DOE–EM/GJ3003

Office of Environmental Management – Grand Junction

UMTRA Project

Moab UMTRA Project
Health Physics Plan

Revision 3

July 2015

U.S. Department of Energy

Office of Environmental Management

Prepared by the Technical Assistance Contractor under contract number DE-EM0002067
and the Remedial Action Contractor Contractor under contract number DE-DT0002936
for the U.S. Department of Energy Office of Environmental Management, Grand Junction, Colorado.

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Revision History

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1	November 2011	Minor changes and clarifications made following biannual review.
2	November 2013	Addition of Addendum A, "Radiological Evaluation, Decontamination, and Final Unrestricted Release Survey of the Fernald Trackmobile."
3	July 2015	Revision update includes deletion of Addendum A, "Radiological Evaluation, Decontamination, and Final Unrestricted Release Survey of the Fernald Trackmobile."

Contents

Section Page
Acronyms and Abbreviations v
1.0 Introduction 1
2.0 General Requirements 2
2.1 ALARA 2
2.2 Radiological Work Permit Process 2
2.3 Radiological Monitoring 3
3.0 Existing Radiological Conditions 3
3.1 General Conditions 3
3.2 Radionuclide Constituents 3
3.3 Dosimetry Parameters of Select RRM Radionuclides 6
3.4 Radon and Thoron 9
3.4.1 General Information and Terminology 9
3.4.2 Current Regulations and Units of Measurement 10
4.0 RWPs and RCT Job Coverage 10
5.0 General Area Air Monitoring 11
5.1 Airborne Particulate Isotopes of Concern 11
5.2 Radon 12
5.3 Factors and Development of a Modified Air Particulate DAC 13
5.3.1 Factors Associated with Air Particulate Sampling 13
5.3.2 Modified DAC Equation 13
5.3.3 Determining the Components of the LLGA 14
6.0 Individual Monitoring 15
6.1 Radon Dosimetry 15
7.0 Internal Dose Assessments 15
8.0 Exposure Control and Personnel Monitoring 17
9.0 Contamination Control 18
10.0 Personnel Responsibilities 18
10.1 Operations/Site Manager 18
10.2 RCM 18
10.3 RCTs and Supervisors 19
11.0 Radiological Incidents and Reporting 19
12.0 Personnel Entry and Exit Protocols 19
12.1 Entering and Exiting Controlled Areas 19
12.2 Entering the Radiological Area Access Control Point 19
12.3 Exiting the Radiological Area Access Control Point 20
13.0 PPE and Anti-C Clothing 21
14.0 Posting and Labeling 21
15.0 Receipt and Control of Radioactive Material and Sources 23
15.1 RAM Control 23
15.2 Source Control 23
16.0 References 23

Section Page

Figures

Figure 1. U-238 Decay Chain....4
Figure 2. U-235 Decay Chain....5
Figure 3. Th-232 Decay Chain....6

Tables

Table 1. Dosimetry and Monitoring Information Summary....7
Table 2. Dosimetry Information for Select Radionuclides....8
Table 3. Radon and Thoron Terminology....9
Table 4. Radium and Thoron Limits, Descriptions, and Notes....10
Table 5. Radionuclide Contribution to LLGA at 7 Days Post-sample Collection....14
Table 6. Fixed + Removable Radioisotope Limits....20

Attachment

Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques

Acronyms and Abbreviations

Ac	actinium
ACL	administrative control level
ALARA	as low as reasonable achievable
ARA	airborne radioactivity area
Bi	bismuth
CA	Contamination Area
CED	committed effective dose
CFR	Code of Federal Regulations
DAC	derived air concentration
DOE	U.S. Department of Energy
DOE G	DOE Guide
DOE STD	DOE Standard
F	equilibrium factor
HPP	Health Physics Plan
hr	hour
ICRP	International Commission on Radiological Protection
LLGA	long-lived gross alpha
MeV	megaelectron volts
mrem	millirem
Pa	protactinium
Pb	lead
PCM	personnel contamination monitor
Po	polonium
PPE	personal protective equipment
Ra	radium
RAC	Remedial Action Contractor
RAM	radioactive material
RAP	Remedial Action Plan
RCM	Radiological Control Manager
RCT	Radiological Control Technician
RMA	radiological materials area
Rn	radon
RRM	residual radioactive material
RWP	radiological work permit
Th	thorium
TLD	thermoluminescent dosimeter
TQT	Task Qualified Technician
U	uranium
UMTRA	Uranium Mill Tailings Remedial Action
WL	working level

1.0 Introduction

The Health Physics Plan (HPP) describes the radiological controls specifically planned for the Moab Uranium Mill Tailings Remedial Action (UMTRA) Project located in Moab and Crescent Junction, Utah. This Plan also addresses the technical basis related to characterization of radiological hazards and controls for the Project. Detailed information can be found in Attachment 1 of this Plan. For detailed information related to terminology, please refer to Appendix A of the *Moab UMTRA Project Internal Dosimetry Technical Basis Manual* (DOE-EM/GJRAC1913).

The HPP documents the requirements of the U.S. Department of Energy (DOE) and the Remedial Action Contractor (RAC) that apply to radiological protection on the Project. Reference to the Project in this document refers to both the Moab and Crescent Junction sites. The HPP documents the Radiological Protection Program elements and services provided to the Project by the RAC and its subcontractors. The principal requirements for radiological protection at DOE facilities are specified in Title 10 Code of Federal Regulations Part 835 (10 CFR 835), "Occupational Radiation Protection." The radiological protection requirements defined by 10 CFR 835 that are implemented at the Project are specified in the *Moab UMTRA Project Radiation Protection Program* (DOE-EM/GJ610).

The HPP identifies radiological hazards and controls specific to maintenance and operations of the Project and satisfies requirements related to occupational as low as reasonably achievable (ALARA) planning and review. Materials and/or equipment that have been received from another site that contain a different isotopic mix from that of the Project must undergo a comprehensive technical review. If there are inconsistencies in the HPP and the new isotopic mix, an addendum to the Plan must be generated. The addendum must clearly describe all differences associated with new isotopic source terms and address all the necessary radiological controls to ensure compliance with 10 CFR 835. This document will be revised as the scope of the Project matures.

The following elements are addressed herein:

- Project operations
- Radiological hazards associated with site maintenance
- Radiological hazards associated with bulk waste retrieval from the pile and bulk waste transfer to load-out containers
- Radiological hazards associated with packaging and staging waste material before and/or during transfer to the disposal site
- Engineering controls, administrative controls, and personal protective equipment (PPE) or anti-C clothing implemented to mitigate radiological hazards
- Estimated radiological dose rates and personnel exposure
- Radiological Control Technician (RCT) coverage requirements
- ALARA goals

2.0 General Requirements

2.1 ALARA

Radiological requirements for maintenance and operations ensure all functions and activities are performed in such a way as to keep internal and external exposure to ionizing radiation ALARA. The ALARA philosophy requires any exposure to ionizing radiation by general employees or the public to be minimized to the extent that social, technical, economic, practical, and public policy considerations allow.

The Project is committed to keeping exposure ALARA through engineering (design), management (administrative controls), and supervision (procedures). This principle is implemented by the following six key elements:

1. Reducing time spent within radiological areas
2. Reducing exposure to the source(s) of radioactivity
3. Increasing the distance from source(s) of radioactivity
4. Providing containment of and shielding from sources of radioactivity
5. Minimizing internal exposures through monitoring and use of PPE
6. Reducing labor requirements for operations in radiological areas

These six key elements are weighed against economic factors, technical feasibility, practicality, public policy, and social needs to implement the best design and operational parameters.

Three approaches are incorporated in design, construction, and operations:

1. Operational layouts (designs) and exposure causing activities are systematically evaluated (with radiological and other safety considerations as the highest priorities) to keep internal and external exposures to individuals and contamination releases to the environment ALARA.
2. Personnel are trained in ALARA principles and practices. Additionally, personnel shall adhere to radiological control requirements during operations, maintenance, and support activities to minimize internal and external radiation exposures.
3. Personnel and facilities at the Project are monitored for radiation hazards, including internal and external exposure and contamination levels. This monitoring is documented to verify exposures are ALARA.

2.2 Radiological Work Permit Process

The RAC controls radiological work through a radiological work permit (RWP) process. An RWP is used to designate the specific radiological controls, precautions, surveillance, and/or instructions to personnel. Training requirements, PPE, exposure limitations, dosimeter requirements, steps to minimize the spread of contamination, steps to limit radiation exposure to adjacent personnel, and provisions for augmented monitoring and surveillance are all specified by Radiological Control on RWPs. In addition, RWPs provide a means to trend job exposures by inclusion of an entry log and dose record sheet. Employees performing radiological work are required to read, understand, sign, and abide by the requirements prescribed on the RWP.

2.3 Radiological Monitoring

Radiological monitoring is performed at the Project to assess changes in radiological conditions, assess airborne concentrations of radon (Rn) and radioparticulates, prevent the spread of radioactive contamination, and limit personnel exposure.

Radiological monitoring for the Project will be performed by the following:

- Personnel contamination monitoring
- Area radiation monitoring
- Contamination monitoring
- Air sampling (boundary, general area, and breathing zone)

3.0 Existing Radiological Conditions

3.1 General Conditions

Significant quantities of by-product material called residual radioactive material (RRM) are produced during milling operations of uranium (U) ore. This material is generally collected in a slurry and pumped to tailings ponds. These ponds are dewatered via evaporation, resulting in mill tailings piles. The Moab pile was subsequently covered by earth from the surrounding site. Because the radioactive constituents in the tailings mix are greatly diluted by the non-radioactive portion of the ore, the specific activity of RRM is very low, generally on the order of nanocuries per gram or less. The contaminants in the soil making up the cover and surrounding the pile are even less concentrated.

3.2 Radionuclide Constituents

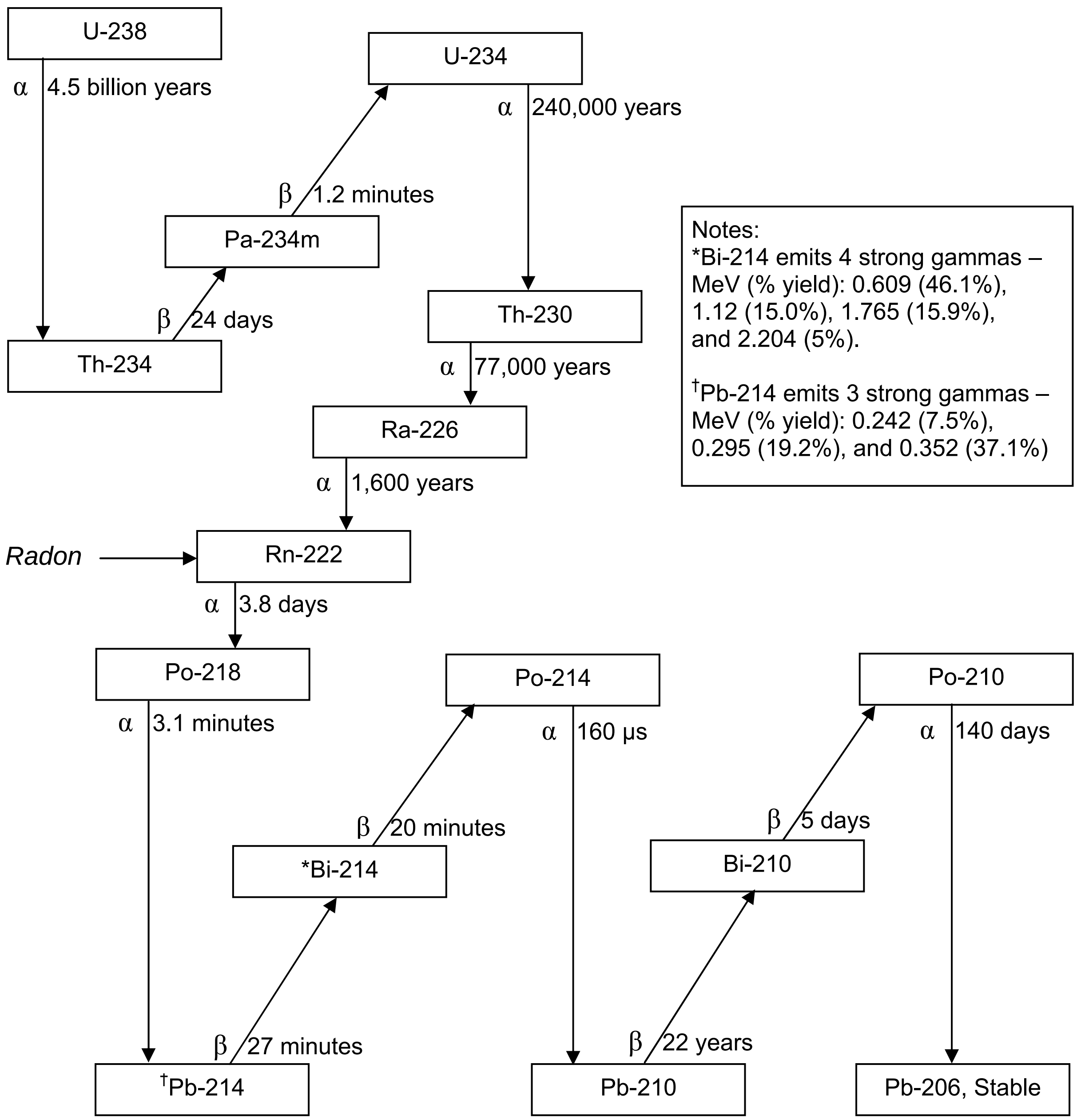
The RRM consists of radioactively inert crushed rock, water, residual milling chemicals, residual uranium, and small quantities of thorium (Th)-230, Th-232, radium (Ra)-226, and their decay progeny. The presence of Th-232 is dependent on the ore body mined to produce the uranium ore and may be found in varying ratios throughout the pile (as multiple ore bodies may have been used to supply the mill). Due to the low specific activity of mill tailings and the large particle size of the dust (5 to 10 micron activity mean aerodynamic diameter), surface contamination in tailings handling areas do not present an acute exposure scenario, although standard precautions should be taken to keep the materials from becoming airborne. In mill tailings, approximately 92 percent or more of the activity due to the uranium isotopes has been removed in the milling process. Except for U-235, the actinium (Ac) decay chain radionuclides are assumed to be in equilibrium with each other and exist at 4.7 percent of the U-238 decay chain activity.

