Monitoring Program

Tritium, curium, plutonium, and activation products workgroup monitoring programs as given in Chapter 10 should be instituted. These programs require the following:

- urine – one sample every month for tritium; one sample every year for plutonium
- chest count – one count every year on the germanium chest counter
- whole-body count – one count every six months

Internal Dosimetry Surveillance Program for Plutonium Storage Facility (PSF)

The Plutonium Storage Facility (PSF) is located in the 200-F Separations Area and is used to receive, store, monitor, retrieve, and ship packaged plutonium as needed[11]. Plutonium in solid and powder form (such as scrap), contained in approved shipping containers may be temporarily stored there prior to shipment.

Description of Process

The central area for loading and unloading shipping containers is known as the vault. Within the vault are areas known as the staging area where the loading and unloading takes place, the primary shipping container storage area, and the nondestructive assay station where the containers both incoming and outgoing are tested for material quantity.

The Receipts Assay Facility is the primary location for material accountability. Specifically, it is the area used for the unpacking of offsite material. Material from offsite is unpacked in a shielded glovebox, the inner package is removed, and it is passed on to the can loading station where the inner package is weighed, labeled and sealed then sent to the nondestructive assay laboratory.

Material in the PSF is ultimately transferred somewhere in the interim; material is stored using an automated retrieval system. This system places material into storage, retrieves material from storage, and maintains an inventory of material so that criticality cannot occur due to human error.

Radionuclides of Concern

The radionuclide of concern in PSF is plutonium. Many different mixtures of plutonium isotopes are possible, but we will assume either 6% Pu or 12% Pu (the percent refers to the mass percent of Pu-240).

The activity fractions of alpha-emitting plutonium (α-Pu) isotopes found in a typical 6% plutonium button from 221-F B-Line[12] are

Isotope	Activity Fraction
Pu-238	0.048
Pu-239	0.776
Pu-240	0.176

The ratio of Pu-241 to α-Pu is 8.0 to 1.

The activity fractions of alpha-emitting plutonium (α-Pu) isotopes found in a typical 12% plutonium button from 221-F B-Line[12] are

Isotope	Activity Fraction
Pu-238	0.266
Pu-239	0.466
Pu-240	0.268

The ratio of Pu-241 to α-Pu is 23 to 1.

Recommended Bioassay Programs

The following programs apply to all personnel assigned to the Plutonium Storage Facility.

Worker Monitoring Program

A plutonium worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every year for plutonium
- feces - one sample every year for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

A plutonium workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every year for plutonium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Program for the F/H Effluent Treatment Facility (ETF)

The F/H Effluent Treatment Facility (ETF) consists of large impermeable storage basins with lift stations located in both F and H Areas and a treatment facility, Building 241-84H, located in H Area[13]. ETF is designed to remove hazardous chemical and radioactive contaminants from the 200-Area liquid effluent waste water streams and concentrate them for disposal. The effluent processing steps include feed adjustment, filtration, reverse osmosis, ion exchange, and evaporation.

Description of Process

This section gives an overview of the processes that must be monitored at the F/H Effluent Treatment Facility.

Feed Adjustment

The 200-Area effluent stream is pretreated to enhance the ETF process and eliminate potential process problems. Pretreatment operations consist of oxidation, grit collection and removal, and pH adjustment.

Filtration

The filtration system consists of three parallel trains with three stages of filtration. Feed is pressurized by the filter pumps and passes through the tubular, ceramic, crossflow elements. At appropriate intervals, the filters are backpulsed with air to remove the deposited materials.

Reverse Osmosis

The bulk of the solid material in the 200-Area liquid effluent wastes is removed using a three-stage reverse osmosis process. Reverse osmosis concentrates dissolved solids without adding material to the flow stream. The water that passes through the membranes is directed to the ion exchange pH adjustment tank. The concentrated liquid on the upstream side of the third membrane, containing the dissolved material which the membranes have concentrated, is either recycled or directed to the evaporator feed tank.

Ion Exchange

The permeate from the reverse osmosis process is the feed solution to a system of four cation columns and two anion columns. The cation and anion resins remove ionic species of radionuclides and serve as a final polishing step for the process. One of the cation columns is specifically designed for mercury removal. The anion columns are normally offline and are operated only when F/H Area operations generate large amounts of iodine that must be removed.

Evaporation

Feed for the evaporation process consists of concentrate from the filtration and reverse osmosis systems, regeneration solution from the ion exchange process, and

the back flushes from the filtration and reverse osmosis processes. Evaporator condensate is collected in a condensate receiver and pumped to one of two overhead tanks for sampling and analysis. The overheads are transferred to the treated water tanks or recycled to the wastewater collection tanks for reprocessing. The evaporator bottoms are sent to the H-Area waste tank farm for transfer to the Saltstone Facility.

Radionuclides of Concern

The Safety Assessment for F/H Effluent Treatment Facility was reviewed and the radionuclide distribution was determined to be representative of the F/H effluent material. Lung solubility classes were chosen to maximize the missed committed dose.

The majority of the annual and committed dose from the F/H effluent material comes from plutonium, cerium, and ruthenium, as shown in the following table:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$2.855 \cdot 10^{-04}$	$7.251 \cdot 10^{-02}$	$3.188 \cdot 10^{-01}$
Pu-239 Y	$2.515 \cdot 10^{-06}$	$5.788 \cdot 10^{-04}$	$2.905 \cdot 10^{-03}$
Pu-241 Y	$8.401 \cdot 10^{-04}$	$1.733 \cdot 10^{-04}$	$1.876 \cdot 10^{-02}$
Ce144 W	$4.690 \cdot 10^{-01}$	$5.955 \cdot 10^{-01}$	$3.972 \cdot 10^{-01}$
Cs134 D	$1.340 \cdot 10^{-02}$	$4.572 \cdot 10^{-03}$	$2.373 \cdot 10^{-03}$
Cs137 D	$2.723 \cdot 10^{-02}$	$6.265 \cdot 10^{-03}$	$3.354 \cdot 10^{-03}$
Ru106 Y	$7.763 \cdot 10^{-02}$	$1.910 \cdot 10^{-01}$	$1.435 \cdot 10^{-01}$
Zr95 Y	$2.382 \cdot 10^{-01}$	$4.348 \cdot 10^{-02}$	$2.109 \cdot 10^{-02}$
Sr89 Y	$1.601 \cdot 10^{-01}$	$5.207 \cdot 10^{-02}$	$2.526 \cdot 10^{-02}$
Sr90 Y	$1.335 \cdot 10^{-02}$	$3.389 \cdot 10^{-02}$	$6.680 \cdot 10^{-02}$

The annual dose conversion factor (DCF) for the mixture is $1.260 \cdot 10^{-01}$ mrem/nCi. The committed DCF for the mixture is $2.598 \cdot 10^{-01}$ mrem/nCi.

Recommended Bioassay Programs

The following bioassay programs were designed for and apply to all personnel assigned to the F/H Effluent Treatment Facility.

Worker Monitoring Program

Plutonium and fission/activation product worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Workgroup Monitoring Program

Plutonium and fission/activation product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Internal Dosimetry Program for the F/H Seepage and Retention Basins

The F and H Areas consist of chemical separations plants and appropriate support facilities for the production of purified radionuclides, principally uranium, plutonium, and tritium[14]. The seepage and retention basins are support facilities for the separations areas. Low-level radioactive liquid wastes from the separations areas are routed to seepage and retention basins.

Description of Process

The following outlines the processing of radioactive material at the Seepage and Retention basins.

Seepage Basins

Both F and H Areas use seepage basins for low-level liquid waste disposal. These seepage basins are operated in a cascade arrangement. In F Area, liquid waste is discharged from the terra-cotta transport line to the basin. Liquid then overflows from Basin 1 to Basin 2 to Basin 3.

In H Area, liquid waste is discharged into Basin 1, overflows into Basin 2 to Basin 4 because Basin 3 is no longer used. As water migrates from the basins by seepage, the movement of most radioactive elements is slowed by ion exchange and sorption of the soil. Tritium, which flows with the ground water, is a major exception.

Retention Basins

Both F and H Area use a retention basin when it is necessary for temporary storage of process cooling water or storm drainage that may be contaminated.

When radioactivity is encountered in a cooling water system, immediate action is taken to divert water and isolate the leak. When flushing has reduced the level of radioactivity to within the guidelines for release to seepage basins, diversion to the retention basin is terminated.

Storm water is automatically diverted to the retention basin when radiation monitors detect contamination levels greater than that specified in process operating limits. After sampling and analysis, the water in the retention basin may be processed further or transferred to a seepage basin or to Four Mile Creek.

Radionuclides of Concern

The liquid waste sent to the seepage and retention basins comes from several different facilities and processes such as, cesium removal columns, tritium facilities, and overhead condensate. The Safety Analysis Report for the Seepage and Retention Basins was reviewed and a radionuclide distribution for each basin was obtained.

The majority of the annual and committed dose from the F-Area basins comes from plutonium and uranium:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$7.007 \cdot 10^{-03}$	$8.592 \cdot 10^{-02}$	$1.122 \cdot 10^{-01}$
Pu-239 Y	$2.803 \cdot 10^{-02}$	$3.115 \cdot 10^{-01}$	$4.644 \cdot 10^{-01}$
U238 Y	$6.206 \cdot 10^{-02}$	$5.708 \cdot 10^{-01}$	$4.113 \cdot 10^{-01}$
Ce144 Y	$1.001 \cdot 10^{-03}$	$9.973 \cdot 10^{-05}$	$2.101 \cdot 10^{-05}$
Cs137 D	$7.047 \cdot 10^{-01}$	$7.831 \cdot 10^{-03}$	$1.246 \cdot 10^{-03}$
Ru106 Y	$7.508 \cdot 10^{-02}$	$8.918 \cdot 10^{-03}$	$1.990 \cdot 10^{-03}$
Sr90 Y	$1.221 \cdot 10^{-01}$	$1.498 \cdot 10^{-02}$	$8.768 \cdot 10^{-03}$

The activity fraction is the fraction of the total activity represented by a particular radionuclide. The annual dose fraction is the fraction of the annual dose conversion factor (DCF) represented by a particular radionuclide. The committed dose fraction is the fraction of the DCF represented by a particular radionuclide.

The annual DCF for this mixture is 2.610 mrem/nCi. The committed DCF for this mixture is $1.811\ 10^{+01}$ mrem/nCi.

The majority of the dose from the H-Area basins also comes from plutonium and uranium:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$1.403 \cdot 10^{-02}$	$3.948 \cdot 10^{-01}$	$4.352 \cdot 10^{-01}$
Pu-239 Y	$1.303 \cdot 10^{-02}$	$3.322 \cdot 10^{-01}$	$4.180 \cdot 10^{-01}$
U238 Y	$9.018 \cdot 10^{-03}$	$1.904 \cdot 10^{-01}$	$1.158 \cdot 10^{-01}$
Ce144 Y	$1.403 \cdot 10^{-02}$	$3.208 \cdot 10^{-03}$	$5.702 \cdot 10^{-04}$
Cs137 D	$7.335 \cdot 10^{-01}$	$1.871 \cdot 10^{-02}$	$2.511 \cdot 10^{-03}$
Ru106 Y	$2.405 \cdot 10^{-02}$	$6.557 \cdot 10^{-03}$	$1.235 \cdot 10^{-03}$
Sr90 Y	$1.924 \cdot 10^{-01}$	$5.414 \cdot 10^{-02}$	$2.675 \cdot 10^{-02}$

The annual DCF for this mixture is 1.137 mrem/nCi. The committed DCF is 9.349 mrem/nCi.

Recommended Bioassay Programs

For the purpose of designing an internal dosimetry program, the Seepage and Retention Basin area is defined as the three seepage basins (identified as 904-41G, 904-42G, and 904-43G) located near F Area, the four seepage basins (identified as 904-44G, 904-45G, 904-46G, and 904-56G) located near H Area, and the retention basins in F Area (281-8F) and H Area (281-8H). The following programs apply to all personnel assigned to these areas.

Worker Monitoring

Uranium and plutonium worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every six months for isotopic uranium, and elemental uranium; one sample every 12 months for plutonium
- feces - one sample every 12 months for uranium and plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Cs-137 is present in sufficient quantities in both basins to make it useful as a tracer for whole-body counts. Annual whole-body counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Cs-137	2	2
F-Area Basin, Cs-137 as tracer	200	1400
H-Area Basin, Cs-137 as tracer	84	690

Workgroup Monitoring

Uranium and plutonium workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium, isotopic uranium, and elemental uranium
- chest count - one count every year on the germanium chest counter

F/H-Area Tank Farms

High level liquid radioactive wastes from SRS are received and managed in large underground tanks in areas called waste tank farms located in 241-F and 241-H[15]. The two farms contain 51 large subsurface tanks and related facilities required for safe handling, processing and temporary retention of the liquid wastes. The liquid waste comes primarily from fuel reprocessing operations in F and H Areas, with smaller quantities from miscellaneous sources. Liquid wastes are classified as either high heat waste (HHW) or low heat waste (LHW) depending on whether or not the concentration of radioactivity is such that forced cooling is required to maintain waste temperature within operating guidelines.

Radioactive Materials Processed

The following outlines the processing of radioactive materials at the F- and H-Area tank farms.

Types and Sources of Liquid Waste

Liquid waste received in the waste storage facilities comes primarily from the fuel reprocessing operations in F and H Areas with quantities from RBOF-RRF (Receiving Basin for Offsite Fuel and Resin Regeneration Facility) Operations, SRL and the 100 Areas.

F-Area and H-Area Reprocessing Waste

Liquid waste from buildings 221-F and 221-H is made of many waste streams generated during the recovery and purification of the reprocessing operation. The HHW from 221-F is transferred to pump Tank 3 and then to receipt Tank 33 or 34. The F-Area LHW is transferred to pump Tank 2 and to evaporator feed Tank 26. In H Area, the HHW from 221-H is transferred to pump Tank 5 and then to receipt Tank 39. The H-Area LHW is transferred to pump Tank 6 to evaporator feed Tank 43.

RBOF-RRF Liquid Waste

An additional source of liquid waste comes from RBOF (244-H) and RRF (245-H); this waste is produced during regeneration of ion-exchange beds, backwashing filters, cleaning and handling of fuel and target elements, and similar incidental operations. The radioactivity is low, about 0.0004 Ci/gal, but the volume ranges from 1.0 to 2.3 million gal/yr. Liquid waste from the RBOF-RRF operations in H Area is pumped to Tank 21 or hold Tank 23. Waste liquid that is too high in radioactivity for adequate decontamination by the cesium removal column is routed to Tank 21 for subsequent evaporation.

SRL and 100-Area Waste

Batches of liquid waste from SRL, which are not suitable for discharge to SRS seepage basins, are delivered to 211-F in tank trailers. This waste is evaporated either in the laboratory waste evaporator in 221-F (high heat) or in the 211-F general evaporator (low heat). The waste is adjusted for alkalinity, if necessary, and transferred to the wastes storage tanks as LHW. Aqueous wastes from the reactor areas are handled much the same as laboratory wastes except when they contain insoluble materials, such as sludges or slurries. When solids are present, the waste is unloaded from the reactor area trailer directly to Tank 47.

Waste Processing Operations

Liquid wastes from separations areas and other locations are routinely processed through a series of operations allowing segregation and consolidation of the waste components prior to interim storage. More information on this stuff can be found in the chapters on S Area and Z Area.

Waste Evaporation

Radioactive waste received in the waste tank facilities is reduced by evaporation, and the concentrated solutions produced are immobilized by solidification of the residual salts. Evaporator concentrate is steam lifted from the evaporator and dropped by gravity either directly or pumped to Tanks 25, 27, 28, 44 and 45 in F Area and Tanks 29, 31, 36 and 37 in H Area. The evaporator condensate is continuously monitored for radioactive contamination. If the radioactivity level exceeds 1500 dpm/ml gamma, the condensate is diverted to waste Tanks 25, 27, and 28 in F Area and Tanks 23 and 43 in H Area for recycle.

Radionuclide Content of Waste

Radionuclide contents depend on the age of the waste and vary with the source stream and prior processing of wastes. The effects of the several waste processing operations on radionuclide contents of the wastes are shown in Tables 13-2 through 13-5. Representative radionuclide contents of wastes as composite sludge, supernate and salt in all tanks is shown in Table 13-2. Comparison of the radionuclide contents of composite sludge, supernate, and salt wastes (Table 13-3) with supernate, in high heat receiver tanks (Table 13-4) after 1.5 years aging illustrates the lower concentration of radionuclides in the salt supernate components of the waste. Radionuclide content of representative evaporator bottoms in Table 13-5 reflects the increased concentrations resulting from dewatering the salt supernate solutions.

Air Monitoring Systems

The air monitoring systems at F- and H-Area Tank Farms are designed to measure radioactive releases from the tanks into the atmosphere. The systems were not designed to monitor the breathing air in the workplace.

A fraction of the tank exhaust air, after filtration, is passed at 3 to 5 cpm through a 3 inch diameter filter paper. The filter paper is monitored by a thin-window GM tube detector whose signal is amplified and sent to the tank farm control room. The detector alarms at an increase in radioactivity above background (approximately 1500 cpm beta-gamma), and alerts operating and HP personnel to check for an abnormal condition. The filter paper is routinely changed weekly, if no abnormal conditions occur, and processed through the HP counting room to measure and maintain records of low level radioactive releases from the tank. The sensitivity of the monitor is 1800 cpm beta-gamma and the range is 1800 to 7000 cpm.

Radionuclides of Concern

The radionuclide contents of the waste tanks shown in Tables 13-2 to 13-5 were reviewed and the content of the combined tanks (Table 13-2) was chosen as being a representative sample for design purposes. The radionuclides of concern were identified on the basis of those that would deliver the majority of the dose and those that could be used as tracers. Lung solubility classes were chosen to maximize the missed dose.

F-Area Combined Tank Waste

The majority of the annual effective dose equivalent is delivered by Sr-90, Ce-144, and Cm-244. The majority of the committed effective dose equivalent is delivered by Sr-90, Pu-241, and Cm-244. The following table illustrates this.

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Am241 W	$1.319 \cdot 10^{-04}$	$1.391 \cdot 10^{-02}$	$5.859 \cdot 10^{-02}$
Cm244 W	$9.341 \cdot 10^{-04}$	$1.035 \cdot 10^{-01}$	$2.305 \cdot 10^{-01}$
Pu-238 W	$2.198 \cdot 10^{-05}$	$2.318 \cdot 10^{-03}$	$8.462 \cdot 10^{-03}$
Pu-239 W	$1.099 \cdot 10^{-04}$	$1.043 \cdot 10^{-02}$	$4.557 \cdot 10^{-02}$
Pu-241 W	$1.022 \cdot 10^{-02}$	$7.546 \cdot 10^{-02}$	$1.009 \cdot 10^{-01}$
Ce144 W	$1.538 \cdot 10^{-01}$	$1.298 \cdot 10^{-01}$	$3.342 \cdot 10^{-02}$
Cs137 D	$4.286 \cdot 10^{-01}$	$6.555 \cdot 10^{-02}$	$1.354 \cdot 10^{-02}$
Ru106 W	$1.055 \cdot 10^{-02}$	$6.120 \cdot 10^{-03}$	$1.250 \cdot 10^{-03}$
Sr90 Y	$3.956 \cdot 10^{-01}$	$6.676 \cdot 10^{-01}$	$5.077 \cdot 10^{-01}$

The annual dose conversion factor (DCF) for the mixture is $1.896 \cdot 10^{-01}$ mrem/nCi. The committed DCF is 1.013 mrem/nCi.

H-Area Combined Tank Waste

The majority of the annual effective dose equivalent is delivered by Sr-90, Ce-144, and Pu-239. The majority of the committed effective dose equivalent is delivered by Sr-90 and Pu-239. The following table illustrates this.

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Am241 W	$7.346 \cdot 10^{-05}$	$5.416 \cdot 10^{-03}$	$1.106 \cdot 10^{-02}$
Cm244 W	$6.121 \cdot 10^{-06}$	$4.739 \cdot 10^{-04}$	$5.118 \cdot 10^{-04}$
Pu-238 W	$5.632 \cdot 10^{-03}$	$3.737 \cdot 10^{-01}$	$7.911 \cdot 10^{-01}$
Pu-241 W	$5.693 \cdot 10^{-03}$	$2.938 \cdot 10^{-04}$	$1.904 \cdot 10^{-02}$
Ce144 W	$2.632 \cdot 10^{-01}$	$1.553 \cdot 10^{-01}$	$1.937 \cdot 10^{-02}$
Cs137 D	$3.489 \cdot 10^{-01}$	$3.730 \cdot 10^{-02}$	$3.734 \cdot 10^{-03}$
Ru106 W	$2.142 \cdot 10^{-02}$	$8.688 \cdot 10^{-03}$	$8.599 \cdot 10^{-04}$
Sr90 Y	$3.550 \cdot 10^{-01}$	$4.188 \cdot 10^{-01}$	$1.544 \cdot 10^{-01}$

The annual dose conversion factor (DCF) for the mixture is $2.712 \cdot 10^{-01}$ mrem/nCi. The committed DCF is 2.990 mrem/nCi.

Recommended Bioassay Programs

Definition of Area

For the purpose of designing an internal dosimetry surveillance program, the F- and H-Area tank farms are defined as follows:

F-Area Tank Farm. The F-Area tank farm is defined as all the buildings and waste tanks in the 241-F area.

H-Area Tank Farm. The H-Area tank farm is defined as all the buildings and waste tanks in the 241-H and 242-H areas.

Worker Monitoring

The following sections outline the worker monitoring programs at the F- and H-Area tank farms.

F Area

Plutonium, strontium, curium, and fission product worker monitoring programs, as given in Chapter 10, should be instituted. Cs-137 is present in sufficient quantities to make it useful as a tracer for whole body counts. Ce-144 is present in sufficient quantities to make it useful as a tracer for chest counts. These programs require the following:

- urine - one sample every three months for curium; one sample every six months for strontium; one sample every 12 months for plutonium
- feces - one sample every six months for curium; one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium counter
- whole-body count - one count every six months

H Area

Plutonium, strontium, and fission product worker monitoring programs, as given in Chapter 10, should be instituted. Cs-137 is present in sufficient quantities to make it useful as a tracer for whole body counts. Ce-144 is present in sufficient quantities to make it useful as a tracer for chest counts. These programs require the following:

- urine - one sample every six months for strontium; one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium counter
- whole-body count - one count every six months

Missed Doses with Tracers

Class D Cs-137 and class W Ce-144 are used as a tracers for the mixtures in the following tables. Annual whole body counts give the following missed doses in mrem:

Mixture	Annual	Committed
F-Area, Cs-137 tracer	24	130
F-Area, Ce-144 tracer	1500	7800
H-Area, Cs-137 tracer	42	460
H-Area, Ce-144 tracer	1200	13000

Annual chest counts give the following missed doses in mrem:

Mixture	Annual	Committed
F Area, Ce-144 tracer	950	5100
H Area, Ce-144 tracer	800	8800

Semi-annual chest counts give the following missed doses in mrem:

Mixture	Annual	Committed
F Area, Ce-144 tracer	110	580
H Area, Ce-144 tracer	92	1020

Workgroup Monitoring

The following sections outline the workgroup monitoring programs for the F- and H-Area tank farms.

F Area

Plutonium, strontium, curium, and fission product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every six months for strontium; one sample every year for curium and plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

H Area

Plutonium, strontium, and fission product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every six months for strontium; one sample every year for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Table 13-2. Radionuclide Content of Combined Supernate, Salt, and Sludge in all Tanks in the Waste Tank Farm in Ci/gal

Combine Tanks (TFA)

	F-Area Tanks			H-Area Tanks		
Radionuclide	Composite	High	Low	Composite	High	Low
H	----------		$4.1 \cdot 10^{-3}$		----------	
Sr-89	$8.8 \cdot 10^{-2}$	$1.1 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$9.4 \cdot 10^{-2}$	$1.9 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Sr-90	$3.6 \cdot 10^{+0}$	$1.8 \cdot 10^{+2}$	$5.5 \cdot 10^{-3}$	$5.8 \cdot 10^{+0}$	$3.5 \cdot 10^{+1}$	$1.1 \cdot 10^{-3}$
Y-90	$3.6 \cdot 10^{+0}$	$1.8 \cdot 10^{+2}$	$5.5 \cdot 10^{-3}$	$5.8 \cdot 10^{+0}$	$3.5 \cdot 10^{+1}$	$1.1 \cdot 10^{-3}$
Y-91	$1.5 \cdot 10^{-1}$	$1.9 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$1.7 \cdot 10^{-1}$	$3.5 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Zr-95	$2.3 \cdot 10^{-1}$	$2.9 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$2.9 \cdot 10^{-1}$	$5.7 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Nb-95	$5.1 \cdot 10^{-1}$	$6.3 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$6.3 \cdot 10^{-1}$	$1.2 \cdot 10^{+1}$	$0.0 \cdot 10^{+0}$
Ru-106	$9.6 \cdot 10^{-2}$	$7.8 \cdot 10^{-1}$	$9.5 \cdot 10^{-6}$	$3.5 \cdot 10^{-1}$	$5.1 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Rh-106	$9.6 \cdot 10^{-2}$	$7.8 \cdot 10^{-1}$	$9.5 \cdot 10^{-6}$	$3.5 \cdot 10^{-1}$	$5.1 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Cs-137	$3.9 \cdot 10^{+0}$	$1.3 \cdot 10^{+1}$	$2.5 \cdot 10^{-1}$	$5.7 \cdot 10^{+0}$	$1.3 \cdot 10^{+1}$	$4.3 \cdot 10^{-2}$
Ba-137	$3.9 \cdot 10^{+0}$	$1.3 \cdot 10^{+1}$	$2.5 \cdot 10^{-1}$	$5.7 \cdot 10^{+0}$	$1.3 \cdot 10^{+1}$	$4.3 \cdot 10^{-2}$
Ce-144	$1.4 \cdot 10^{+0}$	$1.1 \cdot 10^{+1}$	$0.0 \cdot 10^{+0}$	$4.3 \cdot 10^{+0}$	$7.3 \cdot 10^{+1}$	$0.0 \cdot 10^{+0}$
Pr-144	$1.4 \cdot 10^{+0}$	$1.1 \cdot 10^{+1}$	$0.0 \cdot 10^{+0}$	$4.3 \cdot 10^{+0}$	$7.3 \cdot 10^{+1}$	$0.0 \cdot 10^{+0}$
Pm-147	$9.9 \cdot 10^{-1}$	$6.5 \cdot 10^{+0}$	$1.8 \cdot 10^{-3}$	$4.7 \cdot 10^{+0}$	$3.9 \cdot 10^{+1}$	$9.1 \cdot 10^{-5}$
U-235	$8.4 \cdot 10^{-8}$	$6.1 \cdot 10^{-7}$	$5.6 \cdot 10^{-9}$	$3.3 \cdot 10^{-8}$	$3.7 \cdot 10^{-6}$	$4.5 \cdot 10^{-10}$
U-238	$3.3 \cdot 10^{-6}$	$2.9 \cdot 10^{-5}$	$6.3 \cdot 10^{-8}$	$2.1 \cdot 10^{-7}$	$3.9 \cdot 10^{-6}$	$7.0 \cdot 10^{-11}$
Pu-238	$1.7 \cdot 10^{-4}$	$2.3 \cdot 10^{-3}$	$0.0 \cdot 10^{+0}$	$9.2 \cdot 10^{-2}$	$4.0 \cdot 10^{-1}$	$0.0 \cdot 10^{+0}$
Pu-239	$9.8 \cdot 10^{-4}$	$7.7 \cdot 10^{-3}$	$1.6 \cdot 10^{-5}$	$8.8 \cdot 10^{-4}$	$2.9 \cdot 10^{-3}$	$9.8 \cdot 10^{-8}$
Pu-240	$3.0 \cdot 10^{-4}$	$2.1 \cdot 10^{-3}$	$3.4 \cdot 10^{-6}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Pu-241	----------		$9.3 \cdot 10^{-2}$		----------	
Am-241	----------		$1.2 \cdot 10^{-3}$		----------	
Cm-244	$8.5 \cdot 10^{-3}$	$9.4 \cdot 10^{-1}$	$0.0 \cdot 10^{+0}$	$8.4 \cdot 10^{-5}$	$9.6 \cdot 10^{-4}$	$0.0 \cdot 10^{+0}$

Table 13.3. Radionuclide Content of Combined Supernate, Salt, and Sludge in all Tanks in High Heat Receiver Tanks in Ci/gal

High Heat Waste (HHW)

Radionuclide	F-Area Tanks			H-Area Tanks		
	Composite	High	Low	Composite	High	Low
H-3	----------		$5.0 \cdot 10^{-3}$	----------		
Sr-89	$5.9 \cdot 10^{-1}$	$1.1 \times 10^{+0}$	$4.4 \cdot 10^{-2}$	$6.9 \cdot 10^{-1}$	$1.9 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Sr-90	$2.5 \cdot 10^{+0}$	$4.1 \times 10^{+0}$	$8.6 \cdot 10^{-1}$	$1.5 \cdot 10^{+1}$	$1.8 \cdot 10^{+1}$	$1.3 \cdot 10^{+1}$
Y-90	$2.5 \cdot 10^{+0}$	$4.1 \times 10^{+0}$	$8.6 \cdot 10^{-1}$	$1.5 \cdot 10^{+1}$	$1.8 \cdot 10^{+1}$	$1.3 \cdot 10^{+1}$
Y-91	$1.0 \cdot 10^{+0}$	$1.9 \times 10^{+0}$	$1.1 \cdot 10^{-1}$	$1.2 \cdot 10^{+0}$	$3.5 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Zr-95	$1.6 \cdot 10^{+0}$	$2.9 \times 10^{+0}$	$2.2 \cdot 10^{-1}$	$2.0 \cdot 10^{+0}$	$5.7 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Nb-95	$3.4 \cdot 10^{+0}$	$6.3 \times 10^{+0}$	$4.7 \cdot 10^{-1}$	$4.3 \cdot 10^{+0}$	$1.2 \cdot 10^{+1}$	$0.0 \cdot 10^{+0}$
Ru-106	$4.1 \cdot 10^{+0}$	$7.8 \times 10^{+0}$	$3.9 \cdot 10^{-1}$	$2.2 \cdot 10^{+0}$	$5.1 \cdot 10^{+0}$	$3.3 \cdot 10^{-1}$
Rh-106	$4.1 \cdot 10^{+0}$	$7.8 \times 10^{+0}$	$3.9 \cdot 10^{-1}$	$2.2 \cdot 10^{+0}$	$5.1 \cdot 10^{+0}$	$3.3 \cdot 10^{-1}$
Cs-137	$3.9 \cdot 10^{+0}$	$5.4 \times 10^{+0}$	$2.4 \cdot 10^{+0}$	$8.2 \cdot 10^{+0}$	$9.8 \cdot 10^{+0}$	$5.0 \cdot 10^{+0}$
Ba-137	$3.9 \cdot 10^{+0}$	$5.4 \times 10^{+0}$	$2.4 \cdot 10^{+0}$	$8.2 \cdot 10^{+0}$	$9.8 \cdot 10^{+0}$	$5.0 \cdot 10^{+0}$
Ce-144	$8.4 \cdot 10^{+0}$	$1.1 \times 10^{+1}$	$5.3 \cdot 10^{+0}$	$2.8 \cdot 10^{+1}$	$7.3 \cdot 10^{+1}$	$2.2 \cdot 10^{+0}$
Pr-144	$8.4 \cdot 10^{+0}$	$1.1 \times 10^{+1}$	$5.3 \cdot 10^{+0}$	$2.8 \cdot 10^{+1}$	$7.3 \cdot 10^{+1}$	$2.2 \cdot 10^{+0}$
Pm-147	$4.3 \cdot 10^{+0}$	$6.5 \times 10^{+0}$	$2.2 \cdot 10^{+0}$	$2.0 \cdot 10^{+1}$	$3.9 \cdot 10^{+1}$	$9.2 \cdot 10^{+0}$
U-235	$7.2 \cdot 10^{-8}$	8.0×10^{-8}	$6.2 \cdot 10^{-8}$	$2.6 \cdot 10^{-8}$	$3.5 \cdot 10^{-8}$	$1.1 \cdot 10^{-8}$
U-238	$3.5 \cdot 10^{-6}$	4.8×10^{-6}	$2.6 \cdot 10^{-6}$	$1.8 \cdot 10^{-8}$	$2.9 \cdot 10^{-8}$	$5.4 \cdot 10^{-10}$
Pu-238	$2.2 \cdot 10^{-5}$	3.1×10^{-5}	$1.6 \cdot 10^{-5}$	$3.3 \cdot 10^{-1}$	$4.7 \cdot 10^{-1}$	$2.6 \cdot 10^{-1}$
Pu-239	$4.2 \cdot 10^{-4}$	6.0×10^{-3}	$2.9 \cdot 10^{-4}$	$2.5 \cdot 10^{-3}$	$3.3 \cdot 10^{-3}$	$1.9 \cdot 10^{-3}$
Pu-240	$9.7 \cdot 10^{-5}$	1.4×10^{-4}	$6.5 \cdot 10^{-5}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Pu-241	----------		$1.1 \cdot 10^{-1}$	----------		
Am-241	----------		$6.5 \cdot 10^{-4}$	----------		
Cm-244	----------		$2.1 \cdot 10^{-5}$	----------		

