

ENGINEERING DESIGN FILE

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<p>This Engineering Design File is the INEEL M&O Contractor Technical Basis for Internal Dosimetry. The DOE Technical Standard Internal Dosimetry (DOE-STD-1121-98) provided the necessary topical structuring that was followed in developing this as well as facility internal dosimetry technical basis documents. This TBD (technical basis document) addresses organizational agreements, bioassay program design, detection and confirmation of intakes, internal dose evaluation, internal dose management, records and reporting and medical response. The facility specific TBDs address those facets of the internal dosimetry program that are germane to their facility; such as monitoring in the workplace, basis for participation in the facilities bioassay program, and their facility's bioassay program protocol. The present INEEL Contractor as well as other past site Contractors have conducted extensive internal dosimetry programs that have functioned primarily as quality assurance of the radiological control practices. This has shown that intakes of radiological contamination have been adequately controlled to the extent that uptakes exceeding 100 mrem CEDE do not occur. The INEEL Contractor radiological controls program has generally prevented an intake of radionuclides and has created an environment where exposures from internally deposited radionuclides greater than 100 mrem CEDE are very unlikely. Consequently, there are no employees or visitors who fall into the categories defined in 10 CFR 835.402.c. who would then be required to participate in a bioassay program as a result of their routine activities. However, contamination controlled work at the various INEEL M&O Contractor facilities do continue to require individuals to be protected by means of engineering controls, administrative controls and personal protection equipment. Therefore, the practice of utilizing a confirmatory bioassay program involving limited surveillance of workers to show that results are, as expected, are being employed at the various facilities. The basis and rationale for this type of program has been addressed.</p>				
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1.0 Internal Dosimetry Program Objective

The technical basis and methods for internal dosimetry is ever evolving. This document identifies some of the more salient technical factors used in administrating the INEEL M&O Contractor internal dosimetry program. As conditions change and new problems arise, additional technical bases will be developed and added to this document.

This technical basis document is rooted in the requirement set forth in Title 10, Code of Federal Regulations, Part 835; the INEEL Radiological Control Manual; and contractual requirements where applicable. 10 CFR 835.402.c. requires that internal dose evaluation programs shall be conducted for: *(1) radiological workers who, under typical conditions, are likely to receive 0.1 rem or more committed effective dose equivalent, from all occupational radionuclide intakes in the year; (2) declared pregnant workers likely to receive an intake resulting in a dose equivalent to the embryo/fetus in excess of 0.05 rem; (3) minors and members of the general public who are likely to receive, in one year, an intake resulting in a committed effective dose equivalent in excess of 0.05 rem.*

The intent of the INEEL internal dosimetry program is to verify that personnel protection is maintained at state-of-the-art levels and to provide a recognized and competent internal dosimetry program. This is accomplished by (1) demonstrating satisfactorily that health and safety protection of the worker through a bioassay and dose assessment program has been adequately integrated with all disciplines of Radiological Controls, and (2) demonstrating compliance with all applicable regulatory policies set forth for an internal dosimetry program.

The INEEL radiological protection program provides standards and methods to reduce unnecessary internal exposure, and provides a "best" estimate of internal exposure from radionuclide intakes that might occur. It also ensures compliance to applicable regulations and procedures governing internal dosimetry, ensures documentation is adequate to demonstrate program compliance, and documents the technical bases that support the procedures and guidelines. Organ dose equivalents and effective dose equivalents resulting from confirmed intakes are established using mathematical models for retention and excretion based on industry standards such as ICRP, NCRP, ANSI, and/or individually determined models as appropriate. Assessed doses are entered into a dose records system in units of Committed Effective Dose Equivalent (CEDE). The external Effective Dose Equivalent (EDE) shall be combined with the CEDE for a Total Effective Dose Equivalent (TEDE). It is the program's intent to ensure that applicable dose management limits are not exceeded and that internal doses be reported as necessary to each employee and to the dosimetry records system.

Current and past M&O Contractors have conducted extensive internal dosimetry programs that have functioned primarily as quality assurance for the radiological control practices that prevented intakes of radiological contamination. The radiological work force has historically participated by receiving annual whole body counts (WBC) when the source term was tagged with photon activity; others submitted semi-annual urine and fecal samples together with annual WBCs when the source term included transuranic and/or uranium, as well as photon activity. For the SMC project, where the source term is depleted uranium, monthly, quarterly or semi-annual urine sampling were collected and analyzed for total uranium. The bioassay frequencies were primarily based upon the minimum detectable dose concept.

In 1996 the numbers of routine bioassays began to decrease. Facility programs were becoming more selective by placing only those radiation workers who frequented contamination areas, or who worked specific job assignments, in their facility's routine bioassay program. The internal dosimetry program was shifting away from a quality assurance type bioassay program to a confirmatory bioassay program involving limited surveillance of workers providing verification that routine bioassays were not required.

The routine bioassay programs at the various INEEL facilities have shown that employees who frequent contamination areas have been adequately protected against intakes of radioactivity because of the radiological control practices that have been implemented. The respiratory protection program is managed in accordance with and meets OSHA requirements 29 CFR 1910.134 and DOE ORDER 440.1A. These regulatory policies are outlined in company procedure MCP-2726. In addition, the respirator protection factors identified in ANSI Z88.2-1992 are specified in company procedure MCP-432. These procedures and regulations govern the use of respiratory protection at INEEL facilities. The air-monitoring program is outlined in MCP-357. In addition, guidance for sampling radioactive iodine is contained in TPR-6430 and for sampling tritium is contained in TPR-6431. These procedures give direction for compliance with Title 10 Code of Federal Regulations (CFR) Part 835 Section (§) 403(a)(1) and 403(a)(2). The air monitoring procedure requires job specific air monitoring if the source term contains transuranic activity $> 2000 \text{ dpm}/100 \text{ cm}^2$, or $> 100,000 \text{ dpm}/100^2$ for mixed fission products containing Sr-90, etc. If an individual's effective DAC-hours total 4 or more, the facility internal dosimetry coordinator is made cognizant of the accumulated DAC-hours and determines the need for a bioassay.

The radiological protection program includes designed engineering controls, air-monitoring, employee training, contamination control measures, respiratory protection, administrative/procedural instructions that are conducted in accordance with 10 CFR 835, implementation guides, good practices, DOE orders and directives designed and intended to prevent intakes of radioactivity. As a result of these practices during the last 11 years, there have been only 7 dose assessments recorded that have resulted in dose assignments greater than 100 mrem CEDE. Of these, five (652, 655, 656, 677 and 678 mrem CEDE) were related to a single event, having occurred in 1996, which was recognized immediately and bioassays were begun the day of the event. The other two (136 and 205 mrem CEDE) were the result of radionuclides being detected during routine bioassay sampling. In both cases a reasonable intake date could not be determined and thus the intake date for the assessments defaulted to the employees previous bioassay sample. During this span of time there were 17,686 *in vitro* and 16,461 *in vivo* bioassays performed. The INEEL M&O Contractor radiological controls program, have generally prevented the intake of radionuclides and have created an environment where exposures from internally deposited radionuclides greater than 100 mrem CEDE are very unlikely. Consequently, there are no employees or visitors who fall into the categories defined in 10 CFR 835.402.c. who would then be required to participate in a routine bioassay program as a result of their routine activities. However, should an employee or visitor be involved in an event where the internal uptake of radionuclides was likely to have occurred, participation in the bioassay program with an internal dosimetry assessment would be mandatory or non-discretionary. The elements of the radiation protection program discussed above would be used

to detect the loss of control of radioactive materials, the presence of airborne radioactivity, or personnel contamination which might be indicative of the potential for an internal uptake.

Therefore, in accordance with DOE-STD-1121-98 section 5 (5.1) The INEEL administers an internal dosimetry program through the use of confirmatory and special bioassay monitoring where there are sufficient quantities of radionuclides present and handled at INEEL facilities in which an accidental intake, resulting in 100 mrem $H_{E,50}$, cannot be ruled out.

2.0 Organizational Agreements

The internal dosimetry program is independent of any sub-contracting radiobioassay support. All *in vitro* and *in vivo* radiobioassays are performed together with the administration of all aspects of the internal dosimetry program by the INEEL site M&O Contractor. The *In Vitro* analytical laboratory program and support staff reside within the INEEL M&O Contractor Analytical Laboratory Department and directly support the internal dosimetry site programs. The *In Vivo* counting laboratory and staff reside within the Radiation Dosimetry and Records (RDR) unit. The RDR unit administers both the internal and external dosimetry programs for the INEEL M&O Contractor. Therefore, contractual agreements are not necessary for the internal dosimetry program since it resides within M&O Contractors organization. However, there are analytical laboratory performance goals established to meet necessary turn-around time for *in vitro* samples. The average turn-around times for routine (or confirmatory) fecal and urine sample analysis has been set at 40 and 60 days respectively. The turn-around time for declared event based samples has been set at 15 working days.

A medical dispensary program is also operated at the INEEL staffed with medical doctors (MD) and nurses. In the event that medical intervention is deemed necessary, the internal dosimetry staff would provide the necessary, dosimetry technical support to the MDs to best help minimize a significant occupational dose in accordance with EDF 2000-08, Established Levels of Radionuclide Intakes for Consideration of Medical Intervention.

3.0 Bioassay Program Design

The internal dosimetry program is structured where Radiological Control Management is responsible for the internal dosimetry program at their various facilities. The seven INEEL facilities have each assigned a radiological engineer as their facility Internal Dosimetry Coordinator (IDC) and together with a supporting staff organize and maintain a facility-specific internal dosimetry program. To support the IDCs, a Radiation Dosimetry and Records (RDR) Internal Dosimetry Program (RDR-IDP) staff is responsible for providing technical guidance to each facility's program and to review, validate and/or prepare any official internal dose assessments.

3.1 Facility Specific Technical Basis Documents

Complementing the INEEL M&O Contractor Technical Basis for Internal Dosimetry is facility specific TBDs. These documents having been prepared by the facility Internal Dosimetry Coordinators are included as supplements to this general program basis document and together makeup the internal dosimetry program technical basis for the INEEL M&O Contractor. Each of

the seven facilities have outlined the basis for each of their facility specific internal dosimetry programs addressing topical structuring that is specific to their facilities' program. The following areas of emphasis have been addressed by the facilities:

- Facility/Operational Practices – The general practices and operations are described that are performed at the respective facilities/areas as they relate to internal dosimetry.
- Workplace Monitoring Programs – Facets of the personnel protection program, such as, air monitoring program, contamination control program, personnel surveying, etc. that are being managed by the radiation protection organization to control, detect and measure radionuclides that may become a source for internal contamination.
- Characterization of Potential Exposure Hazards – Facilities have described the source term(s) as they relate to the different facility operations and practices that are specific to their facilities. The risks and routes of uptake have been described.
- Physical and Chemical Characteristics of the Radioactive Materials – The predominant forms of the radionuclides that are expected, the chemical makeup that one would expect finding related to the major radionuclides of concern and the particle size of the contaminant as it relates to internal dosimetry.
- Justification for Bioassay Monitoring Frequency – Technical Basis for the facilities' radiobioassay program and its sampling/counting protocol.
- Type and Frequency of Bioassay Measurements – Description of each of the facilities' radiobioassay programs and the administration of the facilities program.
- Rationale for Selection of Workers for Bioassay Monitoring – Technical Basis for who will participate in the facilities bioassay program.

3.2 Bioassay Tracking and Records Management

The Radiological Control Organization placed into action during CY-99 a Radiological Control Information Management System (RCIMS). RCIMS is a radiological control records database management program contained within a vendor-coded product called Passport-Total Exposure. The first phase of the program was instituted October 1997 where a database-records and RWP management program was established that provides the capability of controlling employee entries into radiological control areas based upon employee training, ALARA controls, administrative requirements, etc. Included in the program, based on specifications specified by the procurement, is a bioassay tracking, analysis and dose assignment database. This phase of the RCIMS program was placed into operation June 1999. Individuals who were selected to participate in the internal dosimetry program have their bioassays tracked from the point of initial request to final resolution. The RCIMS managed internal dosimetry program provides assurance that all *in vivo* and *in vitro* bioassays are tracked and brought to conclusion as an internal dose assignment or as a bioassay analysis history.

3.3 Training Requirements

Facility Internal Dosimetry Coordinators (IDCs) work closely with the Internal Dosimetrists (IDs), to the degree that any evaluation concerning assignment of an internal dose is reviewed and approved by an RDR-ID before being entered into an individual's official dosimetry record. The IDC's are not expected to provide internal dosimetry services independently. Consequently, it is sufficient that a prospective IDC have a familiarity of internal dosimetry concepts and the need for bioassays similar to that obtained as a result of academic training as a Health Physicist or as an experienced radiological engineer. Therefore, the level of training required for an IDC is no greater than that generally expected of an individual who is qualified to be a radiological engineer. The IDC needs an understanding of the Company Procedure relating to Internal Dosimetry and Airborne Radioactivity Monitoring.

The Internal Dosimetrist(s) (ID) Subject Matter Expert (SME) in the Dosimetry and Records Unit is expected to have a working level knowledge of the concepts associated with internal dosimetry. In addition, practicing IDs need to stay current in new concepts, models and research findings as published by recognized authorities. A prospective ID should have a degree in health physics or related subject or have sufficient experience in the field of internal dosimetry to provide guidance and direction to the overall internal dosimetry program. The internal dosimetry staff needs to have equivalent training and experience as outlined in the DOE Title 10, Code of Federal Regulations, Part 835 and as outlined in DOE-STD-1107-97 Knowledge, Skills, and Abilities for Key Radiation Protection Positions, at DOE Facilities.