Uranium, radium, and thorium occur in three natural decay series, headed by U-238, U-235, and Th-232, respectively. In nature, the radionuclides in these three series are approximately in a state of secular equilibrium, in which the activities of all radionuclides within each series are nearly equal. The radionuclides of these three decay series are shown in Figures 1, 2, and 3, along with the primary modes of decay for each. In mill tailings, the top of the decay chain has been removed, leaving Th-230, Ra-226, and the decay progeny of radium (mainly radon) as the primary radionuclides of concern.

As shown in Figure 1, most of the uranium has been removed in the milling process. The two long-lived radionuclides (Th-230 and Ra-226) may not be co-located in equal activity concentrations due to various chemical extraction and sorting processes that have occurred. Lead (Pb)-210 should be reasonably close to full equilibrium with Ra-226.

A fraction of the Ra-226 progeny is liberated via the airborne release of radon as shown below.

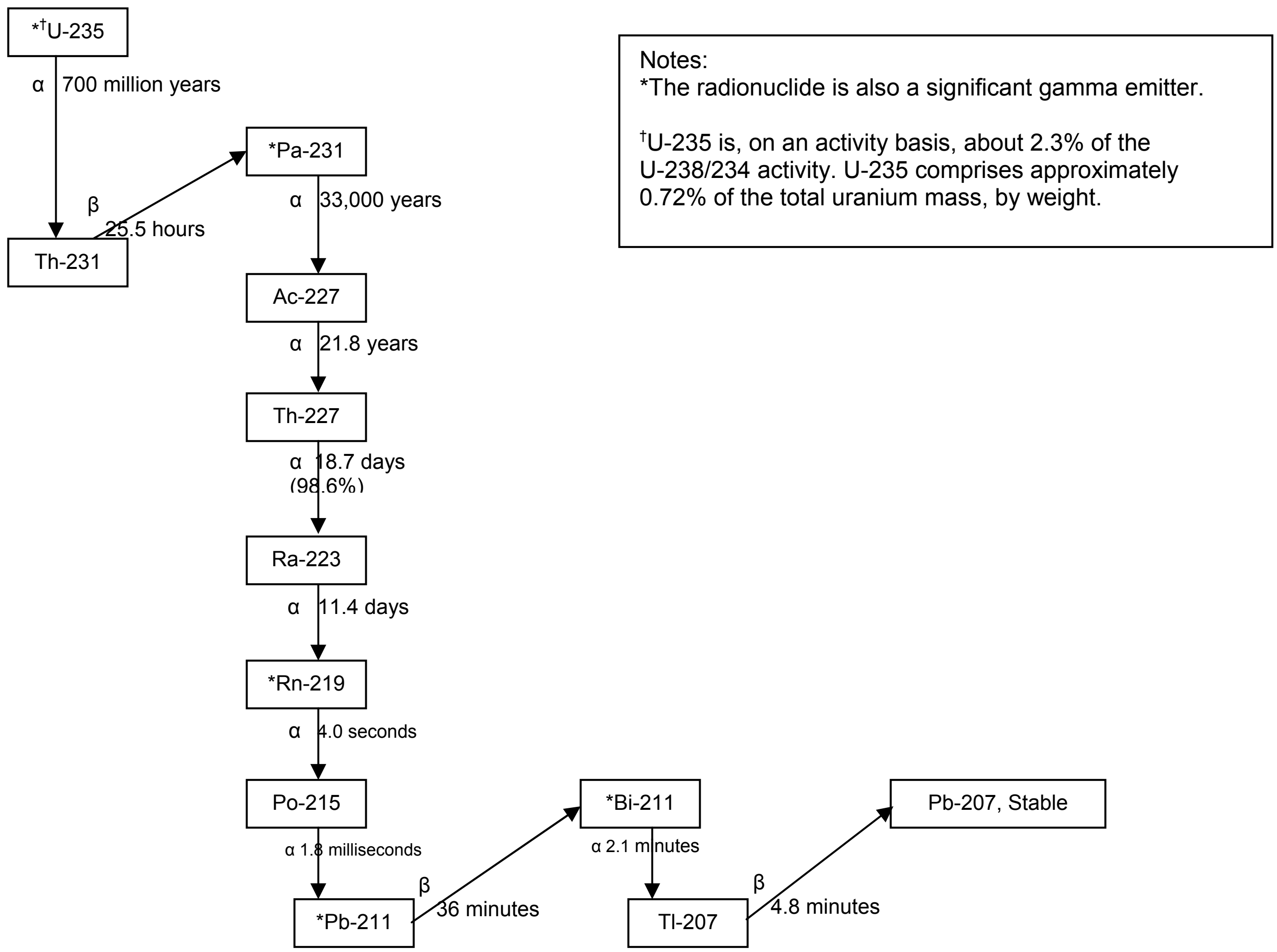
- Bismuth (Bi)-214 has four gammas at 0.609, 1.12, 1.765, and 2.204 megaelectron volts (MeV).
- Pb-214 has three gammas at 0.242, 0.295, and 0.352 MeV.



α = alpha; β = beta; Pa = protactinium; Po = polonium; µs = microseconds

Figure 1. U-238 Decay Chain

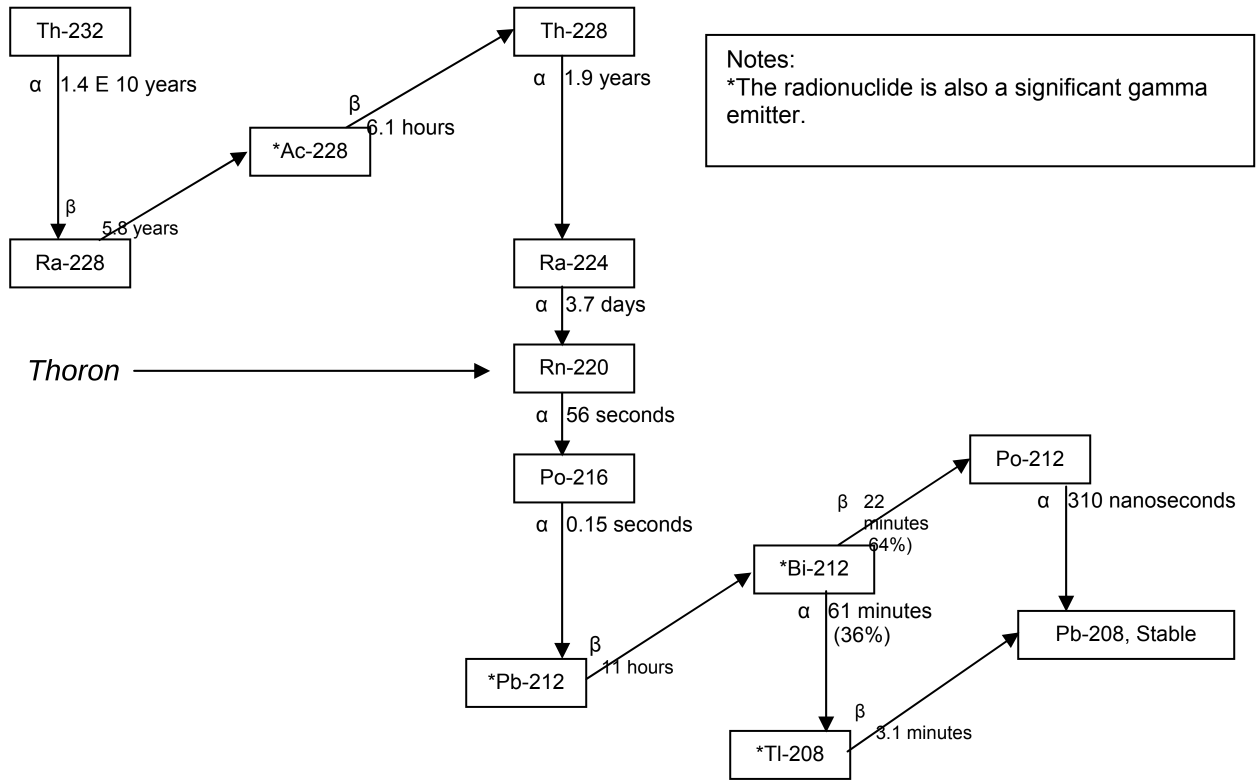
The U-235 shown in Figure 2 is a very small fraction of the total uranium found in nature; it makes up about 0.72 percent, by weight of the total uranium. On an activity basis, U-235 represents about 4.7 percent of the U-238 activity.



α = alpha; β = beta; Pa = protactinium; Po = polonium; Tl = thalium; y = years

Figure 2. U-235 Decay Chain

Th-232 is not a decay product of uranium; it is the lead radionuclide of its own decay chain, depicted in Figure 3. Additionally, it is commonly not a principle element associated with uranium ore bodies, and thus it may or may not be measured above background levels where uranium is found. On the other hand, Th-230 is a decay progeny of U-238 and should naturally be found where uranium is found.



α = alpha; β = beta; Pb = lead; Po = polonium; Tl = thalium

Figure 3. Th-232 Decay Chain

3.3 Dosimetry Parameters of Select RRM Radionuclides

Tables 1 and 2 provide a listing of summary information from Attachment 1, the internal and external technical approach position document, including the *Moab UMTRA Project Remedial Action Plan* (RAP) (DOE-EM/GJ1547) and the International Commission on Radiological Protection (ICRP) Publication 68, "Dose Coefficients for Intakes of Radionuclides by Workers." The derived air concentration (DAC) values provided in 10 CFR 835 are also provided in Table 2 for convenience. This information is pertinent to the design and implementation of the present work place and individual monitoring program used by the RAC.

Table 1. Dosimetry and Monitoring Information Summary

Item/Question	Response/Comment	Source
Radionuclides of concern for internal dosimetry due to inhaled particulates	Primary: Pa-231, Ac-227, Th-230, Ra-226, Pb-210, Po-210 Secondary: Th-227, Ra-223, Bi-214, Bi-210, Th-232, Th-228, Po-214 All others contribute <0.1% on an individual basis to the accumulated dose	RAP HPP
Primary exposure mode for inhaled particulates	Deterministic (organ) dose accumulates at a rate of 10 to 25 times the rate of the non-deterministic (organ effective) whole-body dose. A rate of 10 or more indicates the deterministic (organ specific) dose is most limiting.	HPP
Dose contribution by radionuclide to the committed dose equivalent (maximally exposed organ dose) from tailings	U Isotopes – <1% Pa-231 ~ 15.6% (BS) Ac-227 ~ 67.4% (BS) Th-230 ~ 11.3% (BS) Ra-226 ~ 1.4% (LNG – non-deterministic) Pb-210 ~ 2.9% (BS) Po-210 ~ 1.4% (LNG – non-deterministic) All other radionuclides account for about 0.3%. (These technical assumptions should be confirmed as soil and air data are collected.)	RAP HPP
Dose contribution by radionuclide to the CED (whole-body organ dose) from tailings	U Isotopes – < 1% Pa-231 ~ 8.7% Ac-227 ~ 61.5% Th-227 ~ 0.7% Ra-223 ~ 0.6% Th-230 ~ 16.0% Ra-226 ~ 4.9% Pb-210 ~ 2.4% Po-210 ~ 4.9% All other radionuclides account for about 0.5%. (These technical assumptions should be confirmed as soil and air data data are collected.)	RAP HPP
Aerosol size distribution	Mean particle size of >10 µm over 99% of the particles by mass for tailings. Detailed experimental studies conducted by the Low Dose Radiation Research Program resulting in a conservative assumption of 10 µm is being used, based on the particle size of Th-230 component, which tended to be associated with a smaller AMAD, on average. ICRP 68 values are based upon a 5 µm (AMAD) and are thus conservatively applied without modification. It should be noted that the preamble to Appendix A of 10 CFR 835 allows a modification to the DAC value and/or to dose assessment when the measured AMAD differs significantly from the assumed AMAD.	RAP ICRP 103/68 HPP 10 CFR 835
Air concentration assumptions	Assume 10% of the LLGA component is due to Th-230, and 10% is due to Ac-227 unless radioanalytical data is available to support a modification to this assumption. These components will be confirmed via composite air samples sent for radioisotopic analysis at least every 3 months, while significant radiological work is occurring. Allow 5 to 10 days for air samples to decay before recording the LLGA analysis.	RAP
Retroactive air concentration assumptions	The results of radioanalysis may be used to recalculate the airborne exposure environment experienced by workers.	RAP

µm = micrometer; AMAD = activity median aerodynamic diameter; BS = bone surface; CED = committed effective dose; LNG = lung; LLGA = long-lived gross alpha; Pa = protactinium; Po = polonium; TBD = technical basis document
ICRP 68, “Does Coefficients for Intakes of Radionuclides by Workers”
ICRP 103, “The 2007 Recommendations of the International Commission on Radiological Protection”

Table 2. Dosimetry Information for Select Radionuclides

Radionuclide	Dosimetry Parameters	Source
U Isotopes	Soluble chemical form [Class F] Decay Mode: Alpha DAC: 5E-10 μCi/mL U-nat, weighted, DCF BS = 38,907.4 mrem/μCi U-nat, weighted, DCF Eff = 2,257.1 mrem/μCi Comprises approximately 8.2% of the Th-230 activity	RAP HPP ICRP 68 10 CFR 835
Pa-231	Conservatively assumed as Class M – most restrictive Decay mode: Alpha DAC: 1E-12 μCi/mL DCF, BS = 1.63E7 mrem/μCi (most limiting) DCF, Eff = 3.30E5 mrem/μCi Comprises approximately 2.2% of the Th-230 activity	HPP ICRP 68 10 CFR 835
Ac-227	Conservatively assumed as Class F – most restrictive Decay Mode: Beta DAC: 2E-13 μCi/mL DCF, BS = 1.63E07 mrem/μCi DCF, Eff = 3.30E05 mrem/μCi Comprises approximately 2.2% of the U activity or about 0.182% of the Th-230 activity	RAP HPPICRP 68 10 CFR 835
Th-230	Insoluble chemical form [Class S in ICRP 103, "2007 Recommendations of the International Commission on Radiological Protection," terms] Decay Mode: Alpha DAC: 4E-11 μCi/mL DCF, BS – 518,518.5 mrem/μCi DCF, Eff = 26,666.7 mrem/μCi Comprises approximately 16% of the LLGA or about 10% of the total activity in tailings	RAP HPP 10 CFR 835
Ra-226	Class M (Only Class) Decay Mode: Alpha DAC: 2E-10 μCi/mL DCF, LNG 62963.0 mrem/μCi DCF, Eff = 8148.1 mrem/μCi (most limiting) Comprises approximately 16% of the LLGA or about 10% of the total tailings activity	RAP HPP ICRP 68 10 CFR 835
Pb-210	Class F (only class listed) Decay Mode: Beta DAC: 1E10 μCi/mL DCF, BS = 1.33E5 mrem/μCi (most limiting) DCF, Eff = 4.07E3 mrem/μCi Comprises 0% of the LLGA and about 49% of the tailings LLG beta activity (as might be found on an air particulate sample)	HPP ICRP 68 10 CFR 835
Po-210	Class M (most conservative) Decay Mode: Alpha DAC: 2E-10 μCi/mL DCF, LNG = 6.30E4 mrem/μCi DCF, Eff = 8.15E3 mrem/μCi (most limiting) Comprises 16% of the LLGA or about 10% of the total tailings activity	HPP ICRP 68 10 CFR 835

μCi = microcurie; BS = bone surface; DCF = dose correction factor; Eff = effective dose;= LLG = long-lived gross; LLGA = long-lived gross alpha; LNG = lung; mL = milliliter; mrem = millirem; Pa = protactinium; Po = polonium; TBD = technical basis document; U-nat = natural uranium
ICRP 68, "Does Coefficients for Intakes of Radionuclides by Workers"
ICRP 103, "The 2007 Recommendations of the International Commission on Radiological Protection

3.4 Radon and Thoron

3.4.1 General Information and Terminology

Moab UMTRA Project workers are occupationally exposed to radon gas (Rn-222/Rn-220), herein referred to as radon and thoron, respectively, throughout the remainder of this document (see Figures 1 and 3). A worker's exposure to this hazard is a result of his or her proximity to and handling of tailings materials bearing the parents, Ra-226 and Th-232.