Table 13.4. Radionuclide Content of Combined Supernate in High Heat Receiver Tanks in Ci/gal

High Heat Supernate (HHS)

Radionuclide	F-Area Tanks			H-Area Tanks		
	Composite	High	Low	Composite	High	Low
H-3	----------		$5.0 \cdot 10^{-3}$	----------		
Sr-89	$1.4 \cdot 10^{-2}$	3.0×10^{-2}	$1.2 \cdot 10^{-3}$	$1.0 \cdot 10^{-2}$	$4.0 \cdot 10^{-2}$	$0.0 \cdot 10^{+0}$
Sr-90	$6.1 \cdot 10^{-2}$	1.1×10^{-1}	$2.2 \cdot 10^{-2}$	$3.2 \cdot 10^{-1}$	$3.7 \cdot 10^{-1}$	$2.7 \cdot 10^{-1}$
Y-90	$6.1 \cdot 10^{-2}$	1.1×10^{-1}	$2.2 \cdot 10^{-2}$	$3.2 \cdot 10^{-1}$	$3.7 \cdot 10^{-1}$	$2.7 \cdot 10^{-1}$
Y-91	$2.4 \cdot 10^{-2}$	5.0×10^{-2}	$2.8 \cdot 10^{-3}$	$1.9 \cdot 10^{-2}$	$7.3 \cdot 10^{-2}$	$2.8 \cdot 10^{-8}$
Zr-95	$3.8 \cdot 10^{-1}$	7.6×10^{-1}	$5.6 \cdot 10^{-2}$	$3.1 \cdot 10^{-1}$	$1.2 \cdot 10^{+0}$	$1.3 \cdot 10^{-6}$
Nb-95	$8.1 \cdot 10^{-1}$	$1.7 \times 10^{+0}$	$1.2 \cdot 10^{-1}$	$6.7 \cdot 10^{-1}$	$2.6 \cdot 10^{+0}$	$2.8 \cdot 10^{-6}$
Ru-106	$1.5 \cdot 10^{-1}$	2.1×10^{-1}	$9.9 \cdot 10^{-2}$	$3.7 \cdot 10^{-1}$	$1.1 \cdot 10^{+0}$	$7.4 \cdot 10^{-2}$
Rh-106	$1.5 \cdot 10^{-1}$	2.1×10^{-1}	$9.9 \cdot 10^{-2}$	$3.7 \cdot 10^{-1}$	$1.1 \cdot 10^{+0}$	$7.4 \cdot 10^{-2}$
Cs-137	$4.8 \cdot 10^{+0}$	$6.7 \times 10^{+0}$	$3.0 \cdot 10^{+0}$	$8.4 \cdot 10^{+0}$	$9.6 \cdot 10^{+0}$	$5.0 \cdot 10^{+0}$
Ba-137	$4.8 \cdot 10^{+0}$	$6.7 \times 10^{+0}$	$3.0 \cdot 10^{+0}$	$8.4 \cdot 10^{+0}$	$9.6 \cdot 10^{+0}$	$5.0 \cdot 10^{+0}$
Ce-144	$5.4 \cdot 10^{-1}$	7.5×10^{-1}	$3.4 \cdot 10^{-1}$	$1.2 \cdot 10^{+0}$	$3.8 \cdot 10^{+0}$	$1.3 \cdot 10^{-1}$
Pr-144	$5.4 \cdot 10^{-1}$	7.5×10^{-1}	$3.4 \cdot 10^{-1}$	$1.2 \cdot 10^{+0}$	$3.8 \cdot 10^{+0}$	$1.3 \cdot 10^{-1}$
Pm-147	$2.7 \cdot 10^{-1}$	4.3×10^{-1}	$1.4 \cdot 10^{-1}$	$9.2 \cdot 10^{-1}$	$2.0 \cdot 10^{+0}$	$5.2 \cdot 10^{-1}$
U-235	$1.9 \cdot 10^{-9}$	2.2×10^{-9}	$1.6 \cdot 10^{-9}$	$5.5 \cdot 10^{-10}$	$7.3 \cdot 10^{-10}$	$2.2 \cdot 10^{-10}$
U-238	$9.5 \cdot 10^{-8}$	1.3×10^{-7}	$7.1 \cdot 10^{-8}$	$3.9 \cdot 10^{-10}$	$5.9 \cdot 10^{-10}$	$1.1 \cdot 10^{-11}$
Pu-238	$6.0 \cdot 10^{-7}$	8.1×10^{-7}	$4.3 \cdot 10^{-7}$	$7.0 \cdot 10^{-3}$	$9.9 \cdot 10^{-3}$	$6.0 \cdot 10^{-3}$
Pu-239	$1.1 \cdot 10^{-5}$	1.6×10^{-5}	$7.9 \cdot 10^{-6}$	$5.3 \cdot 10^{-5}$	$6.8 \cdot 10^{-5}$	$4.3 \cdot 10^{-5}$
Pu-240	$2.6 \cdot 10^{-6}$	3.6×10^{-6}	$1.8 \cdot 10^{-6}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Pu-241	----------		$2.7 \cdot 10^{-3}$	----------		
Am-241	----------		$1.6 \cdot 10^{-5}$	----------		
Cm-244	----------		$5.1 \cdot 10^{-7}$	----------		

Table 13-5. Radionuclide Content of Combined Supernate and Salt in Evaporator Concentrate in Ci/gal

Evaporator Concentrate (ECC)

Radionuclide	F-Area Tanks			H-Area Tanks		
	Composite	High	Low	Composite	High	Low
H-3	----------	$-4.1E^{-3}$		----------		
Sr-89	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Sr-90	$1.1 \cdot 10^{-1}$	$1.5 \cdot 10^{-1}$	$6.3 \cdot 10^{-2}$	$1.1 \cdot 10^{-1}$	$1.8 \cdot 10^{-1}$	$4.5 \cdot 10^{-2}$
Y-90	$1.1 \cdot 10^{-1}$	$1.5 \cdot 10^{-1}$	$6.3 \cdot 10^{-2}$	$1.1 \cdot 10^{-1}$	$1.8 \cdot 10^{-1}$	$4.5 \cdot 10^{-2}$
Y-91	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Zr-95	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Nb-95	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Ru-106	$2.6 \cdot 10^{-4}$	$2.9 \cdot 10^{-4}$	$2.3 \cdot 10^{-4}$	$2.6 \cdot 10^{-3}$	$5.4 \cdot 10^{-3}$	$2.1 \cdot 10^{-4}$
Rh-106	$2.6 \cdot 10^{-4}$	$2.9 \cdot 10^{-4}$	$2.3 \cdot 10^{-4}$	$2.6 \cdot 10^{-3}$	$5.4 \cdot 10^{-3}$	$2.1 \cdot 10^{-4}$
Cs-137	$6.3 \cdot 10^{+0}$	$8.9 \cdot 10^{+0}$	$3.3 \cdot 10^{+0}$	$5.3 \cdot 10^{+0}$	$9.1 \cdot 10^{+0}$	$2.1 \cdot 10^{+0}$
Ba-137	$6.3 \cdot 10^{+0}$	$8.9 \cdot 10^{+0}$	$3.3 \cdot 10^{+0}$	$5.3 \cdot 10^{+0}$	$9.1 \cdot 10^{+0}$	$2.1 \cdot 10^{+0}$
Ce-144	$1.4 \cdot 10^{-4}$	$1.7 \cdot 10^{-4}$	$1.2 \cdot 10^{-4}$	$2.9 \cdot 10^{-3}$	$6.1 \cdot 10^{-3}$	$1.6 \cdot 10^{-4}$
Pr-144	$1.4 \cdot 10^{-4}$	$1.7 \cdot 10^{-4}$	$1.2 \cdot 10^{-4}$	$2.9 \cdot 10^{-3}$	$6.1 \cdot 10^{-3}$	$1.6 \cdot 10^{-4}$
Pm-147	$2.7 \cdot 10^{-2}$	$3.4 \cdot 10^{-2}$	$1.9 \cdot 10^{-2}$	$5.5 \cdot 10^{-2}$	$1.1 \cdot 10^{-1}$	$8.7 \cdot 10^{-3}$
U-235	$2.3 \cdot 10^{-9}$	$4.0 \cdot 10^{-9}$	$1.8 \cdot 10^{-9}$	$9.7 \cdot 10^{-10}$	$1.3 \cdot 10^{-9}$	$3.2 \cdot 10^{-10}$
U-238	$1.3 \cdot 10^{-7}$	$1.3 \cdot 10^{-7}$	$1.3 \cdot 10^{-7}$	$7.0 \cdot 10^{-10}$	$1.1 \cdot 10^{-9}$	$1.5 \cdot 10^{-11}$
Pu-238	$8.8 \cdot 10^{-7}$	$9.7 \cdot 10^{-7}$	$7.9 \cdot 10^{-7}$	$1.1 \cdot 10^{-2}$	$1.6 \cdot 10^{-2}$	$9.0 \cdot 10^{-3}$
Pu-239	$1.7 \cdot 10^{-5}$	$2.0 \cdot 10^{-5}$	$1.4 \cdot 10^{-5}$	$8.5 \cdot 10^{-5}$	$1.1 \cdot 10^{-4}$	$7.6 \cdot 10^{-5}$
Pu-240	$3.8 \cdot 10^{-6}$	$4.7 \cdot 10^{-6}$	$3.2 \cdot 10^{-6}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$	$0.0 \cdot 10^{+0}$
Pu-241	----------	$4.9 \cdot 10^{-3}$		----------		
Am-241	----------	$2.9 \cdot 10^{-5}$		----------		
Cm-244	----------	$9.2 \cdot 10^{-7}$		----------		

Internal Dosimetry Program for the Waste Certification Facility (WCF)

The Waste Certification Facility (WCF), Building 724-8G located in the 643-G Burial Ground, is used to certify and package drums of transuranic (TRU) waste[16]. The functions of the WCF are to assay and certify the contents of TRU waste drums and to package, load, and ship the drums.

Description of Process

The following sections outline the processing of radioactive materials at the WCF.

Material Handling

Only physical handling of sealed drums is performed in the WCF. Drums are not opened in the facility, and drum contents are not exposed in normal operations. Each drum received is inspected for proper packaging. Contaminated drums are not shipped to WCF. Health Protection routinely surveys the WCF to ensure that the area is not contaminated.

Radionuclide Content of Material

Typical generic contents of TRU waste drums include the following contaminated materials:

Combustibles:	paper, gloves, sweepings, cloth rags
Noncombustibles:	HEPA filters, crucibles, glassware
Chemicals:	evaporated sludge, caustic sludge, soda lime, spent resins
Equipment:	glove boxes, motors, scales

Nominally, drums contain either Pu-238 or Pu-239 depending upon the source of the TRU waste.

Radionuclides of Concern

The Safety Analysis Report for the WCF was reviewed and was found insufficient information on the radionuclide content of the waste. It is therefore assumed that the radionuclides of concern are the same as for the Burial Grounds, 643-G; tritium, curium, plutonium, and activation products.

Recommended Bioassay Programs

The Waste Certification Facility, Building 724-8G, is located in the 643-G Burial Grounds. The following programs apply to all personnel assigned to the WCF, Building 724-8G

Worker Monitoring Program

Tritium, curium, plutonium, and activation products worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every month for tritium; one sample every 3 months for curium; one sample every 12 months to be analyzed for plutonium

- feces - one sample every six months for curium; one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Workgroup Monitoring Program

Tritium, curium, plutonium, and activation products workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every month for tritium; one sample every year for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

References

(1) *Supplement 4 Safety Analysis 200 Area Savannah River Plant F-Canyon Operations.* DPSTSA-200-10

(2) *Supplement X Classified SAR.* DPSTSA-200-10

(3) *Safety Analysis 200 Area Savannah River Plant Building 235-F Vaults.* DPSTSA-200-10 Supplement 15

(4) Safety Analysis-200 Area, *New Special Recovery Facility - Building 221-F, B-Line.* DPSTSA-200-10, SUPP-16

(5) Safety Analysis - 200 Area, *F-Area Outside Facility Operations.* DPSTSA-200-10, SUPP 10

(6) Safety Analysis - 200 Area, *PuO_2 Fuel Form Facility.* DPSTSA-200-10-1

(7) *Safety Analysis of the PuO_2 Experimental Facility.* DPSTSA-700-30

(8) *Safety Analysis 200 Area Savannah River Plant Process Control Laboratory Building 772-F.* DPSTSA-200-10 Supplement 12

(9) *Safety Analysis 200 Area Savannah River Plant New Production Control Facilities Building 772-1F.* DPSTSA-200-10 Supplement 14

(10) *Safety Analysis-200 Area Burial Ground Operations.* DPSTSA-200-10 Supplement 8

(11) *Safety Analysis 200 Area Savannah River Plant Building 221-F B-Line Plutonium Storage Facility.* DPSTSA-200-10 Supplement 19

(12) DPSTSA-200-10 Supplement 15, Table 5-4.

(13) Safety Assessment-200 Area, *F/H Effluent Treatment Facility.* DPSTSA-200-5

(14) Safety Analysis - 200 Area, *Seepage and Retention Basins.* DPSTSA-200-10, Supplement 6

(15) *Safety Analysis-200S Area, Liquid Radioactive Waste Handling Facilities.* DPSTSA-200-10 Supplement 18

(16) *Safety Analysis-200 Area, Waste Certification Facility.* DPSTSA-200-10 Supplement 8

Chapter 14

H Area

Chapter 14 Preview

- Internal Dosimetry Program for 221-H, B-Line
- Internal Dosimetry Program for H-Canyon
- Internal Dosimetry Surveillance Program for the 211-H Outside Facilities
- Internal Dosimetry Program for Receiving Basin for Offsite Fuel (RBOF) and Resin Regeneration Facility (RRF)
- Internal Dosimetry Surveillance Program for Tritium Facilities

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H Area

Internal Dosimetry Program for 221-H, B-Line

The new HB-Line is located on top of the 221-H Building and is designed to replace the aging existing HB-Line production facility[1]. The new HB-Line consists of three separate facilities; the Scrap Recovery Facility, Neptunium Oxide Facility, and Plutonium Oxide Facility.

Description of Process

The following sections outline the processing of radioactive materials at the 221-H, HB-Line.

Scrap Recovery Facility

The Scrap Recovery Facility is designed to routinely generate nitrate solutions of Pu-238 or U-235/Pu-239 scrap suitable for purification by anion exchange or solvent extraction in the canyon. Scrap is received and dissolver batches prepared based on assay of the scrap. The solid scrap is dissolved in hot nitric acid containing trace fluoride ion, transferred through a filter bag, collected in a tank, sampled for accountability and process control, then diluted with nitric acid and transferred to the proper canyon vessel as a nitrate solution. Duplicate lines are installed in this recovery facility to provide the flexibility required and maintain recovery operations to return all recoverable products to the mainstreams.

Neptunium Oxide Facility

Neptunium nitrate solutions are received from H Canyon and F Canyon and transferred to a receipt tank located in old HB-Line. Solutions of plutonium will normally be received from the Scrap Recovery Facility. On a limited basis, dilute Pu-239 solutions will be received from H Canyon. The solutions are precipitated to yield neptunium and plutonium oxalate. The neptunium and plutonium oxalate cake is then loaded into a furnace and calcined to NpO_2 or PuO_2 .

Plutonium Oxide Facility

Plutonium nitrate solution from H Canyon is converted to plutonium oxide powder by the oxalate precipitation and calcination method. The product is shipped to Building 235-F or to offsite facilities for fabrication into heat sources.

Radionuclides of Concern

The following sections outline the Radionuclides of Concern at the 221-H, HB-Line.

Scrap Material

The scrap material is composed of either Pu-238, in oxide form, or a U-235/Pu-239 mixture. The U-235/Pu-239 mixture normally contains greater than 20% enriched uranium. The isotopic composition of the Pu-238 scrap material is

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu238 Y	$9.601 \cdot 10^{-01}$	$9.988 \cdot 10^{-01}$	$9.979 \cdot 10^{-01}$
Pu239 Y	$1.200 \cdot 10^{-03}$	$1.131 \cdot 10^{-03}$	$1.290 \cdot 10^{-03}$
Pu241 Y	$3.870 \cdot 10^{-02}$	$3.271 \cdot 10^{-05}$	$8.045 \cdot 10^{-04}$

The annual dose conversion factor (DCF) for the mixture is $3.076 \cdot 10^{+01}$ mrem/nCi. The committed DCF for the mixture is $2.790 \cdot 10^{+02}$ mrem/nCi.

The isotopic composition of Pu-239 scrap is

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu239 Y	$1.003 \cdot 10^{-01}$	$9.920 \cdot 10^{-01}$	$8.522 \cdot 10^{-01}$
Pu241 Y	$8.997 \cdot 10^{-01}$	$7.978 \cdot 10^{-03}$	$1.478 \cdot 10^{-01}$

The annual dose conversion factor (DCF) for the mixture is $2.932E^{+00}$ mrem/nCi. The committed DCF for the mixture is $3.531 \cdot 10^{+01}$ mrem/nCi.

The isotopic composition of U-235 scrap is

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U233 Y	$9.556 \cdot 10^{-01}$	$9.601 \cdot 10^{-01}$	$9.586 \cdot 10^{-01}$
U235 Y	$4.020 \cdot 10^{-02}$	$3.606 \cdot 10^{-02}$	$3.722 \cdot 10^{-02}$
U236 Y	$3.800 \cdot 10^{-03}$	$3.545 \cdot 10^{-03}$	$3.812 \cdot 10^{-03}$
U238 Y	$4.000 \cdot 10^{-04}$	$3.445 \cdot 10^{-04}$	$3.704 \cdot 10^{-04}$

The annual dose conversion factor (DCF) for the mixture is $2.787 \cdot 10^{+01}$ mrem/nCi. The committed DCF for the mixture is $1.29 \cdot 10^{+02}$ mrem/nCi.

Neptunium Facility Material

The Safety Analysis Report for HB-Line was reviewed and the following radionuclide distribution is considered representative of the Neptunium Facility material:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$5.000 \cdot 10^{-02}$	$5.748 \cdot 10^{-01}$	$4.563 \cdot 10^{-01}$
Pu-239 Y	$4.000 \cdot 10^{-02}$	$4.167 \cdot 10^{-01}$	$3.776 \cdot 10^{-01}$
Pu-241 Y	$9.100 \cdot 10^{-01}$	$8.500 \cdot 10^{-03}$	$1.661 \cdot 10^{-01}$

The annual dose conversion factor (DCF) for the mixture is 2.784 mrem/nCi. The committed DCF for the mixture is $3.178 \cdot 10^{+01}$ mrem/nCi. The SAR also assumes that the NpO_2 is 100% Np-237 for the purposes of dose calculations.

Plutonium Oxide Material

According to the Safety Analysis Report for HB-Line, the plutonium oxide material consists of the following:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$9.691 \cdot 10^{-01}$	$9.991 \cdot 10^{-01}$	$9.984 \cdot 10^{-01}$
Pu-239 Y	$8.992 \cdot 10^{-04}$	$8.401 \cdot 10^{-04}$	$9.583 \cdot 10^{-04}$
Pu-241 Y	$2.997 \cdot 10^{-02}$	$2.511 \cdot 10^{-05}$	$6.176 \cdot 10^{-04}$

The annual dose conversion factor (DCF) for the mixture is $3.104 \cdot 10^{+01}$ mrem/nCi. The committed DCF for the mixture is $2.815 \cdot 10^{+02}$ mrem/nCi.

Recommended Bioassay Programs

The following programs were designed for and apply to all personnel assigned to HB-Line, located on top of Building 221-H.

Worker Monitoring Program

The worker monitoring programs vary for each facility.

Scrap Recovery Facility

Plutonium and uranium worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 6 months for isotopic uranium; one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium and uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Neptunium Oxide Facility
Plutonium and neptunium worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 3 months for neptunium; one sample every 12 months for plutonium
- feces - one sample every 6 months for neptunium; one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Plutonium Oxide Facility
A plutonium worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

The workgroup monitoring programs vary for each facility. Read the following sections for details.

Scrap Recovery Facility
Plutonium and uranium workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium and uranium
- chest count - one count every year on the germanium chest counter

Neptunium Oxide Facility
Plutonium and neptunium workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium and neptunium
- chest count - one count every year on the germanium chest counter

Plutonium Oxide Facility
A plutonium workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every 12 months for plutonium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Program for H-Canyon

The H-Canyon building was designed for the separation and recovery of Pu-239 and U-238 from irradiated natural uranium by the Purex process[2]. Over the years, other processes have been added. These operations include the processing of irradiated enriched uranium to recover U-235, processing irradiated neptunium targets to separate and recover Pu-238 and Np-237, and processing of irradiated thorium to recover U-233.

Description of Process

The following sections outline the processing of radioactive materials at the H-Canyon building.

Uranium Cycle

Standard enriched uranium fuel is dissolved in nitric acid catalyzed by mercuric nitrate. The dissolved fuel is clarified in head end, and the adjusted solution is fed to the solvent extraction system. Uranium and neptunium are extracted from the fission products and separated from each other in the first cycle. Each solution then goes through two separate cycles of solvent extraction for further purification. The uranium product solution is transferred out of the building for further processing. The neptunium product solution is transferred to the HB-Line for oxide conversion.

Neptunium Cycle

Irradiated neptunium targets are dissolved in nitric acid. Pu-238 and neptunium are separated from fission products and each other by a series of anion exchange resin columns. The plutonium and neptunium product solutions are then concentrated, precipitated as oxalate, and calcined to oxides. The plutonium oxide is either shipped offsite or sent to Building 235-F for formation into heat sources. The neptunium oxide is sent to Building 235-F for refabrication into billets.

Special Programs

In addition to the main process operations, various special programs have been conducted. For example, a plutonium mixture containing appreciable Pu-238 has been recovered by solvent extraction from the fission product waste stream of the uranium cycle. A thorium nitrate solution has been purified and concentrated for offsite use. Also, irradiated thorium has been processed in the solvent extraction system to recover U-233.

Radionuclides of Concern

The H-Canyon facility separates uranium, neptunium, plutonium, and fission products from irradiated uranium fuel. The radionuclide activity fractions, and therefore the radionuclides of concern, will vary with the age of the material. The Safety Analysis Report for H-Canyon Operations was reviewed and four process streams were selected for analysis.

The majority of annual and committed dose for the First Cycle Stream comes from plutonium and cerium:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$5.719 \cdot 10^{-04}$	$9.827 \cdot 10^{-02}$	$3.888 \cdot 10^{-01}$
Pu-239 Y	$4.981 \cdot 10^{-06}$	$7.757 \cdot 10^{-04}$	$3.503 \cdot 10^{-03}$
Pu-241 Y	$1.679 \cdot 10^{-03}$	$2.344 \cdot 10^{-04}$	$2.283 \cdot 10^{-02}$
Np237 W	$1.587 \cdot 10^{-07}$	$1.363 \cdot 10^{-05}$	$1.376 \cdot 10^{-04}$
U234 Y	$2.398 \cdot 10^{-06}$	$3.477 \cdot 10^{-04}$	$7.309 \cdot 10^{-04}$
U235 Y	$3.321 \cdot 10^{-08}$	$4.458 \cdot 10^{-06}$	$9.342 \cdot 10^{-06}$
U236 Y	$3.136 \cdot 10^{-07}$	$4.379 \cdot 10^{-05}$	$9.558 \cdot 10^{-05}$
U238 Y	$8.671 \cdot 10^{-10}$	$1.117 \cdot 10^{-07}$	$2.439 \cdot 10^{-07}$
Ce144 Y	$5.535 \cdot 10^{-01}$	$7.727 \cdot 10^{-01}$	$4.931 \cdot 10^{-01}$
Cs137 D	$3.321 \cdot 10^{-02}$	$5.171 \cdot 10^{-03}$	$2.491 \cdot 10^{-03}$
Ru106 Y	$2.583 \cdot 10^{-02}$	$4.299 \cdot 10^{-02}$	$2.906 \cdot 10^{-02}$
Zr95 Y	$2.214 \cdot 10^{-01}$	$2.734 \cdot 10^{-02}$	$1.194 \cdot 10^{-02}$
Sr89 Y	$1.531 \cdot 10^{-01}$	$3.371 \cdot 10^{-02}$	$1.472 \cdot 10^{-02}$
Sr90 Y	$1.070 \cdot 10^{-02}$	$1.839 \cdot 10^{-02}$	$3.261 \cdot 10^{-02}$

The annual dose conversion factor (DCF) for the mixture is $1.862 \cdot 10^{-1}$ mrem/nCi. The committed DCF is $4.266 \cdot 10^{-1}$ mrem/nCi.

The majority of annual and committed dose for the Second Uranium Cycle Stream comes from uranium:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$9.016 \cdot 10^{-05}$	$4.837 \cdot 10^{-04}$	$9.151 \cdot 10^{-04}$
Pu-239 Y	$7.889 \cdot 10^{-07}$	$3.836 \cdot 10^{-06}$	$8.284 \cdot 10^{-06}$
Pu-241 Y	$2.630 \cdot 10^{-04}$	$1.146 \cdot 10^{-06}$	$5.338 \cdot 10^{-05}$
Np237 W	$5.071 \cdot 10^{-04}$	$1.360 \cdot 10^{-03}$	$6.568 \cdot 10^{-03}$
U234 Y	$1.878 \cdot 10^{-01}$	$8.502 \cdot 10^{-01}$	$8.546 \cdot 10^{-01}$
U235 Y	$2.630 \cdot 10^{-03}$	$1.102 \cdot 10^{-02}$	$1.104 \cdot 10^{-02}$
U236 Y	$2.630 \cdot 10^{-02}$	$1.146 \cdot 10^{-01}$	$1.197 \cdot 10^{-01}$
U238 Y	$6.950 \cdot 10^{-05}$	$2.796 \cdot 10^{-04}$	$2.919 \cdot 10^{-04}$
Ce144 Y	$4.320 \cdot 10^{-01}$	$1.883 \cdot 10^{-02}$	$5.746 \cdot 10^{-03}$
Cs137 D	$2.442 \cdot 10^{-02}$	$1.187 \cdot 10^{-04}$	$2.735 \cdot 10^{-05}$
Ru106 Y	$2.066 \cdot 10^{-02}$	$1.074 \cdot 10^{-03}$	$3.471 \cdot 10^{-04}$
Zr95 Y	$1.766 \cdot 10^{-01}$	$6.808 \cdot 10^{-04}$	$1.421 \cdot 10^{-04}$
Sr89 Y	$1.202 \cdot 10^{-01}$	$8.263 \cdot 10^{-04}$	$1.725 \cdot 10^{-04}$
Sr90 Y	$8.452 \cdot 10^{-03}$	$4.535 \cdot 10^{-04}$	$3.846 \cdot 10^{-04}$

The annual dose conversion factor (DCF) for the mixture is 5.965 mrem/nCi. The committed DCF is $2.857 \cdot 10^{+1}$ mrem/nCi.

The majority of the annual dose for the Second Neptunium Cycle Stream comes from plutonium and cerium, and the majority of the committed dose comes from plutonium and neptunium:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$3.596 \cdot 10^{-03}$	$5.978 \cdot 10^{-01}$	$7.570 \cdot 10^{-01}$
Pu-239 Y	$3.083 \cdot 10^{-05}$	$4.644 \cdot 10^{-03}$	$6.713 \cdot 10^{-03}$
Pu-241 Y	$1.028 \cdot 10^{-02}$	$1.388 \cdot 10^{-03}$	$4.326 \cdot 10^{-02}$
Np237 W	$4.624 \cdot 10^{-04}$	$3.843 \cdot 10^{-02}$	$1.242 \cdot 10^{-01}$
U234 Y	$1.284 \cdot 10^{-06}$	$1.801 \cdot 10^{-04}$	$1.212 \cdot 10^{-04}$
U235 Y	$1.284 \cdot 10^{-06}$	$1.668 \cdot 10^{-04}$	$1.119 \cdot 10^{-04}$
U236 Y	$1.284 \cdot 10^{-06}$	$1.735 \cdot 10^{-04}$	$1.212 \cdot 10^{-04}$
Ce144 Y	$1.670 \cdot 10^{-01}$	$2.255 \cdot 10^{-01}$	$4.606 \cdot 10^{-02}$
Cs137 D	$1.002 \cdot 10^{-02}$	$1.509 \cdot 10^{-03}$	$2.327 \cdot 10^{-04}$
Ru106 Y	$1.284 \cdot 10^{-02}$	$2.068 \cdot 10^{-02}$	$4.475 \cdot 10^{-03}$
Zr95 Y	$6.936 \cdot 10^{-01}$	$8.287 \cdot 10^{-02}$	$1.158 \cdot 10^{-02}$
Sr89 Y	$9.890 \cdot 10^{-02}$	$2.106 \cdot 10-02$	$2.943 \cdot 10^{-03}$
Sr90 Y	$3.339 \cdot 10^{-03}$	$5.551 \cdot 10^{-03}$	$3.151 \cdot 10^{-03}$

The annual dose conversion factor (DCF) for the mixture is $1.925 \cdot 10^{-1}$ mrem/nCi. The committed DCF is 1.378 mrem/nCi.

The majority of the annual and committed dose for the Work Stream comes from plutonium and cerium:

Nuclide	Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238 Y	$5.719 \cdot 10^{-04}$	$9.831 \cdot 10^{-02}$	$3.892 \cdot 10^{-01}$
Pu-239 Y	$4.981 \cdot 10^{-06}$	$7.760 \cdot 10^{-04}$	$3.507 \cdot 10^{-03}$
Pu-241 Y	$1.679 \cdot 10^{-03}$	$2.345 \cdot 10^{-04}$	$2.285 \cdot 10^{-02}$
Np237 W	$1.476 \cdot 10^{-09}$	$1.269 \cdot 10^{-07}$	$1.281 \cdot 10^{-06}$
U234 Y	$2.214 \cdot 10^{-10}$	$3.211 \cdot 10^{-08}$	$6.754 \cdot 10^{-08}$
U235 Y	$3.690 \cdot 10^{-12}$	$4.955 \cdot 10^{-10}$	$1.039 \cdot 10^{-09}$
U236 Y	$2.952 \cdot 10^{-11}$	$4.123 \cdot 10^{-09}$	$9.005 \cdot 10^{-09}$
U238 Y	$7.380 \cdot 10^{-14}$	$9.514 \cdot 10^{-12}$	$2.078 \cdot 10^{-11}$
Ce144 Y	$5.535 \cdot 10^{-01}$	$7.730 \cdot 10^{-01}$	$4.935 \cdot 10^{-01}$
Cs137 D	$3.321 \cdot 10^{-02}$	$5.173 \cdot 10^{-03}$	$2.494 \cdot 10^{-03}$
Ru106 Y	$2.583 \cdot 10^{-02}$	$4.301 \cdot 10^{-02}$	$2.909 \cdot 10^{-02}$
Zr95 Y	$2.214 \cdot 10^{-01}$	$2.735 \cdot 10^{-02}$	$1.195 \cdot 10^{-02}$
Sr89 Y	$1.531 \cdot 10^{-01}$	$3.373 \cdot 10^{-02}$	$1.473 \cdot 10^{-02}$
Sr90 Y	$1.070 \cdot 10^{-02}$	$1.839 \cdot 10^{-02}$	$3.264 \cdot 10^{-02}$

The annual dose conversion factor (DCF) for the mixture is $1.862 \cdot 10^{-1}$ mrem/nCi. The committed DCF for the mixture is $4.262 \cdot 10^{-1}$ mrem/nCi.