On-going training will take the form of periodic meetings of the personnel involved in the Internal Dosimetry Program. These meetings will be the forums for discussing program changes, lessons learned, software changes and other appropriate topics. Additional training at seminars or classes devoted to internal dosimetry concepts is desirable.

Radiological Control Management in each of the facilities have the responsibility to assure that the employee selected as their program IDC has adequate knowledge and training for the internal dosimetry/bioassay program needs.

3.4 Programmatic Investigation Levels

An *in vitro* bioassay sample is investigated as being positive when the measurable radioactivity is greater than 2 sigma where sigma is the total uncertainty being comprised of both systematic and random counting errors. The calculational derivation of the systematic uncertainty is shown in INT/INEEL-99-00715, Nov-2000, "Gamma and Alpha Analysis User Guide for Sun Sparctstations" while the random counting uncertainty is documented in Analytical Methods Manual procedure ACMM-5978 (Determination of Americium, Plutonium and Uranium).

At SMC, total uranium analysis is assessed as being positive at $> 1.0 \mu\text{g/l}$, with follow-up confirmatory bioassays being requested. Because of the current frequency of routine urine bioassay at SMC (monthly, quarterly, and semi-annual), a urine bioassay with measurable uranium radioactivity of $1.0 \mu\text{g/l}$ results in an occupational dose less than 100 mrem CEDE. This assumes that the intake occurred at the midpoint of the monitoring period as well as additional

default assumptions associated with the SMC internal dosimetry program. See SMC TBD addendum for basis of default assumptions.

An *in vivo* count with measurable radioactivity greater than 2.33 sigma will be evaluated and assessed as a positive result. This statistical test is implemented in the ABOCOS Plus software (See Model 48-0198 Genie-VMS Spectroscopy Systems user's manual).

The basis for the investigation levels has been a programmatic choice that has stemmed from past practices with site contractors and the Department of Energy. The practice has weathered technical inquiry over the years and has been found as an acceptable reasonable decision-level at which to begin an investigation into a possible intake.

3.5 Radiobioassay Laboratory Accreditation

A revised 10CFR835 was issued November 4, 1998 that required Radiobioassay laboratories to become DOELAP accredited within 3 years from the date of issue. Based on ANSI N 13.30 Performance Criteria for Radiobioassay, paragraph 6.1.1.6 frequency of testing, a participating laboratory shall be retested and evaluated by an on-site assessment team at least every 3 years thereafter. *In Vitro* and *In Vivo* Radiobioassay Laboratories providing service to the INEEL are required to comply with this requirement. Both the *In Vitro* and *In Vivo* Radiobioassay Laboratories, supporting the internal dosimetry program, received initial DOELAP accreditation in February 1998 and become re-accredited every two years, thus achieving compliance in accordance with this regulatory change of paragraph 10CFR835.402.

3.6 Default Trigger Levels

The following procedures, MCP-191 Radiological Internal Dosimetry, MCP-356 Passive and Real-Time Air Monitoring, and MCP-357 Job Specific Air Sampling provide some of the administrative guidelines for determining the need for event based bioassays. The following trigger levels are used to help make the determination to assess a suspect intake through radiobioassay techniques:

- Extensive work in contaminated areas where a wide use of protective clothing and respiratory protection is deemed necessary based on the work being performed in an expected airborne environment.
- 4 DAC-hours accumulative air borne exposure or greater.
- Facial contamination that is statistically significant for $\beta\gamma$ contamination or any detectable alpha contamination.
- Air borne work exceeds the protection factor of the respiratory protection device and the expected exposure was likely to have exceeded 4 DAC-hours (respirator protection factor used in the DAC-hour calculation).
- Indication of a positive bioassay result.
- Detectable contamination on nasal swab or oral sample.

- Work where there is likelihood to exceed 100 mrem CEDE.
- Other evidence that radioactive material might have been ingested, inhaled, injected or absorbed into the body.
- Event based where there has been a suspected internal deposition.
- Evidence of an open wound that occurs in a contamination or airborne area.
- Exposure to air borne contamination that exceeds 10 percent of the annual limit of intake.
- Job specific review of bioassay requirement indicates that the facility prescribed bioassay program is not adequate for the work scope.
- Failure of a respiratory protection device in an area having airborne radioactivity where an uptake would be likely to exceed 100 mrem CEDE.
- Prolonged exposure to radionuclides that might be absorbed through the skin, such as tritium and radioiodine.
- Alarming a PCM-1B or comparable personnel contamination monitor where the source does not appear to be external contamination but the possibility for internal contamination exist and the employee had not undergone a medical diagnostic treatment.
- Indications show that any likely intake of radioactive material could cause an individual to exceed the annual administrative dose level of 1500 mrem TEDE.

The internal dosimetry program in conjunction with the many other facets that makeup the comprehensive INEEL radiation protection program (engineering controls, ALARA planning, administrative controls, radworker training, personal protection equipment, radiological control oversight, etc.) have demonstrated that intakes greater than 100 mrem CEDE are being prevented and have not gone undetected if one did occur. Therefore, these trigger levels are expected to adequately identify when there may have been a loss of control or a working condition where an above average risk warrants bioassay evaluation of a potential uptake.

3.7 Preliminary Investigative Actions

In the event that a potential internal intake is indicated from a trigger level or a known loss of control, the following preliminary investigative actions are utilized to begin the investigation to assess the magnitude of the internal dose.

Employees may be placed on access control restriction in order to prevent a second uptake during the course of the investigation.

The radionuclides that make-up the exposed source term are generally known from facility knowledge, but become refined for each specific case through analysis of the encountered contamination, through smears samples, air filters, nose swabs, etc, and eventually assessed by radiobioassay techniques to determine the specific radionuclides of concern.

An *in vivo* count at Radiological and Environmental Sciences Laboratory (RESL) CF-690 is often performed soon after a suspected intake to help establish the magnitude of an event-based intake where photon activity is the primary or principle radionuclide tag that can be ascertained into a reasonable preliminary dose. Affected employees and others with important information are interviewed to help determine conditions surrounding a suspected intake and to establish a date/time when an intake could most likely have occurred.

Where both beta-gamma and/or alpha contamination are suspected, preliminary assessments will often utilize estimated ratios. These are determined from what information is initially available, i.e., WBC, proportional counting of smear or air sample, analytical sample histories, etc. where upon a reasonable preliminary internal dose is estimated, and is refined later through additional radiobioassay results.

Employee counseling continues during the course of an investigation to keep the employee informed of the status of the assessment, to provide an opportunity to answer any questions, and to help the employee understand the fundamental information necessary to establish any internal dose assignment.

The Radiological Control organization establishes the necessary controls at the facility to allow an investigation and the necessary fact finding information to go undisturbed until the event can be assessed to the degree necessary to establish an internal dose assessment. Pertinent Radiological Control surveys are performed, information gathered and samples submitted for analytical analyses to help refine preliminary dose assessments.

The INEEL Medical organization is kept abreast of an internal deposition that may be of occupational concern and are consulted with and informed in accordance with the technical basis (INEL-003) that addresses medical intervention.

3.8 Type and Frequency of Bioassay Measurements

The following time analysis tables are used to determine appropriate routine bioassay sampling frequencies (bioassay programs for employees who are likely to have intakes resulting in a CEDE greater than 100 mrem) based upon a radionuclide's Minimum Detection Activity (MDA). If a sampling frequency is such that the period of time post intake is not frequent enough, then an actual dose greater than 100 mrem CEDE could be received by a worker where the bioassay measurement is reported as being less than MDA. To preclude this from happening, bioassay samples taken in the course of periodic sampling, event related, or follow-up sampling, must be timely enough that potential radioactivity in the sample is detectable (and not below MDA because of excessive time lapse). A time analysis is dependent upon bioassay type and the analytical laboratories reported MDA. The following information highlights the time analyses and how they are used and interpreted:

- The radionuclide of concern is shown, with some examples of compounds with representative chemical classes. The chemical class is determined by the pulmonary clearance classification of inorganic compounds as reported by the Task Group of Lung Dynamics. The chemical class that results in the highest CEDE will be used if the chemical compound can not be determined.

- A bioassay sample type (e.g., fecal, urine, whole-body count, etc.) is identified for the radionuclide. The sample type is dependent on the radionuclide deposition and excretion characteristics and therefore, must be appropriate for the radionuclide(s) that may be present from an intake. (e.g., A dose greater than 100 mrem could potentially exist for a urine sample taken at 3 days post intake and analyzed for Class W Pu-239 where the activity in the sample was evaluated as equal to or slightly less than the analytical laboratories MDA.)
- The Minimum Detection Activity (MDA) as reported by the analytical laboratory is shown for each radionuclide. This value is a threshold at which radioactivity can be detected, however, it is dependent upon many factors such as background, detector efficiency, analytical technique, and age of the detectors, and therefore will have a slight change over time. As well, radioactivity excreted from an intake diminishes over time to a point that the quantity of activity in a sample may not be detectable. A timely bioassay assessment is required following a suspected intake to minimize the chance that an intake cannot be quantified as dose received.
- The time from uptake column identifies the period between when an intake is postulated to have occurred until a radiobioassay was performed. The periods listed stop at 1 year since that period is the practical maximum that would normally exist between bioassays. The tables clearly show that some radionuclides can be adequately detected 12 months post intake, e.g., Cs-137, etc. However, other radionuclides such as Pu-238, Pu-239, Am-241, Ce-144, etc. need to have prompt attention.

ICRP-30 biokinetic models were used to estimate intake and establish the following time analysis tables. The Internal Dosimetrist identifies the appropriate biokinetic model to be used in the event of an actual intake. The ICRP-30 model provides a good estimate; however, there are other models that could be used for specific radionuclides and bioassay sampling methods.

Table 1. Time analysis of analytical minimum detectable activity for radionuclide H-3 (D) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci/mL}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
H-3 (D)—All HTO compounds	Urine	1.4E-6	1	ICRP 30 (REMedy)	3.8E-3
			3		4.83E-3
			5		5.0E-3
			7		5.7E-3
			10		7.0E-3
			30		2.8E-2
			60		2.3E-1
			90		1.8
			If there is an intake of H-3 (D) and the bioassay is submitted 114 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.		
If there is an intake of H-3 (D) and the bioassay is submitted 148 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA			148		100.0
a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a urine excretion of 3.0 ml grab sample upon equilibrium in the body approximately 2.0 hours post intake.					

Table 2. Time analysis of analytical minimum detectable activity for radionuclide Co-60 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (nCi)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE (mrem)
Co-60 (W)—all compounds except those identified as class Y.	Whole-body Count	1.1	1	ICRP 30 (REMedy)	0.06
			3		0.12
			5		0.17
			7		0.20
			10		0.22
			30		0.29
			60		0.41
			90		0.57
			182		1.3
			365		3.2

Table 3. Time analysis of analytical minimum detectable limit for radionuclide Co-60 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (nCi)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE (mrem)
Co-60 (Y)—Oxides, Hydroxides, Halide, nitrates	Whole body Count	1.1	1	ICRP 30 (REMedy)	0.42
			3		0.81
			5		1.3
			7		1.4
			10		1.5
			30		1.6
			60		1.7
			90		1.8
			182		2.0
			365		2.6

Table 4. Time analysis of analytical minimum detectable activity for radionuclide Sr-90 (D) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sampl}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
Sr-90 (D)—All other compounds of Sr except SrTiO ₃ .	Urine	1.9E-6	1	ICRP 30 (REMedy)	7.5E-3
			3		1.2E-2
			5		1.8E-2
			7		2.7E-2
			10		4.4E-2
			30		7.5E-1
			60		2.9
			90		3.6
			182		6.0
			If there is an intake of Sr-90 (D) and the bioassay is submitted 312 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.		
			365		11.9

a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.

Table 5. Time analysis of analytical minimum detectable activity for radionuclide Cs-137 (D) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (nCi)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE (mrem)
Cs-137 (D)—all compounds	Whole-body Count	1.9	1	ICRP 30 (REMedy)	0.098
			3		0.10
			5		0.11
			7		0.11
			10		0.11
			30		0.13
			60		0.16
			90		0.19
			182		0.34
			365		1.1

Table 6. Time analysis of analytical minimum detectable activity for radionuclide Ce-144 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (nCi)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE (mrem)
Ce-144 (W)—All compounds except those identified as class (Y)	Whole-body count	15	1	ICRP 30 (REMedy)	5.5
			3		9.4
If there is an intake of Ce-144 (Y) and the bioassay is submitted 4 day post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			4		11.7
			5		13.3
			7		15.0
			10		15.8
			30		18.6
			60		22.9
			90		27.3
			182		40.5
			365		69.7

Table 7. Time analysis of analytical minimum detectable activity for radionuclide Ce-144 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (nCi)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE (mrem)
Ce-144 (Y)—Oxides, Hydroxides, and fluorides	Whole-body count	15	1	ICRP 30 (REMedy)	9.7
If there is an intake of Ce-144 (Y) and the whole-body count is performed 2 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			2		13.3
			3		19.1
			5		30.5
			7		35.5
			10		37.4
			30		40.2
			60		44.4
			90		49.0
			182		66.4
If there is an intake of Ce-144 (Y) and the whole-body count is performed 310 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			310		101.0
			365		120.0

Table 8. Time analysis of analytical minimum detectable activity for radionuclide U-234 (D) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)				
U-234 (D)— UF_6 , UO_2F_6 , $\text{UO}_2(\text{NO}_3)_2$	Urine	4.1E-8	1	ICRP 30 (REMedy)	5.8E-4				
			3		3.8E-3				
			5		8.4E-3				
			7		1.1E-2				
			10		1.5E-2				
			30		6.4E-2				
			60		2.4E-1				
			90		7.0E-1				
			If there is an intake of U-234 (D) and the bioassay is submitted 171 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			171		10	
							182		13.6
			365		66.2				

a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters for a collection period of 24 hours.