Much of the discussion contained in this section is taken from the DOE document, "Occupational Exposure to Radon and Thoron" (PNNL-14108); however, when possible, the information provided in this document has been summarized and/or simplified, thus losing some of the technical nuances contained in that document. This is done in an effort to provide a concise and usable technical basis for radon and thoron monitoring.

According to the ICRP, studies of uranium miners and other underground mines have shown that high exposure to the short-lived progeny of radon significantly increases the risk of lung cancer in human beings, especially among smokers. Radon and its short-lived progeny are continuously produced by decay of Ra-226. Airborne concentrations of radon progeny (polonium [Po]-218, Pb-214, Bi-214, and Po-214) are of interest due to their potential for deposition in the lung, leading to subsequent irradiation of lung tissue by alpha emission from Po-218 and Po-214.

Thoron and its progeny are continuously produced by the decay of Ra-224 (a decay product in the Th-232 decay chain), hence the name thoron. In contrast with radon, substantially less thoron normally reaches and accumulates in the breathing zone due to its short half-life, which is 56 seconds for thoron versus 3.8 days for radon.

In air, there is a complex and dynamic relationship between radon and thoron and their short-lived progeny. As a result, many physical quantities and units (Table 3) are used when discussing exposure levels, measurement terms, and risk. For instance, because the progeny may not be in radioactive equilibrium with the parent or may be removed from air through deposition and other processes, the equilibrium factor (F) is used to describe the fraction of maximum possible progeny present based on the amount present. Radium dosimetry is sharply skewed by assumptions made concerning the equilibrium factor.

Table 3. Radon and Thoron Terminology

Abbreviation	Term	Description
pCi/L	picocuries per liter	A measure of airborne activity per unit volume air, due to radium and/or thoron.
WL	working level	A measure of the alpha particle energy potentially emitted by any mixture of radium/thoron progeny per unit volume of air. 10 CFR 835 defines a WL as any combination of short-lived decay products in one liter of air, without regard to the degree of equilibrium, that will result in the ultimate emission of 1.3 E+ 05 MeV of alpha energy.
WLM	working level month	One month being 170 hours of exposure at a concentration of one WL.
F	equilibrium factor	The fraction of progeny energy (WL) possible, per unit of radon or thoron.

3.4.2 Current Regulations and Units of Measurement

10 CFR 835, as amended June 8, 2007, adopts exposure limits based on information contained in ICRP Publication 65; "Protection Against Radon-220 at Home and at Work," and in DOE Standard (STD)-1121-98, "Internal Dosimetry." These limits, along with notes and examples meant to clarify their use, are presented in Table 4.

Table 4. Radium and Thoron Limits, Descriptions, and Notes

Isotope	Limit	Description/Notes	Source
Th (Rn-220) Ra (Rn-222)	1E-8 µCi/mL 8E-8 µCi/mL	DAC. This value assumes 100% F with progeny. This would be a measurement of filtered (i.e., non-progeny laden) air in a chamber of some kind. A measurement of this type, in these units, is not commonly performed. 10 CFR 835 Appendix A Footnote 5 allows this DAC value to be modified by assuming a differing value for F via actually measured or demonstrated equilibrium factors by the following equation: $DAC_{modified}$ = DAC x (1.00/fraction actual or demonstrated)	10 CFR 835 Appendix. A
Th (Rn-220) Ra (Rn-222)	2.5 WL 0.83 WL	The DAC values discussed above may be replaced by these WL values for appropriate limiting of decay product conditions. A WL is generally measured by passing a certain volume of air through a filter and collecting the short-lived progeny. The alpha activity on the filter is then measured and, based upon various measurement protocols and algorithms, a WL concentration estimate is provided.	10 CFR 835 Appendix. A
Th (Rn-220) Ra (Rn-222)	30 WLM 10 WLM	2.5 WL x 2040 hrs/WLY x 170 hrs/WLM 0.83 WL x 2040 hrs/WLY x 170 hrs/WLM	IAEA, 1994, Table II-I. Safety Reports Series 33, "Radiation Protection Against Radon in Workplaces Other than Mines"
Th (Rn-220) Ra (Rn-220)	10 pCi/L 80 pCi/L	1 DAC in pCi/l (i.e.,DAC in µCi/mL x 1E6 pCi/µCi x 1E-3 ml/L). The DAC, by definition, equates to an exposure rate of approximately 2.5 mrem/hr and assumes 100% equilibrium conditions. By extension, these "DAC" values are able to be modified based upon actual or demonstrated equilibrium conditions.	IAEA Safety Reports Series 33, "Radiation Protection Against Radon in Workplaces Other than Mines"

F = equilibrium factor; hr(s) = hour(s); IAEA = International Atomic Energy Agency; mrem/hr = millirems per hour; µCi/mL = microcuries per milliliter; pCi/L = picocuries per liter; WL = working level; WLM = working level month; WLY = working level year

4.0 RWPs and RCT Job Coverage

RWPs will be generated by Radiological Control and approved by Radiological Control and Health and Safety. Work may not begin until the appropriate RWP has been approved and is in place. The RWP informs workers of area radiological conditions, work controls, PPE, and entry/exit requirements.

RWPs are required for activities at the Project that include, but are not limited to:

- Entry into any radiological and/or radiological buffer area.
- Opening of enclosures where radon gas and progeny can collect (e.g., RRM containers).
- Any work within the Contamination Area (CA) on contaminated or potentially contaminated equipment.
- Digging or disturbing soil.

Workers will be briefed on the content, requirements, and radiological conditions of an RWP by a supervisor or RCT. Workers shall sign the acknowledgment sheet one time (per revision to the RWP) to indicate an understanding of the requirements of the RWP.

Workers shall sign a daily sign-in sheet on the RWP applicable to the work they are going to perform before entering the work area and shall sign out upon exiting these areas. With reference to the daily sign-in sheet, a worker shall only be signed in on one RWP at any one time.

RCT coverage will be provided as indicated on the applicable RWP. RCTs will perform frequent and timely surveys to ensure detection and characterization of contamination, if present. An RCT will periodically monitor radon concentrations in or around active work areas.

5.0 General Area Air Monitoring

General area air samples will be collected to monitor trends of airborne radioactive particulate activity concentrations and to ensure compliance with good work practices for control of radionuclides. Occupational air monitoring for radioparticulates will be performed routinely at designated locations to evaluate concentrations against the potential for a person to exceed 40 DAC-hours (hrs) in a year and to be trended against 2 percent of the effective Project DAC. When the DAC is based on stochastic effects, this translates to a committed effective dose (CED) of 100 millirems (mrem).

The Project routine monitoring program involves monitoring of both workplace and the workers. Workplace monitoring is the primary means of assessing workplace conditions for implementation of engineering and administrative controls to limit worker exposure.

5.1 Airborne Particulate Isotopes of Concern

The most limiting radioisotopes as determined through sampling data will be based on actinium (Ac)-227, protactinium (Pa)-231, Th-230, and Ra-226, which collectively account for approximately 95 percent of the internal dose scenario (bone surface). A derived DAC, based on the isotopic ratios and individually assigned DACs, will be determined and verified through sampling data and calculations as the Project progresses.

RRM contains elevated activity concentrations of radium, thorium, and associated decay products. These radionuclides also contribute to an elevated, direct penetrating gamma radiation field in the vicinity of soils piles, along with the continual emission of gas and progeny to the atmosphere.

5.2 Radon

Rn-222, which has a 3.8-day half-life, is generated at a rate in secular equilibrium with its Ra-226 parent. The nature of radon, being an inert radioactive gas, results in the continual release of the radionuclide from the tailings into the environment. The actual concentration of radon present within the breathable airspace is determined by the production rate (secular equilibrium) and the loss rate provided by environmental factors, such as wind speed and temperature inversions.

Temporary airborne radioactivity areas (ARAs), due to elevated radon concentrations, can occur as a result of environmental changes and changes in the physical condition of the source term (e.g., surface area exposed, moisture content).

Air sampling and/or radon working level (WL) surveys will be conducted in areas where personnel access is possible and radon is most likely to be anticipated (e.g., soil pile). If sustained average radon concentrations approach posted threshold values, as modified by the UMTRA Radon Exemption codified at 10 CFR 835.1101(a)(2), an ARA will be established, and appropriate respiratory protection equipment or stay times will be established as prescribed in the RWP for the area or task. Air sampling and/or radon monitoring will continue as necessary to determine the extent and duration of the ARA. When radon concentrations no longer exceed posted threshold values, the ARA will be disestablished.

It is possible that some permanent or long-term ARAs will be established in excavation or active soil work areas.

RCTs will perform additional radon WL monitoring for liquid system breaches, isolated soil excavations, and/or as dictated by the RWP. Airborne hazards may exist when there is maintenance performed or if there is a liquid system breach, when the primary airborne hazard will be radon and radon progeny (low particulate potential).

It is anticipated that very few Project operations will be continuously conducted in areas where the DAC concentrations exceed 10 percent of 10 CFR 835 limits, as measured without regard to respirator protection factors. When this is the case, the selection and use of respiratory protection equipment will be designed to prevent internal exposure to levels that are ALARA. Air sampling actions based on results include the following:

- General area air sampling will be performed to monitor and document ambient air concentrations in the workplace and to verify that radiological engineering and administrative controls are adequate.
- General area air sampling will be performed to determine the need for personnel air sampling. If measured concentrations are below 2 percent DAC, it is not expected that a worker can exceed 100 mrem per year CED occupying this area. If measured concentrations exceed 2 percent DAC, supplemental personnel air sampling will be implemented.
- Particulate air sampling data will be evaluated against a modified long-lived gross alpha (LLGA) DAC based upon the measured fractions of the primary dose contributors in the airborne mix.
- Air sampling data will be utilized to implement contingencies if air monitoring results are above expected values.
- ARA boundaries, ensuring concentrations outside the posted areas, are below posting thresholds.

5.3 Factors and Development of a Modified Air Particulate DAC

5.3.1 Factors Associated with Air Particulate Sampling

Air monitoring in and around tailings material is confounded by the following variables that must be addressed to achieve a reasonably accurate assessment of the air quality in the work place (this list is not all inclusive).

- LLGA, which contributes predominately to dose, is completely masked by the short-lived radon progeny for the first several hours or days following sample collection.
- The mixture of the radionuclides contributing to LLGA can change from location to location based on the chemical extraction process and sorting of by-product material that occurred while the mill was active.
- Because a mixture of alpha emitting radionuclides is being assessed using a gross alpha counting instrument, a mixture-modified DAC is required. Due to the uncertainty associated with variations in LLGA from location to location, this modified DAC is likely to be incorrect to some extent. Overly conservative assumptions may result in persons suffering the stress of unnecessary exposure control requirements; overly liberal assumptions may result in unnecessary exposures.

The first factor is addressed by counting the LLGA filters, 5 to 10 days post-collection; this allows much of the Rn-222 and short-lived daughters to decay away; the remaining alpha activity is then only associated with the long-lived particulate radionuclides. The second and third factors are addressed by making reasonably conservative assumptions about the mixture make-up and by sending out periodic air composite samples (i.e., many air samples from similar areas) for radioanalytical isotopic analysis. This information can then be used to develop a more accurate modified DAC and to reconstruct a clearer workplace monitoring profile.

5.3.2 Modified DAC Equation

This section provides an initial modified DAC based on the assumed starting conditions outlined in Attachment 1 of this Plan. It is important to recognize that this modified DAC should be updated as actual site data is collected and analyzed. It is also quite possible that the site may need to apply multiple modified DACs, if areas that differ significantly (relevant to dose) from one another are encountered.

The modified DAC is developed by the following equation.

$$DAC_{\mathrm{mod}ifed} = \frac{1}{\left(\frac{C_A}{DAC_A}\right) + \left(\frac{C_B}{DAC_B}\right) + \left(\frac{C_n}{DAC_n}\right)}$$

Where:

$DAC_{modifed}$ = The LLGA DAC

C_A = concentration (or fraction) of radionuclide A

C_B = concentration (or fraction) of radionuclide B

Cn = concentration (or fraction) of radionuclide N, and so on.

5.3.3 Determining the Components of the LLGA

LLGA particulates and their fractional contribution (as a fraction of the Th-230 activity) that would be expected on a particulate air filter, at 7 days post-sample collection, would be as shown in Table 5.

Table 5. Radionuclide Contribution to LLGA at 7 Days Post-sample Collection

U-238 Decay Chain LLGA	Contribution (fraction)	U-235 Decay Chain LLGA	Contribution (fraction)	Alpha In-growth	Contribution (fraction)
U-238	0.041 (0.0128)	U-235	0.0018 (0.0006)	NA	NA
U-234	0.04 (0.0124)	Pa-231	0.044 (0.0137)	NA	NA
Th-230	1.0 (0.311)	Ac-227[1]	0.044 (NA)	Ac-227[1]	0.027 (0.08399)
Ra-226	1.0 (0.311)	Th-227[2]	0.033 (0.0103)	NA	NA
Po-210	1.0 (0.311)	Ra-223[3]	0.028 (0.0087)	NA	NA
Total:	3.08	NA	0.107	NA	0.027
Grand Total:	3.215				

NOTE: The in-growth associated with Ra-223 is multiplied by 5 to account for the 5 alpha-emitting radionuclides with short half-lives. This results in an apparent, fractional alpha activity for Ac-227 of 0.0275. However, the in-vivo activity fraction would be 0.0137 (the same as Pa-231), and this fraction is used in the modified LLGA DAC below. If it is not used, the short-lived alpha-producing radionuclides associated with Ac-227 would be unaccounted for.