Recommended Bioassay Programs

The H-Canyon Operations are contained in Building 221-H. The following programs apply to all personnel assigned to this building.

Worker Monitoring Program

Uranium, plutonium, neptunium, and fission product worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 3 months for neptunium; one sample every six months for isotopic uranium; one sample every 12 months for plutonium
- feces - one sample every six months for neptunium; one sample every 12 months for uranium and plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Workgroup Monitoring Program

Uranium, plutonium, neptunium, and fission product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for neptunium, uranium, and plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Missed Doses with Tracers

Ce-144 and Cs-137 are present in sufficient quantities in the First Cycle Stream and the Waste Stream to make them useful as tracers for whole-body and chest counts. Annual whole-body counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Cs-137	2	2
Pure Ce-144	190	260
1st Cycle Stream; Cs-137 tracer	300	690
1st Cycle Stream; Ce-144 tracer	280	720
2nd U Stream; Cs-137 tracer	13000	63000
2nd U Stream; Ce-144 tracer	16000	78000
2nd Np Stream; Cs-137 tracer	1000	7400
2nd Np Stream; Ce-144 tracer	1200	9600
Waste Stream; Cs-137 tracer	300	690
Waste Stream; Ce-144 tracer	280	720

Annual chest counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Ce-144	120	170
1st Cycle Stream; Ce-144 tracer	180	470
2nd U Stream; Ce-144 tracer	11000	51000
2nd Np Stream; Ce-144 tracer	810	6300
Waste Stream; Ce-144 tracer	180	470

Internal Dosimetry Surveillance Program for the 211-H Outside Facilities

H-Area Outside Facilities include a number of processes and services that support the separations function of the 200 H-Area[3]. The main processes and services of radiological concern are A-Line, Water Handling Facilities, Acid Recovery Unit (ARU), General Purpose Evaporator, Segregated Solvent Facilities, Transfer tanks, Sump Collection Tanks, Recycle Sumps, Auxiliary Systems, and Support Facilities.

Many of these processes and services do not have any connection to each other in terms of function in the Outside Facilities. Due to this it will, in some cases, be necessary to treat each process separately in terms of radiological protection.

Description of Process

The general location of 211-H Outside Facilities is on the east side of separations Building 221-H.

A-Line

The H-Area A-Line receives a dilute aqueous solution of uranyl nitrate hexahydrate (UNH) enriched in the isotope U-235 from the canyon. This solution is temporarily stored in A-Line tanks. From there it is pumped into tank trucks for offsite shipment.

Water Handling Facilities

The water handling facilities are located in the Building 211-H complex between the chemical storage area and the Acid Recovery Unit (ARU). The primary equipment for water handling consists of tanks, skimmers and coolers all of which can contain radioactive materials. The facility is operated to provide process water and recycle water for the canyons, retain recycle water in tanks prior to analysis to permit disposal, decant and discharge skimmed solvent to a solvent hold tank, and store weak acid feed for ARU.

Acid Recovery Unit

Nitric acid is concentrated in the ARU for reuse. The ARU is located in the Building 211-H complex between the control house and the water handling facilities. All material in this location can contain radioactive material.

General Purpose Evaporator

The General Purpose (GP) evaporator concentrates low-radioactivity aqueous waste therefore reducing the volume of waste prior to sending the waste over to the tank farm for storage. The GP evaporator is located in the Building 211-H complex between the control house and the segregated solvent facilities.

Segregated Solvent Facilities

The segregated solvent facilities is where the final purification and storage of the solvent occurs prior to it being sent back to the Building 221 canyons for reuse. The segregated solvent facilities are located in Building 211 north of the GP

evaporator in H-Area. Activity in spent solvent is caused primarily by degradation of n-paraffin dilutant and Tri Butyl Phosphate (TBP) when solvent is exposed to high levels of radiation. The degradation produces complex zirconium and ruthenium which subsequently contaminate both solvent and product streams. One of the purposes of the segregated solvent facilities is to remove the degradation end products and radioactive contaminants from the solvent.

Transfer Tanks

Seven transfer tanks are located along the east side of Building 221-H. These tanks provide intermediate pumping stations for solutions transferred from Building 221-H where there is not enough power to transfer it in a single operation. The seven tanks are located in four open basins, each containing a sump and a pump. Generally leaks are pumped to the tanks that feed the GP evaporator, but occasionally the pump discharge must be sent to the seepage basin (i.e. periods of heavy rain). The basins, therefore, must be considered as a place of potential exposure to airborne radioactivity.

Sump Collection Tank

The Sump Collection Tank is located in an underground vault. This tank receives radioactive waste condensate by steam jet from sumps or tanks related to the canyon air exhaust system and and sends it by steam jet to other tanks that feed the hot or warm canyon waste evaporators.

Recycle Sump

The recycle sump is a underground stainless steel lined concrete sump that is located near the southwest corner of the original Building 211 control house. The sump collects the drainage and overflow for all 211-H tanks that contain contaminated or recycled liquids. Being underground and shielded, the sump can receive radioactive solutions that are above the limits for unshielded tanks.

Recycle Vent System

The recycle vent system provides a means to filter contaminated air form tanks and vessels prior to discharge to the atmosphere. This system services tanks in H-Area A-Line, Building 221 Facilities and first and third levels of Building 221. The vent system travels through Building 292-H where it is filtered and sent to the Building 291 stack.

Auxiliary Systems and Support Facilities

The only auxiliary systems and support facilities that are of concern are those that either directly or indirectly involve radioactive materials. Therefore, the process well water system is the only system in this section that is of concern. There are two cooling water streams that are are of concern. The first stream is the segregated cooling water which is discarded and the second is the recirculating cooling water stream which is pumped through a cooling tower and reused. Both streams are continuously monitored for radioactivity and can be diverted into retention or seepage basins to prevent a release of radioactive material. Segregated water is monitored in Building 281-6. If there is confirmation of a monitor alarm by Health Protection the water will be diverted. If there is not confirmation the monitor alarm is reset. Circulated water is monitored in Building 281-4. Since

the supply wells cannot maintain the demand of the cooling system, operations are immediately shut down to avoid pumping contaminated water through the system.

Radionuclides of Concern

The radionuclide of concern for A-Line is the enriched uranium shown below.

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$8.055 \cdot 10^{-01}$	$8.117 \cdot 10^{-01}$	$8.061 \cdot 10^{-01}$
U235	Y	$8.055 \cdot 10^{-03}$	$7.515 \cdot 10^{-03}$	$7.441 \cdot 10^{-03}$
U236	Y	$1.859 \cdot 10^{-01}$	$1.804 \cdot 10^{-01}$	$1.860 \cdot 10^{-01}$
U238	Y	$5.164 \cdot 10^{-04}$	$4.625 \cdot 10^{-04}$	$4.770 \cdot 10^{-04}$

The activity fraction is the fraction of the total activity represented by a particular radionuclide. The annual dose fraction is the fraction of the annual dose conversion factor (DCF) represented by a particular radionuclide. The committed dose fraction is the fraction of the committed DCF represented by a particular radionuclide. The annual DCF for this mixture is 26.80 mrem/nCi. The committed DCF for this mixture is 129.9 mrem/nCi. The specific activity of the uranium is approximately 120 nCi/mg. For this material, an annual chest count has a missed 12-month effective dose equivalent of 3200 mrem based on detection of the U-235.

The radionuclides of concern for the other facilities are Nb-95, Zr-95, Ru-103, Ru-106, and Cs-137. The mixture given for the Water Handling System will be considered typical.

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Cs137	D	$9.990 \cdot 10^{-04}$	$1.626 \cdot 10^{-03}$	$1.717 \cdot 10^{-03}$
Ru106	W	$7.992 \cdot 10^{-02}$	$4.934 \cdot 10^{-01}$	$5.150 \cdot 10^{-01}$
Zr95	W	$4.096 \cdot 10^{-01}$	$3.678 \cdot 10^{-01}$	$3.519 \cdot 10^{-01}$
Nb95	W	$5.095 \cdot 10^{-01}$	$1.372 \cdot 10^{-01}$	$1.313 \cdot 10^{-01}$

The annual DCF for this mixture is $1.782 \cdot 10^{-2}$ mrem/nCi. The committed DCF for this mixture is $1.862 \cdot 10^{-2}$ mrem/nCi.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the 221-H, HB-Line facility.

Worker Monitoring Program

A fission product worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- whole-body count - one count every six months

In addition, for workers in A-Line, a uranium worker monitoring program should be instituted.

- urine - one sample every six months for isotopic uranium
- feces - one sample every year for isotopic uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

A fission product workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- whole-body count - one count every six months

In addition, for workers in A-Line, a uranium worker monitoring program should be instituted.

- urine - one sample every year for isotopic uranium
- chest count - one count every year on the germanium chest counter

Internal Dosimetry Program for Receiving Basin for Offsite Fuel (RBOF) and Resin Regeneration Facility (RRF)

The Receiving Basin for Offsite Fuel (RBOF) and Resin Regeneration Facility (RRF) include a number of processes, utilities and services that support the Separations Areas[4]. Building 244-H, RBOF, and Building 245-H, RRF, are adjoining buildings.

The RBOF facility handles casks and fuels of various shapes, sizes, and content. The RBOF equipment and procedures provide for the safe receipt and storage of irradiated nuclear fuels in a system of water filled basins connected by canals. Included are facilities to inspect the fuel by mechanical and optical means, disassemble fuel, cut fuel by cross cutting or slitting, and weigh fuel before and after packaging or other alterations.

The RRF is capable of regenerating both anion and cation type ion exchange resins. Normally, it is utilized to regenerate the resin used for deionization of Building 244-H water and the resin used in the portable deionizer to deionize the 100-Area water systems, but it can also regenerate any other resin that can be brought to the facility. Other operations performed at RRF include the removal and disposal of the filter cake from the portable 100-Area filter, chemical cleaning of the filter stones in the portable 100-Area filter, and chemical cleaning of target slugs prior to processing in Building 232-H.

Description of Process

The following sections outline the processing of radioactive materials at the RBOF of the RRF.

RBOF Process Description

The RBOF receives and stores fuel elements from various offsite reactors and, on occasion, from SRS reactors. After storage, the fuel elements may be sent to either of the two SRS separations plants for processing.

Casks of spent fuel are received and washed in the cask basin to remove any dirt that may have accumulated during transit. While no decontamination should be necessary, the wash pit is equipped to perform decontamination procedures. After the cask lid bolts are loosened, the cask is lowered into the cask basin. The lid of the cask is removed and all or part of the loaded fuel basket within the cask may be transferred to a fuel transfer bucket. The fuel transfer bucket can be moved underwater, via a canal system, to either the inspection basin, disassembly basin, repackaging basin or storage basin.

The shipment of fuel entails a reversal of some of the operations previously described. Fuel bundles are transferred from the storage basin, by the canals, to the cask basin where cask loading takes place. After the cask lid is in place, the cask is transferred to the cask wash pit where it is surveyed and decontaminated as necessary.

RRF Process Description

RRF, located in Building 245-H, receives a mixture of anion and cation exchange resins from the basin water deionizer. Resins are transferred from the deionizer

to a tank in the RRF. The mixed resin is first depleted with NaOH and then the cation and anion resins are separated. The anion resin is transferred to a separate vessel where it is regenerated. Similarly, the cation resin is also regenerated. After the regeneration, the resins are rinsed and returned to the deionizer. The RRF is also employed to regenerate either mixed or unmixed resins shipped from the reactor facilities.

Target elements are also cleaned at RRF. Irradiated aluminum-clad target elements of Li-Al alloys are brought by truck to the RRF in a cask which is connected to the hose box for in cask cleaning to remove hydrated alumina from the target elements.

Radionuclide Content of Material

The following sections outline the radionuclide content for the different materials.

Irradiated Fuel

The irradiated fuel received at RBOF contains uranium and the usual fission and activation products. However, since the work is performed underwater, the inert gases that escape the fuel bundles are a primary radiological concern. If a Mark 22 Fuel Assembly (cooled 90 days) is used as a reference fuel processed at RBOF, the following radionuclide distribution could be expected:

Nuclide	Activity fraction
I-131	0.102
Kr-85	0.89
Xe-133	0.0056

• Isotopes with half-lives greater than one day

Spent Ion Exchange Resins

The mixture of cation and anion exchange resins received at RRF contain fission and activation products with approximately the following radionuclide distribution:

Nuclide	Activity fraction
Cs-134	0.15
Cs-137	0.15
Co-60	0.70

Irradiated Target Elements

Irradiated target elements that are cut and cleaned in RRF contain tritium.

Radionuclides of Concern

In addition to receiving fuel, RBOF performs other operations such as basin purification, resin regeneration, and target cleaning. After reviewing the Safety Analysis Report for RBOF, it was determined that tritium, fission/activation products, and plutonium could be present during these activities.

Recommended Bioassay Programs

For the purpose of designing an internal dosimetry program, the RBOF includes Building 244-H and Building 245-H. The following programs apply to all personnel assigned to these areas.

Worker Monitoring Program

Tritium, plutonium, and fission/activation products worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every month for tritium; one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Workgroup Monitoring Program

Tritium, plutonium, and fission/activation products workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every month for tritium; one sample every 12 months for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Internal Dosimetry Surveillance Program for Tritium Facilities

The Tritium Facility includes Buildings 232-H, 234-H, and 238-H. The Replacement Tritium Facility (RTF) is an underground facility adjacent to the Tritium Facility in H Area which will take over tritium operations.

Description of Process

Irradiated lithium targets are processed at the Tritium Facility to recover, purify, and package tritium.

Radionuclides of Concern

Elemental tritium and tritiated water are assumed to be the primary radionuclides of concern at the tritium facilities. In addition, the activation product Zn-65 is associated with the processing of tritium targets.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for the 221-H, HB-Line facility.

Worker Monitoring Program

The tritium worker monitoring program, as given in Chapter 10, should be instituted. This program consists of monthly urine bioassay and is designed to detect intakes of tritiated water. If routine exposures to elemental tritium are anticipated the urine bioassay frequency should be increased to one sample every two weeks. These programs are based on missed dose alone. If urine bioassay is used to detect exposures of workers to tritium and control their exposures then a higher frequency may be necessary.

The radionuclides of concern other than tritium are considered to be a source independent of tritium. A fission product worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- whole-body count - one count every six months

Workgroup Monitoring Program

The tritium workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires monthly urine bioassay for tritium.

The radionuclides of concern other than tritium are considered to be a source independent of tritium. A fission product worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- whole-body count - one count every six months

References

(1) Safety Analysis - 200 Area, *Separations Area Operations, Building 221-H, B-Line*. DPSTSA-200-10-2

(2) Safety Analysis - 200 Area, *H-Canyon Operations*. DPSTSA-200-10, SUPP-5

(3) Safety Analysis 200 Area, *Savannah River Plant Separations Area Operations Building 211-H Outside Facilities*. DPSTSA-200-10, Supplement 11

(4) Safety Analysis-200 Area, *Separations Area Operations - Receiving Basin for Offsite Fuel*. DPSTSA-200-10-3

Chapter 15

M Area

Chapter 15 Preview

- Internal Dosimetry Program for Uranium Target Fabrication Facility, 313-M
- Internal Dosimetry Program for Facilities 320-M, 322-M, and 341-M
- Internal Dosimetry Program for Fuel Fabrication Facility, 321-M

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M Area

Internal Dosimetry Program for Uranium Target Fabrication Facility, 313-M

The Uranium Target Fabrication Facility located in Building 313-M was designed and built to manufacture aluminum clad targets for irradiation in SRS reactors.

Description of Process

Depleted uranium metal, received from the Fernald plant, is fabricated into target rods that are used for transmutation of U-238 into plutonium.

Radionuclides of Concern

No information has been made available on the radionuclide distribution of the 313-M material. Therefore, the following distribution is assumed:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$8.399 \cdot 10^{-02}$	$9.345 \cdot 10^{-02}$	$9.036 \cdot 10^{-02}$
U235	Y	$1.450 \cdot 10^{-02}$	$1.494 \cdot 10^{-02}$	$1.440 \cdot 10^{-02}$
U238	Y	$9.015 \cdot 10^{-01}$	$8.916 \cdot 10^{-01}$	$8.952 \cdot 10^{-01}$

The majority of the annual and committed effective dose equivalent from this uranium mixture is delivered by U-238. The annual dose conversion factor (DCF) for the mixture is $2.427 \cdot 10^{+1}$ mrem/nCi. The committed DCF for the mixture is $1.208 \cdot 10^{+2}$ mrem/nCi.

Recommended Bioassay Programs

For the purpose of designing an internal dosimetry program, the Uranium Target Fabrication Facility is defined as Building 313-M. The following programs apply to all personnel assigned to this area.

Worker Monitoring Program

The uranium worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every six months for elemental uranium
- feces - one sample every 12 months for uranium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

The uranium workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every 12 months for elemental uranium.
- chest count - one count every year on the germanium chest counter

Other Considerations

When calculating the missed dose for uranium, it is necessary to determine the ratio of uranium isotopes. For 313-M, the uranium ratios were not made available so an assumed distribution was used. It should also be noted that since the processes contain depleted uranium, the possibility exists for a chemical toxicity problem.

Internal Dosimetry Program for Facilities 320-M, 322-M, and 341-M

Process descriptions for 320-M, 322-M, and 341-M are unavailable at this time.

Description of Process

Uranium and/or enriched uranium is assumed to be processed.

Radionuclides of Concern

No information has been made available on the radionuclide distribution of the material. Therefore, the following distributions of depleted uranium are assumed:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$8.399 \cdot 10^{-02}$	$9.345 \cdot 10^{-02}$	$9.036 \cdot 10^{-02}$
U235	Y	$1.450 \cdot 10^{-02}$	$1.494 \cdot 10^{-02}$	$1.440 \cdot 10^{-02}$
U238	Y	$9.015 \cdot 10^{-01}$	$8.916 \cdot 10^{-01}$	$8.952 \cdot 10^{-01}$

The majority of the annual and committed effective dose equivalent from this uranium mixture is delivered by U-238. The annual dose conversion factor (DCF) for the mixture is $2.427 \cdot 10^{+1}$ mrem/nCi. The committed DCF for the mixture is $1.208 \cdot 10^{+2}$ mrem/nCi.

The following distribution of enriched uranium is assumed.

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$8.500 \cdot 10^{-1}$	$8.552 \cdot 10^{-1}$	$8.508 \cdot 10^{-1}$
U235	Y	$1.100 \cdot 10^{-2}$	$1.025 \cdot 10^{-2}$	$1.016 \cdot 10^{-2}$
U236	Y	$1.380 \cdot 10^{-1}$	$1.337 \cdot 10^{-1}$	$1.381 \cdot 10^{-1}$
U238	Y	$1.000 \cdot 10^{-3}$	$8.943 \cdot 10^{-4}$	$9.239 \cdot 10^{-4}$

The majority of the annual and committed effective dose equivalent from the uranium mixture is delivered by U-234 as shown in the following table. The annual dose conversion factor (DCF) for the mixture is $2.684 \cdot 10^{+1}$ mrem/nCi. The committed DCF is $1.299 \cdot 10^{+2}$ mrem/nCi.

Recommended Bioassay Programs

The following programs apply to all personnel assigned to Buildings 320-M, 322-M, and 341-M.

Worker Monitoring Program

The uranium worker monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every six months for elemental and isotopic uranium
- feces - one sample every 12 months for uranium

- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

The uranium workgroup monitoring program, as given in Chapter 10, should be instituted. This program requires the following:

- urine - one sample every 12 months for elemental and isotopic uranium
- chest count - one count every year on the germanium chest counter

Other Considerations

When calculating the missed dose for uranium, it is necessary to determine the ratio of the uranium isotopes. For the facilities in this report, the uranium ratios were not made available, so an assumed distribution was used. It should also be noted that since the processes may contain depleted uranium, the possibility exists for a chemical toxicity problem.

Internal Dosimetry Program for Fuel Fabrication Facility 321-M

The Fuel Fabrication Facility located in Building 321-M was designed and built to manufacture aluminum clad fuel elements for irradiation in SRS reactors.[1] In this process, enriched uranium metal is alloyed with aluminum in concentrations required for reactor irradiation. The alloy is cast into hollow cylindrical ingots from which pre-extrusion billet cores are machined. These cores are extruded into logs, which are then machined into sections, encased in aluminum, and coextruded into tubes. Cores for other types of tubes are fabricated and assembled into billets in other SRS facilities (Li-Al in Building 320-M and NpO_2-Al in Building 235-F) before they are received at 321-M.

Description of Process

The following sections outline the processing of material processed at 321-M.

Charge Preparation

Enriched uranium metal, received from Oak Ridge Y-12 plant, is weighed in an exhausted hood on an analytical balance in the charge preparation room and then transferred to a critically safe storage container.

The process monitor computer selects the correct amounts of uranium, aluminum, and U-Al scrap for each casting charge. After the uranium is weighed, it is wrapped in aluminum foil, placed in a charge can. U-Al scrap is weighed on scales in the casting and machining rooms and stored in the U-Al scrap rooms until needed.

Two hoods, one containing the analytical balance, are exhausted at 4000 cfm through four parallel pairs of HEPA filters. This is the only exhaust for this room.

U-Al Alloy Storage

The U-Al alloy storage facilities consist of two rooms containing metal storage racks that provide spacing for nuclear criticality safety. Large pieces, such as ingots and extrusion billet cores, are stored directly on the racks in one storage room. Smaller pieces, such as lathe turnings, are stored in metal cans on racks in the other storage room. The casting area contains a hooded press for crushing large U-Al scrap for remelting. The hood is exhausted at 1500 cfm through a single HEPA filter.

U-Al Casting

Uranium metal, aluminum, and U-Al scrap are blended and cast into hollow cylinders. Melting is done in graphite crucibles in induction furnaces. The alloy is cast in graphite molds. Used graphite crucibles are stripped of adhering U-Al particles, assayed, and buried. The U-Al scrap is stored in cans in the U-Al scrap storage rooms for recovery or returned to Oak Ridge Y-12 plant.

The facility has three induction-heated furnaces for alloy and scrap recovery, two recirculation air electric resistance furnaces for heating graphite molds, and four induction heaters for graphite core removal. It also has scales for weighing castings, a material cooling hood, a filter cleaning glove box, a pulse height analyzer and a Californium shuffler for determining the amount of U-235 scrap and waste, and materials-handling carts.

The casting furnaces are installed in hoods. Two hoods are exhausted at 5900 cfm and one at 5400 cfm through five pairs of HEPA filters connected in parallel. The glove box and cooling hood are exhausted at 5600 cfm through three pairs of HEPA filters in parallel.

The room exhaust system consists of six HEPA filters in parallel exhausted at 8500 cfm.

Core Machining

Enriched U-Al castings are machined to dimensions for pre-extrusion cores on lathes. Machining scrap is collected and stored for recovery in the casting process.

The core machine shop contains two hooded lathes. The hood covering each lathe is exhausted at 4000 cfm through three pairs of HEPA filters in parallel. The facility is also equipped with a power hacksaw, a tube shearer, a balance for weighing cores and scrap, an ultrasonic cleaner, a mechanical chip compactor, and work stations for handling scrap. All this equipment except the tube shearer and ultrasonic cleaner is located in hoods exhausted at 1500 cfm through single HEPA filters.

The room exhaust system consists of three pair of HEPA filters in parallel exhausted at 12,000 cfm.

Billet Assembly

U-Al cores are assembled into extrusion billets by encasing them in aluminum components. Li-Al target billets are brought into the facility at this point in the process. The exterior aluminum joints are welded and the weld integrity checked. Fumes from the processes in this facility are exhausted through two roof stacks.

Billet Outgassing

Billets are vacuum-outgassed in outgassing furnaces at a temperature greater than the extrusion temperature. NpO_2-Al billets are brought into the facility at this point in the process. U-Al and Li-Al billets are outgassed in all the outgassing furnaces, NpO_2-Al billets are restricted to those furnaces equipped with HEPA filtered exhaust systems.

The billet evacuation area consists of six electric resistance furnaces, three of which are in metal enclosures with filtered exhaust systems. The exhaust systems for two enclosures consist of two single HEPA filters in parallel. The other system has two pairs of HEPA filters in parallel. The total airflow through each enclosure and the vacuum pump exhaust is 1800 cfm.

Radionuclide Content of Material

PuO_2 was processed in Building 321-M in 1980. Current forecasts do not include any PuO_2 production. The radionuclide distribution of the material presently being processed in Building 321-M is

Nuclide	Mass Fraction	Activity Fraction(%)
U234	0.015	85.0
U235	0.580	1.1
U236	0.245	13.8
U238	0.160	0.1
	1.000	100.0
Np237	1.0	100.0

Radionuclides of Concern

The Safety Analysis Report for 321-M was reviewed and three radionuclide sources were identified. They are the uranium, plutonium, and neptunium mixtures. Lung solubility classes were chosen to maximize the missed dose.

The majority of the annual and committed effective dose equivalent from the uranium mixture is delivered by U-234 as shown by

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
U234	Y	$8.500 \cdot 10^{-1}$	$8.552 \cdot 10^{-1}$	$8.508 \cdot 10^{-1}$
U235	Y	$1.100 \cdot 10^{-2}$	$1.025 \cdot 10^{-2}$	$1.016 \cdot 10^{-2}$
U236	Y	$1.380 \cdot 10^{-1}$	$1.337 \cdot 10^{-1}$	$1.381 \cdot 10^{-1}$
U238	Y	$1.000 \cdot 10^{-3}$	$8.943 \cdot 10^{-4}$	$9.239 \cdot 10^{-4}$

The annual dose conversion factor (DCF) for the mixture is $2.684 \cdot 10^{+1}$ mrem/nCi. The committed DCF is $1.299 \cdot 10^{+2}$ mrem/nCi.

Recommended Bioassay Programs

For the purpose of designing an internal dosimetry program, the Fuel Fabrication Facility is defined as Building 321-M. The following programs apply to all personnel assigned to this facility.

Worker Monitoring Program

Uranium, plutonium, and neptunium worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every three months for neptunium; one sample every six months for isotopic uranium; one sample every 12 months for plutonium
- feces - one sample every six months for neptunium; one sample every 12 months to be analyzed for uranium and plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter

Workgroup Monitoring Program

Uranium, plutonium, and neptunium workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for neptunium, isotopic uranium, and plutonium
- chest count - one count every year on the germanium chest counter

References

(1) Safety Analysis-300 Area, *Fuel Fabrication Facility - Building 321-M*, DPSTSA-300-3A

Chapter 16

S and Z Areas

Chapter 16 Preview

- Defense Waste Processing Facility
- Saltstone Facility

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S and Z Areas

Internal Dosimetry Program for S-Area (DWPF)

Radioactive waste sludges are stored in large underground tanks at SRS. At the Defense Waste Processing Facility (DWPF), this waste is vitrified to a borosilicate waste glass in a slurry-fed joule-heated melter[1]. The waste glass is encapsulated in stainless steel canisters and stored onsite. DWPF is located in the 200-S Area, north of the existing 200-F and 200-H separations areas.

Description of Process

Radioactive wastes are stored in existing H-Area waste tank facilities in the form of settled sludge, supernatant liquid, and salt cake. The insoluble solids (sludge) that have separated from the aqueous fraction of the waste by settling, undergo aluminum dissolving and washing operations in preparation for the feed to DWPF. The aqueous fraction undergoes treatment with sodium tetraphenylborate to precipitate the cesium and potassium salts. Also, a slurry of sodium titanate is added to absorb the soluble stronium and plutonium. The precipitate is washed and concentrated in preparation for feed to the DWPF.

Following the recovery of the organic fraction of the tetra-phenylborate precipitate, the aqueous product is blended with the formatted sludge feed and adjusted with glass formers. The adjusted feed is vitrified to a borosilicate waste glass in a slurry-fed joule-heated melter. The waste glass is encapsulated in stainless steel canisters and stored onsite in the glass waste storage building (GWSB).

Aqueous wastes generated in the DWPF are chemically treated with caustic and recycled to the H-Area tank farm.

Radionuclides of Concern

The process streams listed in the Safety Analysis Report for DWPF were reviewed. Two process streams representing the high and low ends of the range of Pu-238/Sr-90 concentrations were selected to design monitoring programs.

The sludge-slurry feed stream (Stream 1) has a specific activity of 56 Ci/gal. The majority of the annual and committed effective dose equivalent is delivered by Sr-90 and Pu-238 as the following table shows.

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Cm244	W	$1.682 \cdot 10^{-03}$	$4.616 \cdot 10^{-02}$	$3.910 \cdot 10^{-02}$
Pu-238	W	$2.326 \cdot 10^{-02}$	$6.080 \cdot 10^{-01}$	$8.436 \cdot 10^{-01}$
Pu-241	W	$2.684 \cdot 10^{-02}$	$4.911 \cdot 10^{-04}$	$2.496 \cdot 10^{-02}$
Ce144	W	$1.557 \cdot 10^{-01}$	$3.255 \cdot 10^{-02}$	$3.185 \cdot 10^{-03}$
Cs137	D	$2.326 \cdot 10^{-02}$	$8.816 \cdot 10^{-04}$	$6.922 \cdot 10^{-05}$
Ru106	W	$3.578 \cdot 10^{-02}$	$5.144 \cdot 10^{-03}$	$3.993 \cdot 10^{-04}$
Sr90	Y	$7.335 \cdot 10^{-01}$	$3.068 \cdot 10^{-01}$	$8.869 \cdot 10^{-02}$

The activity fraction is the fraction of the total activity represented by a particular radionuclide. The annual dose fraction is the fraction of the annual dose conversion factor (DCF) represented by a particular radionuclide. The committed dose fraction is the fraction of the committed DCF represented by a particular radionuclide. The annual DCF for the mixture is $7.651 \cdot 10^{-01}$ mrem/nCi. The committed DCF is $1.075 \cdot 10^{+01}$ mrem/nCi. This is essentially a plutonium stream.

The chemical forms of the principle radionuclides in the sludge-slurry feed stream are given below. The chemical composition is insoluble unless otherwise noted.

