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EPRI Alpha Monitoring and Control Guidelines for Operating Nuclear Power Stations, Revision 2

2013 TECHNICAL REPORT

EPRI Alpha Monitoring and Control Guidelines for Operating Nuclear Power Stations, Revision 2

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Product Description

In 2006, the Electric Power Research Institute (EPRI) first published the *EPRI Alpha Monitoring Guidelines for Operating Nuclear Stations* (EPRI report 1013509), to provide standardized guidance for monitoring alpha contamination. Minor revisions were made to the guidelines, which were re-issued in 2009 (1019500). Most US and some international utilities have implemented the guidelines, and several areas of improvement were identified to enhance information to support the monitoring and protection of workers. EPRI has performed a major revision of the document to provide the necessary additional guidance.

Background

Due to the restrictive annual intake limits for alpha emitting radionuclides, the nuclear power industry has developed a standardized, graded approach to monitoring and protecting workers. This report provides guidelines, developed by a group of international radiation safety professionals, for monitoring and protecting workers from alpha emitting radionuclides at pressurized water reactor (PWR), boiling water reactor (BWR) and pressurized heavy water reactor (PHWR) nuclear power plants. The project team developed these guidelines to demonstrate compliance with US regulatory standards for monitoring and protecting workers. In addition, the guidelines are written in consideration of the values and quantities specified by the International Atomic Energy Agency (IAEA) so they may also be applied to international facilities.

Objective

To provide guidelines for a standardized and graded approach to monitoring alpha contamination in the workplace, and for protecting workers against alpha hazards.

Approach

A group of radiation safety professionals representing twenty six US and six international nuclear companies shared their collective experiences and expertise to develop, by consensus, these guidelines.

Results

This document provides guidance on how to identify the presence of alpha emitting radionuclides in operating nuclear reactors; a risk-informed, graded approach to monitoring alpha emitting radionuclides based on the relative abundance of alpha emitters compared to the beta-gamma emitters; and guidelines on how to protect and train workers and how to monitor individuals for exposure to alpha emitting radionuclides. To support guideline implementation, the document has several appendices, including a summary of the monitoring guidelines, information on source term assessments, the technical bases of the guidelines, radon compensation, and instrumentation with examples for work control and internal dose assessment.

Applications, Value, and Use

This document replaces the previous versions of the *EPRI Alpha Monitoring Guidelines for Operating Nuclear Power Stations* (2006-1013509, 2009- 1019500) and supersedes the old companion document, *Program Considerations for Addressing Alpha Emitting Radionuclides at Nuclear Power Plants* (1003126). It addresses an industry need to provide specific technical guidance for monitoring and protecting workers in the presence of alpha contamination at nuclear power plants. Each utility should examine its plant-specific situation to determine how to best implement this guidance.

Keywords

Air sampling
Alpha particle
Alpha radiation
Contamination surveys
Derived Air Concentration
Dosimetry
Radiation protection
Transuranics

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List of Acronyms

ALI	Annual Limit on Intake
ALARA	As Low as Reasonably Achievable
AMAD	Activity Median Aerodynamic Diameter
ANSI	American National Safety Institute
BWR	Boiling Water Reactor
CAM	Continuous Air Monitors
CANDU	Canada Deuterium Uranium Reactor
CDE	Committed Dose Equivalent
CED	Committed Effective Dose
CEDE	Committed Effective Dose Equivalent
CRD	Control Rod Drives
DAC	Derived Air Concentration
GM	Geiger Mueller
HEPA	High Efficiency Particulate Air
HP	Health Physics
ICRP	International Commission on Radiation Protection
IRF	Intake Retention Fraction
INPO	Institute of Nuclear Operations
JIT	Just In Time Training
LLD	Lower Limit of Detection
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
NCRP	National Council on Radiation Protection and Measurements
NRC	United States Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guide, issued by NRC
OE	Operating Experience
OEI	Occupationally Exposed Individual
PAS	Personal Air Sampler
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
PPE	Personal Protective Equipment
RCA	Radiation Controlled Area
RWP	Radiation Work Permit