Table 9. Time analysis of analytical minimum detectable activity for radionuclide U-234 (W) based upon inhalation.

Radionuclide and Chemical Class	Bioassay Type	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Intake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
U-234 (W)— UO_3 , UF_4 , UCl_4	Urine	4.1E-8	1	ICRP 30 (REMedy)	7.6E-3
			3		7.4E-2
			5		1.2E-1
			7		1.4E-1
			10		1.8E-1
			30		4.4E-1
			60		7.7E-1
			90		1.2
			182		3.4
If there is an intake of U-234 (W) and the bioassay is submitted 284 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			284		10.0
			365		29.1
a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.					

Table 10. Time analysis of analytical minimum detectable activity for radionuclide U-234 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (μCi/Sample)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
U-234 (Y)—UO ₂ , U ₃ O ₈	Urine	4.1E-8	1	ICRP 30 (REMedy)	2.4
If there is an intake of U-234 (Y) and the bioassay is submitted 2 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			2		11.8
			3		25.9
			5		40.9
			10		63.3
If there is an uptake of U-234 (Y) and the bioassay is submitted 17 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			17		98.0
			30		164.0
			60		249.0
			182		295.0
			365		293.0

a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.

Table 11. Time analysis of analytical minimum detectable activity for radionuclide U-235 (D) based upon inhalation.

Radionuclide and Chemical Class	Bioassay Type	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)			
U-235 (D)— UF_6 , UO_2F_6 , $\text{UO}_2(\text{NO}_3)_2$	Urine	3.8E-8	1	ICRP 30 (REMedy)	5.0E-4			
			5		7.3E-3			
			7		9.7E-3			
			10		1.3E-2			
			30		5.5E-2			
			60		2.1E-1			
			90		6.1E-1			
			If there is an uptake of U-235 (D) and the bioassay is submitted 176 days or more post intake, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			176	10	
							182	11.8
			365	57.5				

a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.

Table 12. Time analysis of analytical minimum detectable activity for radionuclide U-235 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
U-235 (W)— UO_3 , UF_4 , UCl_4	Urine	3.8E-8	1	ICRP 30 (REMedy)	6.6E-3
			3		6.4E-2
			5		1.0E-1
			7		1.3E-1
			10		1.6E-1
			30		3.8E-1
			60		6.7E-1
			90		1.0
			182		3.0
If there is an intake of U-235 (W) and the bioassay is submitted 286 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			286		10.0
			365		25.2
a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.					

Table 13. Time analysis of analytical minimum detectable activity for radionuclide U-235 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (μCi/Sample)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
U-235 (Y)—UO ₂ , U ₃ O ₈	Urine	3.8E-8	1	ICRP 30 (REMedy)	2.1
If there is an intake of U-235 (Y) and the bioassay is submitted 2 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			2		10.2
			3		22.4
			7		43.3
			10		54.9
If there is an intake of U-235 (Y) and the bioassay is submitted 20 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			20		99.0
			30		142.0
			60		216.0
			182		256.0
			365		254.0

a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.

Table 14. Time analysis of analytical minimum detectable activity for radionuclide U-238 (D) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
U-238 (D)— UF_6 , UO_2F_6 , $\text{UO}_2(\text{NO}_3)_2$	Urine	3.0E-8	1	ICRP 30 (REMedy)	3.8E-4
			30		4.2E-2
			60		1.6E-1
			90		4.6E-1
If there is an intake of U-238 (D) and the bioassay is submitted 182 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			182		9
			365		43.7
<p>NOTE: ICRP 54—“Intakes of the more transportable uranium compounds are limited by considerations of chemical toxicity, rather than radiation dose. This limits the intake by inhalation of soluble compounds in any day is 2.5 mg regardless of isotopic composition. For uranium of natural composition, a daily intake of 2 mg corresponds to a daily intake of 30 Bq Uranium-238. Thus for natural uranium of class D and W, the limits for chemical toxicity are more restrictive. For class Y natural uranium and all classes of highly enriched uranium, radiological considerations are more limiting.” This time-line is based upon radiological dose only and not chemical toxicity.</p> <p>a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.</p>					

Table 15. Time analysis of analytical minimum detectable activity for radionuclide U-238 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
U-238 (W)— UO_3 , UF_4 , UCl_4	Urine	3.0E-8	1	ICRP 30 (REMedy)	5.0E-3
			30		2.9E-1
			60		5.1E-1
			90		7.6E-1
			182		2.3
If there is an intake of U-238 (W) and the bioassay is submitted 310 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			310		10.0
			365		19.1
<p>NOTE: ICRP 54—“Intakes of the more transportable uranium compounds are limited by considerations of chemical toxicity, rather than radiation dose. This limits the intake by inhalation of soluble compounds in any day is 2.5 mg regardless of isotopic composition. For uranium of natural composition, a daily intake of 2 mg corresponds to a daily intake of 30 Bq Uranium-238. Thus for natural uranium of class D and W, the limits for chemical toxicity are more restrictive. For class Y natural uranium and all classes of highly enriched uranium, radiological considerations are more limiting.” This time-line is based upon radiological dose only and not chemical toxicity.</p> <p>a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.</p>					

Table 16. Time analysis of analytical minimum detectable activity for radionuclide U-238 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (μCi/Sample)	Time from Uptake (Days)	Biokinetic Model	Inhalation CEDE (mrem)
U-238 (Y)—UO ₂ , U ₃ O ₈	Urine	3.0E-8	1	ICRP 30 (REMedy)	1.6
If there is an intake of U-238 (Y) and the bioassay is submitted more than 2 days post intake, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			2		7.7
			3		17.0
			5		26.9
			10		41.6
If there is an intake of U-238 (Y) and the bioassay is submitted 27 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			27		99.0
			30		108.0
			182		194.0
			365		192.0
NOTE: ICRP 54—“Intakes of the more transportable uranium compounds are limited by considerations of chemical toxicity, rather than radiation dose. This limits the intake by inhalation of soluble compounds in any day is 2.5 mg regardless of isotopic composition. For uranium of natural composition, a daily intake of 2 mg corresponds to a daily intake of 30 Bq Uranium-238. Thus for natural uranium of class D and W, the limits for chemical toxicity are more restrictive. For class Y natural uranium and all classes of highly enriched uranium, radiological considerations are more limiting.” This time-line is based upon radiological dose only and not chemical toxicity.					

Table 17. Time analysis of decision level excretion of uranium based upon inhalation of assumed Class W depleted uranium (DU). Particle size is assumed to be 2.4 μm AMAD.

Radionuclide	Type Analysis	Decision Level($\mu\text{g/l}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem ^b)
Depleted Uranium Assuming U-238 (W) predominant radionuclide	Urine	1.0	1	ICRP 30 (REMedy)	6.8E-2
			30		6
If the bioassay is submitted 55 days post intake or more, a dose smaller than 10 mrem from an acute occupational inhalation intake of DU would not be distinguishable from non-occupational excretion using 1.0 $\mu\text{g/l}$ as the decision level			55		10.0
			60		12
			90		20
			182		66
If the bioassay is submitted 215 days post intake or more, a dose smaller than 100 mrem from an acute occupational inhalation intake of DU would not be distinguishable from non-occupational excretion using 1.0 $\mu\text{g/l}$ as the decision level.			215		100
			365		544

NOTE: ICRP 54—"Intakes of the more transportable uranium compounds are limited by considerations of chemical toxicity, rather than radiation dose. This limits the intake by inhalation of soluble compounds in any day is 2.5 mg regardless of isotopic composition. For uranium of natural composition, a daily intake of 2 mg corresponds to a daily intake of 30 Bq Uranium-238. Thus for natural uranium of class D and W, the limits for chemical toxicity are more restrictive. For class Y natural uranium and all classes of highly enriched uranium, radiological considerations are more limiting." This time-line is based upon radiological dose only and not chemical toxicity. See the SMC TBD for a discussion of toxicity concerns.

a. Committed Effective Dose Equivalency (CEDE) is based on ICRP 23 reference man and a urinary excretion of 1.4 liters from a collection period of 24 hours.

b. Based on historical data, the SMC project has established an excretion decision level of 1.0 $\mu\text{g/l}$ total uranium or greater as a performance indicator, with excretion between 0.175 and 1.0 $\mu\text{g/l}$ evaluated on a case-by-case basis. The median excretion of uranium from nonoccupational sources is assumed to be 0.034 $\mu\text{g/l}$, which is subtracted from measured amounts prior to dose assessments. (See EDF 3256 "Established Both a Uranium Background and Positive Action Level in Urine") Following correction for non-occupational excretion, an excretion level of 1.0 $\mu\text{g/l}$ equates to 3.19E-10 $\mu\text{Ci/ml}$ or 4.5 E-7 $\mu\text{Ci/day}$ assuming reference man excretion of 1.4 liter per day. This equates to an assessed dose of 100 mrem CEDE at 215 days post intake SMC uses an assumed particle size of 2.4 μm AMAD based on prior assessments and applies a correction factor of 1.16 to U-238 doses to account for the contribution of U-235 and U-234 to the total dose (see SMC TBD for further discussion).

Table 18. Time analysis of analytical minimum detectable activity for radionuclide Pu-238 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (μCi/Sample)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
Pu-238 (W)—All compounds of Pu except PuO ₂ .	Fecal	2.2E-8	1	ICRP-30 (REMedy)	0.23
			7		1.1
			10		4.7
			27		10.0
			30		10.4
			60		15.7
			90		23.7
			182		83.6
			195		100.0
			365		905.0

a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a fecal excretion of 135 grams from a collection period of 24 hours.

Table 19. Time analysis of analytical minimum detectable activity for radionuclide Pu-238 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
Pu-238 (Y)—PuO ₂	Fecal	2.2E-8	1	ICRP 30 (REMedy)	0.13
			3		0.05
			5		0.19
			7		0.83
If there is an intake of Pu-238 (Y) and the bioassay is submitted 10 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			10		6.8
			30		50.7
			60		52.9
			90		55.1
			182		62.7
			365		80.8

a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a fecal excretion of 135 grams from a collection period of 24 hours.

Table 20. Time analysis analytical minimum detectable activity for radionuclide Pu-239 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (μCi/Sample)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)		
Pu-239 (W)—All compounds of Pu except PuO ₂ .	Fecal	2.7E-8	1	ICRP 30 (REMedy)	0.32		
			7		2		
			10		7		
			13		10		
			If there is an intake of Pu-239 (W) and the bioassay is submitted 13 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.				
			30		14		
			60		22		
			90		33		
			172		102		
			If there is an intake of Pu-239 (W) and the bioassay is submitted 172 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.				
182	116						
365	1260						
a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a fecal excretion of 135 grams from a collection period of 24 hours.							

Table 21. Time analysis of analytical minimum detectable activity for radionuclide Pu-239 (Y) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (μCi/Sample)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
Pu-239 (Y)—PuO ₂	Fecal	2.7E-8	1	ICRP-30 (REMedy)	0.18
			5		0.25
			7		1.1
If there is an intake of Pu-239 (Y) and the bioassay is submitted 10 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			10		9.2
			30		68.4
			60		71.2
			90		74.2
			182		84.2
If there is an intake of Pu-239 (Y) and the bioassay is submitted 310 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			310		100.0
			365		108.0
a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a fecal excretion of 135 grams from a collection period of 24 hours.					

Table 22. Time analysis of analytical minimum detectable activity for radionuclide Am-241 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
Am-241 (W)—All compounds	Fecal	2.3E-8	1	ICRP-30 (REMedy)	0.28
			5		0.36
			7		1.4
			10		5.8
If there is an intake of Am-241(W) and the bioassay is submitted 15 days post intake or more, a dose smaller than 10 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			15		10.0
			30		12.6
			60		19.0
			90		28.7
If there is an intake of Am-241 (W) and the bioassay is submitted 182 days post intake or more, a dose smaller than 100 mrem would not be identifiable as a result of the radioactivity in the bioassay being less than the analytical laboratories detection capabilities e.g., MDA.			182		101.0
			365		1090.0
a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a fecal excretion of 135 grams from a collection period of 24 hours.					

Table 23. Time analysis of lung count minimum detectable activity for radionuclide Am-241 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA (nCi)	Time from Uptake (Days)	Biokinetics Model	Inhalation CEDE ^a (mrem)
Am-241 (W)—All compounds	Lung count	0.14	1	ICRP-30 (REMedy)	350
			5		511
			30		720
			60		1,060
			90		1,550
			182		5,080
			365		54,700
a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man.					