Radionuclide	Chemical form
Sr-90	$Sr(NaTi_2O_5)_2$, $Sr(OH)_2$(sol.), $SrCO_3$, SrO
Ru-106	Na_2RuO_4(sol.)
Cs-137	Cs_2O, CsOH (sol)
Ce-144	unknown
Pu-238	PuO_2, $PuO_2(NaTi_2O_5)_2$, $Na_2PuO_2(OH)_4$(sol)
Pu-241	Same as Pu-238
Cm-244	unknown

The precipitate-slurry feed stream (Stream 201) has a specific activity of 37 Ci/gal. The majority of the annual dose comes from the Cs-137 and Ce-144 but Pu-238 still delivers over half of the committed dose as the following table shows:

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Cm244	W	$2.537 \cdot 10^{-05}$	$8.086 \cdot 10^{-03}$	$2.969 \cdot 10^{-02}$
Pu-238	W	$3.171 \cdot 10^{-04}$	$9.626 \cdot 10^{-02}$	$5.790 \cdot 10^{-01}$
Pu-241	W	$2.537 \cdot 10^{-04}$	$5.390 \cdot 10^{-03}$	$1.189 \cdot 10^{-02}$
Ce144	W	$1.839 \cdot 10^{-01}$	$4.466 \cdot 10^{-01}$	$1.894 \cdot 10^{-01}$
Cs134	D	$3.594 \cdot 10^{-03}$	$2.345 \cdot 10^{-03}$	$7.740 \cdot 10^{-04}$
Cs137	D	$7.611 \cdot 10^{-01}$	$3.350 \cdot 10^{-01}$	$1.140 \cdot 10^{-01}$
Ru106	W	$4.228 \cdot 10^{-02}$	$7.059 \cdot 10^{-02}$	$2.375 \cdot 10^{-02}$
Sr90	Y	$8.457 \cdot 10^{-03}$	$4.107 \cdot 10^{-02}$	$5.147 \cdot 10^{-02}$

The annual dose conversion factor (DCF) for the mixture is $6.589 \cdot 10^{-02}$ mrem/nCi. The committed DCF is $2.136 \cdot 10^{-01}$ mrem/nCi. This may be considered to be essentially a fission product stream.

The chemical forms of the principle radionuclides in the precipitate-slurry feed stream are given below. The chemical composition is insoluble unless otherwise noted.

Radionuclide	Chemical form
Sr-90	Sr(NaTi O) , Sr(OH) (sol.), SrCO
Cs-134	Cs O, CsOH (sol.), CsB(C H)
Cs-137	Same as Cs-134.
Pu-238	Same as sludge-slurry.
Pu-241	Same as sludge-slurry.
Cm-244	unknown

The information contained in the previous tables, pertaining to the radionuclide distribution in various process streams, is useful in obtaining ratios that may be used in internal dosimetry calculations. However, the radionuclide composition of the incoming DWPF feed stream may vary depending on which tanks of waste are being processed. Therefore, if a loss of containment incident occurs, it is recommended that an immediate or recent sample of that process stream be analyzed to obtain more accurate radionuclide ratios.

Recommended Bioassay Programs

The following sections outline the recommended bioassay programs for S Area.

Worker Monitoring Program

Plutonium, strontium, and fission product worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every six months for strontium and one sample every year for plutonium
- feces - one sample every year to be analyzed for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Ce-144 and Cs-137 are present in sufficient quantities to make them potentially useful as tracers for whole-body and chest counts. Annual whole-body counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Cs-137	2	2
Pure Ce-144	190	260
S-1,Cs-137 tracer	1800	25000
S-1,Ce-144 tracer	5800	82000
S-201,Cs-137 tracer	5	15
S-201,Ce-144 tracer	420	1400

Monthly whole-body counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Cs-137	<1	<1
Pure Ce-144	190	260
S-1,Cs-137 tracer	1800	25000
S-1,Ce-144 tracer	5800	82000
S-201,Cs-137 tracer	5	15
S-201,Ce-144 tracer	110	370

Annual chest counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Pu-238	>250000	>250000
Pure Ce-144	120	170
S-1,Ce-144 tracer	3800	53000
S-201,Ce-144 tracer	280	900

Monthly chest counts give the following missed doses in mrem:

Nuclide	Annual	Committed
Pure Pu-238	>250000	>250000
Pure Ce-144	<12	<12
S-1,Ce-144 tracer	260	3700
S-201,Ce-144 tracer	24	60

Workgroup Monitoring Program

Plutonium, strontium, and fission product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every six months for strontium and one sample every year for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

Ce-144 and Cs-137 may be useful as tracers.

Internal Dosimetry Program for Z Area

Waste tank supernatant (low level waste) is transferred from H-Area to Z-Area via pipeline. At Z-Area, the salt solution (supernatant) is mixed with cement, flyash and a set-retarding agent and the resulting grout is transferred into trenches where it sets into a saltstone monolith[2]. The trenches are capped with a low-permeability clay, made from native soil and bentonite, and eventually covered with a minimum of 5 meters of soil.

Radioactive Materials Processed

The following outlines the processing of Radioactive materials at the Saltstone Facilities.

Decontaminated Salt Solution

The chemical and radionuclide content of the decontaminated salt solution will vary according to which tank(s) of waste are being processed. Average compositions have been determined from laboratory tests with actual wastes from F-Area and H-Area, along with calculations of fission product generation determined from the nuclear properties of SRS fuels and reactor exposure. The main radionuclides present are as follows:

Radionuclide	nCi/g (15-yr aged)
H-3	10
Sr-90	0.7
Tc-99	30
Ru-106	30
Sb-125	9
Cs-137	20
Pu-238	0.05
Pu-239	0.0005
All TRU elements	0.2

The compositions given are based on the tetraphenyl-borate(TPB) and the sodium titanate($NaTi_2O_5$) precipitation processes for removing Cs-137 and Sr-90 respectively, from the salt solutions.

Saltstone

Saltstone is the solidified low level waste material formed from mixing the basic ingredients discussed above. The main radionuclides present in the saltstone are as follows:

Radionuclide	nCi/g (15-yr aged)
H-3	4
Sr-90	0.3
Tc-99	12
Ru-106	12
Sb-125	4
Cs-137	9
Pu-238	0.02
Pu-239	0.0002
All TRU elements	0.1

Radionuclide Distribution Information

The information contained in the previous tables, pertaining to the radionuclide distribution in various process streams, is useful in obtaining ratios that may be used in internal dosimetry calculations. However, the radionuclide composition of the incoming feed stream may vary depending on which tanks of waste are being processed. Therefore, if a loss of containment occurs it is recommended that an immediate or recent sample of that process stream be analyzed to obtain more accurate radionuclide ratios.

Radionuclides of Concern

The process streams listed in DPSP 83-1019, Basic Data Report-200Z Area were reviewed and two process streams were identified on the basis of the radionuclides that would deliver the majority of the dose and the radionuclides that could be used for tracers. The process streams are the decontaminated salt solution and the saltstone. Lung solubility classes were chosen to maximize the missed dose.

Decontaminated Salt Solution

The majority of the annual and committed effective dose equivalent from the salt solution is delivered by Pu-238 and Ru-106 as shown by

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238	W	$4.907 \cdot 10^{-03}$	$5.492 \cdot 10^{-01}$	$9.498 \cdot 10^{-01}$
Cs137	D	$3.925 \cdot 10^{-01}$	$6.371 \cdot 10^{-02}$	$6.235 \cdot 10^{-03}$
Ru106	W	$5.888 \cdot 10^{-01}$	$3.625 \cdot 10^{-01}$	$3.507 \cdot 10^{-02}$
Sr90	Y	$1.374 \cdot 10^{-02}$	$2.460 \cdot 10^{-02}$	$8.865 \cdot 10^{-03}$

The annual dose conversion factor (DCF) for the mixture is $1.787 \cdot 10^{-01}$ mrem/nCi. The committed DCF is 2.015 mrem/nCi. This is essentially a plutonium stream.

Saltstone

In the saltstone, the majority of the annual and committed effective dose equivalent is delivered by Pu-238 and Ru-106 as shown by

Nuclide		Activity Fraction	Annual Dose Fraction	Committed Dose Fraction
Pu-238	W	$5.602 \cdot 10^{-03}$	$5.887 \cdot 10^{-01}$	$9.567 \cdot 10^{-01}$
Cs137	D	$4.202 \cdot 10^{-01}$	$6.402 \cdot 10^{-02}$	$5.887 \cdot 10^{-03}$
Ru106	W	$5.602 \cdot 10^{-01}$	$3.238 \cdot 10^{-01}$	$2.944 \cdot 10^{-02}$
Sr90	Y	$1.401 \cdot 10^{-02}$	$2.355 \cdot 10^{-02}$	$7.973 \cdot 10^{-03}$

The annual dose conversion factor (DCF) for the mixture is $1.903 \cdot 10^{-01}$ mrem/nCi. The committed DCF is 2.284 mrem/nCi. This is also essentially a plutonium stream.

Description of Process

The following bioassay programs were designed for and apply to all facilities within the Z-Area boundary.

Worker Monitoring Program

Plutonium and fission product worker monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium
- feces - one sample every 12 months for plutonium
- PAS - in lieu of or in addition to fecal bioassay
- chest count - one count every year on the germanium chest counter
- whole-body - one count every six months

Cs-137 is present in sufficient quantities to make it useful as a tracer for whole body counts. Class D Cs-137 is used as a tracer for the salt solution and saltstone mixtures in the following table. Annual whole-body counts give the following missed doses in mrem:

Nuclide		Annual	Committed
Salt sol.	Class W	25	280
Salt sol.	Class Y	49	240
Saltstone	Class W	24	290
Saltstone	Class Y	47	250

Workgroup Monitoring Program

Plutonium and fission product workgroup monitoring programs, as given in Chapter 10, should be instituted. These programs require the following:

- urine - one sample every 12 months for plutonium
- chest count - one count every year on the germanium chest counter
- whole-body count - one count every six months

References

(1) *Safety Analysis-200S Area, SRS DWPF Operations.* DPSTSA 200-10 Supplement 20

(2) *Basic Data Report - 200Z Area.* DPSP 83-1019

Chapter 17

A Area

Chapter 17 Preview

17

A Area

Internal Dosimetry Program for SRL Building 773-A, Main Laboratory Buildings 735-A and 735-11A, and Building 776-A, Liquid Waste Handling Facility

Description of Processes

The following sections outline the processing of radioactive material in A Area[1].

Building 773-A

Building 773-A has many processes; this section outlines them.

Actinide Technology Division
The Actinide Technology Division provides technical support for the Pu and enriched uranium production facilities; Building 221-F, FB-Line, Building 221-H, and HB-Line. The work in the laboratories is primarily process troubleshooting and some independent research.

Analytical Development Division
The Analytical Development Division is responsible for 32 of the 91 laboratories in Building 773-A. It provides analytical support for research programs at SRL and conducts research and development in the area of process control and analyzer development.

Defense Waste Processing Technology Division
The Defense Waste Processing Technology Division (DWPTD) provides support and conducts research and development in the area of waste processing. Research

is conducted using glass, saltstone, and cement to fix or entrap radionuclides and toxic metals into a matrix, thereby preventing or reducing the rate of release to the environment.

Hydrogen and Fuels Technology Division
The Hydrogen and Fuels Technology Division is developing hydride process technology and testing materials for use in corrosive or hydrogen environments.

Interim Waste Technology Division
The Interim Waste Technology Division is developing the technology for the Defense Waste Processing Facility (DWPF) and the Effluent Treatment Facility (ETF). Work is being performed on supernatant processing; precipitation of strontium and cesium and filtration and acid hydrolysis of the resulting sludges. The characterization of waste and the investigation of the treatment of effluents by ion exchange and reverse osmosis is also being developed.

Environmental Technology Division
The Environmental Technology Division processes environmental samples. Iodine-129 is extracted from neutron activated environmental samples and separated from radiobromine by fractional distillation. Carbon-14 is extracted from plant stack samplers and the extracted C-14 is counted in 735-A

Laboratory Services Division
The Laboratory Services Division includes various services such as the chemical storeroom and glass shop; however, the areas of primary interest are the High Level Cells and the High Bay Experimental Area. The High Level Cells provide the shielding and confinement necessary for analysis and testing of highly radioactive material from onsite and offsite facilities. The High Bay Experimental Area has

research and pilot scale production facilities for work with alpha and neutron emitters such as Pu-239, Cm-244, Am-243, and Cf-252.

Robotic and Fabrication Technology Division
The Robotic and Fabrication Technology Division consists of a Robotics Development Laboratory, Powder Metallurgy Facility, Onsite Uranium Recycle Process Facility, and the Hot Machine Shop. The last three facilities are of radiological interest. The purpose of the Powder Metallurgy Facility is to develop a new process for fabricating reactor fuel tube cores by replacing the current casting technique used in Building 321-M. The Onsite Uranium Recycle Process Facility is a prototype facility to assist in the design and operation of a plant scale Fuel Production Facility. The Hot Machine Shop provides a variety of functions and services including machining uranium metals.

Buildings 735-A and 735-11A

Buildings 735-A and 735-11A house the Radiological and Environmental Science facilities. Groups operating laboratories in these buildings include Environmental Monitoring, Environmental Technology, Environmental Radiometrics, and Health

Protection Technology. Their work involves the radioanalysis of low-level environmental and bioassay samples.

Building 776-A Liquid Waste Handling Facility

The SRL Liquid Waste Handling Facility, located in Building 776- A, collects the aqueous wastes from Building 773-A. Facilities at Building 776-A are provided to strain solids from the waste streams, collect waste in batch receiving tanks, sample and adjust the pH of the waste tank contents, and transfer the waste to tank trailers for disposal.

Radionuclides of Concern

Specific radionuclide information for each division was not found in the Safety Analysis of Savannah River Laboratory Technical Area. General guidelines are given for each division based on the type of work that is performed. The

SRL Health Protection Manager may make changes based on experience and current operations.

Actinide Technology Division

The material used by the Actinide Technology Division should be similar to that found in 221-F, FB-Line, 221-H, and HB-Line. Specific radionuclide information can be found in the Internal Dosimetry Technical Basis Manual sections for these facilities. These radionuclides include plutonium, neptunium, and uranium.

Analytical Development Division

The material used by the Analytical Development Division may contain fission products, plutonium, uranium, neptunium, curium, and/or californium.

Defense Waste Processing Technology Division

The material used by the Defense Waste Processing Technology Division is assumed to be similar to the material processed in S- Area and Z-Area and should contain principally fission/activation products and plutonium.

Hydrogen and Fuels Technology Division

The types of radioactive material used by the Hydrogen and Fuels Technology Division is unknown at this time.

Interim Waste Technology Division

The material used by the Interim Waste Technology Division is assumed to be similar to the material processed in S-Area, Z- Area, and the Effluent Treatment Facility in H-Area. The material should contain principally fission/activation products and plutonium.

Environmental Technology Division

The material used by the Environmental Technology Division may contain I-129 and C-14.

Laboratory Services Division

The material used by the Laboratory Services Division in the High Level Cells and High Bay Experimental areas may contain any of the radionuclides listed above. In addition, Am-243 may be present in some areas.

Robotic and Fabrication Technology Division

The material used by the Robotic and Fabrication Technology Division may contain fission/activation products, depleted, and/or enriched uranium.

Buildings 735-A and 735-11A

A complete listing of the types of radioactive material used throughout the Radiological and Environmental Science facilities is unknown at this time. It is suspected that mainly low level radioactive material is used.

Building 776-A Liquid Waste Handling Facility

The types of radioactive material handled at the Liquid Waste Handling Facility is believed to be a composite of the material that is used in the Building 773-A operations listed above.

Recommended Bioassay Programs

The three SRL facilities described above, Building 773-A, Buildings 735-A and 735-11A, and Building 776-A are located in the SRL complex in A Area. The following programs were designed for and apply to all personnel assigned to these facilities.

Worker Monitoring Program

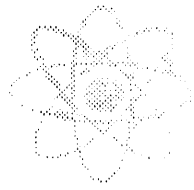
The worker monitoring program may differ between and within divisions, depending on the radioactive material processed. The general radionuclide guidelines given above in conjunction with the worker monitoring programs given in Chapter 10 may be used to establish a worker monitoring program if more accurate information is not available. SRL Health Protection should establish a worker monitoring program based on their knowledge of the material in the facility and on the worker monitoring programs given in Chapter 10.

Workgroup Monitoring Program

Like the worker monitoring, the workgroup monitoring program may differ between and within divisions, depending on the radioactive material processed. The general radionuclide guidelines given above in conjunction with the programs given in Chapter 10 may be used to establish a workgroup monitoring program if more accurate information is not available. SRL Health Protection should establish a workgroup monitoring program based on their knowledge of the material in the facility and on the workgroup monitoring programs given in Chapter 10.

References

(1) Safety Analysis, *Savannah River Laboratory Technical Area*, DPSTSA-700-38.



Part III

Part III - *In Vitro* Bioassay

Chapter 18

Introduction

Chapter 18 Preview

- Overview of Part III

18

Introduction

Chemical bioassay of urine and fecal samples has been carried out at the Savannah River Site since the 1950's to determine the individual's uptake of uranium, tritium, and actinide elements such as americium, plutonium, and californium.

This technique of monitoring radioactive material being excreted by the individual can be much more sensitive than whole body counting or other "monitoring" techniques for certain radionuclides. This is especially true in the case of low energy beta emissions from tritium, or in primarily alpha emitting radionuclides such as the actinides. This technique also allows for the detection and monitoring of low level acute or "chronic" exposures in a way not available to the other methods typically employed for personnel monitoring and surveillance.

Over the years, several techniques have been employed for these types of analyses. Methods ranging from autoradiography to high resolution, solid state alpha spectroscopy have all been used at one time or another. In each case, the main difficulty has been isolating the particular radionuclide of interest from other materials present in the sample, which would either mask the presence of the radionuclide by blocking the emitted radiation, or interfere with the analysis because of their chemical characteristics or radioactive emissions.

As a consequence, many different methods of sample preconcentration, treatment and preparation have been evaluated and utilized over the years. Each technique had some particular advantage at the time it was being used, either in terms of time required for analysis, the number of samples that could be processed simultaneously, the physical form of the final material to be counted , compatibility with other analyses that could be carried out sequentially or in parallel with the main analysis, compliance with more stringent detection levels, etc.

This section describes and explains current attempts to resolve the various factors and demands placed on the laboratory in order to carry out the required analyses. The reasoning behind the current chemical procedures is presented along with a historical background of how the methods evolved at SRS.

Chapter 19

Chemical Preparation

Chapter 19 Preview

- Plutonium Bioassay
- Enriched Uranium Bioassay
- Neptunium Bioassay
- Trivalent Actinide Analyses
- Strontium Bioassay

19

Chemical Preparation

Plutonium Bioassay

Routine Pu bioassay was initiated at SRS in the 1950's. One of the early methods (1954) employed bismuth phosphate coprecipitation, lanthanum nitrate as a carrier, followed by solvent extraction of Pu(IV) with TTA, electrodeposition, and autoradiography with Kodak NTA film for 10,000 minutes.

A later (1959-60) method used chemical ashing, ion exchange on Amberlite CG-400 with 8N and 4N nitric acid, electrodeposition, and autoradiography with Kodak NTA film.

Development of low background solid state alpha counters in the 1960's led to the replacement of autoradiography by gross alpha counting. In 1968, liquid ion exchange by TIOA (tri-isooctyl-amine) and planchetting on a special planchet replaced resin ion exchange and electrodeposition. These changes, in general, were made to improve sample turn around time as the plant population increased, and to reduce the number of samples lost during the laboratory processing.

In 1981 a method modelled after that developed by Kressin at Los Alamos National Laboratory[1] was developed which involved the use of Pu-242 tracer, oxalate precipitation, anion exchange, electrodeposition, and alpha spectrometry. Experience with this method indicated that there was considerable variability with the coprecipitation step. As a consequence, this portion of the procedure was dropped and the Pu-242 was added directly to the acidified sample which was then wet ashed. Sample digestion was later added to the routine procedure instead of wet ashing since it increased sample throughput, and subsequent studies have indicated very favorable results when comparing the two sample preparation methods[2].

Chemical Procedures-Theory and Practice

The following sections give an overview of the chemical procedures for plutonium bioassay.

Wet Ashing and Acid Digestion

The addition of nitric acid and hydrogen peroxide to the urine sample and subsequent heating acts to break down many of the organic materials naturally present in the urine. Both chemicals are strong oxidizers. In addition, the use of a strong acid helps in the dissolution of materials that have precipitated and/or crystallized from the urine sample.

As the volume of acidified solution decreases, the acid concentration increases until the sample that is left is essentially concentrated nitric acid. In this media, the plutonium is present primarily (but perhaps not exclusively) in the +4 oxidation state. In order to convert all the Pu over to the Pu(IV) state, nitrite ions are added to the solution.

The steps are based on the following facts

- Nitrate oxidizes Pu(III) to a mixture of Pu(IV) and Pu(VI) in acid media.

- Nitrite ions oxidize Pu(III) to the Pu(IV) state in acid media. The reaction becomes greatly accelerated as the temperature increases (100 °C).

Thus, in our digestion mixture the overall steps are for nitrate and nitrite to oxidize Pu(III) to Pu(IV). Pu(VI) is reduced by nitrite to Pu(V) which rapidly disproportionates into a mixture of Pu(III), Pu(IV) and Pu(VI). The Pu(III) is oxidized to Pu(IV) by the nitrite.

This leaves all the plutonium in the IV state which is stable in nitric acid media for extended periods.

In addition to destroying the organic material, hydrogen peroxide reduces any Am(VI) to Am(III) in aqueous solution, especially on heating. Any passing presence of Am(V) is short lived as it disproportionates in acid media to the III and VI oxidation states.

This behavior becomes important in the later separation of Pu from Am. This same process can occur for other transplutonium elements that might be in higher valence states (Cf, Cm).

In 1990, J. Hurley of HPT added the additional step of treating the digest with HF acid. There has been some evidence of silicon based inorganic compounds forming in the digest which may later clog the resin/filter apparatus. Column flow has in fact improved following the adoption of this technique.

"Specials", those bioassay samples that have been delivered to the lab based on a special request, are evaporated until dry to insure the complete and total destruction of all organic materials present in the sample. This is carried out as an additional precaution to reduce interferences for these high priority samples. No other modifications are made to routine procedures for the processing of "specials".

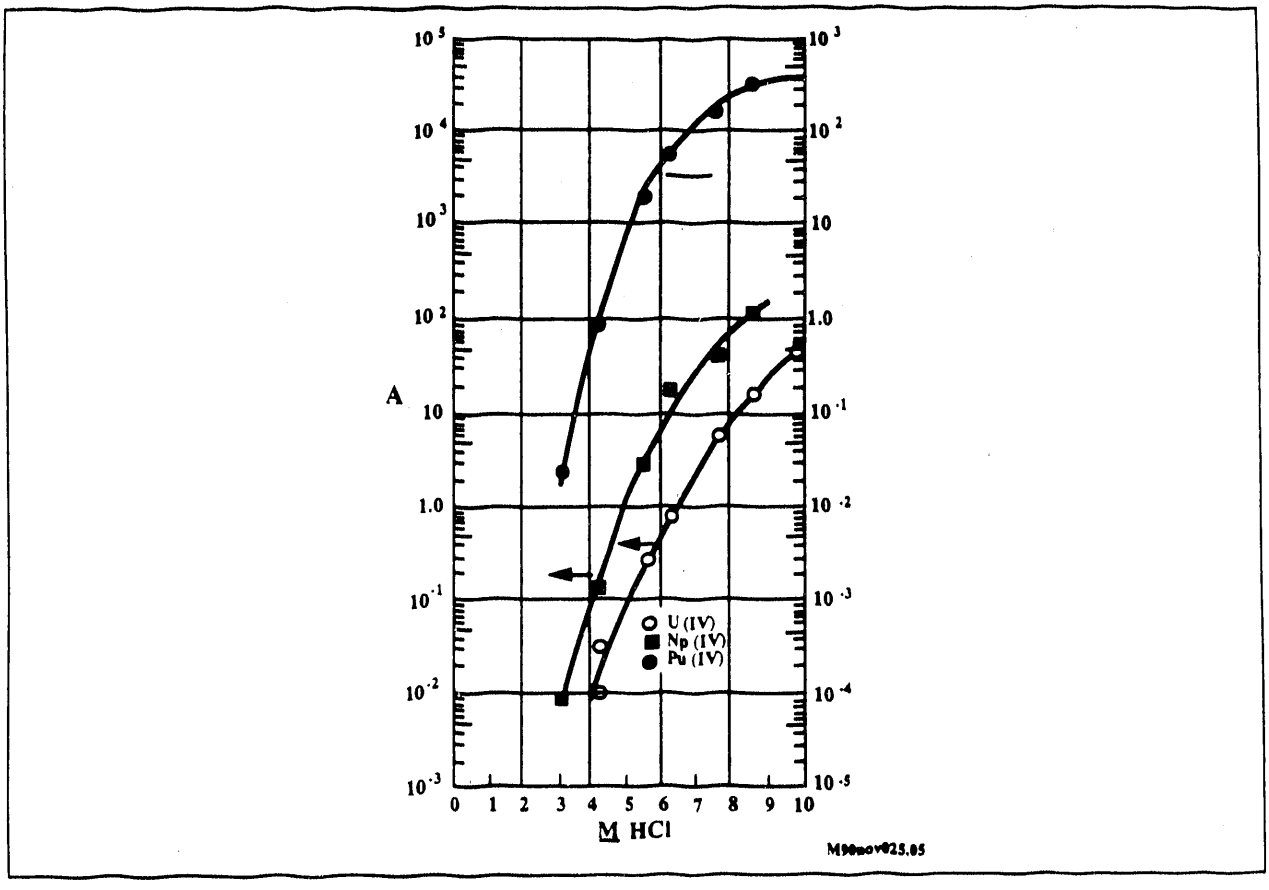


Figure 19-1. Extraction of U(IV) and Np(IV) by 10% TOA and Pu(IV) by 1.0% TOA from NCI Solutions

TIOA Solvent Extraction Separation of Pu

Tri-isooctylamine (TIOA) is a tertiary amine with properties similar to a related compound—tri-n-octylamine (TOA). Both have been referred to as "liquid" ion exchangers and have seen considerable use in the separation of actinides from both hydrochloric and nitric acid media by solvent extraction techniques.

The amines react with acid to form an ion-association complex, which is soluble in the organic phase (usually xylene or toluene)

$$R_3N_{(o)} + H+ + A- = R_3NH+..A-_{(o)}.$$

In analogy to the strong base anion exchange system, Pu(IV) and Pu(VI) are strongly extracted by amines from chloride media while Pu(III) is not. This can be seen in figures 19-1 and 19-2[3].

This forms the initial basis for the plutonium/americium/neptunium/curium/californium separation by TIOA extraction. The following steps are illustrative of the sequential analysis of a sample for multiple actinides, including Pu.

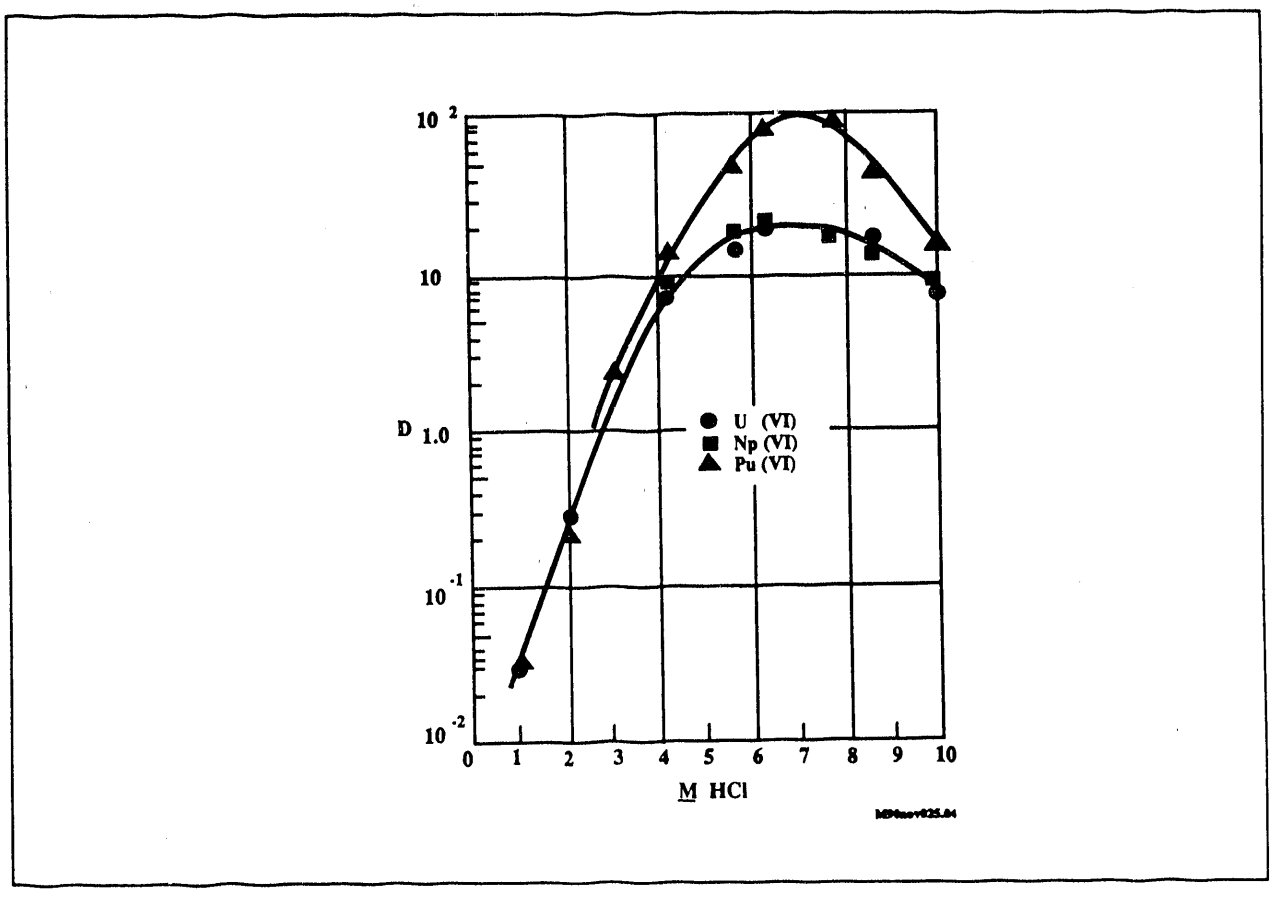


Figure 19-2. Extraction of Hexavalent U, Np, and Pu from HCI Solution with 1.0% TOA in Xylene

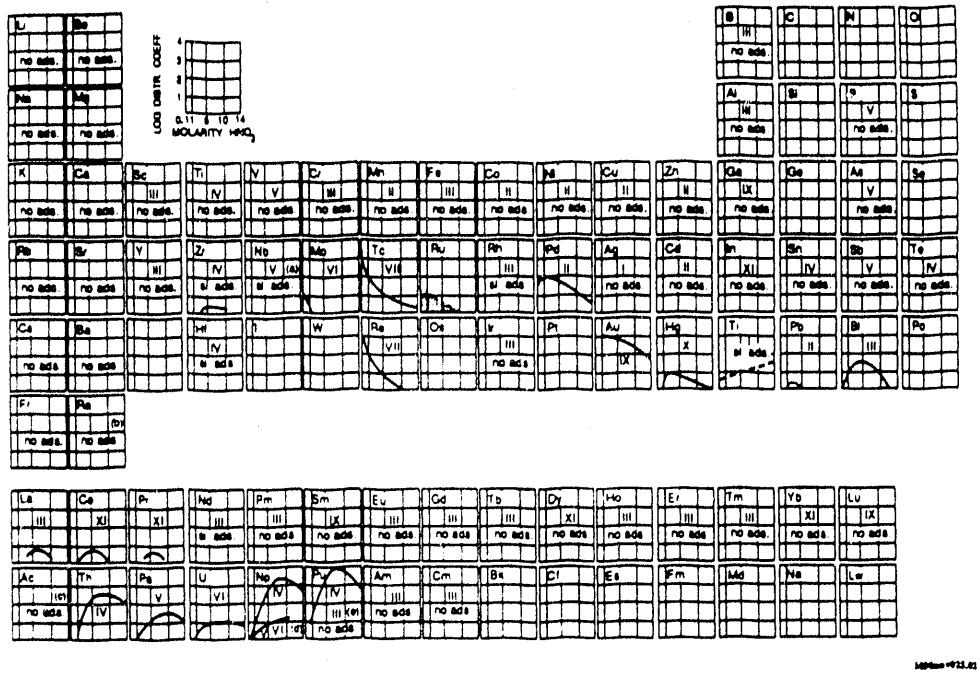
Pu(IV)/Pu(VI) is initially removed by the amine extractant. All the trivalent actinides [Am(III), Cm(III), Cf(III)and Th(IV)] are uncomplexed by the amines. Thus, actinides, such as Am(III), remain in the aqueous phase for collection and later processing.