RWCU	Reactor Water Clean Up
SER	Significant Event Reports
SOER	Significant Operating Experience Reports
TEDE	Total Effective Dose Equivalent
TRU	Transuranic Elements
WBC	Whole Body Count

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Section 1: Introduction

These guidelines pertain to alpha monitoring and control in nuclear power plants. Alpha emitting radionuclides have a significantly lower Annual Limit on Intake than beta-gamma emitting nuclides typically encountered in a nuclear power plant and, when not properly identified and controlled, can result in significant dose to workers. Since it takes orders of magnitude less of ingested or inhaled alpha emitters to produce the equivalent risk as beta-gamma emitting radionuclides commonly encountered in nuclear power plants, the rigor in detecting and controlling alpha is to be commensurate with the risk.

The content of these guidelines consolidates the best practices from member plants represented by the committee working groups that developed this document.

These guidelines use some words with specific meanings. The following identifies how these words should be interpreted:

Shall indicates a requirement. **Requirements are in bold text and underlined**.

Recommended or **Should** indicates a recommendation. The item is recognized as good practice. In the event that the station elects not to implement the stated recommendation, they will need to provide a justification in accordance with site procedures and guidelines. **Recommendations are in bold text.**

May or ***Consider*** denotes a practice that some members have found beneficial. Justification is not required to discount it. *Items to be considered are in italicized text.*

All guideline statements which are requirements or recommendations are numbered, using the format [GS-x].

Appendices contain supplemental information intended to aid implementation of these guidelines. Nothing in the Appendices is to be construed as a requirement, unless explicitly denoted in the guidelines.

These guidelines for monitoring alpha emitting radionuclides at nuclear power plants comply with US regulatory requirements. In addition, the guidelines are written in consideration of the values and quantities of the International Atomic Energy Agency so they may be applied to international committee member facilities.

The processes for identifying, monitoring, and controlling alpha contamination are fundamentally no different than they are for beta-gamma contamination. Likewise, the hazard from alpha contamination is solely a function of the magnitude of the level of the contamination. However, in operating nuclear power plants, a proportionally significant amount of the contamination present in the source term mixture is beta-gamma contamination. As such, the controls applied to beta-gamma contamination are often adequate to control the alpha contamination present.

These guidelines provide a risk-informed, graded approach to monitoring based on the relative abundance of alpha emitters as compared to the beta-gamma emitters. This graded approach is not to replace identification and monitoring for the alpha hazard, but to allow for circumstances in which beta-gamma controls will be adequate to protect workers from both the beta-gamma and potential alpha contamination hazard.

These guidelines recommend that plant areas and systems be classified according to the abundance of loose alpha contamination relative to the presence of loose beta-gamma contamination as follows:

- **Level I Areas: Minimal**
- **Level II Areas: Significant**
- **Level III Areas: Elevated [GS-1]**

In Level I Areas, the relative abundance of loose alpha contamination compared with beta-gamma contamination is minimal. Internal exposure from loose alpha emitters is not likely to exceed 10% of the total internal dose. Actions are recommended to verify the low abundance of alpha emitters when high contamination or high airborne radioactivity is present to confirm that alpha has not become a more significant risk than beta-gamma contamination.

In Level II Areas, the relative abundance of loose alpha contamination compared with beta-gamma contamination is significant. Alpha emitters are likely to contribute more than 10% of the internal dose based on inhalation. Loose contamination survey and airborne activity action levels are intended to alert radiation safety personnel to protect workers from the presence of alpha emitters. These action levels trigger specific monitoring for loose contamination and airborne activity to allow for the alpha hazard to be monitored and controlled appropriately.

In Level III Areas, the relative abundance of loose alpha contamination compared with beta-gamma contamination is elevated. Internal exposure from the loose alpha emitters is likely to exceed 90% of the total internal dose based on inhalation. Therefore, alpha is the primary inhalation hazard to be monitored and controlled.

Appendix A contains a one-page matrix listing trigger levels and recommended actions for Level I, II, and III Areas.

The risk-based approach involves monitoring for the presence of alpha emitters once their presence is suspected; however, controls for alpha are required relative to the overall risk. This includes adequate monitoring to assess the level of the hazard, adequate work controls to prevent exposure, appropriate response to radiological incidents and adequate monitoring of exposed personnel.