Table 24. Time analysis of analytical minimum detectable activity for radionuclide Am-241 (W) based upon inhalation.

Radionuclide	Type Analysis	Analytical MDA ($\mu\text{Ci}/\text{Sample}$)	Time from Uptake (Days)	Software program Biokinetics Model	Inhalation CEDE ^a (mrem)
Am-241 (W)—All compounds	Urine	2.3E-8	1	CINDY (Durbin Excretion)	42
			5		130
			15		330
			30		540
			60		720
			90		900
			182		1500
			365		3000

a. Committed Effective Dose Equivalent (CEDE) is based on ICRP 23 reference man and a fecal excretion of 1400 ml from a collection period of 24 hours.

3.9 Technology Shortfall

There has been an extensive history of analyzing both urine and fecal samples for actinides at the INEEL (primarily INTEC, WROC and RWMC) facilities. Radiation workers who participated in the routine *in vitro* bioassay program throughout the 1980s and into the 1990s were having their samples analyzed on a semi-annual frequency based on a detection level of 100 mrem CEDE shown to be obtainable with a fecal bioassay. Fecal bioassays have been very successful in detecting transuranic depositions while a urine sample does not possess the necessary sensitivity. Fecal samples had historically detected some areas where intake problems had occurred, however, in the last eleven years the few routine fecal samples that have shown detectable transuranic radionuclides have seldom been detectable in a follow-up sample. The dose assessment on these few positive fecal bioassays have indicated low level ingestion or inhalation intakes where the CEDE is less than 100 mrem with most of them less than 10 mrem. Therefore, the utilization of fecal bioassays rather than urine samples where transuranic radionuclides are expected, avoids a technology shortfall.

An equivalent bioassay program had been performed for Sr/Y-90 in urine samples at INTEC, RWMC and TAN during this same period of time. While Sr/Y-90 is readily detectable in a urine bioassay, history has shown that rarely if ever has there been a significant uptake of only Sr/Y-90, without having Cs-137 present as well.

Second, most of the facilities where there could be a technology shortfall with a specific radionuclide have their source terms tagged with photon energies mixed within the fission, activation and corrosion products that make up the facilities' source of radioactive contamination.

The following tables demonstrate the adequacy of using a whole body count for Cs-137 with an MDA of 2.0 nCi, to trigger a follow-up *in vitro* bioassay for other radionuclides of internal interest. The appropriate following-up *in vitro* bioassay sample, when Cs-137 is detected, is time dependent and therefore a critical factor in determining frequency in a job-related bioassay program. The tables below are based on an acute inhalation intake. Comparative data based on ICRP-30 dose modeling and compiled using REMedy 3.01 internal dosimetry software.

Various delays have been used between intake and sample dates and various ratios are assumed to envelop a wide range of possibilities.

Airborne radioactivity having a Sr/Y-90 to Cs-137 intake ratio of 1 to 1

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Sr/Y-90	Cs-137 ⁽³⁾	Sr/Y-90	Cs-137	Sr/Y-90	Cs-137
10	2.73E-8 uCi/ml	2 nCi	3.72 E-3	3.72E-3	0.9	0.1
30	1.83 E-9 uCi/ml	2 nCi	4.25E-3	4.25E-3	1.0	0.1
33 ⁽¹⁾	1.36E-9 uCi/ml ⁽²⁾	2 nCi	4.29E-3	4.33E-3	1.0	0.1
90	< MDA	2 nCi	6.23 E-3	6.23E-3	1.5	0.2
180	< MDA	2 nCi	1.10E-2	1.10E-2	2.6	0.4
365	< MDA	2 nCi	3.58E-2	3.58E-2	8.6	1.1

- (1) Sr/Y-90, Minimum Detectable Activity at 33 days post-intake, based on a Sr/Cs intake ratio of 1 with capability of detecting Cs-137 at 2 nCi.
- (2) Approximate Sr/Y-90 MDA 1.9E-6 uCi/sample, assume 1400 ml reference man urine excretion, 1.36E-9 uCi/ml 24 hour urine
- (3) WBC with detectable Cs-137 radioactivity at 2.0 nCi, approximate MDA for 10 minute count with CF-690 40 cm arch geometry using 133% HpGe detector.

Airborne Radioactivity having a Sr/Y-90 to Cs-137 intake ratio of 2 to 1

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Sr/Y-90 uCi/ml	Cs-137	Sr/Y-90	Cs-137	Sr/Y-90	Cs-137
10	5.48E-8 uCi/ml	2 nCi	7.44E-3	3.72E-3	1.8	0.1
30	3.66E-9 uCi/ml	2 nCi	8.50E-3	4.25E-3	2.0	0.1
46 ⁽¹⁾	1.36E-9 uCi/ml ⁽²⁾	2 nCi	9.41E-3	4.71E-3	2.3	0.2
90	< MDA	2 nCi	1.25E-2	6.23E-3	3.0	0.2
180	< MDA	2 nCi	2.20E-2	1.10E-2	5.2	0.8
365	1.96E-9 uCi/ml	2 nCi	7.16E-2	3.58E-2	17.2	2.2

- (1) Sr/Y-90, Minimum Detectable Activity at 46 days post-intake, based on a Sr/Cs intake ratio of 2 to 1 with capability of detecting Cs-137 at 2 nCi.
- (2) Approximate Sr/Y-90 MDA 1.9E-6 uCi/sample, assume 1400 ml reference man urine excretion, 1.36E-9 uCi/ml 24 hour urine

Airborne Radioactivity having a Sr/Y-90 to Cs-137 intake ratio of 5 to 1

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Sr/Y-90	Cs-137	Sr/Y-90	Cs-137	Sr/Y-90	Cs-137
10	1.36 E-7 uCi/ml	2 nCi	1.86E-2	3.72E-3	4.5	0.1
30	9.16E-9 uCi/ml	2 nCi	2.13E-2	4.25E-3	5.1	0.1
90	2.79E-9 uCi/ml	2 nCi	3.12E-2	6.23E-3	7.5	0.2
180	2.98E-9 uCi/ml	2 nCi	5.50E-2	1.10E-2	13.2	0.4
365	4.89E-9 uCi/ml	2 nCi	1.79E-1	3.58E-2	42.9	1.1

Airborne Radioactivity having a Sr/Y-90 to Cs-137 intake ratio of 10 to 1

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Sr/Y-90	Cs-137	Sr/Y-90	Cs-137	Sr/Y-90	Cs-137
10	2.74 E-7 uCi/ml	2 nCi	3.72E-2	3.72E-3	9.0	0.1
30	1.83E-8 uCi/ml	2 nCi	4.25E-2	4.25E-3	10.2	0.1
90	5.56E-9 uCi/ml	2 nCi	6.23E-2	6.23E-3	14.9	0.2
180	5.96E-9 uCi/ml	2 nCi	1.10E-1	1.10E-2	26.4	0.4
365	9.78E-9 uCi/ml	2 nCi	3.58E-1	3.58E-2	85.8	1.1

Hypothetically, assuming a source term having a 1 to 1 Sr/Y-90 to Cs-137 ratio, if multiple acute intakes were to occur at the first day of every month and with the Cs-137 intake amount being just less than the WBC detection threshold of 2 nCi each time, then an accumulative internal dose of 12 mrem CEDE is estimated for the Sr/Y-90. Likewise, with a ratio of 10 to 1 the estimated Sr/Y-90 dose would be 122 mrem CEDE.

Conclusion – if multiple acute intakes could go undetected through field RadCon survey practices, that is intakes of Cs-137 just less than the WBC detection threshold, the Sr/Y-90 would not result in a significant dose even for ratios of 10 to 1. Consequently the following protocol will be used. When Sr/Y-90 is thought to be present and the ratio is unknown or has been reasonably characterized to be greater than 5 to 1, a urine sample should be taken to determine the Sr/Y-90 dose. For ratios of 5 to 1 or less, a positive detection of Cs-137 in a WBC may be used as a trigger, for following up with a urine bioassay analysis for Sr/Y-90.

Airborne Radioactivity having a Pu-238 (W)/Cs-137 ratio of 0.02

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-238 (W) and a follow up Pu-238 fecal analyses when a positive WBC is identified. The table indicates that a Pu-238 (W)/Cs-137 ratio of 0.02 results in a maximum internal dose of 331 mrem CEDE when the WBC is received 365 days post intake. After 210 days post intake, the Pu-238 activity is shown to fall below the MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-238 (W)	Cs-137	Pu-238 (W)	Cs-137	Pu-238 (W)	Cs-137
10	1.16 E-9 uCi/gm	2 nCi	7.44 E-5	3.72E-3	34.4	0.1
30	6.08 E-10 uCi/gm	2 nCi	8.50 E-5	4.25E-3	39.2	0.1
90	3.90 E-10 uCi/gm	2 nCi	1.25 E-4	6.23E-3	57.6	0.2
180	2.01 E-10 uCi/gm	2 nCi	2.20 E-4	1.10E-2	102.0	0.4
210	1.63 E-10 uCi/gm	2 nCi	2.69E-4	1.34E-2	124.0	0.4
365	5.87E-11 uCi/gm < MDA	2 nCi	7.16E-4	3.58E-2	331.0	1.1

* Approximate Pu-238 MDA 2.2E-8 uCi/sample, assume 135 grams reference man fecal excretion, 1.63E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-238 (W)/Cs-137 ratio of 0.01

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-238 (W) and a follow up Pu-238 fecal analyses when a positive WBC is identified. The table indicates that a Pu-238 (W)/Cs-137 ratio of 0.01 results in a maximum internal dose of 165 mrem CEDE when the WBC is received 365 days post intake. After 114 days post intake, the Pu-238 activity is shown to fall below the MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-238 (W)	Cs-137	Pu-238 (W)	Cs-137	Pu-238 (W)	Cs-137
10	5.80 E-10 uCi/gm	2 nCi	3.72E-5	3.72E-3	17.2	0.1
30	3.04 E-10 uCi/gm	2 nCi	4.25E-5	4.25E-3	19.6	0.1
90	1.95 E-10 uCi/gm	2 nCi	6.23E-5	6.23E-3	28.8	0.2
114	1.63E-10 uCi/gm	2 nCi	7.24E-5	7.25E 3	33.5	0.2
180	1.00E-10 uCi/gm < MDA	2 nCi	1.10E-4	1.10E-2	51.0	0.4
365	2.97 E-11 uCi/gm < MDA	2 nCi	3.58E-4	3.58E-2	165.0	1.1

* Approximate Pu-238 MDA 2.2E-8 uCi/sample, assume 135 grams reference man fecal excretion, 1.63E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-238 (Y)/Cs-137 ratio of 0.02

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-238 (Y) and a follow up Pu-238 fecal analyses when a positive WBC is identified. The table indicates that a Pu-238 (Y)/Cs-137 ratio of 0.02 results in a maximum internal dose of 225 mrem CEDE when the WBC is received 365 days post intake. After 12 days and up until 180 days post intake, the Pu-238 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-238 (Y)	Cs-137	Pu-238 (Y)	Cs-137	Pu-238 (Y)	Cs-137
10	5.50E-10 uCi/gm	2 nCi	7.44E-5	3.72E-3	23.4	0.1
12	1.63E-10 uCi/gm ⁽¹⁾	2 nCi	6.46E-5	3.78 E-3	20.3	0.1
30	8.45E-11 uCi/gm < MDA	2 nCi	8.50 E-5	4.25E-3	26.7	0.1
90	1.13 E-10 uCi/gm < MDA	2 nCi	1.24E-4	6.23E-3	39.2	0.2
180	1.77 E-10 uCi/gm	2 nCi	2.20E-4	1.10E-2	69.2	0.4
365	4.46E-10 uCi/gm	2 nCi	7.16E-4	3.58E-2	225.0	1.1

(1) Approximate Pu-238 MDA 2.2E-8 uCi/sample, assume 135 grams reference man fecal excretion, 1.63E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-238 (Y)/Cs-137 ratio of 0.01

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-238 (Y) and a follow up Pu-238 fecal analyses when a positive WBC is identified. The table indicates that a Pu-238 (Y)/Cs-137 ratio of 0.01 results in a maximum internal dose of 113 mrem CEDE when the WBC is received 365 days post intake. After 10 days and up until 365 days post intake, the Pu-238 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-238 (Y)	Cs-137	Pu-238 (Y)	Cs-137	Pu-238 (Y)	Cs-137
10	2.75 E-10 uCi/gm	2 nCi	3.72E-5	3.72E-3	11.7	0.1
30	4.22E-11 uCi/gm < MDA	2 nCi	4.25E-5	4.25E-3	13.4	0.1
90	5.65E-11 uCi/gm < MDA	2 nCi	6.23E-5	6.23E-3	19.6	0.2
180	8.85 E-11 uCi/gm < MDA	2 nCi	1.10E-4	1.10E-2	34.6	0.4
365	2.22E-10 uCi/gm	2 nCi	3.58E-4	3.58E-2	113.0	1.1

Approximate Pu-238 MDA 2.2E-8 uCi/sample, assume 135 grams reference man fecal excretion, 1.63E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-239 (W)/Cs-137 ratio of 0.02