U(VI) and Np(IV)[4] are extracted along with the Pu(IV)/Pu(VI) ions into the organic phase. Only Pu(IV)/Pu(VI) is selectively reduced to a trivalent state by iodide, which is then back extracted into the aqueous phase in a later step.

Similarly, Np(IV) is removed into the aqueous phase by a competing complexing agent (F-). This ion forms complexes with the actinides to an extent even greater than nitrate or chloride.

The high chloride concentration used with the F- enables the U(VI) to remain in the organic phase even in the presence of fluoride ion (which is used at an admittedly low concentration during the previously described step). This is due to the very high distribution coefficient that U exhibits in chloride media for the amine organic layer.

Finally, adjusting the chloride concentration to a lower level (< 0.5N HCl) forces the U(VI) back into the aqueous layer[5].



No. Ads. - No Adsorption From 0.1 - 14 M HNO_3 ; Sl. Ads. - Slight Adsorption. (Reprinted with Permission from Faris, J.P., Buchanan, R.F.: Anal. Chem. 36, 1158 (1964) Copyright 1964 American Chemical Society)

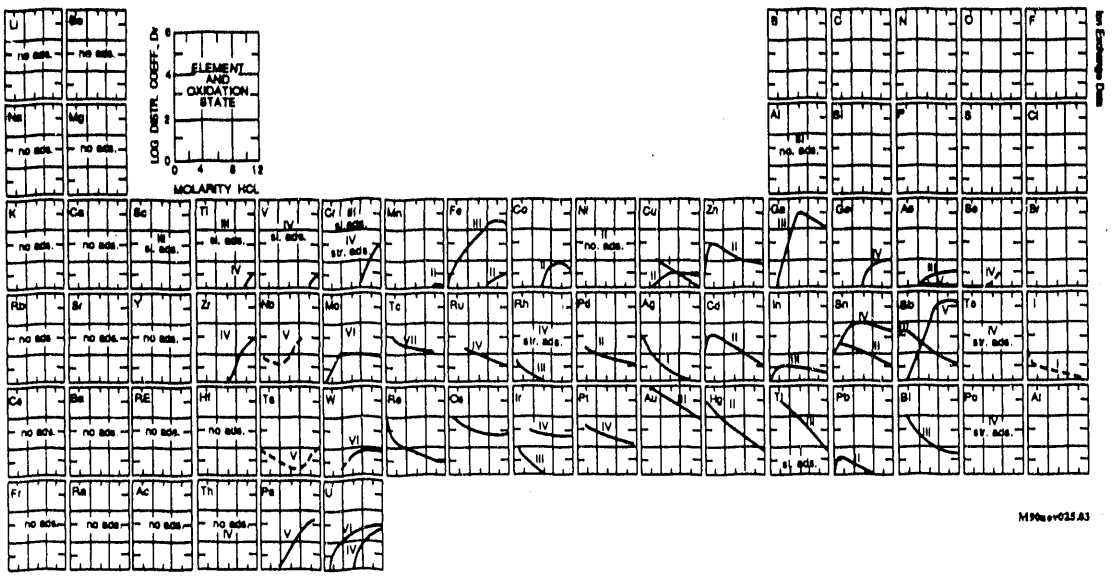
Figure 19.3. Distribution Coeffecients (D) with Strong-Base Anion Exchange Resin (Doweex 1-X10) in Nitric Acid

The aqueous phase containing Am, Cm, and Cf also contains other radioactive nuclides and non radioactive contaminants (including residual TIOA from the earlier extraction). This must be additionally purified before any attempt is made to specifically monitor the Am, Cm, or Cf activities.

This additional purification is made possible by extracting the actinides listed above with dibutyl N,N-diethyl carbamylphosphonate (bidentate) from 12 N nitric acid. By later adjusting the nitrate concentration to only 2N, the actinides are again forced into the aqueous phase.

While this process purifies the actinides in question, it does not isolate them from one another, and one must have some knowledge of the sample being analyzed in order to correctly identify which actinide is being monitored.

Subsequent sections in this chapter on the different actinides give more detail on the exact extraction behavior of each with TIOA, along with some of the potential interferences.



No Ads. -No Adsorption 0.1 < M HCl < 12; Sl. Ads.- Slight Adsorption in 12 M HCL (0.3<Dv<1); Str. Ads. - Strong Adsorption Dv. 1. (Reprinted with Permission from Kraus, K. A., K.A., Nelson, F.: Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva 1955, Vol. 7, P. 118, New York, United Nations 1956)

Figure 19-4. Distribution Coeffecients (Dv) with Strong-Base Anion Exchange Resin in Hydrochloric Acid

Ion Exchange Resin Separation of Pu

The reasoning behind the ion exchange separation of plutonium is a result of the formation of a hexanitrate complex $[Pu(NO_3)_6]^{2-}$ in nitric acid media that is strongly absorbed onto the anion exchanger. The absorption occurs even relatively weak acids, but has its highest K_d at about 7.7 M. Similarly, the K_d is greater at room temperature (25°C) than at elevated temperatures (60°C), but at the higher temperature the rate of sorption increases.

Other species that may be present in the acidified sample behave as indicated in figure 19-3[6, 7].

It should be noted that the only interference comes from Np(IV). However, in acid media of only 4-5 M nitric acid, the oxidation of Np(IV) is almost complete to Np(V) which is not absorbed.

An analogous sorption process occurs with plutonium in the +4 state when it is in hydrochloric acid solutions. The K_d for the process being 1400 in 8 M acid and increasing as the acid strength is increased to 12 M. In this manner, once the Pu(IV) is captured by the anion column from either strong nitric or hydrochloric media, the column can be washed with an 8M HC1 solution without any danger of stripping the Pu.

Figure 19-4 is similar to figure 19-3, but is applicable for HCl media[8, 9].

The addition of iodide as either NH_4I or HI causes the reduction of both Np(V) and Pu(IV) to Np(IV) and Pu(III) respectively. In the case of Pu(III), a cation complex is formed which is not retained on the anion resin. As a result, it is readily eluted. Np(IV) is retained until the acid concentration is reduced.

A strong HCl media is maintained in order to assist in solubilizing the Pu(IV) which has a strong tendency to polymerize in weak acid or neutral media. This polymeric form strongly adheres to glass and other media and can be quite difficult to break down once it has formed.

Electrodeposition

In preparation for electrodeposition, the Pu(IV) must be maintained in solution and kept from polymerizing. The addition of sulfate or bisulfate accomplishes this by the formation of stable plutonium-sulfate anionic complexes.

During the electrolysis of plutonium solutions, the local acidity near the cathode is decreased. Pu is thereby deposited on the cathode as hydroxide of variable composition. The main factors affecting the course of the deposition are the current density on the cathode, and the acidity of the solution. At higher current densities, the H^+ concentration is more depressed which causes the Pu deposition.

The hydroxide precipitates are generally very fragile. Treating the planchets with ammonia greatly enhances their strength[10].

The electrolytic method has several advantages. Films are homogeneous. Moreover, during electrolysis, Pu may be separated from many interfering elements (K, Na, Ca), and elements that do not separate out at the cathode as a metal at the separation potential of hydrogen (Cr, La, Mn). Iron is a potential interference however, and may be present as a result of hemoglobin breakdown from the urine[11].

Yields on platinum can reach 95-98%, and 89% on stainless steel[12]. For reasons of expense and availability, stainless steel is most often used.

Additional Plutonium Analytical Procedures

Coprecipitation of the actinides with calcium phosphate or oxalate, and subsequent ashing of the precipitate has been employed as a means of preconcentration prior to both ion exchange and solvent extraction procedures.

Several solvent extraction techniques have been developed for Pu separation. Compounds such as tri-n-butylphosphate (TBP) or tri-n-octylphosphine oxide (TOPO) in nitrate media have proven useful for the separation of Pu(IV).

The use of TBP in chloride media has also been studied for the separation of all the tetra- and hevavalent actinides. In this media, the trivalent species (such as Pu(III) and Am(III)) are essentially unextracted.

Additionally, bis-2-ethylhexyl phosphoric acid (HDEHP) in both chloride and nitrate media has been studied in detail for the separation of Pu(IV)[13].

Colorimetric procedures based on one or more of the following also have been employed

- The presence of sharp absorption bands due to the reaction of plutonium ion of a particular valency with inorganic ions such as chloride ion. The sensitivity is low on the order of milligrams.

- Colored complexes of Arsenazo I, II ,III, Thoron I with Pu(IV). Sensitivity is in the range of micrograms per liter.

- Solid phase reactions between plutonium and certain organic dyes such as Rhodamine 3B or butylrhodamine. Poor reproducibility is the major drawback of these methods. Sensitivity can be as low as 0.5 microgram per liter.

Alpha spectroscopy is currently the method of choice due to its high sensitivity (0.01 d/m/L is approximately $5.0 \cdot 10^{-14}$ g Pu-239/L).

Enriched Uranium Bioassay

Enriched uranium was determined in the mid-1950's by alkaline earth phosphate coprecipitation, muffling the sample, and ion exchange separation on Dowex 1-X10 with 8N HCL. The final material was electrodeposited and autoradiographed on Kodak NTA emulsion.

Uranium bioassay was made by the Oak Ridge fluorophotometric method until 1980, when it was replaced by an extraction and sintering method (Jarrel-Ash fluorophotometer finish). In the early 80's delayed neutron counting was used for all species of uranium until the reactor activation facility (RAF) was shutdown. Analyses resumed with the Jarrel-Ash method for depleted uranium and to the TIOA method for enriched uranium. In 1986-1987 the kinetic phosphorimeter analysis (KPA) was begun for depleted uranium. In 1990, a new technique based on ion exchange separation of uranium followed by electrodeposition was developed by J. Hurley of HPT. This technique uses U-232 as an internal tracer for each sample being analyzed and has an improved MDA.

Chemical Procedures-Theory and Practice

The following sections give an outline of the chemical procedures for the Enriched Uranium Bioassay.

Wet Ashing and Acid Digestion

The treatment of the sample by strong oxidizing agents has the same affect as in the the section of Pu Bioassay. In the case of U, both U(III) and U(IV) are readily oxidized to U(VI) simply in the presence of room oxygen. As a result, the U is almost certainly present in the +6 state following treatment with strong oxidizers, most probably as UO_2^{2+} when in simple solution. "Specials", those bioassay samples that have been delivered to the lab based on a special request, are evaporated to dryness to insure the complete and total destruction of all organic materials present in the sample. This is carried out as an additional precaution to reduce interferences for these high priority samples. No other modification are made, however, to any procedure for the processing of these samples.

TIOA Extraction Separation of Uranium with Other Actinides

TIOA is currently used in the sequential analysis of several actinides from a single initial sample.

In solutions high in HCl, the uranyl ion forms negatively charged complexes of the type $[UO_2Cl_3]^-$. This complex is rapidly absorbed onto either an anion exchanger resin or into TIOA extractant solutions (K_d = 1800 in 9M HCl).

All the trivalent actinides are uncomplexed by the amines, as is Th(IV). Thus, actinides such as Am(III) remain in the aqueous phase.

Pu(IV)/Pu(VI) and Np(IV) are extracted along with the U(VI) ions into the organic phase. Only Pu(IV)/Pu(VI) is selectively reduced to a trivalent state by iodide which can be back extracted into the aqueous phase.

At lower hydrochloric acid concentrations, the distribution coefficient for U(VI) is much lower (K_d = 1 in 1M HCl) and the U(VI) is either not absorbed or is re-extracted into the aqueous phase.

Due to the distribution coefficient still being slightly positive for the re-extraction, either several rinses are required to remove all the U(VI) from the organic extractant phase, or an additional agent must be added.

Several studies have been made of the behavior of U(VI) in HCl-HF solutions with regards to its ion exchange properties on strong anion exchangers. At high HCl concentrations, the presence of HF has little influence. At low HCl concentrations, the sorption of the U(VI) is considerably lower than that seen in pure HC1 solutions due to the complexing by F^- which results in a uranium fluoride complex with a positive charge[14].

Ion Exchange Resin Separation of Uranium

U(VI) is absorbed strongly by anion exchanger resins from hydrochloric acid media when the molarity is >6M. The process involved was previously discussed in the section on Solvent Extraction of Uranium for TIOA extractions since the TIOA in chloride media is acting as a strong anion exchanger.

Electrodeposition of Uranium

Uranium(VI) migrates to the cathode in uranyl salt solutions of HCl, KCl, NaCl, $NaSO_4$, etc., but to the anode in solutions of excess H_2SO_4, H_3PO_4, etc. This indicates that both anionic and cationic complexes are formed depending on the anion and its concentration.

The precipitation of U is due to the formation of hydrated uranium oxides in the cathodic zone caused by the rise in the concentration of OH- ions. These oxides are transported to the cathode, reduced, and precipitated as partially dehydrated hydroxides of variable composition.

The precipitation of uranium appears to be unaffected by the cathode material, and occurs only after the pH value required for the precipitation of the hydroxide has been reached in the cathodic zone , irrespective of the pH of the original solution.

Additional Uranium Analytical Procedures

Uranium analyses have been determined by several alternate techniques over the years. Both neutron activation and fission track methods have been employed in the determination of 10^{-6} - 10^{-12} grams of uranium in a variety of matrices following reactor irradiations[15].

Solvent extraction techniques for the isolation of uranium have employed TOPO, thenoyltrifluoroacetone (TTA), diethyldithiocarbamate, and 8-hydroxyquinoline[16].

Luminescence of the complexes of the uranyl ion with uranium present as U(VI) under excitation by ultraviolet radiation has been utilized (see the section on KPA). This method is specific for uranium and is the most sensitive known at present for practical work with a sensitivity of 10^{-12} grams.

Colorimetric analysis of uranium is also possible by the formation of complexes with such materials as Arsenazo I. This method has a working range of 0.2–10 microgram/mL of uranium.

Alpha spectroscopy is utilized at the present time due to its ability to differentiate between the various isotopic forms of uranium (not just the total uranium metal content) as well as its high sensitivity.

Neptunium Bioassay

Np was determined by coprecipitation with calcium-magnesium ammonium phosphate, ion exchanged on Dowex-2 with 8N HCL, and autoradiographed with Kodak NTA emulsion for 10,000 minutes as early as 1959 at SRS.

The current approach to Np analyses employing TIOA as an extractant and electrodeposition has been in use at SRS since the mid-1960's. the actual counting of the samples has been accomplished using a variety of counting methodologies including autoradiographpy with Kodak NTA film (early 60s) and solid state detector systems (present.)

Chemical Procedures-Theory and Practice

The following sections give an outline of the chemical procedures for Neptunium Bioassay.

Wet Ashing and Digestion

In the wet ashing of Np ,Am, Cm, and Cf samples, a single sample is utilized since the same sample is processed sequentially for each of the indicated actinides. The volume used is determined by the minimum detectable amount needed by the dosimetrists when the sample recoveries and counting geometries are considered.

In the acidic and oxidizing environment of nitric acid, the Np ions are almost exclusively in the Np(V) state. Subsequent treatment by peroxide reduces the Np(V) to Np(IV).

"Specials", those bioassay samples that have been delivered to the lab based on a special request, are evaporated to dryness to insure the complete and total destruction of all organic materials present in the sample. This is carried out as an additional precaution to reduce interferences for these high priority samples. No other modification are made, however, to any procedure for the processing of these samples.

TIOA Extraction of Np

The extraction of Np(IV) in hydrochloric acid media by TIOA increases as the acid concentration is increased. In 8M HCl, the distribution coefficient is 300. Np(V) on the other hand is only mildly extracted with a distribution coefficient of 1.

Trivalent actinides (Am, Cm, Cf) and Th(IV) are not extracted.

Pu(IV) and U(VI) are extracted strongly but can be separated from Np by the methods previously described.

Ion Exchange Separation of Np

Anion exchange of Neptunium is frequently used. Like the other actinides, the sorption decreases in the sequence Np(IV) > Np(VI) > Np(III) > Np(V). Penta and trivalent anions are practically not sorbed from any HCl solutions. Np(VI) is sorbed from 6M HCl and Np(IV) from 4M HC1. These are probably sorbed as the anions $NpCl_6^{-2}$ and $Np0_2Cl_4^{-2}$ for Np(IV) and Np(VI) respectively.

The Np is desorbed from the anion exchanger by the addition of HCl with HF due to the complexing power of the fluoride ion.

Thorium (IV) is a potential interference, but is not expected to be present to any significant extent in the sample, and due to prewashing of the ion exchange column by HCl.

Electrodeposition of Neptunium

Currently, Np is evaporatively deposited on a planchet for counting. Plans call for the eventual switch to electrodeposition for all counting. This section is included as an analysis of these future plans.

Regardless of the initial acidity of the actinide solutions, the electrodeposition of actinides occurs when the value of the pH of the solution in the cathodic layer approaches that of the precipitation of the actinide hydroxides. The actinides themselves are too electronegative to be deposited as the bare metals from aqueous solution.

The actinide yield is dependant of the actinide concentration. Quantitative yields are obtained when the actinide concentration is about 1 μg/mL. At concentrations below 0.01-0.1 μg/mL, the yield drops to 80-90%. The yield is also reduced in the presence of more than 50 mg nitrate and sulfate, and 10 μg Fe and Al when the media in saturated ammonium chloride solutions containing 0.1M HCl.

These and several other elements deposit hydroxides under these conditions which may build a precipitate which hinders the measurement of the Np.

Yields may be increased to 99% by the slow addition of carrier (such as uranium) in 10 μg quantities 2-3 times during the course of the deposition.

The hydroxide precipitates can be fixed by the addition of ammonium hydroxide to prevent dissolution of the hydrated actinide hydroxides.

Additional Neptunium Analytical Procedures

In the analytical determination of Np, extraction of Np(IV) and Np(VI) by tri-butyl-phosphate (TBP) , trioctylamine (TIOA), and thenoyltrifluoroacetone (TTA) have found widest application[1].

Additionally, neutron activation of Np-237 to Np-238 ($t_{1/2}$ = 2.1 days) has been used for the analytical determination of the neptunium, followed by chemical separation and counting. This yields a sensitivity of 10^{-2} micrograms Np.

Spectrophotometric methods have also been employed for Np determinations, such as the determination of Np(V) by the intense 983 mm absorption band. A sensitivity of 10 micrograms Np following hydroxide precipitation and preconcentration was reported.

Alpha spectroscopy is used again as the method of choice due to its high sensitivity (0.1 d/m/L Np-237 is approximately $6.0 \cdot 10^{-12}$ g Np-237/L).

Trivalent Actinide Analyses

In 1969 Americium (Curium and Californium) were analyzed by re-ashing the Pu(TIOA) raffinate, extracting with HDEHP, and gross counting on low-level alpha counters. In the early 70's the HDEHP extraction was replaced by DDCP extraction, a method developed by Roscoe Hall and Frank Butler, and published in Analytical Chemistry. The HDEHP extraction was highly sensitive to pH adjustment, whereas the DDCP extraction takes place from strong acid solution.

Chemical Procedures-Theory and Practice

The following sections give an outline of the chemical procedures for Trivalent Actinide Analyses.

Wet Ashing and Acid Digestion

In the wet ashing of Np ,Am, Cm, and Cf samples, a single sample volume of 300 mL is utilized since the same sample is processed sequentially for each of the indicated actinides. The volume used is determined by the minimum detectable amount needed by the dosimetrists when the sample recoveries and counting geometries are considered.

As previously described in the section of plutonium wet ashing and digestion, the use of strong oxidizing agents serve to destroy the bulk of the organic matrix present in the sample.

Additionally, hydrogen peroxide reduces any Am(VI) to Am(III) in aqueous solution, especially on heating. Any passing presence of Am(V) is short lived as it disproportionates in acid media to the III and VI oxidation states.

This behavior becomes important in the later separation of Pu from Am. This same process can occur for other transplutonium elements that might be in higher valence states (Cf, Cm).

"Specials", those bioassay samples that have been delivered to the lab based on a special request, are evaporated to dryness to insure the complete and total destruction of all organic materials present in the sample. This is carried out as an additional precaution to reduce interferences for these high priority samples. No other modification are made, however, to any procedure for the processing of these samples.

Solvent Extraction of the Trivalent Actinides by TIOA

All the trivalent actinides (Am(III), Cm(III), Cf(III), Pu(III)) are uncomplexed by the amines such as TIOA, as is Th(IV). Thus, actinides such as Am(III) remain in the aqueous phase for collection and later processing.

The aqueous phase containing Am, Cm, and Cf also contains other radioactive nuclides and non radioactive contaminants (including residual TIOA from the earlier extraction). This must be additionally purified before any attempt is made to specifically monitor the Am, Cm, or Cf activities.

Solvent Extraction of the Trivalent Actinides by DEDECP

The additional purification required (of the aqueous phase of the previous TIOA extraction) is made possible by extracting the actinides with dibutyl N,N-diethyl carbamylphosphonate (bidentate) from 12 N nitric acid.

Bifunctional organophosphorus compounds containing two P=O or one such group and =C=O are of considerable utility in the extraction of trivalent transplutonium elements and rare earth elements(REE) from sufficiently concentrated nitric acid media. Unlike the corresponding monofunctional reagents, the bifunctional reagents DO NOT extract nitric acid along with the elements of interest. Since the nitric acid is no longer competing for available sites, the trivalent species are more readily extracted. By the use of dibutyl-N-N-diethyl carbamylphosphonate, DBDECP, quantitative (>98%) extractions of Am(III) [and Cf(III),Cm(III)] are possible from solutions >9 M HNO_3.

By later adjusting the nitrate concentration to only 2N, the above actinides are again forced into the aqueous phase.

While this process purifies the actinides in question, it does not isolate them from one another and one must have some knowledge of the sample being analyzed in order to correctly identify which actinide is being monitored.

Electrodeposition of the Trivalent Actinides

Currently, Am, Cm, and Cf are deposited by evaporation on a planchet for counting. Plans call for the eventual switch to electrodeposition for all counting. This section is included as an analysis of these future plans.

As in the case of the other actinides elements, the trivalent actinides may also be electrodepositied onto suitable substrates for counting purposes. The same mechanism previously described (see the Pu, Np sections) also applies in this case.

Curium has been separated from Am by the selective oxidation of Am(III) to Am(VI). DMSO was utilized as the solvent medium[17].

Electrolytic precipitations have been carried out for Am and Cm on platinum, nickel, aluminum and stainless steel from solutions of potassium carbonate, ammonium bisulfate, nitric and hydrochloric acid, formic acid, and alcohol-acetone mixtures[18]. Recoveries from 90-99.8% have been reported.

Additional Analytical Procedures for Trivalent Actinides

Cation exchange resins strongly absorb Am(III) and other trivalent and tetravalent ions from acid solution such as 0.25 M nitric acid. This process is commonly used as a preconcentration step.

Chelex 100 complexing resin has also been used to preconcentrate Am(III), with a reported distribution coefficient greater than 10^{19}.

The inherent absorption bands that occur from the presence of Am in a solution of other inorganic ions has also been employed. The method has a sensitivity that makes it suitable for only miligram determinations however.

Mass spectroscopy has also been used for the determination of transplutonium elements. The main advantage of the technique is the small amount of sample typically required, approximately 0.1 microgram. A sensitivity of approximately 10^{-17} gms is obtainable.

An interesting titrimetric method based on the displacement of cerium from a cerium EDTA titrant by americium (or any species that forms a more stable EDTA complex) has been reported. The liberated Ce(III) is then titrated by EDTA. Sensitivity is at the tens of microgram level.

Alpha spectroscopy is again the method of choice due to its high sensitivity (0.1 d/m/L Am-241 is approximately $1.3 \cdot 10^{-14}$ g Am-241/L).

Strontium Bioassay

In the late 1950's-early 1960's, Sr-89/90 was determined by alkaline earth phosphate coprecipitation and beta counting on automatic and GM counters along with zirconium, rare earths, and niobium.

Since 1969, strontium has been determined by the method now in use. In the 50s and 60s, a gross fission product method was used which included strontium.

Sample Pretreatment

Wet ashing and acid digestion of the sample, as previously described, acts to destroy any organic species present.

"Specials", those bioassay samples that have been delivered to the lab based on a special request, are evaporated to dryness to insure the complete and total destruction of all organic materials present in the sample. This is carried out as an additional precaution to reduce interferences for these high priority samples. No other modification are made, however, to any procedure for the processing of these samples.

Extraction by D2EHPA di-(2-ethylhexyl)phosphoric acid

The current Sr procedure is primarily one of purification and separation of the Sr from other interfering radionuclides, particularly rare earth elements and actinides. A common reagent used for the separation of REE is D2EHPA, sometimes described as a liquid cation exchanger. D2EHPA does not extract the alkaline earth elements (such as Ca, Sr).

Degrees of extraction for the rare earth element (REE) ions with D2EHPA increase with increasing Z due to contraction of the ionic radii, the smaller ion being bound tighter to the extractant.

Over the range of 0.1-2.0M [H+], the following formula most likely applies

$$M^{+3}aq^{+3}(HR)_{2org} \quad <> \quad M[H(R)2]3org+3H^{+}aq$$

where R is the main body of the organic complexing agent.

There is a profound affect on the extraction process by the carrier solvent, sometimes altering the distribution coefficient by several orders of magnitude from solvent to solvent. Extraction of Am+3, for example, had measured Kd values that varied by over 1000 changing from iso-octane to chloroform. The trend, which is shared by rare earth and actinide elements, is for extractability to depend on the polarity of the solvent. The more polar solvents suppressing the formation of an extractable complex[20].

Deviations from the above reaction occur when competing complexes from materials, such as nitric and hydrochloric acid, occur. Typically when the concentration of the materials are > 4M. The distribution coefficient, Kd, for most lanthanions has a minimum in 4-6 M nitric acid which then rapidly increases up to an acid concentration of 16M. A similar minimum is seen in 6M HCl solutions.

The current process for the measurement of Sr-90 involves the removal of all interfering (i.e., radioactive) rare earth elements from the sample solution by extraction of all REE from a 0.08N HCl solution[21].

The Y-90 which results from the decay of Sr-90 is then allowed to grow in and reach equilibrium. The resulting Y-90 is then extracted again from 0.08N HCl into D2EHPA, and stripped from the organic phase by 3N nitric acid. The purified Y-90 is counted on a gas proportional counter by environmental monitoring to determine the Sr-90 content.

References and Notes

(1) *Analytical Chemistry 1981* (53), 1270-1274

(2) Memo J. A. Hurley to File, Influence of DTPA on Plutonium Bioassay, 11/22/88

(3) The Radiochemistry of Plutonium, Coleman, George, 1965, US Atomic Energy Commission, NAS-NS 3058, p 55

(4) Hydrogen peroxide reduces higher valence Np to the IV state.

(5) Failure to carry out the preceding steps for Pu and Np mean that these can interfere in the U measurement at this point.

(6) U(VI) is sorbed by anion exchangers to an insignificant amount in nitric acid media by itself. However, its sorption increases markedly in a solution that is low in nitric acid content but relatively high in Al, Li, Ca, or similar nitrates (for example: 0.3M acid/1.6M Al).

(7) *Enrichment Techniques for Inorganic Trace Analysis-Chemical Laboratory Practice*. Mizuike, Atsushi, Springer Berlag, New York, 1983, p 136

(8) The lack of sorption by Th(IV) from HC1 solutions is the basis for the pre-wash used to condition the column prior to sample loading. Thorium would otherwise by retained on the column in nitric acid media > 4M.

(9) Enrichment Techniques for Inorganic Trace Analysis-Chemical Laboratory Practice, Mizuike, Atsushi, Springer Berlag, New York, 1983, p 135

(10) Khlebnikov "GI and EP Dergunov," *Atomnaya energiya*, 4, P376, 1958

(11) Some Fe contamination is occasionally observed, but is not typically a problem.

(12) Milyukova, MS, Gusev, NI, Sentyurin, IG, and Sklyarenko, IS. *Analytical Chemistry of the Elements Plutonium*, 1969.

(13) Coleman, George. *The Radiochemistry of Plutonium*. 1965, US Atomic Energy Commission, NAS-NS 3058, p 45

(14) Palei, PN. *Analytical Chemistry of Uranium*, 1970

(15) Palei, PN. *Analytical Chemistry of the Elements: Uranium*. Ann Arbor-Humphrey Publishers, Ann Arbor, 1970, p 241

(16) Palei, PN., *Analytical Chemistry of the Elements: Uranium*. Ann Arbor-Humphrey Publishers, Ann Arbor, 1970, p 293

(17) Mikhailov, VA. *Analytical Chemistry of the Elements: Neptunium*. Halsted Press, New York, 1973, pp 70-126

(18) Handley, Th. H. and J. H. Cooper. *Analyst. Chem.* 41:381,1969

(19) Translated by N. Kanzer. *Analytical Chemistry of the Elements: Transplutonium Elements*. Halsted Press, pp 105-6, 1974

(20) The trend is iso-octane > cyclohexane > carbon tet > toluene > benzene > chloroform.

(21) Chloride ions hinder the extraction of both the REE and transplutonium elements into D2EHPA, but it suppresses the second group more. This group separation is most useful for the isolation of REE from Am and Cm.

Chapter 20

Analytical Measurements and Results

Chapter 20 Preview

- Kinetic Phosphorimetry Analysis
- Liquid Scintillation Counting
- Alpha Spectroscopy Counting
- High Resolution Ge Counting
- Gamma Screening
- Proportional Counting
- Offsite Contractor Laboratory Analyses

20

Analytical Measurements and Results

Kinetic Phosphorimetry Analysis of Natural or Depleted Uranium

These sections give an overview of the kinetic phosphorimetry analysis of natural or depleted Uranium.

KPA Theory of Operation

The KPA currently in use is a commercial instrument which operates on the emission and detection of uranyl ion fluorescence as a function of time[1, 2].

The nitrogen dye laser is used as a source of excitation energy due to the extremely short duration of the associated pulse (3 nanoseconds) and the fact that its emission is in the region (400-700 nanometers), which is optimal for absorption by the uranyl ion.

The pulse of laser light is assigned an occurrence time of zero. Intensity measurements are taken at fixed time intervals following the pulse. Intensity values from each interval are summed for the number of pulses used in each measurement. In this way, enhanced statistics and precision can be obtained.

The first order kinetic decay of the excited phosphor, for example uranyl ion, can be expressed as

$$\ln U^*_t = \ln U^*_0 - (k_p + k_q)t$$

where U^*_t = population of uranyl ion at time i = 0 or t, k_p= rate constant for phosphor decay, and k_q = rate constant for all other relaxation processes.

A least squares fit to the log of the measured intensity vs. time-after pulse yields a straight line whose intercept is independent of nonradioactive effects such as quenching. The decay lifetime is defined as the inverse of the quantity $(k_p + k_q)$.

Radiationless processes involve collisional interactions and the transfer of energy. These processes may be minimized by stabilizing or otherwise protecting the molecules of interest. For the current process, complexation of uranyl ion with a commercial complexing agent, URAPLEX, is employed. Lifetimes as long as 340 microseconds may be observed under optimum conditions; lifetimes greater than 100 microseconds are required for the best measurements.

In practice instrumental background must be measured and subtracted from the gross sample intensity measurement because background luminescence can be significant and does not follow the first order kinetics equation.

Sample Treatment

Since the KPA is highly specific for uranium and has a very high sensitivity, routine analyses utilize a sample volume that is convenient to manipulate but not large enough to occupy time preconcentrating. Only a sample volume large enough for the initial analysis and a backup analysis is required.

A 2 mL aliquot of acidified urine is therefore wet ashed with nitric acid and hydrogen peroxide. The residue is heated to 500°C to destroy any residual organics, cooled, and covered. Great care is taken to prevent contamination of the sample by organic materials which would adversely affect the measurements.

The 2 mL final volume used for analysis was determined based on the acid volume needed to dissolve the muffled residue, and the quantity which has been found easiest to handle in transferring to the cuvette.

Due to its high sensitivity, there is no need to differentiate between routine and special samples in the initial sample preparation.