The action levels and practices described in these guidelines are not a substitution for regulations, but reflect best practices identified and approved by industry consensus. **As part of implementation of these guidelines, plant radiation protection organizations should assess their programs against the contents of this document. [GS-2]**

Section 2: Defining the Alpha Source Term (Facility Characterization)

In operating nuclear plants, the primary source of alpha emitters is irradiated fuel leaking from fuel pin cladding defects and, to a lesser extent, irradiated tramp fuel materials. Transuranic nuclides, such as americium, plutonium and curium are formed in irradiated uranium fuel by neutron activation and decay predominantly by alpha emission in the energy range of 4 to 6 MeV. Alpha contamination is therefore most commonly associated with systems and components associated with fuel such as the reactor coolant system, spent fuel pool, and the associated radioactive waste systems.

Each operating nuclear facility shall establish procedures to characterize its alpha source term. [GS-3]

Characterizing the alpha source term at a nuclear power plant includes:

- Assessing historical and current fuel cladding defects to identify if there is likely to be transuranic activity in the crud layers in the primary reactor system or associated components, systems, or equipment,
- Determining the radionuclide distribution of alpha emitting radionuclides in loose alpha contamination or airborne activity, when detected,
- Defining the beta-gamma to alpha ratios in loose contamination in the plant areas, in systems, or in airborne activity, and
- Determining the alpha contamination levels in plant areas and systems.

2.1 Assessing Historical and Current Fuel Cladding Defects

Fuel cladding defects or events involving fuel can result in persistent alpha contamination in the primary circuit even when these defects occurred much earlier in the plant's history. See INPO Significant Operating Experience Report 90-02, "Nuclear Fuel Defects" along with other related industry experience reports referenced in Chapter 2.6. Fuel cladding defects or incidents or events involving fuel in the reactor, either in the past or recently, can result in higher radiation, contamination, airborne radioactivity levels, higher beta radiation energies and a higher relative abundance of alpha activity either contained in crud layers or loose in the system. Most alpha emitters are long lived and will not be removed by decay. They therefore remain in the systems for a long time.

Radiation Protection should be cognizant of the fuel failure history of the plant back to the commencement of reactor operations. [GS-4]

An assessment of fuel performance during the plant's operating life can provide insights into potential fixed and loose alpha contamination levels. **If historical information is not known or clear (e.g. fuel cladding failures occurred earlier in the reactor's history or prior to the adoption of radiochemistry programs, which began to detect smaller defects), then it should be assumed that fuel cladding defects have occurred and that there is potential fixed or loose alpha contamination in the primary system crud layers. [GS-5]** It is important that there is alignment in understanding of the significance of nuclear fuel defects from a radiation protection perspective between Nuclear Fuels, Chemistry and Radiation Protection Departments. For example, pinhole leaks may not be considered a significant defect as far as fuel and chemistry performance is concerned, but could release low levels of transuranics into the primary system. Operating experience has shown transuranics can be present without evidence of other fission products.

2.2 Determining the Alpha Radionuclide Distribution

The nuclide distribution of irradiated fuel has been thoroughly studied. Appendix B, Table B-2 provides an example for the distribution of transuranic nuclides in spent fuel one-year and ten-years after discharge from a reactor core for a light water reactor. The principal transuranic nuclides of interest for radiation safety include isotopes of curium, plutonium, and americium. At discharge from the core, curium-242 is the most abundant alpha emitting nuclide. However, ten years after discharge, curium-242 has largely decayed. The net effect is that "older" alpha contamination has a more restrictive effective derived air concentration (DAC) and annual limit on intake (ALI).

An alpha derived air concentration should be established. [GS-6] Three alternatives for determining the alpha nuclide distribution at a facility include:

- Assuming the most restrictive nuclide in the mixture;
- Determining a site-specific alpha nuclide distribution for use throughout the site, and
- Determining the alpha nuclide distribution for specific areas of the plant

A generally conservative approach is to assume an alpha DAC equal to the most restrictive nuclide expected to be present. In the US, this will typically be americium-241, with a DAC[1] of 3E-12 μCi/cc (0.11 Bq/m^3). In newer revisions of ICRP, applicable in other countries, Pu-239 is typically the most restrictive DAC[2].