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-239 (W) and a follow up Pu-239 fecal analyses when a positive WBC is identified. The table indicates that a Pu-239 (W)/Cs-137 ratio of 0.02 results in a maximum internal dose of 371 mrem CEDE when the WBC is received 365 days post intake. After 180 days post intake, the Pu-239 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-239 (W)	Cs-137	Pu-239 (W)	Cs-137	Pu-239 (W)	Cs-137
10	1.16 E-9 uCi/gm	2 nCi	7.44 E-5	3.72E-3	34.4	0.1
30	6.08 E-10 uCi/gm	2 nCi	8.50 E-5	4.25E-3	39.3	0.1
90	3.90 E-10 uCi/gm	2 nCi	1.25 E-4	6.23E-3	57.6	0.2
180	2.01 E-10 uCi/gm ⁽¹⁾	2 nCi	2.20 E-4	1.10E-2	102.0	0.4
365	5.91E-11 uCi/gm < MDA	2 nCi	7.16E-4	3.58E-2	371.0	0.1

(1) Approximate Pu-239 MDA 2.7E-8 uCi/sample, assume 135 grams reference man fecal excretion, 2.01E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-239 (W)/Cs-137 ratio of 0.01

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-239 (W) and a follow up Pu-239 fecal analyses when a positive WBC is identified. The table indicates that a Pu-239 (W)/Cs-137 ratio of 0.01 results in a maximum internal dose of 186 mrem CEDE when the WBC is received 365 days post intake. At approximately 90 days post intake, the Pu-239 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-239 (W)	Cs-137	Pu-239 (W)	Cs-137	Pu-239 (W)	Cs-137
10	5.80 E-10 uCi/gm	2 nCi	3.72E-5	3.72E-3	19.3	0.1
30	3.04 E-10 uCi/gm	2 nCi	4.25E-5	4.25E-3	22.0	0.1
90	1.95 E-10 uCi/gm < MDA	2 nCi	6.23E-5	6.23E-3	32.2	0.2
180	1.0E-10 uCi/gm < MDA	2 nCi	1.10E-4	1.10E-2	56.6	0.4
365	2.97E-11 uCi/gm < MDA	2 nCi	3.58E-4	3.58E-2	186.0	1.1

Approximate Pu-239 MDA 2.7E-8 uCi/sample, assume 135 grams reference man fecal excretion, 2.0E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-239 (Y)/Cs-137 ratio of 0.02

This Table assumes a Cs-137 Whole-body Count as a tag for Pu-239 (Y) and a follow up Pu-239 fecal analyses when a positive WBC is identified. The table indicates that a Pu-239 (Y)/Cs-137 ratio of 0.02 results in a maximum internal dose of 241 mrem CEDE when the WBC is received 365 days post intake. After 10 days and up until approximately 180 days post intake, the Pu-239 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-239 (Y)	Cs-137	Pu-239 (Y)	Cs-137	Pu-239 (Y)	Cs-137
10	5.51E-10 uCi/gm	2 nCi	7.44E-5	3.72E-3	25.3	0.1
30	8.45E-11 uCi/gm < MDA	2 nCi	8.50 E-5	4.25E-3	28.9	0.1
90	1.13E-10 uCi/gm < MDA	2 nCi	1.24E-4	6.23E-3	42.9	0.2
180	1.77E-10 uCi/gm	2 nCi	2.20E-4	1.10E-2	74.3	0.4
365	4.46E-10 uCi/gm	2 nCi	7.16E-4	3.58E-2	241.0	1.1

* Approximate Pu-239 MDA 2.7E-8 uCi/sample, assume 135 grams reference man fecal excretion, 2.0E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Pu-239 (Y)/Cs-137 ratio of 0.01

This table assumes a Cs-137 Whole-body Count as a tag for Pu-239 (Y) and follow up Pu-239 fecal analyses when a positive WBC is identified. The table indicates that a Pu-239 (Y)/Cs-137 ratio of 0.01 results in a maximum internal dose of 120 mrem CEDE when the WBC is received 365 days post intake. After 10 days and up until approximately 365 days post intake, the Pu-239 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Pu-239 (Y)	Cs-137	Pu-239 (Y)	Cs-137	Pu-239 (Y)	Cs-137
10	2.75 E-10 uCi/gm	2 nCi	3.72E-5	3.72E-3	12.6	0.1
30	4.22E-11 uCi/gm < MDA	2 nCi	4.25E-5	4.25E-3	14.5	0.1
90	5.65E-11 uCi/gm < MDA	2 nCi	6.23E-5	6.23E-3	21.5	0.2
180	8.85 E-11 uCi/gm < MDA	2 nCi	1.10E-4	1.10E-2	37.2	0.4
365	2.22E-10 uCi/gm	2 nCi	3.58E-4	3.58E-2	120.0	1.1

* Approximate Pu-239 MDA 2.7E-8 uCi/sample, assume 135 grams reference man fecal excretion, 2.0E-10 uCi/gram in a 24 hour fecal sample

Airborne Radioactivity having a Am-241 (W)/Cs-137 ratio of 0.02

This table assumes a Cs-137 Whole-body Count as a tag for Am-241 (W) and follow up Am-241 fecal analyses when a positive WBC is identified. The table indicates that an Am-241 (W)/Cs-137 ratio of 0.02 results in a maximum internal dose of 379 mrem CEDE when the WBC is received 365 days post intake. After 180 days post intake, the Am-241 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Am-241 (W)	Cs-137	Pu-241 (W)	Cs-137	Pu-241 (W)	Cs-137
10	1.16E-9 uCi/gm	2 nCi	7.44E-5	3.72E-3	40.0	0.1
30	6.09 E-10 uCi/gm	2 nCi	8.50 E-5	4.25E-3	45.0	0.1
90	3.9E-10 uCi/gm	2 nCi	1.24E-4	6.23E-3	66.0	0.2
180	2.0E-10 uCi/gm	2 nCi	2.20E-4	1.10E-2	116.0	0.4
365	5.9E-11 uCi/gm < MDA	2 nCi	7.16E-4	3.58E-2	379.0	1.1

Approximate Am-241 MDA 2.3E-8 uCi/sample, assume 135 grams reference man fecal excretion, 1.7E-10 uCi/gram in a 24 hour fecal sample.

Airborne Radioactivity having a Am-241 (W)/Cs-137 ratio of 0.01

This table assumes a Cs-137 Whole-body Count as a tag for Am-24 (W) and follow up Am-241 fecal analyses when a positive WBC is identified. The table indicates that an Am-241 (W)/Cs-137 ratio of 0.01 results in a maximum internal dose of 189 mrem CEDE when the WBC is received 365 days post intake. At approximately 90 days post intake, the Am-241 activity is shown to be less than MDA for a fecal bioassay sample with this ratio.

# days post intake	Bioassay		Intake (uCi)		Dose (mrem CEDE)	
	Am-241 (W)	Cs-137	Pu-241 (Y)	Cs-137	Pu-241 (Y)	Cs-137
10	5.8E-10 uCi/gm	2 nCi	3.72E-5	3.72E-3	20.0	0.1
30	3.04E-10 uCi/gm	2 nCi	4.25E-5	4.25E-3	23.0	0.1
90	1.95 E-10 uCi/gm	2 nCi	6.23E-5	6.23E-3	33.0	0.2
180	1.0E-10 uCi/gm < MDA	2 nCi	1.10E-4	1.10E-2	58.0	0.4
365	2.95E-11 uCi/gm < MDA	2 nCi	3.58E-4	3.58E-2	189.0	1.1

Approximate Am-241 MDA 2.3E-8 uCi/sample, assume 135 grams reference man fecal excretion, 1.7E-10 uCi/gram in a 24 hour fecal sample

Conclusion - Using Cs-137 as a tag for Pu-238, Pu-239 or Am-241 is not recommended. The tables above demonstrate that multiple internal acute intakes of recordable quantities greater than 100 mrem CEDE could occur if a Cs-137 tag were to be relied upon. Based on an acute inhalation uptake, these two radioisotopes of plutonium or americium can be readily detected in a fecal bioassay sample at Minimum Detectable Activity (MDA) thresholds level of approximately 100 mrem CEDE or less when bioassay sampling frequencies are no greater than 180 days. When the Cs-137 to actinide ratio has been characterized and found to be in excess of 1000 to 1, the need to bioassay for the actinides (Pu-238, Pu-239/240 or Am-241) be excluded in a facilities bioassay program.

3.10 Method and Rationale for Bioassay Measurements

The source terms encountered at the various facilities, for the most part, do not have to contend with short physical half-life radionuclides. The radioactive contaminants encountered are long-lived mixed fission, activation, and corrosion products with associated daughters; Cs/Ba-137m, Sr/Y90, Pu-238, Pu-239/240, Co-60, depleted uranium, Am-241. One exception is the Test Reactor Area (TRA) where short-lived radionuclides can be present resulting from the operation of the Advanced Test Reactor (ATR); Hf-175 and 181, Tc-99, I-131, Ni-59, Co-58, etc. However, the reactor source-term also has predominant long-lived Cs-137 and Co-60 high-energy photon tags that can be used to reference the presence of short-lived radionuclides against. As well, some laboratories and hot cells may have specific work that deals with an acquired or separated short-lived radionuclide. The Radiological Control facility programs deal with these types of source terms on a case by case bases and determines supporting internal dosimetry by way of job specific bioassay evaluations. Routine bioassay programs conducted through out the INEEL facilities have resulted in detecting few internal intakes.

The effective half-lives of the radionuclides encountered at the various INEEL facilities, that have contributed meaningful internal dose, are on the order of several days while others are months with some transuranic radionuclides being in years. Thus, the window that radionuclides remain detectable if there has been an intake is generally several months. Co-60 and Cs-137 are both predominant photon tags in facility source terms that are easily detectable via whole body counts. Cs-137 remains detectable several months to years after an intake and may still result in a dose assessment that remains less than 100 mrem CEDE. The few exceptions where uranium or transuranic radioactivity can be encountered and not have high-energy photon tags (i.e., Co⁶⁰, Cs¹³⁷) is the SMC project, RWMC Transuranic Storage Area, and some facility specific laboratory assaying. The few intakes that have occurred over the last Twelve years have been small in activity and small in dose.

Annual Summary of Recorded Internal Doses, Facilities involved and assessed Radionuclides

Year of dose assignment	Facility	Dose Assessment (mrem CEDE)	Radionuclides of concern
2003	INTEC	11, 34	Pu-238, Pu-239
2002	SMC	12, 15, 18, 36 and 36	U-238
2002	INTEC	13, 16, 15 and 38	Pu-238 and Pu-239
2001	SMC	11, 13, 17, 18 and 24	U-238
2000	SMC	10, 31, 20, 10, 13 and 21	U-238
2000	INTEC	11	Pu-238 and Pu-239/240
1999	SMC	15, 48, 13 and 12	U-238
1999	INTEC	16	Pu-239/240
1998	SMC	16	U-238
1997	TRA	10* and 10*	Eu-152 and Eu-154
1997	INTEC (ICPP)	24	Pu-238 and Pu-239/240
1997	TAN	13*	Am-241, Cs-137, Sr-90, U-233, U-238, Pu-238 and Pu-239/240
1997	SMC	16 and 20	U-238
1996	RWMC	43	Pu-239
1996	INTEC (ICPP)	15, 87, 136, 652*, 655*, 656*, 677* and 678*	Am-241, Cs-137, Sr-90, Pu-238 and Pu-239/240
1996	SMC	10, 10, 12, 16, 17, 18, 20, 20 and 23	U-238
1995	INTEC (ICPP)	10, 13, 14, 15, 23, 28, 29, 42, 45 and 53	Am-241, Cs-137, Sr-90, Pu-238 and Pu-239/240
1995	SMC	10, 12, 14, 15, 19 and 26	U-238
1994	INTEC (ICPP)	14, 20, 25 and 29	Am-241, Cs-137, Pu-238 and Pu-239/240
1994	SMC	10, 10, 12 and 15	U-238
1993	INTEC (ICPP)	14, 35, 36, 39, 50 and 53	Am-241, Pu-238 and Pu-239/240
1993	SMC	11	U-238
1992	RWMC	20 and 205	Am-241, Pu-238 and Pu-239/240
1992	SMC	11, 12, 12, 14, 15, 15, 16, 20, 32 and 52	U-238

* Internal doses (mrem CEDE) denoted with an asterisk were assessments that resulted from a known event where bioassays were utilized to assess the employee's internal dose.

3.11 Type of Radiobioassays and associated MDAs

Whole body Counts - The whole body counter located in CFA-690 is designed to identify and quantify common fission and activation product radionuclides that emit photons. The whole body counter system consists of a 133% high-purity germanium p-type coaxial low-background detector arranged in 40-cm arc bed geometry. The detector is housed in a shielded vault constructed of 9' x 9' x 7" steel, 1/4" lead, 1/50" cadmium, and 1/50" copper. The counter is controlled and spectra are analyzed by the ABACOS Plus software package. The WBC is calibrated for an energy range of 86.5 keV to 1836 keV. The approximate MDAs for this counting system (MDAs vary with background, counting time and calibration effects) is given in the following table for major radionuclides found at the INEEL.