Table 20-1. KPA Analysis/Reporting Operation Guides

Operating Parameters

Laser pulses:	2000
First time gate:	6
Max # of time gates:	20
Dilution factor for self-test:	1.0
Dilution factor for samples, blanks and synthetics:	1.8
Mode:	LOW RANGE

The final reported result is determined in the following manner

$\mu g/L$ Uranium = ABCD/EF

where

A = sample reading, ng/mL

B = sample volume taken for analysis, mL (2)

C = adjustment for 20% sample acidification (1.2)

D = acid volume used to dissolve sample, mL (2)

E = conversion to L (1000)

F = conversion to μg (1000)

Reporting Guides

If lifetime is less than 150 μS, the sample is a rerun. If final result is less than 5 ng/mL, record <5. If the result is >(or equal to)5 ng/mL, rerun the sample. If concentration is <0.4 ng/mL, spike with 100 lambda of 10 mg/L uranium standard, then reanalyze. If concentration is >10 ng/1.5L, 6.7 ng/L, rerun the analysis on a 0.2mL aliquot.

Operation Guides

Lifetime has to be above 150 μS for acceptance. Background intensity cannot be above 1600. Reference intensity 40,000 - 100,000 counts after every 20 samples, instrument is recalibrated. If the instrument sits idly more than 30 minutes, it is recalibrated.

When the prescribed factors are combined, the result in $\mu g/L$ is obtained by multiplying the sample reading by 1.2.

Sensitivity

The current set of instrumental parameters used in the analysis are based on recommendations of B. Bushaw (the developer of the commercial instrument) and are designed to meet the current acceptable reporting level of 3.3 $\mu g/L$.

The MDA detection limit for this technique based on measurements of the blanks currently measured is approximately 0.30 $\mu g/L$ of uranium with a blank "mean" of 0.17 $\mu g/L$ uranium. The formulas used are

- Decision Level (DL): The amount of a count or final instrument measurement of a quantity of analyte at or above which a decision is made that a positive quantity of the analyte is present.

$$MDA = \frac{[4.65(S_{Blk}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

$$DL = \frac{[DB \cdot B + 1.645\ S][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For DL, DB is the fractional systematic error bound in the determination of a blank population . The absolute magnitude of the maximum positive systematic error in the blank population should be used. B is the mean value of the blank population. S is the standard deviation in the net count of B.

- Minimum Detectable Amount (MDA): The smallest amount of a radionuclide in a sample that shall be detected with a probability of a non-detection (Type II error) while accepting an a probability of erroneously detecting that radionuclide in an appropriate blank sample (Type I error). for this subcontract, the a and b probabilities are both set to 0.05.

$$MDA = \frac{[4.65(S_{BIK}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{REcovery} \cdot \text{Volume}}$$

For MDA , S_{Blk} is the standard deviation of the count from a set of acceptable blanks adjusted to the sample count time, Time is the sample count time, and the units correction factor adjusts the DL or MDA to a 1000 ml urine sample volume by taking into account the initial aliquot and any dilution (such as by acidification) of the urine sample prior to aliquoting. DL and MDA are based on trial use ANSI N13.30.

Liquid Scintillation Counting of Tritium

In the early-mid fifties, tritium was measured with the vibrating reed electrometer method down to the microCurie/liter level. In this method hydrogen and tritium gas generated by running the sample over Ca turnings was collected in an ionization chamber. The potential difference across the chamber correlated to the amount of tritium in the sample.

Liquid scintillation counting of tritium was performed in the late fifties using a commercial Packard liquid scintillation counter with PPO/POPOP cocktails.

Currently, liquid scintillation counting at WSRC employs three Beckman 6000 low level liquid scintillation counters and two Beckman 9800 liquid scintillation counters. A biodegradable cocktail (Ecolite +) from ICN Biochemical is used to alleviate the problems in the waste disposal of standard toluene based cocktails.

LSC Theory of Operation

The need to use liquid scintillation counting for tritium arises due to the extremely low energy of the tritium beta particle (18.6 KeV) along with the fact that there are no other associated emissions with the decay.

Liquid scintillation counting employes the intimate contact between the sample (often a liquid) and a scintillator dissolved in a suitable solvent (often toluene). The scintillator is electronically excited by the energetic particles emitted during radioactive decay and some of its electrons are raised to a higher electronic state. These electrons either lose their energy by returning to their ground state with the simultaneous emission of a photon, or transfer energy by way of collisions with other molecules.

An additional constituent of the scintillation mixture is often a wavelength shifter. This absorbs the excitation energy from the primary scintillant, and subsequently re-emits a photon at a different wavelength. The re-emitted photon is closer in wavelength to that required for optimal detection by the light sensitive detectors surrounding the sample in the counting chamber.

Since the overall process depends on the emission, absorption, re-emission and subsequent detection of photons, any process which can interfere with these steps can bias the final results.

The two main interferences occur from

- color quenching- where the photons are actually absorbed by other colored materials in the sample
- chemical quenching- where chemical species present in the sample interfere with the transfer of the excitation energy

These are both corrected for by means of a quench curve. Operationally, the observable affects of the two forms of quench are indistinguishable from one another; therefore, if a set of known standards are progressively treated with a quenching agent and then counted, the degree of quenching can be monitored and corrected for.

The exact method employed for the quench measurement and correction varies with the manufacturer of the instrument in question. Currently, the bioassay laboratory is using commercial instrumentation that measures the shift in the Compton edge as an indication of the degree of quench (H number). The quench correction is then automatically taken into account by the instruments.

Sample Counting and Data Evaluation

In our current routine methodology, 2 mL of untreated urine are added to 17 mL of scintillation cocktail and counted for a 30 second count. The amount of sample that is used is determined by the vendor's published capacity of the cocktail for aqueous samples and measured efficiency values for various sample loadings. The counting time of 30 seconds is determined arbitrarily to allow for routine processing of large numbers of samples while maintaining the detection levels required by dosimetry.

Samples that are measured to have > 8 μCi/L tritium are further analyzed by the "spike" or internal standard technique in which a known quantity of tritium is added to the sample in question. The sample is then recounted. The advantage of the technique is that there is no quench correction involved. The scintillation unit acts merely as a counter.

In this procedure, 0.5 mL of the sample and 1.5 mL of distilled water is run in two sets of triplicates. One set is spiked with a known amount of tritium while the other is counted "as is" for 1 minute.

The μCi/L for the spiked sample is calculated as

Step 1 Calculate the CPM per μCi factor for tritium

$$Q = (A-B)/C$$

Q = spike CPM/μCi
A = sample + spike CPM
B = sample CPM
C = spike activity in μCi tritium added

Step 2 Calculate the mean of the Q values for all samples (F).

Step 3 Determine the tritium concentration per liter

$$T = L/(FV)$$

T = tritium concentration in μCi/L
L = sample CPM
F = average Q value
V = sample volume, Liters

Table 20-2. Tritium Analysis Conditions and Reporting

Preset Time	0.5 Minutes
Channel Settings	
Channel 1	40-400 (tritium)
Channel 2	400-1000
Sample Volume	2 mL
Scintillation Cocktail Volume	17 mL
Dark Adaptation Interval	20 Minutes

Reporting

Samples > 8.0 μCi/L tritium or samples which do not fall on the calibration quench curve are analyzed by standard addition.

All analyses less than 0.1 μCi/L are reported as < 0.1 μCi/L

Sensitiv ty

The ultimate sensitivity of the method depends on the length of time the sample is counted and the degree of quenching encountered in the sample. Currently measured MDAs for the technique are shown in the attached table based on current blank sample analyses.

Decision Level (DL): The amount of a count or final instrument measurement of a quantity of analyte at or above which a decision is made that a positive quantity of the analyte is present.

$$DL = \frac{[DB \cdot B + 1.645\ S][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For DL, DB is the fractional systematic error bound in the determination of a blank population . The absolute magnitude of the maximum positive systematic error in the blank population should be used. B is the mean value of the blank population. S is the standard deviation in the next count of B.

Minimum Detectable Amount (MDA): The smallest amount of a radionuclide in a sample that shall be detected with a probability of a non-detection (Type II error) while accepting an a probability of erroneously detecting that radionuclide in an appropriate blank sample (Type I error). For this subcontract, the a and b probabilities are both set to 0.05.

$$MDA = \frac{[4.65(S_{Blk}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For MDA , S_{Blk} is the standard deviation of the count from a set of acceptable blanks adjusted to the sample count time, Time is the sample count time, and the units correction factor adjusts the DL or MDA to a 1000 ml urine sample volume by taking into account the initial aliquot and any dilution (such as by acidification) of the urine sample prior to aliquoting. DL and MDA are based on trial use ANSI N13.30.

Samples analyzed by the standard additions method are more precise due to the elimination of the quench curve correction. Any factors affecting the measurement of the tritium in the original sample will affect the measurement of the standard addition equally.

Low Level Tritium Counting

Low level counting of tritium samples is carried out mainly on those samples from offsite residents.

The tritium sample is distilled with potassium permanganate/sulfuric acid to destroy all trace organic species which might quench the sample. The distillation step also helps to remove interferences from such species as K-40 and other naturally occurring radionuclides.

The 5 mL samples (in triplicate) are then counted for two 50 minutes cycles using an internal standard technique.[3] This cancels out any residual quenching that may occur in the sample without the need of a quench curve and its associated errors. Counts are repeated if the variation between the two sets is more than 8% (which is 3 sigma for low level counts).

$$\% \text{ difference} = 2(A-B)/(A+B)$$

where A and B are the first and second cycle counts

Calculations are as follows:

Step 1 Calculate the CPM per nCi factor for tritium

$$Q = (A-B)/C$$

where:

Q = spike CPM/nCi
A = sample + spike CPM
B = sample CPM
C = spike activity in nCi

Step 2 Calculate the mean of the Q values for all samples (F).

Step 3 Determine the tritium concentration per liter

$$T = 1000 \cdot L/(FV)$$

where:

T = tritium concentration in nCi/L
L = sample CPM-blank water
F = average Q value
V = =sample volume

All blank water samples utilize a special low level well water (designated as P4A) that has been geologically isolated from naturally occurring tritium contamination.

Alpha Spectroscopy Counting with Solid State Detectors

Solid state alpha detectors are utilized because of their high resolution (14 - 20 KeV), low voltage requirements (<100V), and ease of of setup. Although moderate vacuum (10-50 micron) may need to be sustained in the counting chamber for maximum detector resolution, some counting applications can operate the detectors at atmospheric pressure if a "gross" alpha count is all that is desired.

The current system at WSRC is composed of 120 ion-implanted solid state alpha detectors (450 mm^2, 20 Kev FWHM) interfaced to three Canberra Series 90 multichannel analyzers. Spectral data is transferred over RS-232 serial communications to a AT style microcomputer where the final results are calculated by an in-house program.

Sample Pretreatment

The quantity of acidified urine used in the various procedures is determined by the MDA requirements for the various radionuclides.

Due to the strong interaction of charged alpha particles with the material making up the bulk of the sample, and the minute quantities of radionuclides with which we are concerned, the samples to be analyzed must be chemically destroyed and concentrated prior to their analysis.

The previous sections on the analysis of plutonium and the other nuclides of interest detail the specifics of how the samples are treated.

Planchet Preparation

In order to minimize the absorption of the emitted alpha particles by surrounding air and other materials, the sample is destroyed and the remaining residue mounted on a planchet. The planchet is then placed directly under the solid state alpha detector.

There are two methods of planchet preparation in use, direct evaporative mounting and electrodeposition.

Evaporative Mounting

Evaporative mounting is the simplest and easiest of the sample mounting methods. The original sample is reduced to 1-2 mL of solution which is evaporated onto the surface of a stainless steel disk. This planchet then becomes the sample to be counted.

Advantages of the technique are

- minimal additional equipment is required for sample preparation
- rapid preparation of planchets

Disadvantages of the technique are

- any dissolved salts add mass to the planchet obscuring the alpha particles by absorption and scattering
- material is not firmly fixed to the planchet and may rub off with handling

Due to the degradation of the spectral information by scattering and self absorption, this method of mounting is useful when one is sure of isolating only one alpha emitting radionuclide per planchet, or when all the occurring alpha emitters can be summed into a single "alpha activity". As a result, the use of internal spikes for these samples is ruled out.

Current species counted in this way are Np, Am, Cf, and Cm.

Electrodeposition

Electrodeposition of samples is more complex than evaporative mounting, but is usually considered more desirable.

Advantages to the technique are

- a planchet of less mass with a corresponding decrease in sample absorption and scattering
- a much more durable sample due to the strength of the electroplate
- the removal of potential interferences by tailoring the electrodeposition parameters

Disadvantages to the technique are

- additional time and equipment are required for sample processing and electrodeposition
- additional sample losses may occur during electrodeposition

A detailed description of the electrodeposition process is described in the earlier section under each applicable radionuclide.

The plating that is obtained has essentially no "mass" as far as the alpha emissions are concerned. Thus there is minimal energy degradation of the alpha's due to self absorption and scattering. Internal spikes may therefore be added to the samples in order to determine individual chemical recoveries.

Both Pu and U are currently processed in this manner. Pu analyses are especially aided by this method in that the resulting planchet is typically lower in naturally occurring alpha activities. This leads to a lower minimum detectable activity.

Sample Counting and Data Evaluation

In our current methodology, 300-600 mL of untreated urine are acidified, concentrated, plancheted and counted for a 14-50 hour count. The amount of sample and the counting time is determined by the detection levels required by dosimetry for their current sampling schedule, as well as in-house rules that have been developed over the years. The following table summarizes these factors.

Table 20-3. Alpha Spectroscopy Counting Conditions and Reporting

Full Spectrum Range

3.10-6.21 MeV/256 channels

Count Time

1000 Minutes Pu-238/Pu-239 isotopic
1440 Minutes Gross Alpha Enriched Uranium/Am/Cm/Cf/Np
3000 Minutes (ALL Recounts of positive samples)

Vacuum Setting

500 micron

Sample Volumes

Isotopic Pu	600 mL
	Am,Cm,Np,Cf 300 mL
Enriched U	50 mL for isotopic analysis
	300 mL for gross alpha

Regions of Interest

Isotopic

Pu-242	3.68-4.94 MeV
Pu-239	4.95-5.21
Pu-238	5.28-5.54
U-234*	4.00-4.80

Gross Alpha

Am	3.69-6.10 MeV
Np	"
Cf	"
Cm	"
Enriched U	3.69-5.55 MeV

Reporting Guides

Analysis	Guide for a "POSITIVE" Analysis
Am,Cm,Cf	> 0.13 DPM/L
Pu-239	> 0.05
Pu-238	> 0.05
Np	> 0.13
Enriched U	> 0.90

* includes other U isotopes

Batch Counting Technique

In the batch counting technique, 3 laboratory blanks are "spiked" with a known quantity of the radionuclide of interest. These 3 spiked samples are included in a batch of 17 routine urine and urine blank samples. All the samples are chemically processed together and counted.

As long as suitable control is demonstrated over the entire batch such that the individual samples do not vary significantly one from the next, then these 3 "spikes" are representative of the entire batch.

After counting, the data is processed with the assumption that the recovery of radionuclide observed in the spiked samples is representative of the batch as a whole such that

$$\text{d/m/planchet} = (A - B)/(T \cdot E \cdot R)$$

where:

A = radionuclide integral count over the energy range appropriate to the alpha being measured

B = integral background counts for time interval T

T = time of count, minutes

E = detector counting efficiency

R = average recovery of all spiked blanks in batch

The final blank corrected activity is obtained by subtracting the mean of a suitable set of blank samples[4] from the above sample activity.

The individual spiked blank recovery is determined by

$$R = (D - C)/(F \cdot G)$$

where:

D = d/m/planchet of spike

C = blank correction, average d/m/planchet

F = volume of spike added, mL

G = activity concentration of spike, d/m/mL

An acceptable yield range for the batch method is taken as 20–120%. The lower limit is to insure an acceptable statistical error (20% at 2 sigma) on the number of net counts typically obtained following chemical processing and counting. It also allows one to meet the necessary DL limits for the isotopes in question. The upper limit is to guard against accidental "double spiking" of the samples in the laboratory.

Internal Spike Technique

In the internal spike technique, a known quantity of element of interest is added to each sample. The individual radionuclide that is added not one of those normally found in the sample being analyzed. After equilibration, the sample is processed chemically and counted.

The sample spectrum is analyzed by a computer program that has data on each individual detector with regards to its energy response and background counting characteristics. The program looks for the peak of the added radionuclide (such as Pu-242 for Pu analyses) and determines its peak limits. These limits are then used as representative of the overall spectral quality and peak shape associated with this sample.

Using these peak characteristics and the known energies of the other alpha emissions of interest, the appropriate alpha energy regions are integrated. Limits are imposed on such observables as the expected versus measured location of the peaks, number of total counts, etc. These are used as quality control factors. Deviations from the expected response are reported on the final report.

The final calculation of sample activity is

$$d/m/L = (C \cdot A \cdot K)/(D \cdot V \cdot T) - BL$$

where

C = counts in region of interest of assay radionuclide corrected for background

D = d/m spike added to sample

K = conversion factor to 1.0 L volume based on 20% acidification

A = counts in spike region of interest corrected for background

V = volume of acidified sample taken for analysis

T = count time in minutes

BL = mean of a suitable set of blank samples[5] corrected for the sample volume in question

An acceptable yield range for the method is taken as 20-120%. The lower limit is to insure an acceptable statistical error (20% at 2 sigma) on the number of net counts typically obtained following chemical processing and counting. It also allows one to meet the necessary DL limits for the isotopes in question. The upper limit is to guard against accidental "double spiking" of the samples in the laboratory.

Table 20-4. Current Measurable MDAs

Batch Alpha Count Method with solid state detectors using average recovery/batch with a 22 hr count

Cf-252	6.12-6.08 Mev	0.2 pCi/L
Am-241	5.48	0.2
Cm-244	5.80	0.2
U isotopes	4.77-4.19	0.4
Np-237	4.79	0.2

Beta proportional counter using average recovery/batch and a 20 minute count (via Y-90)

Sr-90	2.28 MeV(Y-90)	6.0

Liquid Scintillation counting for 0.5 minute count

H-3	0.0186 MeV	20,000

Liquid Scintillation counting for 50 minute count

H-3 (sample distilled before counting)	800

Electrodeposition with solid state detectors and internal tracer with a 16 hr count

Pu-238	5.49	0.015
Pu-239	5.15	0.015
U-234*	4.77-4.19	0.20

* includes other U isotopes

Decision Level (DL): The amount of a count or final instrument measurement of a quantity of analyte at or above which a decision is made that a positive quantity of the analyte is present.

$$DL = \frac{[DB \cdot B + 1.645\ S][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For DL, DB is the fractional systematic error bound in the determination of a blank population . The absolute magnitude of the maximum positive systematic error in the blank population should be used. B is the mean value of the blank population. S is the standard deviation in the net count of B.

Minimum Detectable Amount (MDA): The smallest amount of a radionuclide in a sample that shall be detected with a probability of a non-detection (Type II error) while accepting an a probability of erroneously detecting that radionuclide in an appropriate blank sample (Type I error). For this subcontract, the a and b probabilities are both st to 0.05.

$$MDA = \frac{[4.65(S_{Blk}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Eefficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For MDA , S_{Blk} is the standard deviation of the count from a set of acceptable blanks adjusted to the sample count time, Time is the sample count time, and the units correction factor adjusts the DL or MDA to a 1000 ml urine sample volume by taking into account the initial aliquot and any dilution (such as by acidification) of the urine sample prior to aliquoting. DL and MDA are based on trial use ANSI N13.30.

Future Development of High Resolution Ge Counting

Although several gamma emitting radionuclides were measured in urine during the 1950s-1960s (iodine, cesium, thorium) , there are currently no samples are processed in the bioassay laboratory for gamma emitters. These functions have been taken over by the Canberra Fast Scan units employed out in the areas. Future projects, however, intend to utilize Ge detectors for

- developing suitable calibration standards to allow the counting of dried fecal samples on a high purity Ge counter (This will allow more positive identification of radionuclides.)

- developing standards and procedures for counting tritium samples for medical or other radioisotopes when the liquid scintillation counters indicate excessive counts in a higher energy window above tritium (This will aid in reducing the number of "false" positive tritium samples resulting from this source.)

Gamma Screening of Dried Fecal Samples

The following sections outline the gamma screening of dried fecal samples.

Sample Pretreatment and Packaging

Fecal samples are placed in a waxed paper carton and dried in a microwave oven. This reduces the mass available to attenuate the low energy gamma emissions from plutonium-238 and americium-241. It also produces a sample suitable for shipment offsite to a contract laboratory for destructive actinide analysis.

Sample Counting and Data Evaluation

Dried fecal samples are counted on the phoswich counter interfaced to a Canberra S-100 multichannel analyzer for one hour. This counting time was chosen in order to rapidly obtain a screening value for the sample, as well as provide the necessary detection levels for this purpose. Although the counter has limited resolution, it is capable of resolving the 17 KeV group from plutonium-238/239 and the 60 KeV gamma from americium-241.

The phoswich counter has a greater overall efficiency than the higher resolution Ge detector, and as such allows samples to be screened in a more rapid manner.

Following the 60 minute count, the region of interest summations for the plutonium and americium regions are processed through a computer program which contains

- The measured phoswich detector efficiency for Am-241 of 11%.
- An empirically determined correction for sample mass determined from the counting of previous fecal samples of known activity, and a correction for Am-241 interference in the 17 Kev region such that

$$\text{Pu-238 nCi} = (C-Bkg)/(2220 \cdot T \cdot (0.026 \cdot EXP(-0.0081 \cdot W))) - 2.8 \cdot EXP(-0.0089 \cdot W) \cdot A$$

where

W = dry fecal weight, gms
C = sample counts in 17 Kev region
Bkg = 17 Kev background count for time T
T = count time, min
A = Am-241 nCi in sample

- An assumed ratio of 2.7 between the Pu-239/Pu-238 which has been determined based on relative emission intensities of the interfering energy lines.

Table 20-5. Fecal Analysis Conditions and Reporting

Count Time 60 Minutes

Region of Interest

Pu-238	5.90-27.8 KeV
Am-241	46.3-77.5 KeV

(Pu-239 determined from Pu-238 present)

Reporting

- any activity greater than the Decision Level (DL)
- any activity > 2.0 nCi/gm is flagged as being in excess of the DOT guideline

Sensitivity and Decision Levels

Calculated sensitivity (DL) based on background activities is in the range of 2-10 nanoCuries for Am-241, and 20-80 nanoCuries of Pu-238 and Pu-239. The formula used to determine DL is

Decision Level (DL): The amount of a count or final instrument measurement of a quantity of analyte at or above which a decision is made that a positive quantity of the analyte is present.

$$DL = \frac{[DB \cdot B + 1.645\ S][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For DL, DB is the fractional systematic error bound in the determination of a blank population . The absolute magnitude of the maximum positive systematic error in the blank population should be used. B is the mean value of the blank population. S is the standard deviation in the net count of B.

Minimum Detectable Amount (MDA): The smallest amount of a radionuclide in a sample that shall be detected with a probability of a non-detection (Type II error) while accepting an a probability of erroneously detecting that radionuclide in an appropriate blank sample (Type I error). For this subcontract, the a and b probabilities are both set to 0.05.

$$MDA = \frac{[4.65(S_{Blk}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For MDA , S_{Blk} is the standard deviation of the count from a set of acceptable blanks adjusted to the sample count time, Time is the sample count time, and the units correction factor adjusts the DL or MDA to a 1000 ml urine sample volume by taking into account the initial aliquot and any dilution (such as by acidification) of the urine sample prior to aliquoting. DL and MDA are based on trial use ANSI N13.30.

Proportional Counting of Sr-90/Y-90

Sample Treatment

The original urine sample is wet ashed as previously described and then extracted to isolate the daughter product Y-90. The Y-90 is then used to measure the quantity of Sr-90 present in the sample using the assumption that the two are in equilibrium. The final solution containing the extracted Y-90 is evaporated onto a steel counting planchet for counting under a gas proportional beta counter by environmental monitoring.

Sample Counting and Data Evaluation

Proportional counters are used in the measurement of Sr-90/Y-90 because these are pure beta emitters. Additionally, the low background obtainable (< 1 c/m) when compared to liquid scintillation counting (approximately 14 c/m) allows for a lower level of detection.

Count data obtained from environmental monitoring for the samples in question is analyzed by the "batch" counting technique previously described under the Alpha Spectroscopy Section. All samples are currently counted for 20 minutes using a 2 pi proportional counter which allows for an acceptable MDA as long as chemical recoveries are greater than 10%.

Two blank samples are spiked with known amounts of Sr-90 and are processed along with the routine urine analyses. The calculated recovery for these two samples are considered to be indicative of the entire batch. Recoveries as high as 80-90% are measured routinely.

Two additional planchets are prepared by pipetting a known amount of Sr-90 directly to the counting planchets. These sources are counted and are used to determine the counter's efficiency for the beta emissions at the time that the entire batch is counted. Typical efficiencies are 45-48%.

Initially, the extracting solution from the first separation is counted by environmental monitoring (referred to as count 1). Corrections to the count data account for the decay of the Y-90 following the first extraction step until it is counted. Samples that show no activity above the current action levels are reported as such since no extractable radionuclides (hence no Y-90 or Sr-90) are present.

In the case where activity above the current action levels is detected, an additional strip of the Sr-90 containing phase is performed. The first separation step should have removed all extractable radionuclides. Any new radionuclides are therefore the result of Y-90 growth from Sr-90. Corrections to the count data take into account the percentage growth of the Y-90 daughter since the time of the first separation step in reporting the Sr-90 activity (referred to as count 2).

The formula used to calculate the activity is as follows

For count 1 data

$$\text{pCi/planchet} = (TC-Bkg)/(t \cdot CF)$$

where:

TC	=	total counts for count 1
Bkg	=	counter background counts for time t
t	=	count time in min
CF	=	conversion factor (counter efficiency · 2.2 d/m/pCi)

where the spike recovery is

Recovery	=	[(pCi/planchet)-(pCi of blank)]/pCi of added spike
pCi/planchet	=	Y-90 on planchet corrected for decay since separation
pCi of blank	=	average of blank controls
pCi of added	=	Sr-90 (pCi) added to control[6] spike

The initial estimate of Sr-90 is then

$$\text{pCi/L} = (\text{pCi/planchet})(1000\ \text{mL})/(\text{recovery})(V)$$

where

pCi/planchet	=	count 1 activity
V	=	aliquot volume,mL
1000	=	reporting volume, mL

For count 2 if it is required, the amount of Sr is determined as

$$MDA = \frac{[4.65(S_{Blk}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

$$\text{Sr-90 nCi/L} = \frac{2 \cdot [(\text{Corrected pCi/planchet}) - (\text{pCi of blank})]}{1000 \cdot \text{Recovery} \cdot (\text{T-90/Sr-90 ratio})}$$

where

2	=	volume correction factor (1000/500)
Corrected pCi/planchet	=	count 2 corrected for decay since second separation
pCi of blank	=	average pCi/planchet of blank control samples
recovery	=	see above
1000	=	pCi/nCi conversion factor
Y-90/Sr-90 ratio	=	degree of equilibrium of Y-90 daughter with Sr-90 in solution at time of second separation

Table 20-6. Sr Analysis Conditions and Reporting

Count Time

20 Minutes with a 2 pi P-10 proportional counter

Typical Analysis Conditions

Sample Volume	500 mL
Chemical recovery	60%
Counting Efficiency	42%

Reporting Guides

>0.27 nCi/L	Notify supervisor
>0.13 nCi/L	Resample
>0.067 nCi/L	Perform second Sr strip of solution

Minimum Detectable Activity and Decision Level

The attached table lists the currently measured MDAs based on a background count rate of approximately 1.4 c/m on the proportional counter utilized for these measurements.

Decision Level (DL): The amount of a count or final instrument measurement of a quantity of analyte at or above which a decision is made that a positive quantity of the analyte is present.

$$DL = \frac{[DB \cdot B + 1.645\ S][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For DL, DB is the fractional systematic error bound in the determination of a blank population . The absolute magnitude of the maximum positive systematic error in the blank population should be used. B is the mean value of the blank population. S is the standard deviation in the net count of B.

Minimum Detectable Amount (MDA): The smallest amount of a radionuclide in a sample that shall be detected with a probability of a non-detection (Type II error) while accepting an a probability of erroneously detecting that radionuclide in an appropriate blank sample (Type I error). For this subcontract, the a and b probabilities are both set to 0.05.

$$MDA = \frac{[4.65(S_{Blk}) + 3][\text{Unit Correction Factor}]}{\text{Time} \cdot \text{Counter Efficiency} \cdot \text{Recovery} \cdot \text{Volume}}$$

For MDA , S_{Blk} is the standard deviation of the count from a set of acceptable blanks adjusted to the sample count time, Time is the sample count time, and the units correction factor adjusts the DL or MDA to a 1000 ml urine sample volume by taking into account the initial aliquot and any dilution (such as by acidification) of the urine sample prior to aliquoting. DL and MDA are based on trial use ANSI N13.30.

Offsite Contractor Laboratory Analyses

The analyses performed by the contract laboratory will include selected radionuclides at the MDAs specified in N13.30.

The laboratory will have demonstrated the ability to perform these analyses in accordance with the precision and bias requirements of N13.30.

QA Requirements

The contract laboratory will have a WSRC approved quality assurance plan to provide assurance that proper control is maintained over the analyses.

Records and Reports

The vendor will document and retain, for their record, the calculations, raw data, lab books, computer files, and final data that pertain to the reliability and quality of the sample analyses results for the duration of the contract. These records (or copies thereof) will then be transferred to EHP within 6 months of the end of the contract.

Summary reports will be provided on a regular basis to document the results and quality control data relating to the analyses. These reports are in both paper and electronic formats.

These records will provide all the necessary short term and permanent documentation to support the quality and documentation requirements of their analyses.

Notes

(1) Fluorescence involves the absorption of a photon which excites the molecule. De-excitation occurs by the emission of a photon with an associated relaxation "lifetime" on the order of nanoseconds.

(2) Phosphorescence involves a similar process, except the relaxation involves an electronic state from which emission of a photon is "forbidden". In this way, relaxation lifetimes range from microseconds to seconds.

(3) The sample volume is based on the maximum loading of the scintillation cocktail that is practically achievable. The count time is chosen to be the maximum achievable and still result in the analysis of a single sample in one 8-hour work shift.

(4) Using the Dixon Test for outliers to define a suitable set of blanks

(5) Determined by the Dixon Test for outliers

(6) Most Sr-90 solutions are in equilibrium with Y90 already, and we want the amount of Sr-90 (not Sr-90 + Y-90) added.

Part IV

Part IV - *In Vivo* Bioassay

21 - Introduction
22 - Facility Description
23 - Chest Counting
24 - Whole Body Counting

Chapter 21

Introduction

Chapter 21 Preview

- Overview of Part IV
- Administration

21

Introduction

Radiation protection programs are mandated for DOE contractors by DOE Order 5480.11, "Radiation Protection for Occupational Workers". It is the policy of DOE to implement radiation protection standards consistent with the Presidential approved guidance to Federal Agencies promulgated by the EPA and based on recommendations by authoritative organizations.

The in vivo measurement program is the accepted good practice method of implementing the applicable portion of this policy. The in vivo measurement program at SRS consists of two facets; chest counting and whole body counting.

Chest Counts

Chest counts are quantitative investigations for the presence of the actinide radionuclides in the lungs. The chest counts are 30 minute counts performed in the steel shielded room with a state-of-the-art germanium detector system and a phoswich detector system for backup. Chest counting provides rapid evaluation for incident evaluation, long term monitoring for known intakes, and monitoring for possible accumulation of highly insoluble chemical forms of the actinides.

Whole body counts

Whole body counting measures photon-emitting fission products and neutron activation products at any location in the body. Whole body counts utilize standup whole body counters, which are Fastscan commercial counters with NaI detectors. Whole body counting is used in place of urine sampling for fission products and activation products.