[1] ICRP30

[2] ICRP68

Alternatively, a site-specific nuclide distribution can be established. Use caution to ensure that this remains accurate during future fuel cycles or when fixed alpha contamination is disturbed. *A facility may also choose to establish area-specific nuclide distributions, appropriate for areas where significant levels of "old/aged" alpha contamination are present.*

The contribution of Pu-241 towards the effective DAC should be included in the development of an appropriate DAC. [GS-7] Appendix B, "Source Term Assessment" provides an example of how a nuclide distribution can be used to calculate an effective DAC and stochastic ALI for inhalation and ingestion. This Appendix also identifies how the quantity of Pu-241 (a very low energy beta emitter not detected by standard counting methods) in the transuranic mix can reduce the composite "alpha" DAC value.

2.3 Defining Beta-Gamma to Alpha Ratios

The significance of alpha contamination depends on its relative abundance compared to beta-gamma contamination. It is therefore convenient to define the term "activity ratio".

ACTIVITY RATIO = beta-gamma activity† / alpha activity[3]

†As determined with a frisker, ion chamber, counter or gamma spectroscopy

Likewise, for air sampling, the relative significance of alpha compared to beta-gamma can be calculated using the "DAC fraction ratio".

$$DAC_{FractionRatio} = \frac{fDAC_{\alpha}}{fDAC_{\beta\gamma}} \qquad \textit{Eq. 2-1}$$

where:

$f\mathrm{DAC}_{\alpha}$ - is the total alpha activity divided by its DAC value

$f\mathrm{DAC}_{\beta\gamma}$ - is the sum of each beta-gamma emitting nuclide's concentration divided by its corresponding DAC value

Alpha DAC-fractions refer to licensed radioactivity (long lived) with no contribution from natural sources. See Appendix C for information on interference from short lived alpha emitting radon daughters.

Defining beta-gamma to alpha ratios in loose contamination or the DAC fraction ratio for air sampling helps identify those areas of the plant where more concentrated alpha monitoring and controls are likely to be required. It also helps to identify an appropriate amount of monitoring required in areas where loose

[3] Activity must be in the same units and on the same smear

alpha contamination hazards or airborne alpha activity are not expected to be present.

Appendix D provides examples of the types of instrumentation and laboratory techniques, which are required for the detection of alpha activity as well as their limitations, including cross talk and MDA considerations. The LLD typically associated with higher count rates may not be appropriate for alpha. See the specific LLD calculation associated with low count rates.

2.4 Determining Contamination Levels

Smears should be taken from primary plant areas and systems within the plant Radiologically Controlled Area, when accessible, concentrating on those areas where transuranics are most likely to be present. [GS-8]

Alpha characterization should not be based solely on the samples of dry activated waste collected for waste classification purposes.

Smears should be analyzed for beta-gamma and alpha activity. [GS-9] *Consider more detailed radionuclide analysis where alpha contamination is present.* **There should be a sufficient number of smears and, where available, sufficient activity on the smears to define contamination levels and classification. [GS-10]**

The extent of the site characterization will depend on the fuel failures that have occurred at the site. Sites that have no or minimal history of fuel failures will not require the same degree of scrutiny as sites with a significant history of fuel failures, although caution should be taken to ensure this has been adequately verified as described in Chapter 2.1.

The loose contamination characterization should include areas with potentially different levels of transuranics or radionuclide distributions relative to the rest of the plant. [GS-11] Appendix B provides examples. The radionuclide distribution on surfaces may be altered with time as a result of various physical and chemical processes.

The characterization should include areas where "old/aged" alpha contamination may be present, which could exhibit lower, more restrictive beta-gamma to alpha ratios. [GS-12]

Examples are:

- Radioactive waste storage buildings containing legacy waste
- Radioactive waste treatment systems
- Tanks and vessels that contained primary fluid
- Pressurizer heater sleeves
- Reactor Head Control Rod Drive Pressure Housing

- Any other system components, items or special tooling which may have been in contact with the primary system or spent fuel pool and have been removed from the reactor for long periods of time.
- Contamination beneath primary system component heat insulation that may have been exposed to primary fluid contamination during a defect fuel cycle
- Contamination which has migrated to other areas, (e.g. floor drains or sumps via spills) or to other systems connected to the original contamination area
- Areas where paint or other fixatives may have been used to affix historical contamination should also be considered.