40 cm Arch Geometry Body Counter approximate MDAs
(600 Second Count)

Nuclide	MDA For 133% HpGe P-type Coaxial Low-background detector
Mn-54	1.3 nCi
Co-58	1.1 nCi
Co-60	1.1 nCi
Fe-59	1.5 nCi
Zn-65	2.0 nCi
Ru-106	7.6 nCi
Cs-134	0.96 nCi
Cs-137	1.9 nCi
Ce-141	3.2 nCi
Ce-144	15.0 nCi
Eu-152	4.0 nCi
Eu-154	2.0 nCi
Eu-155	1.0 nCi
Ga-153	6.5 nCi

Fastscan Whole body count system - located in a mobile trailer is designed to identify and quantify common fission and activation product radionuclides that emit photons and x-rays. The Fastscan system includes two 3" x 5" x 16" NaI (TI) detectors in a linear configuration parallel to the axis of a subject standing erect in the shield. The counter is controlled and spectra are analyzed by the ABACOS Plus software package. The WBC is calibrated for an energy range of 88 keV to 1836 keV.

Fastscan WBC system approximate MDAs
(300 Second Count)

Nuclide	MDA For Fastscan WBC system
Cr-51	32.0 nCi
Mn-54	2.6 nCi
Co-58	2.5 nCi
Co-60	2.5 nCi
Fe-59	4.5 nCi
Zn-65	4.9 nCi
Zr/Nb-95	2.6 nCi
Ru-106	27 nCi
I-131	3.8 nCi
I-133	3.4 nCi
Cs-134	2.7 nCi
Cs-137	3.1 nCi
Ba/La-140	12.0 nCi
Ce-141	9.9 nCi
Ce-144	44.0 nCi

Lung counts – The lung count system located in CFA-690 is designed to identify low energy photon emitters in the lungs. The system consists of an array of 4 High-Purity Germanium N-type Planar detectors. The detectors are housed in a shielded vault constructed of 9' x 9' x 7" steel, 1/4" lead, 1/50" cadmium, and 1/50" copper. The counter is controlled and spectra are analyzed by the ABACOS Plus software package. The lung counting system is calibrated for an energy range of 14 keV to 344 KeV.

Lung Counter approximate MDAs
(2.71 cm chest wall thickness at 3600-sec. count)

Nuclide	MDA For a 4 detector arrangement (2700 mm ² crystals)
Th-234	1.4 nCi
U-235	0.11 nCi
Pu-238	31.0 nCi
Pu-239	82.0 nCi
Am-241	0.14 nCi
Ce-141	0.11 nCi
Ce-144	0.44 nCi
Eu-152	0.18 nCi
Gd-153	0.096 nCi

Thyroid counts – The thyroid counter located in CFA-690 is designed to identify radioactive Iodine deposition in the thyroid gland. The thyroid counter system consists of a 133% high-purity germanium p-type coaxial low-background detector (for I-131) and a High-Purity Germanium N-type Planar detector (for I-125) that is positioned with a 10-cm air gap between the subjects' neck and the detector. The detectors are housed in a shielded vault constructed of 9' x 9' x 7" steel, 1/4" lead, 1/50" cadmium, and 1/50" copper. The counter is controlled and spectra analyzed for I-131 or I-125 by the ABACOS Plus software package.

Nuclide	MDA For the 133% HpGe P-type Coaxial Low-background detector (I-131) and the N-type Planar detector (I-125) arranged for thyroid counting.
I-125	0.44 nCi
I-131	0.13 nCi

Urine samples – Urine bioassay samples are requested when there is the likelihood the source term consists of beta emitters (e.g. Sr/Y-90, tritium), isotopes of uranium or transuranic radionuclides. Composite 24-hour urine samples are requested representative of standard man or woman urine excretion. Urine samples are rejected if the sample is less than 500 ml. (exception – tritium analyses require single 5 ml grab samples). Urine samples are evaporated to dryness and

then ashed in a muffle furnace to destroy the organic material present in the sample. Appropriate tracers are added to the ash, which is then dissolved in acid. Chemical separations are performed to rid the matrix of interfering material and radionuclides. Alpha spectrometry is used to determine the amount of each actinide present in the sample. After at least a seven-day in-growth of the Y-90 daughter, Sr-90 is separated from its daughter and determined by counting the daughter on a low background gas proportional counter.

Urine samples being analyzed for tritium are pipetted into a liquid scintillation vial, a scintillation cocktail is added and the solution is counted on a three-channel liquid scintillation counter.

Fecal samples – Fecal bioassay samples are the preferred method when there is the likelihood the source term consists of transuranic radionuclides. As stated in ICRP-54 fecal analysis has adequate sensitivity for special monitoring for the detection of Pu-238 Pu-239/240 or Am-241 radionuclides. Fecal sample analysis based on a six month frequency has shown to have adequate sensitivity at the recording level of 100 mrem CEDE based on the analytical laboratories reported MDAs and as indicated in the time analysis tables. Neither *in vivo* or urine analysis has adequate sensitivity to detect intakes at the recording level in a routine monitoring program. Again stated in ICRP 54 “The usefulness of urine monitoring lies in the long-term assessment of cumulative intakes of transuranic radionuclides.” Bioassay programs have included extensive routine urinalysis for transuranic radionuclides where their accumulation has not been detected. Composite 24-hour fecal samples are requested representative of standard man or woman fecal voiding. Fecal samples are rejected if the sample is less than 40 grams. Fecal samples are placed in an aluminum pan and dried on a hot plate. The samples are then ashed in a muffle furnace destroying the organic materials present in the sample. Up to five grams of the ash is spiked with appropriate tracers and then dissolved in acids. The actinides are separated from the matrix material by precipitating iron phosphate. The actinides are purified using various precipitations and column chromatography. Alpha spectrometry is used to determine the amount of each actinide present in the sample.

Urine and Fecal bioassay sample approximate MDAs

Radionuclide	MDA Based on analysis of blank samples from 10/1/91 through 1/15/93
Pu-238	2.2 E-8 μ Ci/sample
Pu-239/240	2.7 E-8 μ Ci/sample
U-233/234	4.1 E-8 μ Ci/sample
U-235	3.8 E-8 μ Ci/sample
U-238	3.0 E-8 μ Ci/sample
Am-241	2.3 E-8 μ Ci/sample
Sr/Y-90	1.9 E-6 μ Ci/sample
H-3	1.4 E-6 μ Ci/milliliter

4.0 Detection and Confirmation of Intakes

For *in vivo* radiobioassays, if a gamma peak is identified above background (greater than 2.33 sigma uncertainty), then a recount is performed (after the subject has showered and put on clean coveralls). If the recount still identifies the gamma peak above background (greater than 2.33 sigma uncertainty), then detection is confirmed and a radionuclide intake is suspected.

For *in vitro* radiobioassays, if the initial measured radioactivity in the bioassay sample is greater than 2 sigma, then the sample is re-purified and recounted a second time for conformation of the initial count. If only a portion of the initial sample was used then a complete reanalysis can be performed. If the analysis remains greater than 2 sigma a preliminary assessment is performed and if the dose warrants further investigation, greater than 10 mrem CEDE, then follow-up radiobioassays are requested. For tritium analyses, counts in the first channel are compared against counts in the second and thirds channels of the liquid scintillation counter to detect a false positive analysis. For total uranium a reanalysis of second aliquot, where analysis is based on a 5 ml aliquot of urine, is performed if the measured activity is greater than 1.5 µg/l.

4.2 Investigation levels

The investigation level defined in 10 CFR 835 Implementation Guide for Internal Dosimetry Program is; "the value of the Committed Effective Dose Equivalent (CEDE) from intake(s) of a radioactive material by a worker at or above which, for regulatory purposes, is regarded as sufficiently important to justify further investigation. The DOE adopts an investigation level of 0.1 rem committed effective dose equivalent from intakes occurring in a year for general employees". The investigation level of 0.1 rem CEDE is one of several dose levels for which progressive follow-up action has been defined for the internal dosimetry program. Dose-of-Record resulting from an internal intake and final estimated dose becomes recordable at 10 mrem CEDE. This minimum recording level is the same as the Minimum Reporting Level (MRL) in the external dosimetry program.

Where bioassay measurements are conducted for the purpose of screening personnel prior to job assignments (i.e., baseline bioassays), the presence of radionuclides above natural background are investigated.

The following actions are taken by the IDC/ID when an estimated intake or preliminary dose assessment falls within the specified range:

Less than 10 mrem CEDE, (< 0.002 ALI)

1. No follow-up action required.
2. Preliminary and final dose assessments less than 10 mrem CEDE become recorded as an spreadsheet entry and not assigned as a recordable Dose-of-Record.

10 mrem to 100 mrem CEDE, (0.002 ALI < intake < 0.02 ALI)

1. Confirm results through follow-up *in vivo* and/or *in vitro* bioassays, review employees work history, air sampling data, co-worker exposure and validate sample activity reported by analytical laboratory.
2. Review the possibility that additional radionuclides may be associated with an employee's intake but are not detectable by the sampling technique used or activity could be present but less than the minimum detectable activity for the analytical laboratory.
3. Dose assessment is performed using Reference man (ICRP 23) and appropriate biokinetics models and/or assessment tools e.g., REMedy, CINDY, estimating the occupational internal dose being assigned to the employee.
4. A formal report that includes the Technical Requirements for Dose Assessment identified in the Radiological Control Manual Chapter 5 section 523 is prepared documenting the internal dose assessment. Estimated dose is assigned to the employee's internal Dose-of-Record file.
5. Prepared dose assessment report receives program ID and facility IDC concurrence, technical peer review and management approval upon being recorded as Dose-or-Record.
6. Employee is notified by letter following the finalization of any documented dose assessment as directed by the Radiological Control Manual Chapter 5 Section 521. The employee is notified of the specific radionuclides detected, the rationale that lead to the final conclusion and the estimated Committed Effective Dose Equivalent (CEDE) and the maximum Committed Dose Equivalent (CDE) to an organ.

100 mrem to 500 mrem CEDE, (0.02 ALI < intake < 0.10 ALI)

Action taken as specified in the above section.

Additional bioassay measurements are taken as deemed appropriate to better help confirm results and possible intake routes and chemical classes.

Review of air sampling and other exposure (e.g., skin contamination incidents, injury/wounds, medical history, etc.) data to determine consistency with bioassay results.

Compare worker's current internal and external dose to applicable limits.

Review radiation protection program, ALARA practices and radiological engineering controls if the exposure was unplanned.

500 mrem to 1500 mrem CEDE, (0.10 ALI < intake < 0.3 ALI)

1. Take action as specified in the 100 mrem to 500-mrem section above.
2. If additional internal exposure would substantially interfere with intake determination, temporarily limit the employee's potential for receiving additional exposure.

3. Consider use of alternate bioassay methods to confirm measurements.
4. Review occurrence for reportability

1500 mrem to 5000 mrem CEDE, (0.3 ALI < intake < 1.0 ALI)

1. Take action as specified in the 500 mrem to 1500 mrem section above.
2. Use alternate bioassay techniques to aid in evaluation of the intake, if possible.
3. Restrict further exposure to the individual
4. Use person-specific data to review suitability of assumptions, techniques and models used to assess the intake.
5. Notify medical for consideration of short-term treatment.
6. Consider dose limitation methods for the individual.

5000 mrem to 15000 mrem CEDE, (1.0 ALI < intake < 3.0)

1. Take action as specified in the 1500 mrem to 5000 mrem section above.
2. Thoroughly review the exposure event, dose assessment and potential health risk with the individual exposed.

Greater than 15000 mrem CEDE, (> 3.0 ALI)

1. Take action as specified in the 5000 mrem to 15000 mrem section above.
2. Notify medical for consideration of extended or protracted treatment.

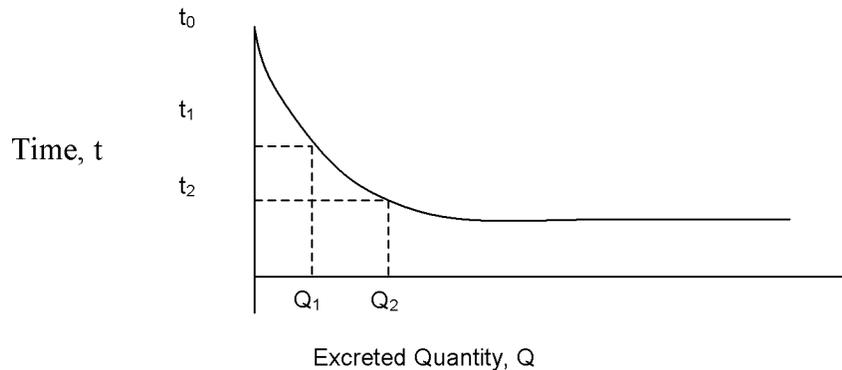
4.3 Evaluating Doses if Time Course of an Intake cannot be Established

When a positive bioassay warrants investigation, the work history of the employee is evaluated, the employee is interviewed, RWPs worked by the employee are assessed, and follow-up bioassays are initiated. From the interview and evaluation of the employees work history, a likely date when an intake could have occurred can often be determined.

Many radiation workers who frequent contamination areas are on routine bioassay programs where the bioassay frequency is such that the midpoint between the radiobioassays is assessed as a reasonable intake date. This practice is a viable choice as reference in ICRP 30 and the DOE Internal Dosimetry Technical Standard.

Deducing time of intake from bioassay data is often utilized for positive Pu-238 or Pu-239/240 analysis. Follow-up urine and fecal bioassay samples are requested when there appears to have been a single acute intake where the initial sample was found to be positive. The excreted quantity as a function of time is determined where a radionuclide excretion function is

superimposed over the sample data points to establish an intake date. CINDY software program is utilized in making this intake date determination where the activity observed in the initial and follow-up samples are fit to the function based on an assumed date of intake. When the follow-up sample activity is a negative value or when the activity excreted in the follow-up bioassay is expected to be less than the analytical detection capability, the reported MDA of the follow-up sample is used as a conservative second data point from which a worst case date of intake is estimated.