Facility

The Whole Body Counter Facility is located in building 735-A. The facility contains a stationary standup whole body counter and the shielded room that is used for chest counting. Two mobile standup whole body counters are located in trucks and are operated at various site locations.

Relation To Other Health Protection Functions

The in vivo counting program is an integral portion of the Health Protection program. Major facets of the in vivo program are routine and special counts. The routine counting program provides prospective monitoring of routine plant operations and the special counting program provides retrospective monitoring for unusual operating conditions. For most photon emitting radioisotopes, measurements of dosimetrically insignificant amounts are straightforward and are accomplished quickly and easily. The uncertainty of body burden calculation by modeling of excreta data is avoided.

Measurements of actinide radioisotopes in the lungs are difficult because the photon emissions are both low yield and low energy. Special counts are part of the investigation following incidents in which employees have the potential for assimilation of radioactive material. Chest counting is the initial assessment and sets an upper limit on the intake. The special in vivo measurements are used in conjunction with in vitro samples to further delineate any assimilation.

Whole body counting using the standup whole body counter is the method of choice at this site for measurement of gamma emitting fission products and neutron activation products. This direct counting method has replaced the urine sampling program for monitoring of gamma-emitting radioisotopes. The standup whole body counter measures lower values than ordinary radiochemical procedures and eliminates modeling the excreta data.

History of *In Vivo* Measurements

The SRS Whole Body Counter was completed in 1960. The detectors available limited gamma measurements to energies above about 100 keV. The initial detector was an 8" diameter by 4" thick NaI(Tl) detector viewed by three 3" photomultiplier tubes. The counting was the 40 cm arc, also known as the Argonne chair. The person sat in a reclining chair and the detector was suspended above the pelvic area. The major portion of the body was located in an arc of about 40 cm, therefore the 40 cm arc position. Limited investigational counts of the lungs were available with the detector placed in contact with the chest. Radionuclides counted were fission products and neutron activation products with a minimum energy of 135 keV from Ce-144 and a maximum energy of 1332 keV of Co-60. Calibrations utilized the Remcal phantom which contained the major organs. Calibration involved filling appropriate organs with an aqueous solution containing the radionuclides. Radioactive material was obtained from the Health Protection environmental monitoring laboratory.

In the late 1960's, the phoswich detectors were developed for low energy photon and x-ray measurement. A dual phoswich system began operation in in the shielded room 1972. The counting geometry was changed to supine. Six detectors were used; 2 phoswich on the chest for the transuranic radionuclides and four 5" diameter by 4" thick NaI(Tl) under the bed for fission product and neutron

activation products. In 1976, side-looking germanium detectors were added to measure the higher energy radioisotopes in the lungs. Calibration used a torso skeleton of Asian origin. Lungs were shaped from molds obtained from the Remcal phantom. Lungs were prepared by mixing lung tissue powder, catalyst, and radioactive material. The lung tissue powder was especially manufactured for this use. This composite material was widely used for lung phantoms in that era. The radioactive material was obtained from Health Protection environmental monitoring laboratory. Phantom fabrication was performed by Whole Body Counter personnel.

The Humanoid torso phantom developed by Lawrence Livermore National Laboratory was major milestone in chest counting. A second generation phantom was acquired by SRS in 1981. The phantom provided an anatomically and radiologically correct phantom for all laboratories. Radioactive organs are compared and traceable so that continuity exists among laboratories.

Germanium detectors have become the detectors of choice for measurement of actinide radioisotopes in the lungs. The SRS germanium detector system became operational in December, 1989, replacing the phoswich detector system. The system consists of 6 detectors placed against the chest. The counting geometry was changed to a reclining chair. Calibration utilized the Humanoid torso phantom.

Administration

Scheduling

Employees who enter areas that have the potential for assimilation of radioisotopes must have routine whole body and/or chest counts as required by the schedule in Chapter 10. The standup whole body counter throughput is adequate so that scheduling is not necessary. Counts are made on a "walk-in" basis. The counting time is 2 minutes and the person must remain in the facility until the computer analysis is complete, nominally 3 minutes. One shielded room limits the number of chest counts to about 3000 total counts annually. In order to effectively utilize the chest counting facilities, an appointment system is necessary. The number of employees scheduled for counting is administratively limited to maximize but not over schedule the use of the chest counter.

Count Requirements

Designated employees must obtain a routine chest count and/or a whole body count. In addition to the routine program, personnel are counted for special, termination, base line, recount, and personal reasons. HPO is responsible for identifying the personnel and count type required.

Routine

The routine program encompasses the requirement that all employees who enter a radiation control area must have a whole body count or a chest count at the intervals required in Chapter 10. The routine chest counts of employees are scheduled as described in Scheduling. The routine whole body counts are available on a "walk-in" basis. The routine counts utilize the major time portion of the counters.

Even though the detection capability of the chest counter does not equal the ability of in vitro bioassay, it is considered good practice to monitor employees by the direct counting method. Routine chest counting monitors for buildup of highly insoluble forms of actinides and ingrowth of daughter species. Regular operation of the counter keeps the equipment and personnel ready for special counts. The routine whole body counts provide the primary radiobioassay for gamma emitting fission products and activation products.

Special

Special counts are for employees who are involved in any incident that has the potential for assimilation of radioactive materials. These counts are the highest priority and supersede the other counting programs. When an incident occurs, the employees are first decontaminated, if necessary, and then brought to a whole body counter or to the chest counter.

All employees involved in any unplanned event which releases radioactivity are counted as soon as possible. This immediately establishes an upper limit of the radioactivity in the body. The upper limit established by the chest count is not acceptably low, therefore additional samples are collected for in vitro analysis.

Baseline

Baseline counts are given all new employees within a few days of their reporting to work. All new employees receive a whole body count and new employees who have prior occupation exposure to actinide elements receive a base line chest count.

Baseline counts establish the amount of natural or man-made radioactivity that the person contains when reporting for work. Of particular interest are those employees who have previously had occupational exposure to radioactive material.

Termination

Termination counts are required by all employees who leave the site. All employees receive a whole body count as a minimum, and a chest count if they have worked in areas with exposure to actinide radioisotopes.

The termination counts along with the termination in-vitro bioassay samples establish the amount of radioactivity that the employee contains when leaving the site.

Visitors and Subcontractors

Visitors and subcontractors who enter radiation controlled areas receive entry and exit whole body counts. In addition, subcontractors who work in the areas with actinide radionuclides receive entry and exit chest counts.

Entry and exit counts along with urine samples verify that visitors and subcontractors have not assimilated radioactive material as a result of their presence at SRS.

Recount

Employees are recounted for several reasons. Employees who have received medical radioisotopes are recounted until the radioisotope is below detectable limits. Technical support personnel may request recounts after the routine count reports are reviewed when there is any indication of uncertainty such as electrical disturbances or unidentifiable energies. Detectable assimilations are recounted to follow the elimination of the material. Internal Dosimetry personnel may request recounts for verification purposes.

Personal

Employees may be counted by submission of a personal request through their management.

Reporting Results

Printed Report

The final report of a count is a printed report that is deposited in the employee's permanent dosimetry record. The report is computer generated and it is evaluated, signed, and dated by whole body counter personnel.

Telephone Report

Technical personnel may report results by telephone to HPO management in the operating area whenever there is immediate concern about the count results. HPO are responsible to communicate results to other plant personnel, as needed. No results are given to other than HPO management.

Personal Explanation

Employees may have any count verbally explained by technical personnel at the counter. In addition, a written copy of the count results with explanation will be given any employee upon written request routed through their management to Dosimetry Management.

Chapter 22

Facility Description

Chapter 22 Preview

- Chest Counter
- Standup Whole Body Counter

22

Facility Description

The Whole Body Counter facility contains a steel shielded counting room, a separate, free standing whole body counter, a small reception area, electronics space, a change room, and a rest room. The facility was built in 1959-60, a period of high fallout from atmospheric weapons testing. All materials used in construction were selected on the basis of minimum radioactivity, and handled protectively in order to minimize fallout contamination. Sand, gravel, and cement for the concrete are especially susceptible to contain thorium and uranium daughters, and were carefully selected and handled. The normally used site wall construction material, transite, contains low levels of natural radioactivity. Masonite was used for the walls. The floor covering was selected solid vinyl, chosen on the basis of minimum radioactivity content.

Chest Counter

The chest counter is the assemblage of the shielded room, radiation detectors, electronics, and analysis software.

Shielded Room

The chest counter radiation detectors are housed in the steel shielded counting room. The room is 8' by 10' with 8' ceilings and is constructed of 12" thick pre-World War II steel. This older material precludes any fallout radioactivity from atomic weapons tests. This steel was rolled for belt armor plate for a battle ship but was not used. The steel room is lined with 1/4" lead with no graded shielding. The room was built before the capability existed to measure low energy x-rays and photons. For optimum low energy counting performance, a shielded room should have a graded liner, typically cadmium and copper, or tin and stainless steel. The new facility shielded rooms, scheduled for operation in April 1993, will have graded liners.

Radiation Detectors

Detectors in use are germanium semiconductors and phoswich scintillators for actinide radioisotopes.

Germanium Detectors

The germanium detectors are low energy extended energy range detectors, 51 mm diameter by 20 mm thick. Each detector has an ultra-thin boron implanted P+ outer contact on the front face and cylindrical wall. Lithium diffused on the rear surface is the N+ contact. This design minimizes the capacitance of the detector compared to a conventional planar detector, resulting in improved energy resolution. The 20 mm detector thickness is optimum for efficient measurement of the energy range of 8 keV to 400 keV. The 51 mm diameter is the largest diameter commonly available. These detector dimensions are the similar to those used at most other DOE chest counters. The detector is fitted with a 0.51 mm selected beryllium entrance window. The beryllium window provides the maximum mechanical protection with the minimum radiation attenuation. The beryllium is specially refined to minimize radioactivity content. Six germanium detectors are arranged in groups of three for use on each side of the chest. Each detector is fitted to an individual 1 liter dewar.

The electronic charge from photon interactions is collected by a charge sensitive preamplifier mounted within the detector/dewar assembly. The preamplifier FET is liquid nitrogen cooled to minimize electronic noise. The negative signal is input through an amplifier, a mixer/router, and an analog-to-digital-converter (ADC) to the multichannel analyzer (MCA).

Phoswich Detectors

Phoswich scintillation detectors are used as backup when the germanium detectors are not operational. Different size phoswich detectors are used for measurement of uranium and for measurement of plutonium. The detectors are sized to most efficiently measure the emitted photons.

Phoswich Operation

The phoswich detector is a phosphor sandwich of the scintillators NaI(Tl) and CsI(Tl) viewed by a single photomultiplier tube. The detector is a right circular cylinder with a radiation entrance window adjacent to the face of the NaI(Tl) and the photomultiplier optically coupled to the face of the CsI(Tl). Radiation impinging on the entrance window of the detector has several probable interactions. The low energies limit interactions to the photoelectric effect and/or Compton scattering. The desired interaction is photoelectric effect with absorption in the NaI(Tl). The thin NaI(Tl) minimizes Compton scattering. A highly probable interaction for higher energies is a Compton scattering in the NaI(Tl) and additional Compton scattering in the CsI(Tl) portion of the detector. A third probability is no interaction in the NaI(Tl) and Compton scattering or photoelectric effect interactions in the CsI(Tl).

The characteristics of the scintillators and the associated electronics permit the acceptance of only signals from either photoelectric effect or Compton scattering in the NaI(Tl) with no interaction in the CsI(Tl). Radiation excitation of the two materials trigger decay with differing times, thus resulting in differing rise times

of the electrical pulses generated from the photomultiplier tube. Signals from the photomultiplier tubes pass through a preamplifier and then to a Pulse Shape Analyzer (PSA). The PSA contains two doubly differentiated amplifiers and a time-to-amplifier converter. A double differentiated amplifier displays a sine wave shape output which is proportional to the rise time of the pulse. The crossover point is the time of maximum value of the input pulse. Differing rise times of pulses display differing crossover points. The entering pulse is input simultaneously to the two amplifiers. The acceptable time between the crossover points is adjusted on the PSA. A logic pulse is output for pulses which satisfy the time requirement. For timing purposes, the short amplifier output from the PSA is routed through a time delay amplifier with 2 microseconds delay. The output of the delay amplifier is input into a linear gate. The logic signal from the PSA triggers the linear gate. The output from the linear gate is input to a Canberra model 8076 ADC for storage in the MCA.

Plutonium Detectors

The plutonium phoswich detectors are designed to most efficiently measure the energy range of 14 keV to 130 keV. The phoswich detectors are 12 cm diameter with a 3 mm thick front detector of NaI(Tl) and a 51 mm thick rear detector of CsI(Tl). The detectors have 0.51 mm thick selected beryllium entrance windows. Beryllium is the window material of choice because it minimizes attenuation of the low energy radiation and provides a light and moisture shield. Selected beryllium is chosen based on minimum radioactivity content.

Uranium Detectors

The uranium phoswich detectors measure radiation from 10 keV to 250 keV, but most efficiently from 60 keV to 250 keV. The uranium phoswich detectors are 12 cm diameter with 12 mm thick detector of NaI(Tl) in front of 51 mm of CsI(Tl). The detectors have 0.51 mm thick selected aluminum entrance windows. The primary energy range of interest is above 60 keV and does not require the use of beryllium. Aluminum is a less expensive, satisfactory material for the window and provides a light and moisture shield with satisfactory attenuation of the radiation.

Electronics

The chest counter electronics include the PDP 11/23+ computer, the MCA, ADCs, mixer-routers, amplifiers, high voltage supplies, and the pulse shape discrimination system. Data from the chest counter is collected by a Canberra Series 90 MCA. The computer controls all MCA collection and transfer operations. Operator input is minimized except for energy alignment and demographic data input for personnel counts.

Germanium measurements collect six spectra, one each of the detectors. These spectra may be displayed separately, and are summed by the computer prior to analysis. Gain adjustment is by an ultra fine gain potentiometer added to each of 6 amplifiers, one of each of the detectors.

The plutonium phoswich collects data into a a 1024 channel spectra. This number of data channels is excessive for sodium iodide data, but it was used historically and has been retained. No detriment exists from the excessive number of data channels and the analysis software would require modification to use a smaller number of data channels. Data from the two detectors are summed at the amplifier into a single spectrum. Gain adjustment is via high voltage to the individual detectors.

The uranium phoswich collects data into a 256 channel spectra. Data from the two uranium phoswich are summed at the preamplifier into a single spectrum. Gain adjustment is by the high voltage to the detectors. The arrangement of different groups for uranium and plutonium permits the changing from plutonium to uranium detectors with no change in the MCA setup.

Standup Whole Body Counter

The standup whole body counter is a commercially available Fastscan counter from Canberra Industries. The counter consists of the shield, detectors, and the electronics and computer contained in the console.

Shield

The counter shield is a shadow shield in which a person stands. The shield is 4" thick selected steel, but not pre World War II steel. The shield is built of 1/4" plates which fit into frames. This design facilitates portability. The largest portion is the inner "tub" which weighs 600 pounds. The complete shield weighs 10,000 pounds. The completed shield has an entrance height of 75 inches high and an entrance width of 14 inches.

Detectors

The counter has 2 NaI(Tl) detectors, one mounted above the other with a 1" space between the detectors. Each detector is 4" by 4" by 16" long, located end to end so as to act as a 33" detector located parallel to the body axis. Each detector is viewed by one photomultiplier on the end of the detector.

Electronics

The electronics consist of a PDP micro 11/23 computer, an MCA, ADC, mixer-router, preamplifiers, amplifiers, and high voltage power supplies. The 2 detectors are operated at the same high voltage. Gain is adjusted by use of a separate amplifier for each detector. Two spectra are collected by use of a mixer-router. The two spectra of 512 data channels are collected separately from the upper and lower detectors. The spectra are displayed separately and may be useful in locating the source of radioactivity which is not distributed uniformly in the body. The spectra are summed for analysis in order to improve sensitivity.

Chapter 23

Chest Counting

Chapter 23 Preview

- Germanium Detector Measurements
- Phoswich Detector Operations
- Calibration

23

Chest Counting

Chest counting is the direct measurement of radioactive material in the lungs and bronchial region by detection of the photons emanating from the chest. The procedure at SRS measures the actinide radionuclides. These radionuclides emit low energy photons and their detection is severely impacted by chest wall tissue and low energy natural background clutter.

Chest counting utilizes the shielded room described in Chapter 22, Page 2. The shielded room is designed to minimize the effect of natural background radiation on the low energy spectra region. Radiation that is degraded in the shield is further minimized by the lead lining. The lead x-ray of 84 keV is not evident in the background energy spectra in the room. A graded lining of tin and stainless steel, or cadmium and copper, is desirable but not present inside the lead lining.

Measurement Geometry

Personnel who are counted sit in a reclining chair whose back is 30 degrees from vertical. This geometry is a change from the previous supine position. The emitted low energy photons are significantly attenuated by the chest wall tissue between the lungs and the detectors. When a person is supine, gravity pulls overlying chest towards the collarbone. Positioning a person in a reclining chair minimizes the tissue thickness in the counting area by utilizing the normal tissue sagging induced by this geometry and aging. With younger persons there is minimal sagging, but there is no increase in thickness with the reclining chair. This reclining counting geometry is the method of choice with most chest counters.

Radionuclides Measured

The chest counter measures the actinide radionuclides. These nuclides primarily decay by alpha emission, but most have associated low energy photons, or decay to excited states with x-rays transitions to ground state. These photons and x-rays are the only radiation which can penetrate the chest wall tissue thickness and be measured directly.

The radionuclides that are measured result from several site programs. The primary radionuclides of concern are the plutonium, uranium, and americium isotopes, which result the site's production of nuclear weapons materials. Special programs in the past also produced the heavier transuranic radionuclides such as Cm-242, Cm-244, Cf-252, and Am-243. The fuel fabrication and uranium recycling programs involve thorium and uranium radionuclides.

The radionuclides are measured by taking advantage of various methods of photon emissions. Plutonium and curium are measured by the L x-rays that are emitted after the primary alpha decay to an excited level. Americium, neptunium, and uranium-235 are measured by low energy gamma emissions. Uranium-238 is measured by equilibrium thorium-234 daughter low energy gamma rays. Table 23-1 shows the radionuclides with their associated energies and yields. The emissions of these radionuclides are directly measurable with the chest counter.

Table 23-1. Radionuclides Measured By Chest Counting

Radionuclide	Energy(keV)	Yield(%)
Th-228	84.40	1.19
"	131.6	0.11
Th-231	25.64	14.8
"	84.21	6.50
Th-232	59.00	0.19
"	106.0	0.12
Th-234	63.29	3.81
"	92.38	2.73
"	92.80	2.69
Np-237	29.37	14.0
"	86.50	12.6
U-234	19.09	1.25
"	53.20	0.12
U-235	143.8	10.8
"	185.7	54.0
U-238	19.09	1.02
Pu-238	17.06	5.57
"	20.29	1.28
Pu-239	17.06	2.09
"	20.29	0.49
Pu-240	17.06	5.32
"	20.29	1.22
Am-241	59.54	35.7
Am-243	43.53	5.54
"	74.67	66.0
Cm-242	18.06	5.20
"	21.55	1.18
Cm-244	18.06	4.83
"	21.55	1.09
Cf-252	19.22	3.24
"	23.00	0.76

The above list of radionuclides are measured when counting with the germanium detectors. The phoswich detectors do not have satisfactory energy range for the the complete list of radionuclides. The plutonium phoswich measures the energy range for Pu-238, Pu-239, Am-241, and Cm-244. The uranium phoswich measures the energy range for U-235, and U-238 by measurement of Th-234.

Personnel Preparation

Personnel to be measured report to the WBC facility for their appointment. They are directed to shower, shampoo, and change into a surgical scrub suit provided

in order to minimize the presence of external contamination. The counter can not differentiate external and internal contamination. Any radiation measured must be considered internal until further counting provides additional information.

In the event any radioactivity other than naturally occurring is measured, the person is asked to shower again, change into another surgical scrub suit, and be recounted. The second shower uses the cleaning agents Tide and Alconox for more thorough cleaning than with conventional soap. Alconox is a a surfactant that enhances the cleaning action of Tide.

Dressed in the scrub suit, the height and weight are measured and recorded for input into the chest wall thickness algorithm in the following section. The person then enters the shielded room and sits in the reclining chair.

The detectors are placed in contact with the chest, one on each side of the chest. The detectors are adjusted with the face of the detectors parallel with and in contact with the surface of the chest wall on each side. The upper edges of the detectors are just below the lower edges of the collar bones. The detectors for each side of the chest will normally contact each other at the middle of the chest, except for persons with broad chests. Years of experience has shown that this placement minimizes the chest wall tissue thickness interference while maintaining proximity to the lung and bronchial region, thereby optimizing the number of photons measured.

Chest Wall Thickness

The low energy x-rays are strongly attenuated by the tissue overlaying the lungs, the chest wall tissue thickness (CWT). Chest measurements are analogous to counting a source through a shield and calculating the activity of the source. The CWT is the shielding and must be accurately determined for quantitative measurements. The desirable method is use of ultrasound with delineation of adipose and muscle tissue. The differing density and thickness of muscle and adipose results in significant differences in attenuation corrections.

Ultrasound measurements are not currently practical on site. The alternative method is use of published biometric relations to establish CWT. Six journal articles were reviewed for CWT measurement relations to weight and height. Four of these showed similar results for male subjects. Based on these published results, a composite relation was selected as the most representative for male subjects[1,2,3,4].

The chosen relation is:

$$CWT = 60 \cdot W/H - 1.0$$

where: CWT = chest wall thickness in mm.
W = weight in kg.
H = height in cm.

The relation is also used at SRS for female subjects. This is not satisfactory, but it is the only choice available with male operators because it not possible to make the measurements as described by Berger,and Lane[5].

This method will change significantly in the future. A project has begun with the University of South Carolina to develop a method to measure chest wall tissue

thickness. The project will have two phases. The initial phase is to develop a better estimation method using CAT scans, x-rays, or other methods. This phase is interim until operating space for ultrasound is available with completion of the New Whole Body Counter Facility in April, 1993. The second phase is to determine the ultrasound hardware and software for the new facility which will measure the tissue thickness and delineate adipose tissue from muscle.

Germanium Detector Measurements

Energy Alignment

The 6 germanium detectors collect spectra individually as described in Chapter 22. The energy alignment is 8 keV to 400 keV in 2048 data channels, approximately 0.2 keV/channel. This energy alignment gives an adequate number of channels within a photopeak for full-width-half-maximum (FWHM) calculation. The FWHM calculation is a Gaussian fit calculation which is most satisfactory with a minimum of seven points in the photopeak area.

Energy alignment consists of the amplifier gain adjustment and the ADC zero adjustment. The zero adjustment sets the ADC to a reproducible zero point for the gain curve. Monthly, this alignment is performed by setting the 121.8 keV photopeak at channel 607.0 and the 344.0 keV peak at channel 1715.0, the calibration gain setting. The zero of the ADC and gain of the amplifier is adjusted for detector 1 for ADC 1 and detector 4 for ADC 2. The zero of the ADC must not be adjusted for the other detectors after it is properly adjusted. The two photopeaks are checked for detectors 2 and 3 for ADC 1 with amplifier gain adjustments only, and detectors 5 and 6 for ADC 2. Both peaks should be satisfactory with no further adjustment of the zero. If any peaks are not correctly located, an iterative adjustment of ADC zero and amplifier gain must proceed.

The energy alignment uses Eu-152 with a 13.6 year halflife. This nuclide is particularly suited for the purpose because of the combination of long halflife and multiple photons.

The energy resolution of the germanium detectors is such that background subtraction is not necessary. The change in count rate over the energy range of a photopeak is not significant. This eliminates the uncertainty of determining the "clean" person background as is required with plutonium phoswich measurements.

Daily Operation Check

The daily operational check is the energy alignment of each of the 6 detectors and measurement of a chest phantom containing natural uranium (Th-234 and U-235).

Alignment Check

The energy alignment uses the Eu-152 source. The primary alignment energy is the 121.8 keV peak. To minimize peak degradation from summing, the alignment must be precise. The centroid channel for the 121.8 keV photopeak must be between channel 606.80 and 607.20 as calculated by the FWHM function of the MCA. The ideal alignment is exactly channel 607.00 for all detectors, however, the alignment limits of ± 0.20 channels are achievable in a practical operation time and produce sum peaks with minimal degradation.

During alignment of the detectors, the MCA is programmed to calculate the FWHM. When the alignment is adjusted for each detector, the FWHM value is entered in the source log book. This data gives a daily check of each detector performance.

Phantom Count

A chest phantom is used as a daily check source for the detectors and software. This phantom is not designed to be a replicate phantom to a human, but is to check operation of the total system. The phantom is a distributed natural uranium source constructed of scattering material similar to tissue with markings for reproducible positioning of the detectors.

The phantom is 13.5" by 12" by 4" and constructed of Plexiglass. It is composed of a front of 5 cm thick, a 4 cm space and a natural uranium planar source distributed on filter paper with 1 cm Plexiglass back. The chest phantom count checks operation of the detector alignment, software summing, and software analysis. Whenever calibration is performed, the phantom is immediately counted 10 times to establish the response and limits of reproducibility. The 3 sigma limits are entered into the software as acceptable response limits. The daily phantom count response must be within these limits or an explanation entered. Limits are established for nuclide amount, FWHM, and centroid channel for the photons at 63.3 keV and 185.7 keV. All values, gained along with the acceptable limits, are maintained in the a software file, which is printed monthly.

Data Analysis

The analysis software ABACOS-LE was developed by Canberra Industries for low activity spectra, typically found in in vivo counters. This method uses a modified peak analysis technique that determines the areas of photopeaks in the sample spectrum after the underlying continuum background has been subtracted. This method resolves multiple overlapping peaks. It uses two consecutive methods for determining peak location; a "library driven" peak search using a nuclide library file; and, a "sliding peak" analysis to locate spectral peaks not included in the library.

Background Determination

The initial operation is an computation of a background spectrum. This is done by a systematic smoothing of the original spectrum in order to erode any photopeaks into the continuum. The smoothing method examines each division of the spectrum and compares it to the average of the two adjacent divisions. If the number of counts in a division is greater than the average of the adjacent divisions, it is reduced to the average. This process is repeated several times for the entire spectrum. This resulting "background" gives a smooth continuum which is then subtracted from the original spectrum to delineate the photopeaks.

Prior to smoothing, the energy axis is transformed into a grid of "bins" with the width of the bins a function of the FWHM of the photopeak centered in that bin. Bin widths are chosen so that a spectral peak will be spanned by five to seven bins. This choice of bin width allows information known about the spectrum to be explicitly folded into the averaging algorithm. The bin widths can change by a factor of ten or more from the low energy to the high energy end of a typical spectrum.

The bins are further assigned a set of four repeating plies such that each bin has an associated ply. Averaging is then done between adjacent bins of the same ply. For example, the counts in any bin are reduced to the average of adjacent bins of the same ply. Several averaging passes are made to establish the continuum background. Thus, spectral peaks will be eroded into the continuum. The background subtraction algorithm is designed to be invariant with respect to counting rates and the slope of the continuum.

There are three cases in which the background determination method can produce a poor estimation of the background continuum: 1) if the continuum is concave downward, 2) if no bin from any ply can entirely fit into a particular "true minimum", and 3) if there are so many peaks in the spectrum that no point in the spectrum is a good estimate of the continuum background. The first two items are adequately handled by the peak determination routine. The third item is a minor problem with germanium detectors, and occurs very rarely in in vivo measurements.

Peak Determination

The peak search methods are the "library driven" and the more general "peak driven". The "library driven" peak search rapidly determines if peaks are present at expected locations in a spectrum. The "peak driven" determines the presence of any unexpected peaks.

The energy tolerance range for "peak driven" versus "library driven" is user adjustable. The vendor recommended value is 1.5 FWHM, i.e. if an unidentified peak is within 1.5 FWHM of a library peak, the unidentified peak is shifted to the library peak, showing the percent gain shift in the energy calculated. (FWHM ranges from 0.5 keV to 1.5 keV for the energy range of 8 keV to 400 keV.)

The peak determination uses a linear least squares fitting algorithm with constraints on peak height and gain shift. No peak with negative area is retained in the fit. If the gain shift of the centroid is greater than $0.05 \cdot FWHM$, the fitting equation is recalculated with a modified energy calibration. The fitting process is then retried until a zero gain shift is found.

The shape of the spectral peak is represented by a Gaussian function because the peak areas can be easily calculated. If multiple unresolved peaks are present, then additional Gaussian terms can be added automatically, one for each peak.

The software for the germanium detectors has an added peak area calculation. In order to improve sensitivity for low level, statistically poor photopeaks, the net counts are summed in the peak with a linear extrapolation of background. This method is noted on the output as "Alternate Peak Area Used".

Analysis

The analysis is a function of calibrations of energy range, FWHM values and efficiency. The above methodology locates the peaks of the spectra and determines the energy via the energy calibration. With the energy determined, the peak is fit to a Gaussian shape as directed by the FWHM calibration data. With the peak energy and peak area determined, the amount of radioactivity is calculated using the appropriate efficiency curve for the chest wall thickness and the photon yield

data from the library file. A peak is reported as an "unknown" for any energy not in the library file. The radioactivity is calculated assuming a photon yield of 100%.

Two printouts are generated, the preliminary and the final. The preliminary report shows parameters of photopeaks found. The error is the counting error at a user selected confidence limit. The vendor suggested operational value is 1.96 sigma, 95% confidence limit. An error value greater than 50% is suspect and must be confirmed by an investigational count. The fit value is a measure of the Gaussian shape of the peak. A value of 1.00 is an exact fit. Acceptable fit values range from 0.5 to 3.0. The gain gives the percentage change in gain to fit the peak to the peak in the nuclide library. A gain value greater than ± 3% is suspect for the peak energy.

Minimum Detectable Amount

The minimum detectable amount is calculated as a function of the gross count and the FWHM. The MDA is defined as:

$$MDA = \frac{A + S \cdot [\sum_{c-w}^{c+w} Y_i']1/2}{T_1 \cdot \beta \cdot @}$$

where:

- A = user variable constant value, set at 2.71
- S = user variable sigma value, set at 4.66
- c = centroid energy
- w = user variable window function, set at 0.64·FWHM
- Y_i' = number of counts in gross spectrum
- T_1 = count time
- β = detection efficiency at centroid energy
- @ = photon yield

This MDA calculation is not consistent with Draft ANSI N 13.30. The MDA calculation will be further evaluated in the future.

Phoswich Detector Operations

Plutonium Phoswich

The phoswich detectors are used as a backup when the germanium detectors are not operational. Plutonium phoswich are used for the measurement of plutonium and americium. Operations with the phoswich detectors differ from the germanium because of different software.

Energy Alignment

Energy alignment consists of the gain adjustment and the ADC zero adjustment to establish the relation of MCA channel number to photon energy. The plutonium phoswich spectrum is a 1024 channel spectrum that measures the energy range of 10 keV to 127 keV. The spectrum has a 25 channel offset with a gain of 0.25 keV/channel. The two phoswich detectors are connected via a single amplifier to ADC number 3. Monthly, the ADC zero is aligned using the Am-241 source. The alignment sets the 17 keV photopeak at channel 93 and the 60 keV peak at channel 265, the calibration gain setting. The zero of the ADC and gain is adjusted with the Am-241 source and detector 1. Gain adjustment is via the high voltage control. When the zero is set for detector 1, it is not changed for detector 2. However, the two peaks are checked for detector 2. Gain adjustment with the high voltage should locate both peaks satisfactorily. If the peaks are not correctly located, an iterative adjustment of ADC zero and gain must proceed.

The number of channels is larger than necessary for the phoswich detectors, but there is no detriment. This is a historical setting and energy alignment changes would necessitate changes in the software.

Daily Operations Checks

Daily operations checks are required to ascertain that the equipment is operating at a specified level of performance. The daily operation checks are essential to ascertain that valid results are produced.