[1]Th-227 at 7 days decay.

[2]Ra-223 at 7 days decay.

[3]Ac-227 – 0.00 α, but has alpha in-growth of:

$A_{in\ growth} = A_p\,[(1-e^{-\lambda p+1}) + 5\,[(1-e^{-\lambda p+1})(1-e^{-\lambda p+2})]]$

Where:
$A_{in\text{-}growth}$ = Activity due to in-growth
A_p = Activity of parent (Ac-227 at 0.044)
λ = Decay probability contestant for radionuclide
$_{p+1}$ = First decay progeny from parent (Th-227)
$_{p+2}$ = Second decay progeny from parent (Ra-223)

Using the data provided in Section 3 and Table 5, the modified DAC is calculated as follows.

$$DAC_{\mathrm{mod}ifed} =$$

$$\frac{1}{\left(\frac{0.0258_{Unat}}{5E-10}\right)+\left(\frac{0.622_{Ra226+Po210}}{2E-10}\right)+\left(\frac{0.0137_{Ac227}}{2E-13}\right)+\left(\frac{0.311_{Th230}}{4E-11}\right)+\left(\frac{0.0137_{Pa231}}{1E-12}\right)+\left(\frac{0.010_{Th227}}{7E-11}\right)+\left(\frac{0.009_{Ra223}}{9E-11}\right)}$$

$= 1\ E - 11\ \mu Ci/ml$

This DAC should be compared to the LLGA air particulate data provided post-radon decay.

6.0 Individual Monitoring

The internal individual monitoring program supplements the workplace monitoring program and is used to: (1) confirm suspected intakes of radioactive material (RAM); (2) provide data for assessing dose for confirmed intakes; (3) verify the integrity of the workplace monitoring program; and (4) demonstrate compliance with the requirements of DOE radiation protection standards and 10 CFR 835.

Monitoring for internal exposure is performed on the following individuals:

- Workers who are likely to receive a CED of 100 mrem or more from all occupational radionuclide intakes in a year.
- Workers who are likely to receive a CED of 500 mrem or more from exposure to radon or thoron in a year.
- Minors and members of the public who are likely to receive a CED of 50 mrem or more from all radionuclide intakes in a year.
- The declared pregnant worker who is likely to receive an intake that results in a dose equivalent of 50 mrem or more to the embryo/fetus during gestation.

The internal dosimetry monitoring program relies on in vitro measurements (indirect bioassay) and air sampling to assess dose from internally deposited radionuclides. The use of a radioanalytical vendor who has been accredited under the DOE Laboratory Accreditation Program in the radionuclides of concern is used to ensure the accuracy of the information provided. Internal dose estimates are based on bioassay data rather than air sampling data unless the bioassay data is: (1) unavailable; (2) inadequate; or (3) internal dose estimates based on air sampling data are determined more accurate.

Routine monitoring of workers who are potentially exposed to radionuclides and do not have satisfactory bioassay detection capabilities at reasonable sampling frequencies (e.g., thorium isotopes) is accomplished by personal air sampling and by using uranium as a surrogate radionuclide. When appropriate, bioassay samples are collected for exposure to those isotopes when assessing intakes or when a significant intake is indicated or suspected.

6.1 Radon Dosimetry

In addition to Bioassay, Personal Air Sampling, and External Dosimetry Programs, a radon dosimetry program consisting of personal radon dosimeters and representative area monitoring may be used to assess individual exposures to radon. This may be required if an individual is likely to receive a dose because of radon exposure in excess of 500 mrem in a year from all sources including background, in accordance with standing requirements and approved 10 CFR 835 exemptions.

7.0 Internal Dose Assessments

For the Project, personal air sampling will be the primary indicator of internal exposure. These air samples, however, are expected to require a delay in counting of up to 7 days for radon and progeny decay to ensure the alpha activity is attributable only to long-lived particulate isotopes.

The Radiological Control Manager (RCM) will be immediately notified and will initiate a dose assessment in the case of:

- A confirmed positive bioassay (urinalysis).
- A DAC-hour (hr) trigger level of 25 mrem CED or greater in 1 week, as determined by personal air sampling results. This assessment would be initiated between 7 and 14 days after the exposure period due to the counting delay and will require collection and processing of a bioassay sample. Dose assignments would be made on review of the bioassay results, which will provide confirmation and quantification.
- Conditions that indicate an intake is suspected or is known to have occurred.

NOTE: An internal particulate dose assignment in excess 10 mrem will be evaluated and approved by a RAC Certified Health Physicist. The Project RCM and Operations/Site Manager will collectively review and approve interim job assignments for the affected employee during the course of the assessment.

Internal dose assignments up to 10 mrem will be reviewed and approved by the Project RCM. External technical expertise may be utilized at the discretion of the Moab RCM.

Fecal analysis is not anticipated to be utilized, but may be required when it is determined it would aid with quantifying the magnitude and nature of a suspected or confirmed intake of radioactive material.

Dose assessments from exposure to radon and its decay products will be based on general area radon WL air sampling results, such as DAC-hr tracking, in cases when concentrations exceed 10 percent of the DAC.

Doses are recorded and assigned for the calendar year in which the intake occurred. All confirmed intake evaluations are maintained in the worker's exposure file for future evaluation. All documentation and information necessary to review or re-calculate each assessed dose is recorded and maintained as part of the worker's permanent record.

Results of internal dose assessments are provided to workers. Those individuals who are monitored are provided with an annual dose report. The report to the employee includes a summary of internal as well as external dose. For visitors, all non-zero doses are reported to the individual within 30 days of the dose determination.

On request, terminating employees are provided a report within 90 days of the last day of employment that summarizes radiation dose for their total period of employment. A written estimate of the radiation dose received by that employee based on available information shall be provided at the time of termination, if requested. On request, a detailed exposure report is available to an employee or visitor.

8.0 Exposure Control and Personnel Monitoring

As part of the Project Radiation Protection Program, RCTs perform routine and special surveys to assess radiation levels in work areas, detect changes, and/or ensure the appropriateness of access controls and radiological postings. These surveys are used to preclude the possibility of exceeding established radiation dose limits and to minimize personnel exposure. Surveys are used to define the boundaries for posting CAs and radiation areas and to advise individual radiological workers of conditions.

Area radiation monitoring will be performed extensively during startup of soil removal and packaging activities. Survey locations will be identified and used to establish baseline radiation levels in all work areas and will be routinely performed to track and trend specific locations of interest, such as the soil pile and container load-out area.

All workers performing work within radiological areas will be assigned individual thermoluminescent dosimeters (TLDs) for monitoring external (e.g., beta, gamma) radiation exposure.

An administrative control level (ACL) will be used by the Project to maintain personnel exposures ALARA. RAC intends to initiate the Project with an assigned ACL for total effective dose (internal and external) of 700 mrem per individual per year. The Project ACL will be reviewed and revised annually based on prior year exposure results.

As Project activities are being conducted, it is not anticipated that the total effective dose ACL will be exceeded, and there is no expectation that the DOE ACL for individuals of 2,000 mrem per year total effective dose will be approached.

On reaching 500 mrem total effective dose in a work year, workers will have their normal job scope evaluated, and a determination will be made to place them in lower exposure positions until a review is completed by the RCM and the Operations/Site Manager.

This review will consist of an evaluation of the individual’s work hours, general work area radiological conditions, modes of exposure (e.g., internal, external), and comparison of coworker exposures. Recommendations will be documented, and the worker will be allowed to return to his or her normal work assignment with the approval of the RCM and Operations/Site Manager.

Respiration assigned protection factors that shall apply for the Project radionuclide particulate concentrations include the following:

- 1,000 for atmosphere supplying, airline, hood, and continuous flow.
- 1,000 for powered air purifying respirators.
- 50 for full face air purifying respirators.
- 25 for hooded air purifying respirators

9.0 Contamination Control

For normal operations, work will be performed in posted CAs, with RBAsseparating the CAs.

The physical nature of the mill tailings is a sand-like aggregate soil with the principle radioactive isotopes Ra-226 and Th-230. This soil has a low potential to adhere to material surfaces under dry conditions, but will adhere when wet. It is also prone to wind dispersion, especially during disturbances such as material loading into containers. CA boundaries will be established so the area is minimized and will be modified based on the ongoing work and potential for spread of the soils.

Established CAs will require an RWP for entry and/or work, implementing the radiological controls and required PPE. Personnel entry into these areas will be limited to the minimum required by operations, maintenance, and oversight demands. Entries will be made only by trained radiation workers.

- A CA will be established in the event of any accidental spills of the RRM. Containment and cleanup of the spill will be conducted within the CA.
- Those radionuclides in the U-238/U-235 decay chains that decay via alpha radiation, primarily Th-230, Ra-226, and Po-210, drive posting, and controls of radiological areas.
- All material and equipment exiting a CA will be surveyed for release by an RCT or Task Qualified Technician (TQT).
- Routine contamination surveys will be performed by RCTs at a specified frequency appropriate for the detection and control of contamination.

10.0 Personnel Responsibilities

10.1 Operations/Site Manager

The Moab and Crescent Junction Operations/Site Managers report to the Project Manager and have overall responsibility at their respective sites to ensure implementation of the requirements of this Plan.

10.2 RCM

The RCM reports to the Environmental, Safety, Health, and Quality Manager and has overall responsibility to ensure Radiological Control personnel assigned to the Project are implementing required contamination, radiation, airborne, and individual monitoring specified in this HPP. The RCM interfaces with the Operations/Site Manager on radiological issues encountered during operations, provides guidance to Project management for corrective actions, and ensures doses on the Project are maintained under the ALARA principle.

10.3 RCTs and Supervisors

RCTs and supervisors report to the RCM and are responsible for collecting, documenting, and reviewing radiological data identifying trends and comparing results against the limits in this HPP. TQT responsibilities can be the same as those for RCTs if they are Task Awareness Safety Card qualified. Radiological Control personnel make recommendations to the RCM and Operations/Site Manager on modifications to the monitoring program or operational activities to enhance performance or implement ALARA principles. Radiological Control personnel notify the RCM if contamination and airborne radioactivity limits are exceeded.

11.0 Radiological Incidents and Reporting

All radiological incidents or abnormal events shall be immediately reported to the RCM and Operations/Site Manager. Examples include, but are not limited to, skin or personal clothing contamination, situations when radioactive material uptake is suspected, and situations when contamination is spread to or discovered in a non-radiological (un-posted) area. Radiological Control supervision will facilitate the documentation and proper notification of the event or condition as required by site procedures and ensure corrective actions are taken as necessary. As required, Radiological Control supervision will assist with generating formal documentation, such as a Condition Report.

12.0 Personnel Entry and Exit Protocols

12.1 Entering and Exiting Controlled Areas

Access to a Controlled Area on the Project is controlled and managed through designated site access control points. TLDs are not required for access to Controlled Areas. Personnel and material monitoring is not required when exiting Controlled Areas.

12.2 Entering the Radiological Area Access Control Point

Access to a CA requires the following:

- Workers will verify their training and qualifications are current before using an RWP for entry and use of assigned PPE (respiratory protection where required).
- Workers shall sign the appropriate RWP for entry into a contaminated work area. Workers shall obtain the prescribed PPE clothing and respiratory protection equipment (as required).
- When wearing protective clothing such that no skin is exposed (e.g., full anti-Cs and respirator), the worker’s TLD must be worn underneath the protective clothing. When protective clothing requirements are such that skin is exposed (e.g., no respirator), the TLD must be worn on the outside of the anti-Cs. This protocol ensures representative beta exposure measurements.
- Before entering the contaminated work area, workers shall contact an RCT or TQT for assignment of a personal air sampler as required by the RWP, either as an individual or for a group.

- When changing work areas or job scope, a worker must sign in on the appropriate RWP and verify that he or she is wearing PPE that is in compliance with the RWP for the new area or work scope. If the worker must change PPE before moving to a new job area, the worker must exit the CA and go through the appropriate steps for re-entry, wearing the correct PPE for the new area, unless PPE modification, such as the addition of plastic sleeves or extra shoe covers, is approved on the RWP.
- Personnel entry into CAs must be through the established control point.
- If required, TLDs shall be worn on the outside of the worker's clothing or PPE, facing forward, between his or her waist and shoulders. Visitors may be allowed to enter the radiological areas on approval of Radiological Control staff with a properly trained and cognizant escort.

12.3 Exiting the Radiological Area Access Control Point

To exit a CA where whole-body personnel contamination monitors (PCMs) are not present, the potentially contaminated outer layer of PPE will be doffed at the CA exit in accordance with RWP instructions, and the workers will perform a minimum of a hand and foot monitoring with the instrumentation provided by the RCT. If no contamination is detected, they will proceed directly to the control point to perform whole-body monitoring through use of the PCM. To exit a CA, personnel must follow the exit requirements as prescribed in the *Moab UMTRA Project Radiological Posting and Access Control Procedure* (DOE-EM/GJRAC1748).

If contamination in excess of the values specified on the RWP is detected or an automated PCM alarms, personnel shall stay in the area and notify an RCT.

Personnel exiting CAs shall be surveyed using instrumentation capable of detecting radioactive contamination at the Fixed + Removable limits for the radioisotope of concern. The limits are specified in Table. 6.

Table 6. Fixed + Removable Radioisotope Limits

Radioisotope	Limit
Uranium and associated decay products (i.e., Th-230, Ra-226)	5,000 dpm/100 cm^2 alpha

cm^2 = square centimeters; dpm = disintegrations per minute

Short-lived radon progeny will, at times, become a factor leading to false positives with monitoring equipment. RCTs will evaluate these conditions after notification of an alarm or elevated frisking results.

Determination of short-lived radon progeny contamination will be determined using decay times or specific instrumentation capable of distinguishing short-lived radon progeny. If short-lived progeny is confirmed, and there is no confirmation of uranium, radium, and/or thorium above release criteria, the person may be released in accordance with approved site procedures.

If a specific monitoring protocol differs from the standard routines, such as in the case of work activities being conducted at remote locations, RCTs or TQTs will brief the workforce and provide instructions on the RWP.