4.4 Biokinetic Models and Internal Dose Evaluation

The internal dosimetry program for the INEEL M&O Contractor will follow Regulatory and National Consensus Codes/Standards as set forth by recognized government agencies and authorized by the Department of Energy. The internal dosimetry program recognizes and follows the following internal dosimetry regulations and standards; 10CFR835 and implementation guides, DOE technical standards, International Commission on Radiological Protection, applicable Nuclear Regulatory Commission reference document, Environmental Protection Agency, Good practice manuals prepared by DOE Site Contractors, National Council on Radiation Protection and Measurements.

Biokinetic models and model parameters together with their default values having acceptance by the DOE and founded in regulation and as national consensus codes and standards are utilized for dosimetric modeling. Internal dose assessments that utilize standard models and reference man assumptions provide a reasonable estimate of the internal dose one receives from a radionuclide uptake. As stated above, alternate bioassay methods will be considered to confirm measurements when a preliminary dose assessment exceed 500 mrem CEDE. Also, if a dose estimate exceeds 1500 mrem CEDE, person-specific data will be reviewed for suitability of assumptions, techniques and models.

When the details (known or assumed) have been established, then appropriate dosimetry models will be used to assess doses from the intake. Internal dosimetry software programs (i.e., CINDY, REMedy, INDOS, etc.) are approved for use with the understanding that no one program or model can serve alone in calculating internal dose. The ICRP 30 models coded into the presently accepted software programs are very acceptable for many of the radionuclides encountered at the INEEL M&O Contractor facilities. However, there are some radionuclides which may require

the use of other available models or conditions which will require worker specific parameters to be utilized.

ICRP 30 Approach to Interpreting Internal Dose

The general equations used in ICRP 30 to calculate the committed dose equivalent (H_{50}) for organs or tissues are described in ICRP 30 Part 1. This approach toward estimating internal dose has become an industry-accepted practice. Software packages such as REMedy and Cindy implement these models. However, the software user must understand that there are limitations with packages. The first understanding by the user is that the dose calculations are based on standard models found in ICRP 23—"Reference Man, Physiological, and Metabolic Characteristics," and not individual characteristics. Second, there are other models that may be more appropriate and should be used with some radionuclides. The selection of a "most appropriate" model (in lieu of individual data) involves reviewing recommendations and screening alternatives. For plutonium, the following NCRP and ICRP guidance and recommendations should be considered:

ICRP-54 contains the following cautions and recommendations concerning the use of Durbin's exponential as well as other functions.

"...however, it should be noted that there is considerable evidence to suggest that this urinary excretion function (Durbin) substantially underestimates urinary excretion at times greater than a few hundred days after uptake...In addition, a review of the original data supports the view that plutonium excretion departs from the power function curve fit as early as 300 days after uptake (Moss and Gautier, 1985). A number of modified excretion functions have been proposed (Jones, 1985; Lawrence, 1987; Parkinson and Henley, 1981) to take account of these discrepancies..."

The NCRP, Report 84, provides the following recommendation (and cautions against the indiscriminate use of ICRP-30 for metabolic modeling) concerning the use of radionuclide excretion models:

"...In some instances, and over limited intervals, excretion is as well, or better, represented by power functions of time, as for example in the first few years after an intake of plutonium or uranium. Consideration of this aspect of internal exposure evaluation is beyond the scope of the present report, and the issue is raised only to caution against the indiscriminate use of the biokinetic models of ICRP 30 for purposes other than those for which they were developed. Interpretation of excretion data for purposes of body burden estimation should be based on models with that application primarily in mind. The models of ICRP Publication 30 were derived for the estimation of organ dose and were not extended to account for excretion."

Therefore, when a more appropriate model is known (e.g., Jones, Langham, Durbin, etc. for transuranic modeling) then the best model should be used. The Internal Dosimetry staff will provide the technical guidance for these situations. The DOE Technical Standard –1128-98 "Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities" Section 5 on Internal Dosimetry provides guidance and accepted protocol to be followed when differing biokinetic models might be used in the event of a plutonium uptake.

The following example calculation shown in Table 24 is a manual approach to estimating internal dose using ICRP models and the method used to validate the REMedy software program.

General Equation:

$$H_{50mT} = 1.6 * 10^{-10} \sum_s \sum_j U_s \sum_i SEE (T - S)_i$$

The ICRP 30 publications provide the SEE and U_s tables for most all radionuclides and their isotopes which would be encountered at the INEEL facilities. They are defined as:

U_s —The total number of nuclear transformations which occurred in the source organ or tissue over a 50 year period.

SEE—The total energy absorbed per unit mass of the target organ per transformation of the radionuclide or Specific Effective Energy.

The units on SEE in the equation for Committed Dose Equivalent (CDE) are MeV per gram per transformation. The constant 1.6E-10 is used to convert grams to kilograms plus a factor that is the number of joules per MeV. The CDE is in terms of sieverts and would normally be converted to rem or mrem.

Table 24. Example ICRP-30 Calculation.

Tissue/Source	Sr-90 (ICRP 30)		Y-90 (ICRP 30)		Reference Committed Dose Equivalent Sv/Bq ^a	Realized Committed Dose Equivalent (from example) Sv/Bq ^b	ICRP Weighting Factor
	SEE	Us	SEE	Us			
R Marrow/Trab Bone	2.2E-4	7.9E+6	4.6E-5	7.9E+6	3.36E-7	2.69E-3	0.12
Lungs/Lungs	9.3E-4	3.4E+3	2.0E-4	1.9E+4	1.11E-9	8.91E-6	0.12
Bone Surfaces/Cort. Bone	1.2E-4	1.9E+7	2.4E-5	1.9E+7	7.30E-7	5.86E-3	0.12
Bone Surfaces/Trab. Bone	1.9E-4	7.9E+6	4.1E-5	7.9E+6			
ULI Wall/ULI Contents	2.1E-3	8.6E+2	4.4E-4	5.2E+3	6.55E-10	5.26E-6	0.12
LLI Wall/LLI Contents	3.5E-3	3.1E+3	7.2E-4	9.3E+3	2.81E-9	2.25E-5	0.12

CEDE

Reference CDE determined from ICRP 30 equation

a. $reference\ CDE = 1.6E - 10 * U_s * SEE$

Realized CDE from bioassay sample activity (test case: Urine bioassay, activity 1.26 Bq/Sample or 3.4E-5 μCi/1.56E-4)

b. $realized\ CDE(Sv) = \frac{bioassay\ activity(Bq / sample)}{Intake\ retention\ fraction} * reference\ organ\ CDE(Sv / Bq)$

Example Calculations

Estimation of Committed Effective Dose Equivalent Due to an Intake of ⁶⁰Co at a Known Time:

An individual, whose routine monitoring frequency is annual, receives an additional whole body count (WBC) 48 hours after a suspected inhalation incident. The result showed 1.0 μCi of ⁶⁰Co. It is assumed that the radionuclide is Class Y and the route of intake was inhalation. Assuming reference man biokinetics and 1.0 μm AMAD particle size, the Intake Retention Fraction from NUREG/CR-4884 for 2 days following inhalation is used to calculate the following intake:

$$\text{Intake} = \frac{\text{bioassay compartment content (uCi)}}{\text{IRF}}$$

$$\text{Intake} = ({}^{60}\text{Co}) = \frac{1.0 \text{ uCi}}{0.424} = 2.36 \text{ uCi}$$

*Convert uCi Intake into Bq (2.36 uCi * 3.7E4 Bq/uCi) = 8.8E4 Bq*

In ICRP-30 the Committed Dose Equivalent in target organs or tissues per intake of unit activity is identified in Sv/Bq. For Co-60 class Y inhalation, the lungs are the only source organ or tissue identified and are shown with 3.4E-7 Sv/Bq as the unit activity. Therefore, the committed dose equivalent (CDE) is calculated as follows:

$$8.8E4 \text{ Bq} * 3.4E-7 \text{ Sv/Bq} = 3.0E-2 \text{ Sv}$$

$$3.0E-2 \text{ Sv} * 1.0E2 \text{ rem/Sv} = 3.0 \text{ rem (CDE)}$$

The weighting factor recommended in ICRP-26 for stochastic risks identifies the lung weighting factor as 0.12. The committed effective dose equivalent (CEDE) is the sum of the entire individual organ CDEs multiplied by the organ weighting factor. Therefore, the CEDE is calculated as follows:

$$3.0 \text{ mrem} * 0.12 = 360 \text{ mrem (CEDE)}$$

Another approach is to use the Annual Limit of Intake (ALI) to estimate CEDE. Since the stochastic dose is limiting for the radionuclide Co-60, ICRP 54 shows an ALI of 1.0E6 Bq for a class Y, Co-60 inhalation. 0.05 Sv (5 rem) per ALI could be used to estimate dose as follows:

$$\text{Committed Effective Dose Equivalent} = 0.05 \text{ Sv/ALI} \frac{\text{intake (Bq)}}{\text{Bq/ALI}}$$

$$\text{For Co-60 CEDE} = 0.05 \text{ Sv/ALI} \frac{8.8E4 \text{ Bq}}{1.0E6 \text{ Bq/ALI}} = 4.4E-3 \text{ Sv}$$

$$4.4E-3 \text{ Sv} * 100.0 \text{ rem/Sv} * 1000.0 \text{ mrem/rem} = 440.0 \text{ mrem (CEDE)}$$

This same calculation when performed using the REMedy software program calculated a Committed Effective Dose Equivalent at 519.0 mrem. Some differences in the estimated dose are observed depending on the method being employed. The reason for some of the discrepancy can be attributed to number rounding that has taken place in some of the reference tables found in regulatory guides e.g., ICRPs, NUREG-4884, etc. As well, the methods themselves have some intrinsic differences with the approach toward dose calculation. In addition, with the REMedy program a standard library of organs is used with additional dose being assigned to these lesser critical organs in some instances. A portion of the additional organ dose identified with the REMedy calculation is founded in what is termed remainder organs. With this example calculation, ICRP-30 only identifies the lungs as the primary deposition site. However, small amounts of dose are contributing to the overall estimated CEDE from these lesser organs. For example, REMedy calculates an endosteal dose of about 20% of the lung dose for DU, and includes this, with its factor, to the effective dose.

In doing a dose assessment, all assumptions, software programs, models, regulatory guides followed, investigative data gathered, etc. are clearly stated in the assessment. Information presented in the assessment will be such that personnel familiar with internal dosimetry practices may reconstruct the values of intake and dose at a later date. Any departures from standard models or parameters will be clearly noted and justified. All computer programs used in the assessment are referenced.

4.5 Summary Criteria of Tritium

This section provides information on the sources and biokinetics of tritium and summarizes the technical basis used for the internal dosimetry of tritium (^3H) at INEEL facilities. This section is intended to provide the approach to be used for routinely encountered tritium exposures.

Tritium exists as part of the natural background of environmental radiation (National Council on Radiation Protection and Measurements [NCRP] 1979a). It can be assumed that the tritium concentration in the body water of non-occupational exposed persons should be reasonably close to that of their drinking water. The Environmental Protection Agency (EPA) has reported that background tritium concentrations in U.S. drinking water range from 100 to 400 pCi/L. In addition, the EPA has promulgated a limit for tritium in drinking water of 20 nCi/L (44 dpm/ml), based on 4 mrem/yr (EPA 1976). The CFA drinking water, available from two wells, has the highest concentration of tritium at INEEL. In July of 1994 these two wells measured 12 and 14 nCi/l tritium activity. The RWMC well measured an order of magnitude less than CFA with TRA and INTEC showing non-detectable tritium activity (results < background + three sigma). Source for the above data was the United States Geological Survey data files at the INEEL.

It is also worth noting that tritium has been widely distributed in the public domain as a source of luminosity for various "glow-in-the-dark" applications, such as the faces of watches, clocks, instruments, and exit signs. Breakage or other loss of containment in such devices could result in tritium levels in urine being substantially above background without occupational exposure.

4.5.1 Chemical Form

Tritium water (HTO) and tritium gas (HT) are the two forms of tritium likely to be encountered at the INEEL M&O Contractor facilities. Tritium gas, upon inhalation, is only slightly incorporated into the body, and it is rapidly removed (exhaled). In contrast, tritium water vapor is readily taken up and retained by the body.

Dosimetry for exposure to other forms of tritium i.e., elemental or organic forms, requires further knowledge of the nature of the material and the circumstances of exposure. The overall contribution from organically bound tritium has been found to be relatively small: less than 5% for acute exposures and 10% for chronic exposures, (DOE Handbook—Primer on Tritium Radiological Fundamentals [Draft DOE-NE-73] 1993). These other forms of tritium are not in widespread use at the INEEL, and therefore are not addressed in this technical basis.