Alignment and source counts are the first operation for each day. The energy alignment adjusts the equipment to the energy alignment used for calibration. The radioactive sources used are Am-241 and Co-57. The Am-241 source provides photopeaks at 17 keV and 60 keV which are aligned at channels 93 and 265, respectively. The Co-57 source checks response at 122 keV to assure that higher energy photons are measured correctly.

Source count response must be within acceptable limits. The counts are compared to values in a file containing acceptable upper limits and lower limits established for the 17 keV, 60 keV, and 122 keV photons. The upper and lower limits are ± 3 sigma of the mean of a series of 10 counts immediately after calibration. The limits are adjusted monthly to account for the decay of Co-57. The operator must respond appropriately for the program to continue when a value is outside the acceptable limits.

The source counts for all detectors are maintained in a data file. When the first source counts are made each day, all of the previous day's counts and the mean values are printed before the current source count values. If additional source counts are made during the day, the accumulated counts for the day are printed.

The accumulated file is printed monthly and is cleared and initialized semiannually. The monthly printout is a permanent record.

After it is determined that the detectors are correctly aligned and responding correctly, the next measurement is the background. The combination of source and background measurements determine the correct operation of the equipment. Background measurements are made at the start of each shift and a minimum of once each 4 hours of operation. The background count data is maintained in a file on the computer and is printed monthly as a permanent record.

Data Analysis

The data from the plutonium phoswich do not have enough definition for spectral analysis and the counts measured are not large enough to delineate photopeaks. In order to maximize efficiency, the spectrum is divided into five regions of interest, four corresponding to a particular photon energy, and one for body potassium content. The initial step in data analysis is the summing of the five regions of interest. The analysis utilizes the sum of these regions of interest for gross counts, background, and calibration. Data analysis consists of subtracting background from gross counts, comparing net counts to the expected counts, and calculating the minimum detectable amount (MDA) and/or lung content of the transuranic radionuclides.

The background is measured minimally each 4 hours, resulting in a minimum of five measurements for the two operating shifts each day. When the first background is measured each morning, the average of the previous day's values are calculated. These average values of the previous day are used for the daily background.

Analysis Software

The analysis software is a method to differentiate the low energy counts measured from the scattering from the naturally occurring potassium-40 (K-40) and any transuranic radionuclides assimilated from occupational exposure. All persons contain naturally occurring K-40. The K-40 gamma rays interact with the body mass and are scattered into the low energy regions that are used for plutonium and/or americium measurement. The analysis method involves having previously counted a group of persons of varying sizes and K-40 content who have had no exposure to transuranic radionuclides. The energy spectrum collected results from their K-40 content with no contribution from plutonium and/or americium. From this data a set of relations was determined relating the measured counts in selected energy regions with the K-40 value, weight, height, and sex. These relations, called prediction equations, are calculated for the four energy regions of interest for male and female. This method with the current relations is described in detail on page 18 of this chapter.

The 14 keV to 22 keV is used to quantify Pu-239 and/or Pu-238. These radioisotopes have the same x-ray energy. The nuclide identification must be accomplished by supplementary material samples from the employee's work location.

The MDA is calculated based on the counting statistics of the gross counts and background counts.

MDA = ((4.66 · (background + gross)1/2) + 2.71) · conv. fact.
where:

background = 30 min gross background count in the energy region of interest
gross = 30 min gross sample count in the energy region of interest
conv. fact. = conversion factor of nCi per counts/30 minutes

The MDA determination method is not completely consistent with the guidance in Draft ANSI N 13.30 and will be reevaluated in the future.

Uranium Phoswich

The uranium phoswich are the backup detectors for measurement of uranium when the germanium detectors are not operational. The differing spectra require differing analysis methods and operating methodology.

Energy Alignment

Energy alignment consists of using the gain adjustment and the ADC zero adjustment to set the energy per channel to the value at the time of calibration. The uranium phoswich uses a 256 channel spectra that is adjusted for 1 keV per channel, for an energy range from 3 keV to 256 keV. The two uranium phoswich detectors are connected through a single amplifier to ADC 4. Monthly the ADC zero is aligned using the natural uranium source.. The alignment sets the 63 keV photopeak at channel 63 and the 186 keV peak at channel 186, the calibration gain setting. The zero of the ADC and gain is adjusted for detector 1. Gain adjustment is via the high voltage controls. The two peaks are checked for detector 2 with gain adjustments only. Both peaks should be satisfactory with no further adjustment. If the peaks are not correctly located, an iterative adjustment of ADC zero and detector gain must proceed.

Operational Check

Operational checks are the energy alignment, source response, and background evaluation. The source response checks operation of the detector, detector positioning, and software operation. The source response and background evaluation are permanent records.

The energy alignment uses a uranium source. Photon energies of 63 keV, 93 keV and 186 keV from Th-234 and U-235 give spectrum peaks centered at channels 63, 93, and 186, respectively.

The source response check is measurement of a phantom as a daily check source. The phantom is not designed as an exact chest phantom, but is a reasonable representative with scattering medium. The phantom is a box that is 18" by 18" by 10", constructed of 1" thick Plexiglass. A 1/16" Plexiglass sheet contains filter paper with natural uranium. This phantom is counted daily to ascertain that the hardware, electronics, and software are operating correctly. The analyzed value is plotted on a QC graph for comparison to values obtained after calibration.

The background spectrum is subtracted from the personnel counts before analysis. The background spectrum is obtained by a weekly 8 hour count. At the start of

each operating shift, a 30 minute background check compares the current background rate to that of the 8 hour. If the values are within ± 3 sigma, backgrounds are considered equivalent.

Analysis Software

The analysis software is based on the Oak Ridge National Laboratory program ALPHA-M. The collected spectrum is analyzed as the summation of a library of standard spectra. The background is first subtracted from the collected spectra. A limitation of the system is that the standard spectra must exist for all radioisotopes encountered. Analysis of the net spectra is accomplished by setting up a matrix consisting of the spectrum to be analyzed and each of the standard spectra. The result is then found by performing a least squares matrix reduction. The resulting coefficients are then used to determine the proportion of each standard present in the sample.

The printed report shows the analysis parameters for construction of the subject spectrum from the library spectra. The parameters of fit, shift, and gain are printed for up to three iterations. The fit is the value of the chi-squared fit of the library spectra to the subject spectra. The spectra will not give a perfect fit of zero, but gives acceptable analysis if the fit is less than 3.0. The shift parameter is the number of channels that are shifted for the best composite. The shift value must be within ±2 for acceptance. Gain is a correction factor to compensate for gain shifts between the subject spectrum and the library spectra. No gain correction is indicated as 1.0, and a value between 0.9 and 1.0 is acceptable. Subjects with unacceptable analysis parameters are recounted.

Calibration

Calibration of the chest counter determines the relationship between radioactivity present in the chest and the response measured by the detectors.

Calibration Phantoms

Phantoms are devices which simulate the human body both anatomically and in the attenuation and scattering of radiation. Phantoms are used for calibration by placing known amounts of radioactivity in locations similar to the anticipated location of the radioactive material in the human body. The radioactive materials in the phantoms are then measured with the phantoms in the same subject-detector relation as humans. This provides a conversion of detector response to radioactivity for the body or body parts. Different phantoms are used for the different radioisotopes and for different counters.

Humanoid Torso

The first successful calibration phantom for the actinide radioisotopes was the Humanoid torso developed at Lawrence Livermore National Laboratory(LLNL) in 1981. The Humanoid phantom is a simulation of a human torso. The original was constructed by LLNL to be identical to a specific cadaver. LLNL constructed a small group of duplicates, one of which is the SRS torso phantom. Later torso phantoms were constructed by commercial vendors.

The phantom is constructed of tissue equivalent plastic with a simulated skeleton for correctness anatomically and radiologically, including both scattering and shielding. The SRS phantom was thoroughly tested for attenuation and thickness by LLNL. Test data are available at SRS. The phantom has lungs, heart, liver, stomach, and tracheobronchial lymph nodes that are anatomically and radiologically correct. The inert organs are replaced with active organs containing known amounts of radioactive material for calibration purposes.

Humanoid Lungs

The lungs of the Humanoid phantom are similar in size and density to human lungs. SRS has pairs of active lungs each containing a known amount of the radioisotopes Pu-238, Pu-239, Am-241, U-238, U-235, Am-241 and Eu-152, or Np-237. The Am-241 and Eu-152 active lungs are placed in the torso for calibration of the germanium detector system. The single radioisotope active lungs are used for calibration of the phoswich detector systems.

The lungs containing plutonium, americium, and U-235 were obtained with radioassay certificated from Lawrence Livermore National Laboratory. The depleted uranium lungs were obtained from Humanoid Systems. The lungs containing Np-237 and those containing the mixture of Am-241 and Eu-152 were obtained from Radiology Support Devices. Lungs from commercial vendors have NIST traceable radioisotope content certificates.

Humanoid Chest Plates

The low energy photons emitted by the actinides are significantly attenuated by chest wall tissue. In order to calibrate for variations in chest wall tissue thickness, the Humanoid phantom has three sets of four chest plates each for differing

combinations of chest wall thicknesses and tissue types. The chest plates are used singly and not in combination. Three sets of the chest plates are available to simulate 1) 87% adipose and 13% muscle; 2) 50% adipose and 50% muscle, and; 3) 100% ICRP muscle. Use of the bare phantom and the chest plates give chest wall thickness calibration ranges of 16 mm to 40 mm. The table below shows the characteristics of the chest plates.

Effective Thickness Of Torso And Chest Plates (mm)

part number	right side	left side	mean	torso + plates
Torso				
C112	14.7	17.2	15.8	15.8
Chest plate; 87% adipose, 13% muscle				
A116-1	6.3	6.4	6.3	22.1
A116-2	12.5	12.3	12.4	28.2
A116-3	17.3	17.3	17.3	33.1
A116-4	24.6	23.9	24.2	40.0
Chest plate; 50% adipose, 50% muscle				
B115-1	6.5	5.8	6.1	21.9
B115-2	12.9	12.3	12.6	28.4
B115-3	17.9	17.7	17.7	33.5
B115-4	25.1	24.2	24.7	40.5
Chest plate; ICRP muscle				
C123-1	6.6	6.5	6.5	22.3
C123-2	12.9	12.8	12.8	28.6
C123-3	17.7	17.8	17.7	33.5
C123-4	25.3	25.1	25.2	41.0

The chest wall thicknesses of the phantom were measured in detail by LLNL. The slight differences noted between right and left sides are manufacturing variations. The B series of chest plates used with the torso for calibration give a range of chest wall thicknesses of 15.8 mm to 40.5 mm.

The overlays of 50% muscle and 50% adipose are used for calibration. This choice is an average value because there is presently no information available on the adipose to muscle ratios of the personnel being counted. Ultrasound can measure this but it is not routinely used.

Humanoid Other Organs

Liver and lymph node phantoms containing known amounts of Pu-238, Pu-239, and Am-241 are available for calibrations. These calibrations are applicable for measurements of personnel who have older uptakes and for whom radioactivity has translocated into these organs. Calibrations have not been performed with organs other than lungs because of insufficient facilities.

Calibration Methods

The calibration methods describe the use of radioactive organs, placement of organs in the phantom, the measurement of the response, and analysis of measurement.

Germanium Detectors

The germanium detector detector system uses the software package Abacos-LE developed by Canberra Industries, Inc. The software calibration method establishes an energy calibration as a function of MCA channel location, a FWHM calibration as a function of energy, and a family of efficiency calibrations as functions of energy and chest wall thickness. The calibration method uses a mixed radiation source with several photon energies to determine the calibration parameters over the complete energy range. The source for the chest counter is a pair of active lungs containing a mixture of Am-241 and Eu-152. This mixture gives nine energy peaks over the energy range. Am-241 gives photons of 13.9, 17.7, 20.9, 45.4, and 59.5 keV. Eu-152 gives photons of 121.8, 244.7, and 344.0 keV.

The initial calibration is the detector energy and FWHM calibration. The two functions are; 1) the photon energy as a function of channel number, and; 2) the FWHM as a function of energy. The peak location, or "centroid", and FWHM are determined from each calibration peak by making a weighted least-squares fit of a Gaussian function to the pulse height data for each peak. The weighting factor is the inverse of the variance of the data. The calibration peaks centroids and FWHM values are fit to polynomial functions by least-squares methods.

The energy versus channel function is:

$$E = A_1 + A_2 \cdot C + A_3 \cdot C^2 + A_4 \cdot C^3$$

where : E = energy in keV
C = peak centroid channel
A_1, A_2, A_3, and A_4 = fitting coefficients
A_1 is energy offset
A_2 is the uncorrected system gain
A_3 and A_4 are correction coefficients for nonlinearity

The FWHM versus energy function is:

$$F \cdot A_2 = B_1 + B_2 \cdot E + B_3 \cdot E^2 + B_4 \cdot E^3$$

where : F = FWHM in channels
A_2 = uncorrected system gain coefficient from energy alignment
E = energy in keV
B_1, B_2, B_3, and B_4 = fitting coefficients

Efficiency responses are directly measured only for the five chest wall thicknesses of the phantom: with no overlay and the phantom with each of the 4 overlays. These values give efficiency measurements for chest wall thicknesses from 16 mm to 40 mm. The significant attenuation of the low energy photons requires efficiency curves for gradations of 1 mm. This gradation was accomplished by plotting the efficiency for each of the energies of 13.9, 17.7, 20.9, 26.4, 59.5, 121.8, 244.7, and 344.0 as a function of the phantom chest wall thickness. Equations were calculated by linear least square methods for each of these energy lines as a

function of chest wall thickness. These equations were used to calculate the efficiency for each energy in one millimeter increments from 16 mm to 40 mm chest wall thickness.

Efficiency data values for each millimeter from 16 mm to 40 mm were input into the software analysis. The data for each chest wall thickness was fit to a fifth order polynomial. These polynomials were plotted to assure that the efficiency function was a smooth function of energy for each chest wall thickness. Each analysis of person data accesses an appropriate an efficiency function for each analysis.

Plutonium Phoswich Detectors

The plutonium calibration is a two step process; measurement of a group of employees who have no occupational exposure, and determination of the response from radioactive lungs in the Humanoid phantom.

All persons contain naturally occurring potassium-40 (K-40). The K-40 gamma rays interact with the body mass and are scattered into the low energy regions that are used for plutonium and/or americium measurement. The data collected in the low energy region from K-40 scattering must be differentiated from that caused by plutonium and/or americium present. The method uses prediction equations derived from counting a group of persons of varying sizes and K-40 content who have had no exposure to plutonium. The energy spectrum collected results from their K-40 content with no contribution from plutonium, americium, and/or uranium.

The prediction equation determines the variables and coefficients that are used to predict the spectrum that was collected. To make this method practical, the spectrum is divided into five energy regions with the data summed over the regions of interest. The personal variables available are K-40 content, height, weight, and sex. The energy segment of 90 keV to 127 keV is higher energy than transuranic emissions and data in it are related directly to the K-40 content. The region is designated potassium scatter. The data measured in the four low energy regions was then evaluated as functions of the person's sex, weight, height, potassium scatter data, and combinations of these variables. The spectrum is divided into energy regions which show the gamma and/or x-rays of the nuclides of interest. The energy regions are: 1)14 keV to 22 keV for the 17 keV x-rays; 2) 22 keV to 47 keV for the 30 keV gamma of Cs-137; 3) 48 keV to 66 keV for the 60 keV gamma of Am-241,and 4) 66 keV to 89 keV for a check for other higher energy unanticipated radioactivity. Energy ranges were chosen on the basis of maximum source response with minimum background.

Prediction equations for the four energy regions were developed by use of the SAS Procedure GLM[6]. The counts that are measured in the four lower energy regions result from the person's K-40 content as modified by the person's weight and height. The GLM software uses least squares multiple linear regression to fit the independent variables to the dependent variable. The dependent variable of net counts in the selected energy region is calculated from combinations of weight, height, and K-scatter values as independent variables. The constant and coefficients for the predicted terms are calculated. The independent variables that have physical significance are a) potassium scatter value, b) the product of potassium scatter value and the quotient of weight divided by height, and c) the product of potassium scatter value and the square root of the quotient of weight

divided by height. These variables were investigated separately for the two sexes. The current prediction equations in the following table are based on counts of 59 males and 32 females who had no occupational exposure to radioactive material.[7] These functions are very similar to those from prior populations. The following tables provides prediction equations from unexposed employees.

Prediction Equations From Unexposed Employees

Reg. of Int.	Function
	Male prediction equations
14 - 22 keV	CTS = 56.1 + (0.0754 · KS)
22 - 47 keV	CTS = 58.0 + (0.625 · KS)
48 - 66 keV	CTS = 130. + (0.374 · KS · SQ)
66 - 89 keV	CTS = 118. + (0.459 · KS · SQ)
	Female prediction equations
14 - 22 keV	CTS = 82.6 + (0.0069 · KS · WH)
22 - 47 keV	CTS = 141. + (0.323 · KS · SQ)
48 - 66 keV	CTS = 34.7 + (0.417 · KS · SQ)
66 - 89 keV	CTS = 109. + (0.669 · KS)

where:

CTS = predicted counts for the 30 minute count
KS = measured counts per 30 minutes in the potassium scatter region
WH = quotient of weight (pounds) divided by height (inches)
SQ = square root of quotient of weight (pounds) divided by height (inches)

The 14 keV to 22 keV region is used to quantify Pu-239 and/or Pu-238. The detectors cannot differentiate these radioisotopes. The nuclide identification must be accomplished by supplementary information or material samples from the employee's work location.

Calibration is accomplished by counting the phantom with the plutonium-238, plutonium-239, and americium-241 lungs in the position within the phantom. The phantom is counted with no chest overlay and with all 4 overlays. Fitting the data of the response of both the 17 keV energy range of plutonium versus the chest wall thickness gives equations of the form:

$$N = \exp(A + D_1 \cdot C + D_2 \cdot C^2)$$

where: N = nCi per counts per 30 minutes
A = constant
C = chest wall thickness in millimeters
D_1, D_2 = fitting coefficients

The 60 keV response from Am-241 gives a response of the form:

$$M = \exp(B + F \cdot C)$$

where:
- M = nCi per counts per 30 minutes
- B = constant
- C = chest wall thickness in millimeters
- F = fitting coefficient

Americium-241 has 17 keV x-rays and 60 keV photons. The scatter of the 60 keV into the 17 keV region must be calculated as a function of chest wall thickness. The scatter relation is of the form:

$$R = \exp(S + T_1 \cdot C + T_2 \cdot C^2)$$

where:
- R = ratio of 17 keV to 60 keV counts
- S = constant
- C = chest wall thickness in millimeters
- T_1, T_2 = fitting coefficients

Uranium Phoswich Detectors

Uranium phoswich detectors have thicker NaI(Tl) detectors and therefore more definition to the energy spectra. This added definition permits a different methodology for uranium than the plutonium. The spectrum collected when counting personnel is constructed using portions of standard spectra of K-40, Cs-137, U-235, and U-238. The K-40 and Cs-137 standard spectra are acquired with the radioisotopes in the Bomab phantom since both radionuclides are total body distributed. Uranium-235 and U-238 spectra are acquired with phantom lungs in the Humanoid phantom with chest plate B115-2. The U-238 lungs contain depleted uranium, and need no correction for the presence of U-235. When the spectra from the calibration standards are acquired, the current software converts the spectra into the usable form after input of the count time and the radioactivity amount of the sources.

Two Bomab phantoms are filled with K-40 and Cs-137. The K-40 calibration source is prepared by dissolving KCl in distilled water. The Bomab with K-40 contains 1283 nCi of K-40. The Cs-137 was obtained from the environmental standards group. The Bomab with Cs-137 calibration source contains 471 nCi of Cs-137 in distilled water. These calibration phantoms are not of primary importance because K-40 and Cs-137 are not the primary radioisotopes of interest. These standards are used to create reference spectra for use in construction of the employee spectra.

The enriched uranium (U-235) active lungs for the Humanoid phantom were obtained from Humanoid Systems. The depleted uranium (U-238) active lungs were obtained Radiology Support Devices. The Calibration Certificates are maintained.

Calibration includes acquisition of a background, and counts of the four sources. All counts are made as long as reasonably possible to minimize statistical uncertainty from counting statistics. Typical count times are 480 minutes for background, Cs-137, and K-40; and 50 minutes for U-238 and U-238.

As noted above the analysis method uses a spectrum composition method. The primary photons are 63 keV and 93 keV from Th-234, the daughter of U-238, and the 185.7 keV from U-235. These photons have much less attenuation from chest wall thickness than lower energies, and therefore efficiencies are not significantly effected by chest wall thicknesses. The calibration is performed with the Humanoid phantom with overlay B115-2 of the 50% adipose and 50% muscle. This is a 28mm chest wall thickness, near average observed for the population. The higher energies of uranium are not strongly attenuated by the chest wall tissue, and therefore, use a single efficiency function.

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Chapter 24

Whole Body Counting

Chapter 24 Preview

- Data Analysis
- Counter Calibration

24

Whole Body Counting

Whole body counting is performed using a commercial counter called the Fastscan, which is manufactured by Canberra Industries, Meriden, Connecticut. The counter is described In Chapter 22, page 6.

Two NaI detectors measure photons in the energy range of 135 keV to 2000 keV. Data from the two detectors are displayed separately but are summed for analysis. The ratio of the sum of counts for the top and bottom detectors to the counts for each detector is calculated. This is of use in locating any localized radioactive material.

The shield is 4" thick and is constructed of steel selected on the basis of minimum radioactivity. The shield is laminated to facilitate portability. The completed shield weighs 10,000 pounds. The completed shield is 75 inches inside height with an entrance width of 15 inches.

Measurement Geometry

Personnel stand in the counter during the count. The two detectors are about 5" from the front of the person. The two 16" NaI(Tl) detectors extend from near the knees to the upper chest area.

Measured Radionuclides

The standup whole body counter measures photon emitting radionuclides. These are naturally occurring radioisotopes, fission products, neutron activation products, and medical radioisotopes. Potassium-40 (K-40) is naturally occurring and is considered a built-in check source. The presence of K-40 is confirmation that both the counter hardware and software are operating. Persons in this geographical area who eat venison sometimes contain Cs-137 from weapons fallout material.

Cs-137 along with several other radionuclides may be present in employees due to work at SRS. The table below shows radionuclides that are commonly measured in the whole body counter.

Radionuclide	Photon energy (keV)	Yield (%)
K-40 (1)	1461	10.7
Cs-137 (2,3)	661.6	85.0
CePr-144 (3)	133.5	10.8
RuRh-106 (3)	511.8	20.6
"	621.8	9.81
"	1050.5	1.46
Zr-95 (3)	724.2	44.5
"	756.7	55.0
Nb-95 (3)	765.8	10
Zn-65 (3)	1116.0	50.7
Co-60 (3)	1173	99.9
"	1332	100
I-131 (3,4)	364.5	81.2
Tl-201 (4)	135.3	2.65
"	167.4	10.0
Tl-202 (4)	439.6	91.4

(1) naturally occurring
(2) weapons testing fallout
(3) SRS operations
(4) medically administered

Personnel Operations

Personnel who are counted stand in the counter with their back in contact with the rear wall of the counter. Personnel are counted in their street clothes, which simplifies the operation and increases personnel throughput by eliminating changing clothes. Experience has shown this to be satisfactory with no counter contamination occurring. The persons must remove their security badge, TLD badge and watch. These items may have been worn in radiation controlled areas and therefore have some potential for contamination.

Persons are routinely counted for 2 minutes. This count time is the minimum required to achieve the desired minimum detectable amounts. All personnel counts register naturally occurring K-40, and many individuals who consume venison produce counts which register Cs-137 from fallout from nuclear weapons testing. If the initial count shows the presence of radioactivity other than K-40 or >25 nCi of Cs-137, the person is then recounted for a 4 minute investigation count. Indicated radionuclides other than K-40 and <25 nCi of Cs-137 are considered to be positives when counting errors are less than 50%. The rules for determining positives are recommended by Canberra Industries, Inc. These rules result in a small number of false positives, and potentially false negatives.

If the 4 minute count shows measurable amounts of radioactivity, the person is assumed positive and Health Protection Operations (HPO) is called to survey his/her clothes. If radioactive material is found, it is decontaminated. Regardless, the person must shower and change into a radioactively clean surgical scrub suit and be recounted. Any activity measured at this time is considered to be internal and reported to an Internal Dosimetrist.

Energy Alignment

Energy alignment consists of setting the amplifier gain and the ADC zero. Monthly, the ADC zero adjustment is checked with the check source of Cs-137 and Co-60. The combination of ADC zero and amplifier gain of detector 1 is used to set the 662 keV peak to channel 169 and the 1332 keV peak to channel 341. When this is complete, the alignment of detector 2 is adjusted with the amplifier gain only. If the peaks of detector 2 are not located correctly, an iterative adjustment of the ADC zero and amplifier gain is necessary.

At the start of each shift, the amplifier gains are checked, and adjusted if necessary. The energy alignment is adjusted with the amplifier gain so that the center of the peak represented by the 1332 keV photopeak of Co-60 is positioned to the data channel 341. The gain is adjusted for both the bottom and top detector. These are commonly accepted alignment methods for instruments of this kind.

Operation Check

In order to ensure the quality of all measurements, the total counter operation is checked at the start of each 8-hour shift. The checks consist of energy alignment, source check response, and background check.

The energy alignment is described above. After the energy alignment is complete, a check radioactive source of Cs-137 and Co-60 is counted in a reproducible position. Parameters of the 3 peaks of 662 keV, 1173 keV, and 1332 keV must be within acceptable limits in order to proceed with personnel counting operations.

The background response is checked to ascertain that there is no radioactivity present in the unoccupied counter. The background subtracted from personnel counts is not this count but a normalized fraction of the monthly 30 minute environmental background. The background check once per shift verifies that there is no change in the background due to contamination present.

Contamination Control

Plant-issued coveralls are not permitted in the standup whole body counter. The coveralls may contain low levels of radioactive material and, over time, could increase the counter background. The counter is checked monthly by Health Protection Operations for any smearable contamination.

Data Analysis

The analysis software ABACOS-II was developed by Canberra Industries for low activity spectra, typically found in in-vivo counters. This software is similar to ABACOS-LE, Chapter 23, page 7. If the alignment can not be adjusted correctly. Slight differences result from parameter selections applicable to NaI(Tl) scintillation spectra rather than to germanium spectra.

Background Determination

The initial operation is an estimation of the subject count background continuum by a systematic smoothing of the subject count spectrum. This smoothed continuum is subtracted from the original subject count spectrum to identify and quantify the gross counts of the photopeaks.

The counter background correction for the identified photopeaks is determined from the monthly 30 minute environmental background. The background correction is not the traditional channel by channel subtraction. Background correction is subtraction of the analyzed nanocurie amounts obtained from the 30 minute background count from the nanocurie amounts of the subject count.

Peak Determination

Peak determination by "library driven" and "peak driven" methods are described in Chapter 23, on page 8. The smoothed continuum is subtracted from the original spectrum. The resulting peaks are fitted to Gaussian functions.

The energy tolerance range for adjusting "peak driven" peaks to "library driven" peaks is user variable. The vendor recommended value for the scintillation detectors is 0.25 FWHM, i.e. if an unknown peak energy is within 0.25 FWHM of a library peak energy, the unknown peak is shifted to the library peak energy and the percent change in gain is shown on the printed report. (FWHM values for the NaI(Tl) detectors range from 15 keV for 120 keV energy photopeaks to 110 keV for 2000 keV photopeaks.)

Analysis

The analysis is a function of calibrations of energy range, FWHM values and efficiency. The above methodology locates the photopeaks of the spectra and determines the energy from the energy calibration. Each peak is fit to a Gaussian shape as directed by the FWHM calibration data. With the peak energy and peak area determined, the amount of radioactivity is calculated using the appropriate efficiency curve for the chest wall thickness and the photon yield from the library file. A peak is reported as an "unknown" for any energy not in the library file and the radioactivity is calculated assuming a photon yield of 100%.

Minimum Detectable Amount

The minimum detectable amount is calculated as a function of the gross count and the FWHM. The MDA is defined as:

$$MDA = \frac{C_1 + C_2 \cdot [\sum_{c-w} Y_i']^{1/2}}{T_1 \cdot \beta \cdot @_{c+w}}$$

where:

C_1 = user variable MDA constant, set at 2.71
C_2 = user variable MDA sigma, set at 4.66
c = centroid energy
w = window function, 0.64*FWHM
Y_1' = number of counts in gross spectrum
T_1 = count time
ß = detection efficiency at centroid energy
@ = photon yield, 1.0 for 'unknown' peaks

This MDA calculation recommended by the vendor is not consistent with Draft ANSI N 13.30. The MDA evaluation will be reevaluated in the future.

The table below shows typical minimum detectable amounts of the radionuclides in the library for two minute counts. These values increase with an increase in the amount of K-40 or other radionuclides.

Radionuclide	nCi
Cs-137	3.5
CePr-144	50
RuRh-106	15
Zr-95	4.8
Nb-95	2.6
Zn-65	5.0
Co-60	2.6
I-131	4.0

Counter Calibration

Calibration Phantoms

Remcal

The first generally accepted phantom was Remcal. This was a butyrate plastic phantom designed to visually resemble a man. It is generally accepted that the detail in Remcal was not necessary. This detail contributed to its cost. Calibration for fission radioisotopes includes neutron activation radioisotopes is for gamma emitting nuclides in the energy range of 125 keV to 2000 keV. The bottle phantom, Bomab, was a simpler alternative to find an acceptable phantom at a reasonable cost.

Bomab

Bomab is the "standard" calibration phantom for fission products. Bomab is a group of polyethylene bottles that are arranged to simulate a body. There are 10 bottles that represent the head, neck, upper body, pelvis, 2 thighs, 2 lower legs, and 2 arms. The phantom contains about 58 liters of liquid. SRS has two of these phantoms. One currently contains KCl for a K-40 standard. The other can be used for any other radionuclide of interest.

Remcal Equivalent

The Remcal equivalent phantom is a solid Plexiglass phantom constructed to respond similarly to Remcal for the energy range of 125 keV to 1800 keV. This phantom was constructed by Canberra for use in calibration of the Fastscan counters. Canberra has documented the equivalence in unpublished reports.

Calibration Method

The standup whole body counter ABACOS-II software includes the calibration methodology. The calibration uses a mixed gamma source containing Cd-109, Hg-203, Y-88, Cs-137, and Co-60. This source gives 10 gamma rays with energies of 88 keV to 1836 keV. This mixed source provides the photons for energy, FWHM, and efficiency calibration. The counter is calibrated for assay of gamma emitting radionuclide from 125 keV to 1800 keV. The Remcal equivalent phantom is used with the source to calibrate the counter to measure radioactivity in the lung, GI tract, or thyroid of an individual. The response with the source in the lung position is very nearly identical to the response with identical quantities of the same radioisotopes distributed uniformly throughout the total body. Therefore, for convenience, the lung calibration is routinely used for whole body analysis. In addition to the in-vivo analysis, the counter is calibrated to measure radioactivity in an unshielded location on the rear wall of the counter. This back wall calibration gives a more clearly defined spectrum for energy and FWHM calibration and may used for analysis of a source of radioactivity on the back wall of the counter.

The radioisotopes, gamma energies, and radioactive halflife of the mixed gamma source are shown below. The calibration sources are NIST traceable and the certificate of calibration for each source is on file. A source is used for a period of one year before radioactive decay renders the source unusable. The following table provides the radionuclide components for the Fastscan Mixed Gamma Calibration Source.

Fastscan Mixed Gamma Calibration Source

Radio-nuclide	Photon energy(keV)	Photon yield(%)	Halflife (days)
Cd-109	88	3.61	464
Co-57	122	85.6	270.9
Ce-139	166	79.1	137.7
Hg-203	279	81.5	46.62
Sn-113	392	64.2	115.0
Cs-137	662	85.0	10950
Y-88	898	93.4	106.7
Co-60	1173	99.9	1925
Co-60	1332	100	1925
Y-88	1836	99.3	106.7

END

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