All material exiting a CA shall be surveyed by an RCT or TQT. Workers shall doff anti-Cs at the appropriate control point whenever their protective clothing is compromised, when non-water-resistant anti-Cs get wet, or workers sweat through their protective clothing. Vehicles, tools, lapel samplers, and other equipment may only be surveyed out of a CA by an RCT or TQT. Workers requiring items of this nature to be removed from the CA shall give the RCT or TQT notice of such in advance.

Workers shall sign out on the RWP on exiting through the area access control point and place their TLDS in the appropriate slot in the TLD storage location.

13.0 PPE and Anti-C Clothing

The Project requires the evaluation, designation, and use of an appropriate level of protective clothing for entry to areas where removable contamination exists at levels exceeding the removable surface contamination values specified in of 10 CFR 835 Appendix D, “Surface Contamination Values.”

Designations will be based on the existing contamination levels in the work area, the anticipated work activity, worker health considerations, areas of the body likely to be exposed to removable contamination, and consideration for non-radiological hazards.

Anti-C clothing is best described as an intervention on the worker’s behalf with respect to potential risks associated with radiological contamination of the skin; however, as with any method of intervention, there are potential risks that must be evaluated against the benefits. Weighing the risks and benefits is the best method by which to reach a prudent decision as to when anti-C clothing is warranted. Relaxation of protective clothing requirements is an acknowledged means of risk reduction in certain situations where heat stress is of equal or greater concern to the workforce in accordance with Article 534 of DOE-STD-1098-99, “Radiological Control.”

Work activities in high contamination areas, soil contamination areas, fixed contamination areas, and ARAs require special consideration based on evaluating the potential risks associated with personnel exposure of material becoming removable during activities.

Designation and use of the appropriate protective clothing will be in accordance with the applicable RWP, after review by Operations, Safety, and Radiological Control personnel.

Cleaned PPE (e.g., face shields, respirators) that come into contact with the wearer’s face will be surveyed before re-use and/or issuance.

14.0 Posting and Labeling

All entrances to radiological areas and radiological materials areas (RMAs) must be clearly and conspicuously posted with the appropriate radiological postings. Signs at entrance points identify the type(s) of radiological areas and the facility/area-specific entry requirements for radiological control, such as RWP and dosimetry requirements.

Only Radiological Control personnel will designate, establish, and maintain radiological posting. No other personnel are authorized to place or remove any radiological posting. In some cases, hazardous waste operations posting designations may be used concurrently with radiological postings if authorized by the Environmental Compliance Manager in consultation with the RCM. Definitions, directions, and limitations can be found in the DOE Guide 441.1C, "Radiation Protection Program."

Workers should be aware of entry requirements and the information provided by radiological posting as applicable in 10 CFR 835. If more than one radiological condition exists in an area and requires posting, each condition must be identified by posting all radiological conditions on one or more signs (e.g., user changeable signs using inserts) using the most stringent heading and listing the radiological areas or other radiologically posted areas in decreasing order of importance. Any supplemental information will follow radiologically posted area designations. From most to least stringent, the hierarchy of posting is listed below. Each area lists whether it applies to the Moab UMTRA Project:

1. Very high radiation area – not applicable
2. High radiation area – not applicable
3. ARA – applicable
4. High CA – not applicable
5. Radiation area – not applicable
6. CA – applicable
7. RMA – applicable
8. Soil CA – applicable
9. Fixed CA – as applicable
10. Radiological buffer area – applicable
11. Underground RMA – applicable

Procedures will require each item or container of radioactive material to have a durable, clearly visible label bearing the standard radiation warning trefoil and the words "Caution, Radioactive Material" or "Danger, Radioactive Material."

The label must provide sufficient information to permit individuals handling, using, or working in the vicinity of the items or containers to take precautions to avoid or control exposures. Labeling and tagging guidance (e.g., definitions, directions, limitations) can be found in DOE G 441.1C. Internally contaminated or potentially internally contaminated material or equipment is individually labeled with the words "Caution, Internal Contamination" or "Caution, Potential Internal Contamination," as applicable. Radiological use vacuum cleaners must be uniquely marked and labeled to identify both their internal and external contamination characteristics.

Sealed and unsealed sources or their associated storage containers are labeled as radioactive material, and storage containers and devices containing a sealed source are clearly marked.

If material or equipment is taken from a radiological area or RMA, placed in the CA, and has not been adequately surveyed to allow unrestricted release, the material and equipment must be tagged as radioactive with a yellow tag.

15.0 Receipt and Control of Radioactive Material and Sources

RAC will receive, manage, and control RAM and sources that are DOE-owned and assigned to RAC. Radioactive sources owned by subcontractors/suppliers and are licensed by the Nuclear Regulatory Commission of Agreement States are excluded from DOE requirements per 10 CFR 835.1(b)(1). Materials are to be controlled at quantities in excess of the levels stated in 10 CFR 835 Appendix E, "Values for Establishing Sealed Radioactive Source Accountability and Radioactive Material Posting and Labeling Requirements."

15.1 RAM Control

The RAC will maintain a RAM control program that will include protocols for the receipt, inventory, labeling, control, storage, transfer, disposal, recordkeeping, training, and surveying. Monitoring received RAM packages is performed as soon as practicable following receipt, but no later than 8 hours following the beginning of the work day following the day of delivery. Monitoring will normally include a review of any accompanying paper work, dose rates, an inspection for physical damage/leaking, and swipe surveys for alpha and beta contamination. Specific instructions and definitions covering receipt inspections of RAM packages are found in DOE G 441.1C.

15.2 Source Control

RAC will maintain an accountable source control program that will include protocols for receipt, inventory, labeling, control, storage, transfer, disposal, recordkeeping, training, surveying, and leak (integrity) testing. Each source will be subject to an initial leak test upon receipt, when damage is suspected, and at intervals not to exceed 6 months or license conditions. These leak tests will be capable of detecting radioactive material at or below 0.005 microcuries. Leaking sources will be removed from service and controlled. The data presented in 10 CFR 835 Appendix E are to be used for identifying accountable sealed RAM and establishing the need for RAM labeling in accordance with 10 CFR 835.605, "Labeling items and containers."

16.0 References

10 CFR 835 (Code of Federal Regulations), "Occupational Radiation Protection."

DOE (U.S. Department of Energy) Guide 441.1C, "Radiation Protection Program."

DOE (U.S. Department of Energy), *Moab UMTRA Project Internal Dosimetry Technical Basis Manual* (DOE-EM/GJRAC1913).

DOE (U.S. Department of Energy), *Moab UMTRA Project Radiation Protection Program* (DOE-EM/GJ610).

DOE (U.S. Department of Energy), *Moab UMTRA Project Radiological Posting and Access Control Procedure* (DOE-EM/GJRAC1748).

DOE (U.S. Department of Energy), *Moab UMTRA Project Remedial Action Plan* (DOE-EM/GJ1547).

DOE (U.S. Department of Energy), "Occupational Exposure to Radon and Thoron" (DOE PNNL-14108).

DOE (U.S. Department of Energy), STD-1098-99, "Radiological Control."

DOE (U.S. Department of Energy), STD-1121-98, "Internal Dosimetry."

ICRP 65 (International Commission on Radiological Protection), "Protection Against Radon-220 at Home and at Work."

ICRP 68 (International Commission on Radiological Protection), "Dose Coefficients for Intakes of Radionuclides by Workers."

ICRP 103 (International Commission on Radiological Protection), "The 2007 Recommendations of the International Commission on Radiological Protection."

IAEA (International Atomic Energy Agency) Report 33, "Radiation Protection Against Radon in Workplaces Other than Mines."

Attachment 1.
Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques

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Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques

Technical Approach Position

Energy*Solutions*
September, 2007

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Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

1.0 Purpose

EnergySolutions is managing the UMTRA Moab Project for the Department of Energy. This will involve the excavation and handling of million-yard quantities of uranium mill tailings, also referred to as by-product material in regulations. These by-products are found in the tailings mixture at low concentrations (1 – 2 nCi/g) for individual radionuclides. These radionuclides consist of U-238, U-235, Th-232 and their decay progeny. The equilibrium conditions of these decay chains have been disturbed by the milling process. Thus, this appendix has been developed to:

1) Document the likely radionuclide mixture to be found on site, and
2) Provide information pertinent to the various internal dose monitoring and modeling strategies that may be applied to workplace and individual exposure assessments.

1.1 General Approach

Radionuclide mixture assumptions are carefully developed.

Uranium is selected to act as a surrogate for other radionuclides because it is highly soluble (in tailings) and shows up in the urine bioassay compartment at much higher concentrations, per unit intake, than several of the other, more dose effective radionuclides such as Pa-231, Ac-227, and Th-230.

The uranium intake in this appendix is modeled assuming the urine sample is collected at various post-acute-intake dates. The intakes of other radionuclides are then developed as a simple ratio corresponding to the radionuclide mixture.

1.2 Precautions

Actual radionuclide mixtures may change daily. Composite analysis via off-site radioanalytical vendor of air samples should provide a reasonable approximation of the average daily in-field radionuclide mixture conditions – however, when gross-activity counting individual samples on-site, large variations in alpha to beta ratios may indicate a mixture variation.

Radioanalytical analysis accuracy of certain radionuclides may be effected by the chemical extraction process (e.g., lead may move into a gaseous state at relatively low temperatures prior to counting) thus attempts to correlate on-site gross activity with radioanalytical results may be confounded by these factors.

Class F Uranium is biokinetically very different from Class M and S radionuclides; thus large intake error terms may arise due to chronic vs. acute intake assumptions. It would be appropriate to consult air sample data and activity logs when developing intake models.

Air sample data will likely be a more accurate, timely, and reliable indicate of actual intake.

Depending on the overall intake or intake-rate found in the field, this general approach should be periodically repeated as better data, by location and work activity, is developed.

Th-232 may or may not be associated with mill tailings and the concentrations may vary considerably.

Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

2.0 Methods and Devices

2.1 Radionuclide Mixture Development

The MK-Ferguson (MKF) UMTRA Project Internal Technical Basis Document (TBD) is used to develop the activity fractions of the various radionuclides in the tailings material. During the development of the MKF TBD, thousands of soil and air place monitoring results from similar tailings sites (Grand Junction, Colorado; Mexican Hat, UT; Durango, CO) were analyzed to develop average exposure parameters which included activity fraction analysis by radionuclide along with solubility and particle size studies. Pertinent findings from the MKF TBD include:

- Uranium is assumed to be 8.3% of the Th-230 activity
- Solubility Class for Uranium is F (Fast)
- Th-230 solubility Class is S (Slow)
- Activity Mean Aerodynamic Diameter (AMAD) of 10 µm is adequately conservative

Sample data provide by the SM Stoller Company collected in 2007 is used to confirm the relative mixture assumptions of the MKF TBD. This soil data is summarized in the Table A-1. This data appears to be a product of gamma spectroscopy and errors (e.g., U-238/-234 not in equilibrium, U-235 is slightly enriched) may be assumed to be a product of low measurement selectivity, low efficiency, and intra-mixture gamma line interference; i.e., normal error associated with gamma spectroscopy. On the whole the soil data set appears to be representative of the tailings mixture and serves to confirm that uranium is largely removed from the mixture and that Ra-226 and Th-230, and their associated decay progeny, are at or near a state of activity equilibrium.

Table A-1
[1], Average Radionuclide Results

Radionuclide CPC	pCi/g
U-238	39
[2], U-235/236	4
U-234	79
Th-230	1025
Ra-226	1464
Rn-222	1464
Po-218	1464
Pb-214	1464
Bi-214	1464
Po-214	1464
[3], Th-232	3.3

1, The values shown for Ra-226, Th-230, U-238, U-234/-235, Th-232, Th-228 are the averages of the sampled results, the remaining radionuclides are decay products assumed to be in secular equilibrium with their parent. As shown in the table Radium-226 and its shorter-lived decay products Rn-222, Po-218, Pb-214, Bi-214, Po-214, Pb-210, Bi-210 and Po-210 provide approximately 90 percent of the activity in the uranium mill tailings.
2, U-235 leads a separate decay chain from U-238.
3, Th-232 leads a separate decay chain.

Working from Table A-1 and the MKF Internal TBD, an assumed tailings mixture is developed in Table A-2. The notes and assumptions used to develop Table A-2 are found directly following Table A-2.

2.1.1 Column-by-Column Discussion of Table A-2

The first column is a simple numbering A1 – A46 used to reference the various rows in the table.

The second column in Table A-2 contains a listing of the three decay chain radionuclides found in tailings. These are headed by U-238, U-235, and Th-232. The U-235 Chain is placed between Pa-234 and U-234 in the U-238 decay chain. This is done to facilitate the handling of "natural uranium" in urine bioassay sample results performed later in this appendix.

Columns 3 and 4 provide the soil activity concentrations, and the fractional activity of the whole, as provided by SM Stoller. This information is generated from the information provided in Table A-1. This data is used to confirm and compare the mixture of radionuclides to the decay mode and fractional analysis performed on the right side of the table.

Columns 5 and 6 provide each radionuclide's numerical half-life and measurement unit (note that the units are as follows: y = year, d = day, h = hour, m = minute, s = second, ms = millisecond, us = microsecond, ns = nanosecond).

Columns 7 and 8 (labeled as α-U, and β-U respectively) are used to sum up the alpha and beta activity associated with the three uranium isotopes (U-238, U-235, U-234). This is done for the purpose of deriving detection sensitivities of natural uranium in urine which is performed later on in this Appendix.

Columns 9 and 10 are used to sum up the alpha and beta activity from the non-uranium radionuclides, decay chain. Column 11 provides the solubility class of each radionuclide. Note that where a class is not listed, one is not provided in ICRP 68; in these cases the radionuclide is assumed to be biokinetically aligned with the parent radionuclide and any additional progeny induced dose contribution is applied within the dose conversion factor (DCF) associated with the parent.

2.1.2 Mixture Derivation

Th-230, Ra-226, and their decay progeny are assumed to be equilibrium. These are each set a dimensionless activity fraction of one (1.0).

The total alpha activity due to uranium in tailings is assumed to be 8.3% of the remaining U-238 decay chain radionuclides; e.g., U-nat = 8.3% of the Th-230 activity. This flows from the MKF Internal Dosimetry TBD. This 8.3% is then subdivided between the 3 isotopes based upon their natural activity abundance in nature (0.495, 0.483, and 0.022 for U-238, U-234, and U-235 respectively). Thus, for U-235 the activity fraction assigned is 0.083 x 0.022 or 0.00183 which in effect means that 0.00183 U-235-alphas is produced per Th-230-alpha.