4.5.2 Metabolism of Gaseous Tritium

Following a brief exposure to tritium gas (HT), small amounts of the gas are dissolved in the bloodstream. The dissolved gas is circulated in the bloodstream for about two minutes. Most of the gas is exhaled along with the gaseous waste products, carbon dioxide and normal water vapor. If the exposure persists, the gas is found in the extra and intercellular body fluids. A small percentage of the gaseous tritium is converted to the oxide form (HTO), most likely in the gastrointestinal tract. For gaseous tritium exposures, there are 2 doses: (1) a lung dose from the tritium in the air in the lung, and (2) a whole body dose from the tritium gas that has been converted to water.

Skin absorption of gaseous tritium has been found to be negligible compared to inhalation. Small amounts of tritium can enter skin through contact with contaminated metal surfaces, which results in elevated organically bound tritium in tissues and urine.

4.5.3 Metabolism of Tritiated Water

The uptake of airborne HTO can be extremely efficient: up to 99% of inhaled HTO is taken into the body within seconds. Ingested liquid HTO is almost completely absorbed by the gastrointestinal tract and quickly appears in the blood. Skin absorption is also important, especially during hot weather, because of the normal movement of water through the skin. For skin temperatures in the range of 30 to 40°C, the absorption of HTO is about 50% of that for HTO by inhalation. Independent of the absorption mechanism, absorbed HTO will be uniformly distributed in all biological fluids within one to two hours. Tritiated water is the most dosimetrically important form of tritium at the INEEL.

Tritiated water or vapor is assumed to be instantaneously and uniformly mixed with body water immediately following intake. From a practical standpoint the process is quite rapid and an approximate equilibrium condition will probably be reached by the time a sample can be collected.

The average biological half-life of tritium is 10 days, but it can vary naturally by 50% or more depending on the body-water turnover rate. For both HTO and HT exposures, a bioassay program that samples body water for HTO is an essential element of a good personnel monitoring program.

4.5.4 Sampling Schedule and Technique

Workers who may be or who have been exposed to tritium are normally required to submit urine samples for bioassay on a periodic basis. The sampling period is daily to biweekly depending on the potential for significant exposure.

Following an incident or a work assignment with a higher potential for exposure, a special urine sample is usually required for each person involved. The preferred method is to wait 1 to 2 hours for equilibrium to be established. The bladder is then voided. A sample submitted soon afterwards should be reasonably representative of the body water concentration. A sample collected before equilibrium is established will not be representative because of dilution in the bladder, or because of initial concentration in the blood. However, any early sample may still be useful as a sign of the potential seriousness of the exposure.

4.5.5 Metabolic Model

The metabolic model used for tritium water (HTO) is described in ICRP 30 (1979). Tritium water is assumed to be uniformly distributed among all soft tissues at any time following intake. Its retention function $R(t)$ is described as a single exponential with an effective clearance half-time of 10 days. Thus, the fraction of tritium taken into the body as tritiated water, which is retained in the body at time t days later, is given by:

$$R(t) = \exp^{(-0.693 / 10)t} = \exp^{-0.0693t} \quad (4.1)$$

This retention function has been well established and is considered appropriate for exposure to tritiated water. It can be expected that the retention of tritiated water in individuals will vary from this. If a dose looks to be significant and if sufficient data is available to establish an alternate model for an individual worker's exposure, the alternate model could be used. In addition to body water, ICRP 30 acknowledges the existence of two organically bound tritium components. However, the ICRP concludes that these could be ignored for radiation protection purposes, and Johnson (1982) estimates that these components would add approximately 10% to the committed dose equivalent. Unless worker data specifically indicate the existence of significantly long-term components, the Internal Dosimetry Program will follow the ICRP recommendation of single compartment retention.

The fraction of the initial uptake eliminated on any given day after the intake is derived by differentiating the retention function. Thus, the elimination function based on Equation (4.1) is given by:

$$e(t) = 0.0693 \exp^{-0.0693t} \quad (4.2)$$

where $e(t)$ is the fraction of uptake excreted on day t and t is the elapsed time (days) post intake.

It must be recognized that equation (4.2) describes the total tritium elimination from the body. This elimination occurs through a number of pathways, notably urine, feces, exhalation and sweat. The ICRP 23 (1974) Reference man water balance indicates approximately 47% of the water loss of the body occurs via urine. In application of the excretion function to uptake estimation based on urine sampling, the fraction of an initial uptake excreted in urine on day (t) post intake is then calculated to be:

$$e_u(t) = 0.47 e(t) = 0.033 \exp^{-0.0693t} \quad (4.3)$$

For airborne exposures, ICRP 30 suggests that the uptake rate for tritiated water by inhalation and skin absorption pathways can be approximated by:

$$U^1 = 1.8E + 6 * C(\text{air}) \quad (4.4)$$

where U^1 is the uptake rate $\mu\text{Ci}/\text{h}$, and $C(\text{air})$ is the air concentration in $\mu\text{Ci}/\text{cc}$ (Technical Basis for Internal dosimetry at Hanford [July 1991]).

4.5.6 Tritium Gas

If pure tritium gas is inhaled, approximately $5E-5$ of the intake is converted to tritiated water. A pure HT exposure can be thought of as a combination of a lung exposure from the HT and a whole body exposure from HTO. The HTO comes from the conversion of HT dissolved in the blood. Since the effective dose equivalents from the lung and whole body exposure are about equal, the total effective dose can be conservatively obtained by multiplying the HTO whole body dose by 2. (DOE Handbook – Primer on Tritium Radiological Fundamentals {Draft DOE-NE-73} 1993).

4.6 Summary Criteria of Uranium

The sensitivity of *in vitro* sampling as a uranium bioassay tool is limited by the presence of environmental levels of uranium, which is subject to some uncertainty in interpretation. In ICRP 30 (1979) the average daily ingestion intake of natural uranium in food and water is estimated to be $1.9 \mu\text{g}$. Based on studies performed at INTEC, the concentration of uranium found in potable water is approximately $3 \mu\text{g}$ per liter of water. This background quantity of uranium is found in subsurface water that is consumed as drinking water by workers. All individuals are exposed to natural uranium found in foods and in the soils/rocks found in their respective communities. These exposures constitute background exposures. An analysis of aquifer water was performed by INTEC for uranium isotopes which identified the following concentrations: U-234— $1.94E-6 \mu\text{Ci}/\text{L}$, or $3.11E-4 \mu\text{g}/\text{L}$; U-235— $7.0E-8 \mu\text{Ci}/\text{L}$, or $3.24E-2 \mu\text{g}/\text{L}$; U-238— $9.54E-7 \mu\text{Ci}/\text{L}$, or $2.84 \mu\text{g}/\text{L}$. This gives a U-234/U-238 activity ratio of 2.0.

Uranium is also found as a by-product in phosphate deposits in southeastern Idaho. INTEC also looked at this by comparing uranium bioassay fecal sample results from employees coming from communities where diverse differences in levels of uranium background are known to exist. The

study was not able to discern any relationship between uranium bioassay results and the residency of the people. Therefore, the contribution of "background" uranium intake from non-nuclear environments is very common in bioassay samples and should be subtracted from the total uranium isotopic intake in order to establish an occupation intake.

INTEC established a technique for which uranium bioassay results are evaluated as being something other than the detection of non-occupational uranium uptakes. Studies performed by INTEC established a U-234/U-238 ratio of 2.0 for evaluating urine bioassay samples and a ratio of 1.7 to evaluate fecal bioassay samples. The basis (See WINCO Internal Dose Technical Basis document [Dec 1990]) can be shared by facilities as a screening tool to evaluate uranium bioassay results. The presence of uranium at other facilities can vary from small quantities found as contamination at some D&D sites, to somewhat larger quantities in contaminated waste at RWMC, with the largest amounts stored at (1) TRA in fuel pools, hot cells and as fuel plates and (2) at INTEC in fuel storage pools, or throughout the uranium extraction processes whenever operational.

The Specific Manufacturing Capability (SMC) project has experienced both low-level chronic and acute uranium exposures. These internal exposures are exclusively from depleted uranium with only trace amounts of U-235 left in the depleted uranium. The material at SMC is not at full secular equilibrium with U-238, with virtually no uranium material present beyond U-234. The facility specific technical basis accounts for the non-equilibrium ratios of uranium isotopes in assessing doses from intakes of depleted uranium. SMC has also determined that the normal background in a urine sample is about 0.16 µg/l, which is subtracted from all sample results before dose calculation.

Future D&D and waste handling projects could have a higher probability of workers encountering uranium isotopes with a much greater concern over worker intakes. For work involving uranium where an intake potential exists, baseline samples submitted by the workers with subsequent follow-up samples would provide the necessary results to assess any occupational intakes.

4.7 DAC-hour Calculations to Assess Exposures

Final dose assessments will be quantified based on bioassay results and not air monitoring data whenever possible. The source terms at the INEEL facilities are inherently distinguishable via *in vivo* and *in vitro* bioassays. However, because of time lag necessary to performing specific radionuclide analysis on a bioassay sample, air-monitoring data may be the initial information available relevant to a significant intake. Decisions involving medical intervention may be necessary based upon air-monitoring data. Where available air-monitoring data may be utilized during initial and preliminary dose assessments, it will be evaluated against the bioassay assessment once the analysis results become available. There may be conditions where air monitoring data may be more representative of a situation and/or a specific radionuclide can be better assessed via an air sample. These conditions, if one ever occurs, will be handled on a case by case basis. As a general approach when there is an air sample and a DAC-hours exposure estimate available, an estimated dose could be projected by assuming 2.5 mrem per DAC-hour.

5.0 Internal Dose Management

Radiological Control Information Management System (RCIMS) is managed by the INEEL Radiological Controls organization. Internal exposure histories are maintained in the RCIMS database as Committed Effective Dose Equivalent and as Committed Dose Equivalent to any organ or tissue of concern in accordance with regulatory requirements. The software program provides the interface between the internal and external dose and sums the doses with respect to annual and lifetime dose totals. This database program maintains the radiation exposure records in accordance with 10CFR835.702. RCIMS is utilized as an administrative tool in helping keep exposures ALARA and managing an individuals' radiation dose within regulatory limits. The reporting of dose history information in accordance with 10CFR835.801 is managed and interfaced with the RCIMS database program.

5.1 Medical Intervention

Engineering Design File titled "Established levels of Radionuclide Intake for Consideration of Medical Intervention" INEEL-003, addresses action levels for administrative response to intakes of radionuclides by workers. The decision to administer treatment and the treatment protocol are solely the responsibility of the physician in charge. The RDR staff provides advice on the dose consequences of performing or not performing a treatment. Time is of most importance in determining if an accident has resulted in exposure to employee(s) that warrants notification of medical for possible treatment. Therefore, notification levels based on work-place indicators (nose swabs, skin contamination, and preliminary results of any air monitoring) are recommended in the TBD for consideration of possible early medical intervention.

5.2 Position Statement on Historical Internal Dose Calculations

Reference to memo (ANT-17-95) about recalculating old internal doses

All internal doses which have been reported to the INEEL Radiation Dosimetry System (IRDS), have now been replaced by RCIMS, have been entered into the computerized database and are carried as committed effective dose equivalent when an individual's dose history is prepared. When an internal dose assessment is reviewed, which happens for a variety of incidental reasons, the review includes an evaluation with more contemporary models and methods and with all the available bioassay results. If a significant improvement is achieved, the assessment is revised accordingly. However, there will be no general attempt to update personnel files with dose assessments for exposures that were below the reporting guidelines applicable at the date of the bioassay results, for which no dose assessment was made or submitted to the IRDS. Such an effort would require substantial resources beyond those currently available to Radiation Dosimetry and Records and would not significantly improve the quality of the database in that most of the internal exposures fitting these criteria will never come into question or be the subject of an inquiry. On the other hand, if we have occasion to review a bioassay history for a specific employee and find bioassay results above today's reporting guide, a re-assessment will be made if it improves the original dose assessment and resources are readily available to complete the improvement.

5.3 Limiting Exposure to Declared Pregnant Workers, Minors, and Students

The INEEL administratively limits radiation exposure to declared pregnant workers and minors to limits specified in table 2.1 of the INEEL Radiological Controls Manual. The Radiological Control Manual states: "that after a female worker voluntarily notifies her employer in writing that she is pregnant, for the purposes of fetal/embryo dose protection, she is considered a declared pregnant worker. MCP-145 provides procedural guidance for radiation protection of the embryo/fetus. As well, visitors to the INEEL site which includes minors under the age of 18 who occasional are allowed to tour some facilities at the site, are limited to an annual sum of internal and external radiation sources of 100 mrem.

6.0 Quality Assurance

Radiochemical laboratories and *In Vivo* counting facilities whose measurements are used by internal dosimetry programs each have quality assurance programs, documented regular equipment calibration programs, use calibration sources traceable to the National Institute of Standards and Testing, and have written procedures that are referenced by internal dosimetry programs. The two radiobioassay laboratories participate in inter-comparison studies thus maintaining quality analyses programs through testing against known standards and laboratory comparisons.

The internal dosimetry program receives periodic assessments by the site radiation protection organization. The radiation protection organization review technical basis, instrumentation, analytical methods, qualifications of personnel, quality assurance program elements, and any other elements of the program necessary to insure that the program maintains the capability to provide a quality radiation protection service to workers.

The Radiobioassay Laboratories have participated in the DOELAP Accreditation Program and have each received their laboratory accreditation. They will again be assessed and re-accredited in accordance with 10CFR835 and on a 3-year rotation frequency per ANSI 13.30.

7.0 References

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10. Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities, Section 5 Internal Dosimetry, DOE-STD-1128-98
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