The U-235 decay chain radionuclides below Pa-231 are assumed to be at their natural activity fraction of U-238 decay chain radionuclides or about 4.4% of any individual radionuclide. This is because U-235 makes up about 2.2% of the natural uranium activity or about 4.4% of either the U-238 or U-234 activity alone[1]. Additionally, Th-230 should be in equilibrium with the pre-milled U-234 activity, and Pa-231 should also be in equilibrium with the pre-milled U-235 activity; thus Pa-231 activity will be 4.4% of the Th-230 activity.

Th-232 and progeny are assumed to be at the same faction as provided in Table A-1.

[1] Several references put the U-235 activity at 4.7%, however dividing 0.022 by the U-238 pr U-234 fractional activity puts the percentage of U-235 activity at 4.6% and 4.4% respectively. 4.4% is chosen as it is more intuitively cohesive with the overall approach - if the activity that U-235 is being compared with is cut in half (approximately) then its relative activity should approximately double (i.e., 0.022 become 0.044).

Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

Table A-2
Tailings Activity Mixture and Select Radiological Parameters

1	2	3	4	5	6	7	8	9	10	11
A1			fraction of			[2,3] α-U				
A2		[1] pCi/g	Total Act.	T1/2		0.083	β-U	A	B	CLS
A3	U-238	39	0.0027	4.47E+09	y	0.041				F
A4	Th-234	39	0.0027	24.1	d		0.04			S
A5	Pa-234	39	0.0027	1.17	m		0.04			S
A6	[5] U235	4	0.0003	7.03E+08	y	0.0018				F
A7	Th-231	4	0.0003	25.5	h				0.0018	S
A8	[5,7] Pa-231	4	0.0003	3.28E+04	y			0.044		M
A9	Ac-227	4	0.0003	21.77	y				0.044	F
A10	Th-227	4	0.0003	18.7	d			0.044		S
A11	[8] Ra-223	4	0.0003	11.43	d			0.044		M
A12	Rn-219	4	0.0003	3.96	s			0.044		
A13	Po-215	4	0.0003	1.78	ms			0.044		
A14	Pb-211	4	0.0003	36.1	m				0.044	
A15	Bi-211	4	0.0003	2.14	m			0.044		
A16	Tl-207	4	0.0003	4.77	m				0.044	
A17	Pb-207	0	0.0000	Stable						
A18	U234	79	0.0055	2.45E+05		0.040				F
A19							U-nat Alpha Activity Fraction:			0.083
A20								α	β	
A21	[4] Th-230	1025	0.0708	7.70E+04	y			1		S
A22	[4] Ra-226	1464	0.1011	1600	y			1		M
A23	Ra-222	1464	0.1011	3.82	d			1		
A24	Po-218	1464	0.1011	3.05	m			1		
A25	Pb-214	1464	0.1011	26.8	m				1	F
A26	Bi-214	1464	0.1011	19.9	m				1	M
A27	Po-214	1464	0.1011	164	us			1		
A28	Pb-210	1464	0.1011	22.3	y				1	F
A29	Bi-210	1464	0.1011	5.01	d				1	M
A30	Po-210	1464	0.1011	138.4	d			1		M
A31	Pb-206	0	0.0000	Stable				0	0	
A32	Th-232	3.3	0.0002	1.40E+10	y			0.0002		S
A33	Ra-228	3.3	0.0002	5.75	y				0.0002	M
A34	Ac-228	3.3	0.0002	6.13	h				0.0002	F
A35	Th-228	3.3	0.0002	1.913	y			0.0002		S
A36	Ra-224	3.3	0.0002	3.66	d			0.0002		M
A37	Rn-220	3.3	0.0002	55.6	s			0.0002		
A38	Po-216	3.3	0.0002	0.15	s			0.0002		
A39	Pb-212	3.3	0.0002	10.64	h				0.0002	F
A40	Bi-212	3.3	0.0002	60.55	m			0.0001	0.0001	M
A41	Po-212	2.112	0.0001	305	ns			0.0001		64%
A42	Tl-208	1.188	0.0001	3.07	m				0.0001	36%
A43	Pb-208	0		Stable				0.0000	0.0000	
A44	Sum:	14474.00	100%		sum:	0.083	0.082	6.2654	4.1347	
A45							Total:	6.35	4.22	
A46							Total	α + β:	10.57	

1, pCi/g derived from "Preliminary Results Report," Moab, VP Remediation (Policaro), RIN: 07060946, 7/12/2007 [Provided by Stoller to ES]
2, alpha (α) or beta (β) activity.
3, Uranium activity is assumed to be 8.3% of the "tailings mixture" activity (MKF UMTRA RAC Internal TBD, 1995); Upon researching this TBD what is meant by this is that the U-natural activity is 8.3% of the remaining U-238 decay chain radionuclides; e.g., 8.3% of the Th-230 activity. Thus if activity of Th-230 = 1, the U-nat activity would equal 0.083.
4, Th-230 and Radium are assumed, in this analysis to be in approximate equilibrium; this assumption relies on the rational that while the ratios may vary by location, the overall average should demonstrate equilibrium across the whole of the by-product material.
5, U-235 and its direct decay progeny exist at 4.4% of the U-238 decay chain activity
Note # 6 is contained in Table A-5.
7, Pa-231, and its progeny are assumed to be in their original natural abundance, as a percentage of the U-238 decay chain radionuclides, resulting in an activity fraction of 0.022 per U-238-decay-chain-radionuclide.

2.2 Derivation of the Uranium Dose Conversion Factor

Bioassay sampling for mass-uranium, reported in µg/sample, allows for a simple, inexpensive, reasonably sensitive, and quick analysis of the uranium content of urine samples. Uranium is also one of the only radionuclides found in tailings systemically eliminated through the urine compartment in sufficient quantities to be reasonably detectable. As an example the intake retention functions (IRFs) at 40 days post intake for a few select tailings radionuclides are listed in Ref. 3 are as follows:

Uranium -	4.27E-4 (Class F)
Actinium –	4.20E-6 (Class F)
Thorium -	5.25E-7 (Class S)

As demonstrated above, uranium shows up at a rate almost 100 times greater than Actinium and 1000 times greater than thorium. Thus, even if uranium only makes up 8.3 percent of the per-unit-Th-230-intake; we would expect to see 83 times more uranium than [2]Th-230 (1000 x 0.083). Under a chronic exposure scenario this becomes even more exaggerated. Thus using uranium as an indicator of intake is appealing for a number of reasons.

One issue that needs to be addressed when using mass (µg) uranium results to determine potential dose is that ICRP 68 does not supply a DCF for mass natural uranium. Thus it must be converted to its isotopic parts, multiplied by each isotope's DCF, and then summed to get a total dose, or alternatively an activity-weighted DCF for natural uranium can be derived. This derivation is performed using the information provided in Table A-3 and the following equation.

$$DCF_{U-Nat} = (f_{U238} * DCF_{U238}) + (f_{U234} * DCF_{U234}) + (f_{U235} * DCF_{U235}) \quad \text{Eq-1}$$

Table A-3
U-Nat DCF Derivation Givens

Uranium Isotope	Natural Activity Fraction	DCF, BS (mrem/µCi)	DCF, Eff, (mrem/µCi)
U-238	0.495	37037.0	2148.1
U-234	0.483	40740.7	2370.4
U-235	0.022	40740.7	2222.2

Using the givens from Table 1, and Eq-1, the U-Nat DCF is thus determined to be:

Target Organ (Bone Surface), Committed Equivalent Dose = 38,907.4 mrem/µCi intake of U-Nat.
Effective (whole body), Committed Effective Dose = 2257.1 mrem/µCi intake of U-Nat.

2.3 Assessment of the Contribution to Total Dose by the Individual Radionuclides

Given the activity ratios and the DCFs of the various radionuclides a weighted-dose-contribution, by radionuclide, is derived by multiplying the activity contribution, as a fraction of the Th-230 contribution, by the DCF of each radionuclide. The calculated dose for each radionuclide is then summed to a total. The individual contribution is then simply the individual dose contribution divided by the total dose. This process is represented in equations 2, 3, and 4 below.

$$\text{WRD}_n = (\text{I}_{f,n}) * (DCF_n) \quad \text{Eq-2}$$

[2] In fact, uranium activity in the urine compartment would be more than 83 times the activity of Th-230 due the differences in specific activity.

Where:

WRD_n = Weighted Relative Dose of Radionuclide n

$I_{f,n}$ = Fractional Intake of Radionuclide n

DCF_n = Dose Conversion Factor of Radionuclide n

$$AWD = \sum WRD_{n1, n2, n3 ect.} \qquad \text{Eq-3}$$

Where:

AWD = Accumulated Weighted Dose

$$PAD_n = 100x\left(\frac{WRD}{AWD}\right) \qquad \text{Eq-4}$$

Where:

PAD_n = Percent Accumulated Dose for Radionuclide n

This process is carried out for all of the tailings radionuclides and presented in Table A-4 and Table A-5.

2.4 Derivation of the Actinium/Thorium Bioassay Screening Level via Uranium Activity

Given that the ratios of Ac-227 and Th-230 to U-natural activity are or can be reasonably well estimated, an assessment of dose due to these radionuclides, and others within the tailings mix can be estimated if the uranium intake can be estimated. This section provides a description of how to obtain a screening level for reviewing uranium bioassay results. For purposes of this derivation the following is assumed:

- Normal background uranium intake results in a background bioassay result ranging from 0.007 to as high as 0.06 µg/sample; better numbers would be calculated based on the actual population to be monitored; however this range is supported by baseline bioassay samples collected at the OU-1 Mound, Ohio Project during 2007.
- An average screening background of 0.01 µg/sample would be subtracted from the reported result prior to comparison against the investigation level. Under field conditions, a person's actual baseline sample result would normally be subtracted.
- The uranium is modeled based upon an acute intake at 40 and 60 days post intake – this is a common method used when implementing a quarterly bioassay program wherein the midpoint of the monitoring period is assumed to be the intake date. This is likely to be very conservative in that urine samples would be exponentially influenced by more recent, possible chronic, intakes that occur to a small degree on daily basis while working with tailings material. Thus, a sample that exceeds a screening level is not necessary an indication of a reportable dose, and even less so of a significant dose.

Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

Table A-4
Committed Effective Dose Contribution
by Radionuclide Given Tailings Mixture Assumptions

Radionuclide	U Activity α	U Activity β	α	B	CLS	mrem/uCi DCF Eff (ICRP 68)	Weighted Rel. Dose EFF	Percent of Accumulated Dose
U238	0.041				F			
Th234		0.04			S			
Pa234		0.04			S	2.15E+00	0.08833275	0.0%
[5] U235	0.0018				F			
Th-231				0.0018	S	1.48E+00	0.00270248	0.0%
[5,7] Pa-231			0.044		M	3.30E+05	1.45E+04	8.7%
Ac-227				0.044	F	2.33E+06	1.03E+05	61.5%
Th-227			0.044		S	2.81E+04	1.24E+03	0.7%
[8] Ra-223			0.044		M	2.11E+04	9.28E+02	0.6%
Rn-219			0.044					
Po-215			0.044					
Pb-211				0.044				
Bi-211			0.044					
Tl-207				0.044				
Pb-207								
U234	0.040				F			
			U-nat alpha Activity:		0.083	2.26E+03	187.3	0.1%
			α	β				
[4] Th-230			1		S	2.67E+04	26666.7	16.0%
[4] Ra-226			1		M	8.15E+03	8150.0	4.9%
Ra-222			1					
Po-218			1					
Pb-214				1	F	1.78E+01	17.8	0.0%
Bi-214				1	M	7.78E+01	77.8	0.0%
Po-214			1					
Pb-210				1	F	4.07E+03	4070.0	2.4%
Bi-210				1	M	2.22E+02	222.0	0.1%
Po-210			1		M	8.15E+03	8150.0	4.9%
Pb-206			0	0				
Th-232			0.0002		S	4.44E+04	10.1	0.0%
Ra-228				0.0002	M	6.30E+03	1.4	0.0%
Ac-228				0.0002	F	1.07E+02	0.0	0.0%
Th-228			0.0002		S	9.26E+04	21.1	0.0%
Ra-224			0.0002		M	8.89E+03	2.0	0.0%
Rn-220			0.0002					
Po-216			0.0002					
Pb-212				0.0002	F	1.22E+02	0.0	0.0%
Bi-212			0.0001	0.0001	M	5.56E+01	0.01	0.0%
Po-212			0.0001		64%			
Tl-208				0.0001	36%			
Pb-208			0.0000	0.0000				
Sum:	0.083	0.082	6.2654	4.1347		sum:	1.67E+05	100.0%

Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

Table A-5
Committed Equivalent Dose Contribution
by Radionuclide Given Tailings Mixture Assumptions

Radio-nuclide	U Act α	U Act β	α	β	CLS	mrem/uCi DCF Max Organ (ICRP 68)	Target	[6]CED div EFF	Weighted Rel. Dose Org	Percent of Accumulated Dose
U238	0.041				F					
Th234		0.04			S					
Pa234		0.04			S	2.81E+01	ET	13.1	1.15E+00	0.0%
[5] U235	0.0018				F					
Th-231				0.0018	S	8.89E+00	ET	6.0	1.62E-02	0.0%
[5,7] Pa-231			0.044		M	1.63E+07	BS	49.4	7.17E+05	15.6%
Ac-227				0.044	F	7.04E+07	BS	30.2	3.10E+06	67.3%
Th-227			0.044		S	2.31E+05	LNG	8.2	0.00E+00	0.0%
[8] Ra-223			0.044		M	1.75E+05	LNG	8.3	0.00E+00	0.0%
Rn-219			0.044							
Po-215			0.044							
Pb-211				0.044						
Bi-211			0.044							
Tl-207				0.044						
Pb-207										
U234	0.040				F					
	U-nat alpha Activity:				0.083	3.89E+04	BS	17.2	3.23E+03	0.1%
			A	b						
[4] Th-230			1		S	5.19E+05	BS	19.4	5.19E+05	11.3%
[4] Ra-226			1		M	6.30E+04	LNG	7.7	6.30E+04	1.4%
Ra-222			1							0.0%
Po-218			1							0.0%
Pb-214				1	F	4.81E+02	ET	27.0	4.81E+02	0.0%
Bi-214				1	M	1.78E+03	ET	22.9	1.78E+03	0.0%
Po-214			1							0.0%
Pb-210				1	F	1.33E+05	BS	32.7	1.33E+05	2.9%
Bi-210				1	M	1.81E+03	LNG	8.2	1.81E+03	0.0%
Po-210			1		M	6.30E+04	LNG	7.7	6.30E+04	1.4%
Pb-206			0	0						
Th-232			0.0002		S	5.19E+05	BS	11.7	1.18E+02	0.0%
Ra-228				0.0002	M	1.33E+05	BS	21.1	3.03E+01	0.0%
Ac-228				0.0002	F	3.11E+03	BS	29.1	7.09E-01	0.0%
Th-228			0.0002		S	7.78E+05	ET&LNG	8.4	1.77E+02	0.0%
Ra-224			0.0002		M	3.52E+03	BS	0.4	8.03E-01	0.0%
Rn-220			0.0002							0.0%
Po-216			0.0002							0.0%
Pb-212				0.0002	F	3.70E+03	ET	30.3	8.44E-01	0.0%
Bi-212			0.0001	0.0001	M	1.56E+03	ET	28.1	3.56E-01	0.0%
Po-212			0.0001		64%					
Tl-208				0.0001	36%					
Pb-208			0.0000	0.0000						
Sum:	0.083	0.082	6.2654	4.1347			sum:		4.60E+06	100.0%

Notes 1 – 5, and 7 are contained in Table A-2.
6, A result > 10 indicates that the Committed Equivalent Dose (CED) is more limiting than the Committed Effective Dose (EFF) for the Radionuclide.
8, The dose due to the radionuclides below Ra-223 are summed up into Ra-223.

2.4.1 Determine the Uranium Intake

It should be noted that if an actually determination of the potential intake were to be performed, it should be done so using the best exposure scenario information available. The process used in this Appendix is conservative in nature and is meant to supply a poor-case assessment of the reasonableness of using uranium as a surrogate. Thus, step one in the process is conservatively estimating the intake of uranium at 40 and 60 days. This is done by taking the sample result (µg/sample) divided by the IRF (Ref. 3) for uranium, Class F.

In Table A-6 and in the charts presented at the end of this section, this is performed for sample results ranging from 0.05 to 0.29 µg/sample uranium.

Table A-6
Uranium Intake as Function of Bioassay Results (µg/sa or dpm/sa)

	µg/sa-->	0.05	0.07	0.09	0.11	0.13	0.15	0.17	0.21	0.25	0.29
	dpm/sa>	0.0749	0.1049	0.1349	0.1648	0.1948	0.2248	0.2547	0.3147	0.3746	0.4346
	U-nat f_u										
1d	1.85E-01	1.82E-07	2.55E-07	3.28E-07	4.01E-07	4.74E-07	5.47E-07	6.20E-07	7.66E-07	9.12E-07	1.06E-06
3d	5.09E-03	6.63E-06	9.28E-06	1.19E-05	1.46E-05	1.72E-05	1.99E-05	2.25E-05	2.78E-05	3.32E-05	3.85E-05
10d	2.67E-03	1.26E-05	1.77E-05	2.28E-05	2.78E-05	3.29E-05	3.79E-05	4.30E-05	5.31E-05	6.32E-05	7.33E-05
20d	1.25E-03	2.70E-05	3.78E-05	4.86E-05	5.94E-05	7.02E-05	8.10E-05	9.18E-05	1.13E-04	1.35E-04	1.57E-04
40d	4.27E-04	7.90E-05	1.11E-04	1.42E-04	1.74E-04	2.06E-04	2.37E-04	2.69E-04	3.32E-04	3.95E-04	4.58E-04
60d	2.26E-04	1.49E-04	2.09E-04	2.69E-04	3.29E-04	3.88E-04	4.48E-04	5.08E-04	6.27E-04	7.47E-04	8.66E-04
80d	1.45E-04	2.33E-04	3.26E-04	4.19E-04	5.12E-04	6.05E-04	6.98E-04	7.91E-04	9.78E-04	1.16E-03	1.35E-03
100d	1.00E-04	3.38E-04	4.73E-04	6.08E-04	7.43E-04	8.78E-04	1.01E-03	1.15E-03	1.42E-03	1.69E-03	1.96E-03
200d	2.44E-05	1.38E-03	1.94E-03	2.49E-03	3.04E-03	3.60E-03	4.15E-03	4.70E-03	5.81E-03	6.92E-03	8.02E-03
300d	8.86E-06	3.81E-03	5.33E-03	6.86E-03	8.38E-03	9.90E-03	1.14E-02	1.30E-02	1.60E-02	1.90E-02	2.21E-02
400d	4.58E-06	7.37E-03	1.03E-02	1.33E-02	1.62E-02	1.92E-02	2.21E-02	2.51E-02	3.09E-02	3.68E-02	4.27E-02

Given the information in Table A-6, the intake of any of the tailings radionuclides can be estimated by scaling the intake to it relative relationship to uranium. An example of how a Th-230 intake represented by a sample result of 0.09 µg/sample at 40 days post intake would be calculated is provided below.

> Example 1 – Th-230 intake as a function of U-Nat intake
> Given:
> bioassay result of 0.09 µg.
> > We assume a post intake date of 40 days; from Table A-6, this equates to an intake of 1.42E-4 µCi of uranium.
> > We know that uranium is 8.3% of Th-230, so inversely Th-230 is 1/0.083 or 12.05 times the uranium activity.
> > The uranium resulte is multiplied by 12.05 to get the Th-230 result
> Thus: 1.43E-4 µCi U-Nat x 12.05 Th-230 (µCi) per U-Nat (µCi) = 0.0017 µCi of Th-230.

In table A-7, the intake concentrations for Th-230 and Ac-227 are developed, based on the example above, over a range of uranium sample results.

Table A-7
Intake of Select Radionuclides as a Function of U-Natural Intake

µg/sa>	0.05	0.07	0.09	0.11	0.13	0.15	0.17	0.21	0.25	0.29
dmp/sa>	0.0749	0.1049	0.1349	0.1648	0.1948	0.2248	0.2547	0.3147	0.3746	0.4346
Th-230 Intake as a Function of U-Nat intake										
1d	2.20E-06	3.08E-06	3.96E-06	4.84E-06	5.71E-06	6.59E-06	7.47E-06	9.23E-06	1.10E-05	1.27E-05
3d	7.99E-05	1.12E-04	1.44E-04	1.76E-04	2.08E-04	2.40E-04	2.72E-04	3.36E-04	3.99E-04	4.63E-04
10d	1.52E-04	2.13E-04	2.74E-04	3.35E-04	3.96E-04	4.57E-04	5.18E-04	6.40E-04	7.61E-04	8.83E-04
20d	3.25E-04	4.55E-04	5.86E-04	7.16E-04	8.46E-04	9.76E-04	1.11E-03	1.37E-03	1.63E-03	1.89E-03
40d	9.52E-04	1.33E-03	1.71E-03	2.10E-03	2.48E-03	2.86E-03	3.24E-03	4.00E-03	4.76E-03	5.52E-03
60d	1.80E-03	2.52E-03	3.24E-03	3.96E-03	4.68E-03	5.40E-03	6.12E-03	7.56E-03	9.00E-03	1.04E-02
80d	2.80E-03	3.93E-03	5.05E-03	6.17E-03	7.29E-03	8.41E-03	9.53E-03	1.18E-02	1.40E-02	1.63E-02
100d	4.07E-03	5.69E-03	7.32E-03	8.95E-03	1.06E-02	1.22E-02	1.38E-02	1.71E-02	2.03E-02	2.36E-02
200d	1.67E-02	2.33E-02	3.00E-02	3.67E-02	4.33E-02	5.00E-02	5.67E-02	7.00E-02	8.33E-02	9.67E-02
300d	4.59E-02	6.43E-02	8.26E-02	1.01E-01	1.19E-01	1.38E-01	1.56E-01	1.93E-01	2.29E-01	2.66E-01
400d	8.88E-02	1.24E-01	1.60E-01	1.95E-01	2.31E-01	2.66E-01	3.02E-01	3.73E-01	4.44E-01	5.15E-01
Ac-227 Intake as Function of U-Nat Intake										
1d	4.84E-08	6.77E-08	8.70E-08	1.06E-07	1.26E-07	1.45E-07	1.64E-07	2.03E-07	2.42E-07	2.80E-07
3d	1.76E-06	2.46E-06	3.16E-06	3.87E-06	4.57E-06	5.27E-06	5.98E-06	7.38E-06	8.79E-06	1.02E-05
10d	3.35E-06	4.69E-06	6.03E-06	7.37E-06	8.71E-06	1.01E-05	1.14E-05	1.41E-05	1.68E-05	1.94E-05
20d	7.16E-06	1.00E-05	1.29E-05	1.57E-05	1.86E-05	2.15E-05	2.43E-05	3.01E-05	3.58E-05	4.15E-05
40d	2.10E-05	2.93E-05	3.77E-05	4.61E-05	5.45E-05	6.29E-05	7.12E-05	8.80E-05	1.05E-04	1.22E-04
60d	3.96E-05	5.54E-05	7.12E-05	8.71E-05	1.03E-04	1.19E-04	1.35E-04	1.66E-04	1.98E-04	2.30E-04
80d	6.17E-05	8.64E-05	1.11E-04	1.36E-04	1.60E-04	1.85E-04	2.10E-04	2.59E-04	3.08E-04	3.58E-04
100d	8.95E-05	1.25E-04	1.61E-04	1.97E-04	2.33E-04	2.68E-04	3.04E-04	3.76E-04	4.47E-04	5.19E-04
200d	3.67E-04	5.13E-04	6.60E-04	8.07E-04	9.53E-04	1.10E-03	1.25E-03	1.54E-03	1.83E-03	2.13E-03
300d	1.01E-03	1.41E-03	1.82E-03	2.22E-03	2.63E-03	3.03E-03	3.43E-03	4.24E-03	5.05E-03	5.86E-03
400d	1.95E-03	2.73E-03	3.52E-03	4.30E-03	5.08E-03	5.86E-03	6.64E-03	8.20E-03	9.77E-03	1.13E-02

2.4.2 Determine the Uranium Dose

While the uranium dose is not directly pertinent to it use as a surrogate, it is provided here for completeness. The dose due to the uranium intake is estimated by multiplying the intake, calculated in the step above, by the DCF for the organ of concern and/or the whole body. In Table A-8, the bone surface (most limiting organ for tailings material) dose, due to uranium, is calculated for a range of sample of results.

Table A-8
Committed Equivalent Dose (mrem) the Bone Surface from Uranium Corresponding to the Intakes Calculated in Table A-6

µg/sa	0.05	0.07	0.09	0.11	0.13	0.15	0.17	0.21	0.25	0.29
Dpm/sa	0.0749	**0.1049**	**0.1349**	**0.1648**	**0.1948**	0.2248	0.2547	0.3147	0.3746	0.4346
1d	7.10E-03	9.94E-03	1.28E-02	1.56E-02	1.85E-02	2.13E-02	2.41E-02	2.98E-02	3.55E-02	4.12E-02
3d	2.58E-01	3.61E-01	4.64E-01	5.68E-01	6.71E-01	7.74E-01	8.77E-01	1.08E+00	1.29E+00	1.50E+00
10d	4.92E-01	6.89E-01	8.85E-01	1.08E+00	1.28E+00	1.48E+00	1.67E+00	2.07E+00	2.46E+00	2.85E+00
20d	1.05E+00	1.47E+00	1.89E+00	2.31E+00	2.73E+00	3.15E+00	3.57E+00	4.41E+00	5.25E+00	6.09E+00
40d	3.08E+00	4.31E+00	5.54E+00	6.77E+00	8.00E+00	9.23E+00	1.05E+01	1.29E+01	1.54E+01	1.78E+01
60d	5.81E+00	8.13E+00	1.05E+01	1.28E+01	1.51E+01	1.74E+01	1.98E+01	2.44E+01	2.91E+01	3.37E+01
80d	9.06E+00	1.27E+01	1.63E+01	1.99E+01	2.35E+01	2.72E+01	3.08E+01	3.80E+01	4.53E+01	5.25E+01
100d	1.31E+01	1.84E+01	2.36E+01	2.89E+01	3.41E+01	3.94E+01	4.46E+01	5.52E+01	6.57E+01	7.62E+01
200d	5.38E+01	7.53E+01	9.69E+01	1.18E+02	1.40E+02	1.61E+02	1.83E+02	2.26E+02	2.69E+02	3.12E+02
300d	1.48E+02	2.07E+02	2.67E+02	3.26E+02	3.85E+02	4.45E+02	5.04E+02	6.22E+02	7.41E+02	8.60E+02
400d	2.87E+02	4.01E+02	5.16E+02	6.31E+02	7.45E+02	8.60E+02	9.75E+02	1.20E+03	1.43E+03	1.66E+03

It is apparent from Table A-7 that the dose consequence from uranium is relatively small. Keep in mind that the dose limit to an organ or tissue is 50 rem (50,000 mrem) so a calculation of a few mrem is insignificant in comparison.

2.4.2 Dose Effect Scaling

The are several way to scale the other tailings radionuclides to the intake or dose of uranium however the approach used to produced the charts of this section is as follows.

The derived uranium DCF (Section 2.2) is compared against the DCFs of Ac-227 and Th-230 and a multiple is developed. This multiple equals the Ac-227 or Th-230 DCF for a particular organ or whole body divided by the derived uranium DCF. An example is provided below.

Example 2 – Dose Scaling Against U-Nat
Givens:
Ac-227 Bone Surface DCF = 7.04E7 mrem/µCi
U-nat Bone Surface DCF = 3.89E4 mrem/µCi
> The dose multiple for Ac-227 is 7.04E7 divided by 3.89E4 or 1809.4. In other words, Ac-227 produces 1809.4 mrem per 1 mrem of uranium dose to the bone. In the same way, a dose multiple of 13.3 is developed for Th-230.
> and since the uranium DCF is in terms of mrem/µC-U-nat
The scaling factor becomes: Ac-227 (Target Organ Dose in mrem) per µCi-U-Nat intake.

Thus, by determining the U-nat intake, the dose due to Ac-227, Th-230, or any radionuclide in a known or knowable ratio may be determined. This approach is used to complete Table A-9 for and is also used to complete the charts found at the end of this section.

Table A-9 provides the committed equivalent dose for Ac-227 and Th-230 to the bone, based upon scaling factors and the dose developed in Table A-7. The overall range is limited to .05 to 0.19 µg/sample.

Table A-9
Committed Equivalent Dose, Bone Surface
from Ac-227 and Th-230 as Estimated based on Uranium Bioassay Results

Post Intake	µg/sa (U-nat)> dpm/sa (U-nat)>	0.050 0.075	0.070 0.105	0.090 0.135	0.110 0.165	0.130 0.195	0.150 0.225	0.170 0.255	0.190 0.285
1	**BS Ac-227 (mrem)**	3.40E+00	4.77E+00	6.13E+00	7.49E+00	8.85E+00	1.02E+01	1.16E+01	1.29E+01
3	**2% is 1.0 rem**	1.24E+02	1.73E+02	2.23E+02	2.72E+02	3.22E+02	3.71E+02	4.21E+02	4.70E+02
10		2.36E+02	3.30E+02	4.25E+02	5.19E+02	6.13E+02	7.08E+02	8.02E+02	8.96E+02
20		5.04E+02	7.05E+02	9.07E+02	1.11E+03	1.31E+03	1.51E+03	1.71E+03	1.91E+03
40		1.47E+03	2.06E+03	2.65E+03	3.24E+03	3.83E+03	4.42E+03	5.01E+03	5.60E+03
60	**10% is 5.0 rem**	2.79E+03	3.90E+03	5.02E+03	6.13E+03	7.25E+03	8.36E+03	9.47E+03	1.06E+04
80		4.34E+03	6.08E+03	7.82E+03	9.56E+03	1.13E+04	1.30E+04	1.48E+04	1.65E+04
100		6.30E+03	8.82E+03	1.13E+04	1.39E+04	1.64E+04	1.89E+04	2.14E+04	2.39E+04
200		2.58E+04	3.61E+04	4.65E+04	5.68E+04	6.71E+04	7.74E+04	8.78E+04	9.81E+04
300	**100% is 50.0 rem**	7.11E+04	9.95E+04	1.28E+05	1.56E+05	1.85E+05	2.13E+05	2.42E+05	2.70E+05
400		1.38E+05	1.93E+05	2.48E+05	3.03E+05	3.58E+05	4.13E+05	4.68E+05	5.23E+05
1	**BS Th-230 (mrem)**	1.14E+00	1.60E+00	2.05E+00	2.51E+00	2.97E+00	3.42E+00	3.88E+00	4.33E+00
3	**2% is 1.0 rem**	4.15E+01	5.80E+01	7.46E+01	9.12E+01	1.08E+02	1.24E+02	1.41E+02	1.58E+02
10		7.90E+01	1.11E+02	1.42E+02	1.74E+02	2.06E+02	2.37E+02	2.69E+02	3.00E+02
20		1.69E+02	2.36E+02	3.04E+02	3.71E+02	4.39E+02	5.06E+02	5.74E+02	6.42E+02
40		4.94E+02	6.92E+02	8.90E+02	1.09E+03	1.29E+03	1.48E+03	1.68E+03	1.88E+03
60		9.34E+02	1.31E+03	1.68E+03	2.05E+03	2.43E+03	2.80E+03	3.17E+03	3.55E+03
80		1.46E+03	2.04E+03	2.62E+03	3.20E+03	3.78E+03	4.37E+03	4.95E+03	5.53E+03
100	**10% is 5.0 rem**	2.11E+03	2.95E+03	3.80E+03	4.64E+03	5.49E+03	6.33E+03	7.18E+03	8.02E+03
200		8.65E+03	1.21E+04	1.56E+04	1.90E+04	2.25E+04	2.59E+04	2.94E+04	3.29E+04
300		2.38E+04	3.33E+04	4.29E+04	5.24E+04	6.19E+04	7.15E+04	8.10E+04	9.05E+04
400		4.61E+04	6.45E+04	8.29E+04	1.01E+05	1.20E+05	1.38E+05	1.57E+05	1.75E+05

2.4.3 Setting the Screening Value and Final Conclusions

Based upon the analysis of this section several charts were produced to allow for a visual inspection of where a screening value should be placed. Keeping in mind that the approach applied herein is likely to be conservative based upon an acute, rather than a chronic, uranium intake scenario. Still, it appears that a screening value can be set between what is detectable above background and what might be considered significant dose. However, this approach will not reach a monitoring objective of 10 – 25 mrem on a quarterly basis – it is standard to set the minimum detectable dose at ≤ 100 mrem divided by monitoring frequency (e.g., 100 mrem/4 to 6 sample points per year). In regards to bioassay urine sampling, it is recognized that a technology shortfall exists for several of the UMTRA Tailings radionuclides; most severely so for Ac-227, Pa-231, and Th-230. Thus, air sampling should be considered first when conducting dose assessment and the bioassay program should be looked as providing confirmation of the effectiveness of work controls. In the absence of confirmed uptakes the bioassay program may also serve to document and produce dose reports.

To support the setting of a reasonable screening value, Chart 1 – 4 were produced to graphically depict the dose implicated due to various uranium results range from 0.05 to 0.29 µg/sample assuming an acute intake 40 and 60 days prior to sample collection. It is apparent that Ac-227 is the most limiting radionuclide under the dose modeling assumptions provided in this appendix; producing between 1 and 10 rem to the bone at 0.15 µg/sample U-nat, post 40 and 60 days intake respectively. Notes and narratives are produced directly on the charts, but overall a screening value between 0.05 and 0.12 µg/sample above background and a collection frequency of 40 to 50 days, or the most exposed group (or a representative sample of the most exposed group), would be reasonable.

Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

Chart 1 – Th-230 Committed Equivalent Dose

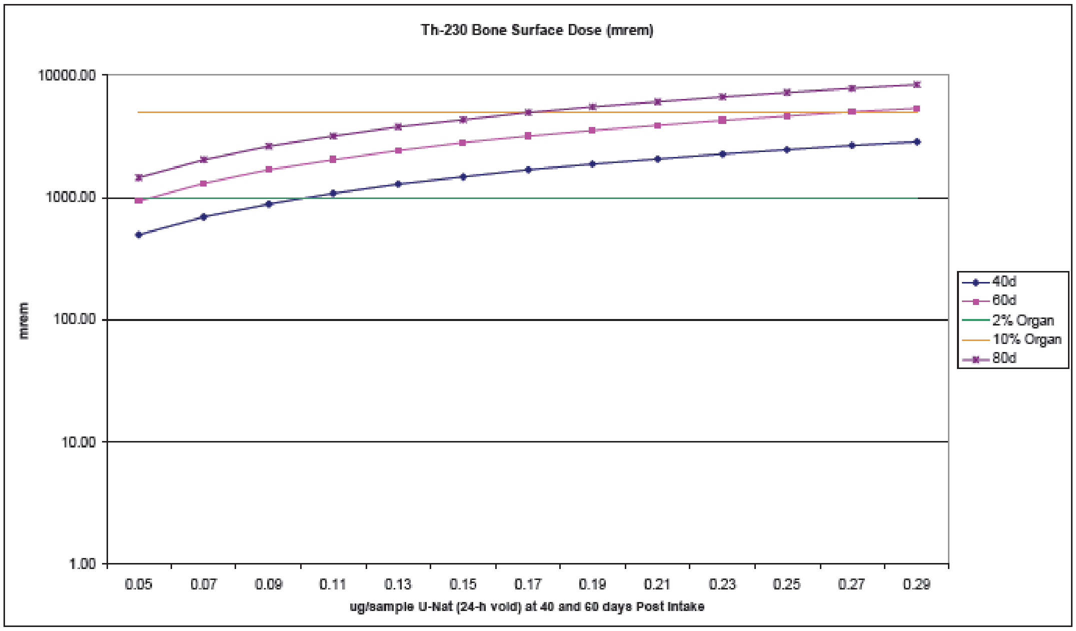
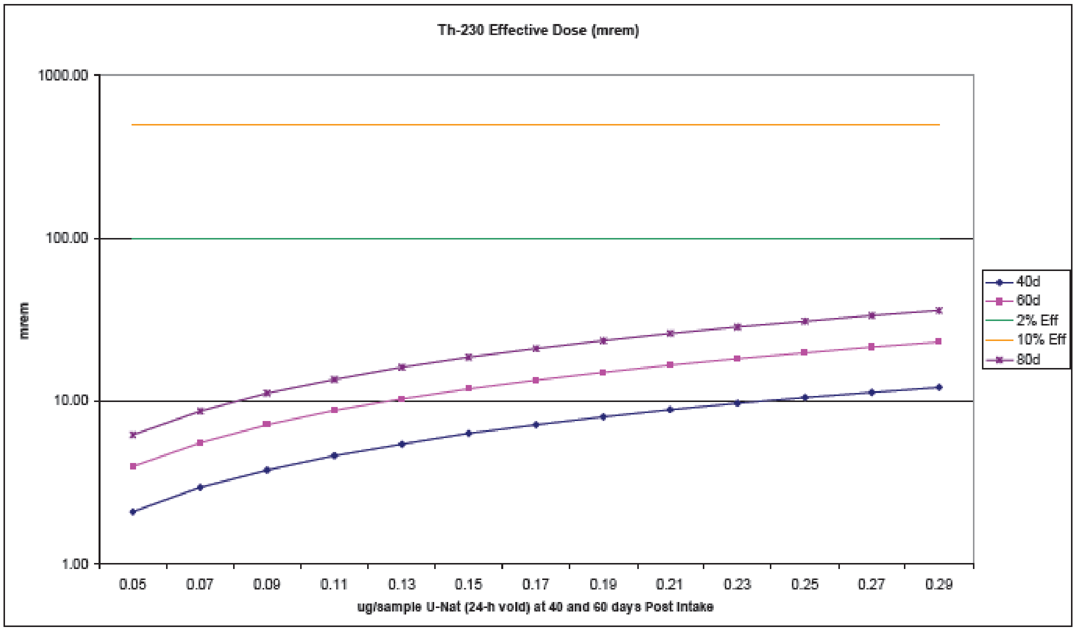


Chart 2 – Th-230 Committed Effective Dose



Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

Chart 3 – Ac-227 Committed Equivalent Dose

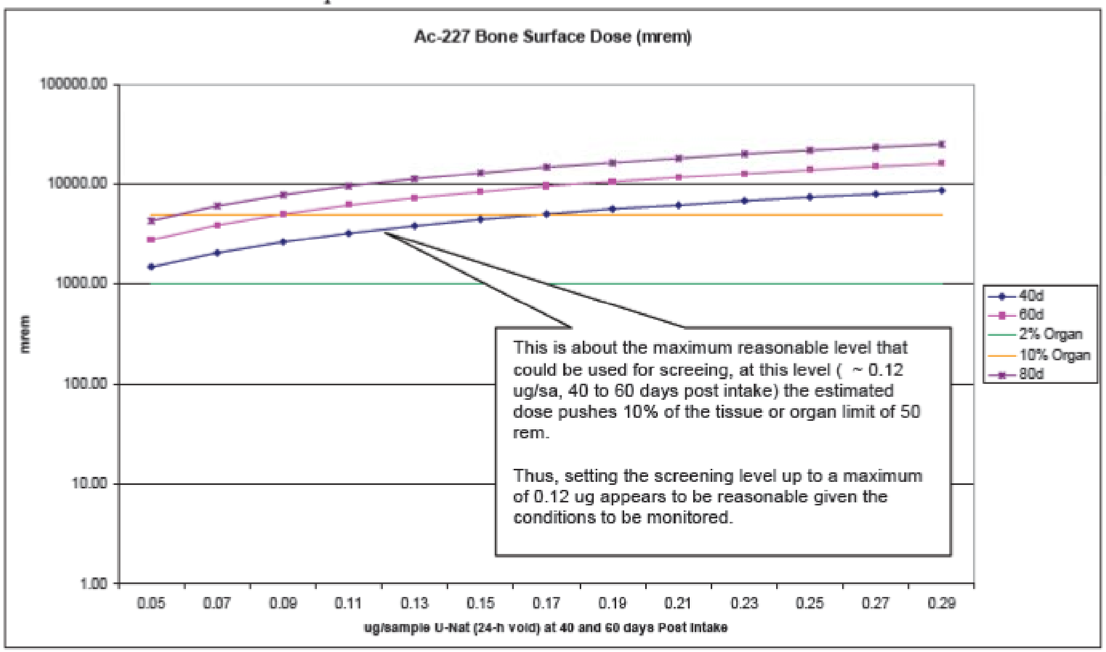
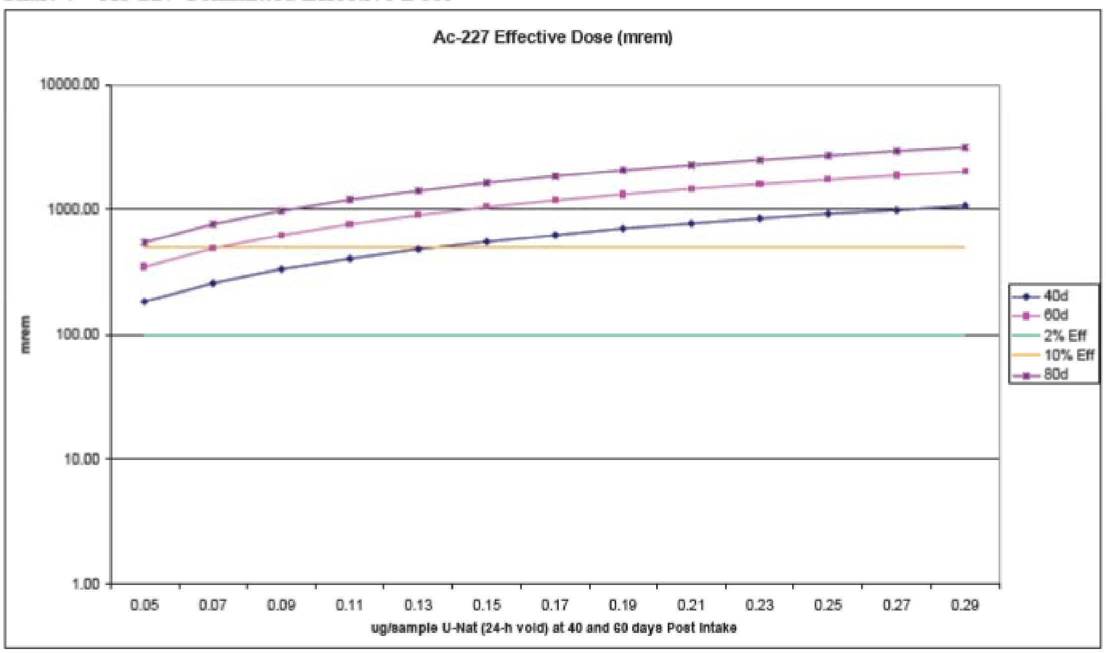


Chart 4 – Ac-227 Committed Effective Dose



Attachment 1. Characterization of Mill Tailings in Terms of Internal Dose Contributions, Bioassay Measurement Restrictions and Requirements, and Screening Level Techniques (continued)

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