As part of the characterization, it may also be useful to examine contamination in crud, or on components removed from the primary system (artifacts).

Another potential source of alpha contamination can be equipment brought to the site from another facility with an unknown alpha contamination history. Sealed alpha sources which have been handled improperly can also be a source of alpha contamination. **Procedures should include alpha assessment of equipment received from other sites. [GS-13]**

Alpha contamination characterization should be maintained current. [GS-14]
This may include verifying the contamination characterization:

- *Following a fuel failure*
- *Once per fuel cycle in a plant system (or in the fueling machine for CANDU reactors)*
- *When entering an unknown area, or working with an unknown component associated with the primary system, especially if there is a history of failed fuel*
- *Following extended plant shutdown (beyond normal outage and including decommissioning), and*
- *Following any system chemical decontamination.*

Indication of alpha abundance higher than previously assessed or expected should be evaluated to determine the extent and additional controls required. [GS-15]

2.5 Alpha Classification of Plant Areas and Systems

Plant areas and systems should be classified as Level I, II, or III Alpha Areas consistent with Table 2-1 and based on the results of loose alpha contamination characterization and on the results of any previous air samples. The number of smears taken to perform characterization should be sufficient to adequately characterize the hazard. [GS-16]

The classification levels in Table 2-1 below are derived from the potential for inhalation of Am-241 compared to Co-60 as shown in Appendix E (using ICRP-30 dose coefficients). Note that if the most representative contaminants are not Am-241 and Co-60, then these levels may no longer be conservative.

In an area where loose contamination has a beta-gamma to alpha ratio of >30,000:1 (a Level I area), the primary hazard is beta-gamma. In an area where loose contamination has a beta-gamma to alpha ratio of <300:1 (a Level III area), the primary hazard is alpha. In a Level II area, the alpha hazard can range between 10 and 90 % of the dose, if inhaled. Therefore, depending on the actual ratio within this category, the main hazard may be alpha or beta-gamma. The range of relative abundance of alpha activity to beta-gamma activity associated with each "Area Level" is defined in Table 2-1 below.

Analysis of air sample data can also provide additional support to the classification through comparison with the DAC fraction ratio as shown in Table 2-1 below.

Table 2-1
Alpha Hazard Classification[1,2]

Activity Ratio ($\beta\gamma/\alpha$)	LEVEL I AREAS (Minimal) >30,000	LEVEL II AREAS (Significant) 30,000 – 300	LEVEL III AREAS (Elevated) <300
% dose from alpha in inhaled material	**<10**	**10-90**	**>90**
DAC Fraction Ratio ($\alpha/\beta\gamma$)	<0.1	0.1 – 10	>10

1. Based on ICRP30 DAC values for Co-60 and Am-241. This may not be conservative for the use of other radioisotopes or other ICRP references.
2. *Areas with low alpha activity levels, such as less than 20 dpm/100 cm^2, may be assigned Level I Areas.*

The results of the alpha characterization should be recorded, and areas of the plant and systems classified as Level I, II or III. The results of the alpha characterization should be made accessible as needed to appropriate RP personnel. [GS-17]

This classification is used only for the purposes of identifying the relative alpha hazard of loose contamination in an area compared with beta-gamma. The classification in itself does not determine work controls because the bands are too broad. The actual ratio for the job at hand and many other factors determine the work controls.

2.6 Operating Experience Reports

Listed below are some useful INPO Operating Experience Reports [OE], Significant Operating Experience Reports [SOER], Significant Event Reports [SER], Just In Time Operating Experience [JIT], and NRC reports, citing industry experiences related to transuranic contamination.

OE 34763	Multiple Alpha Intakes During Pipe Preparation Following Valve Removal, October 18, 2011
OE33431	Unrecognized Alpha Contamination Levels May have Resulted in an Unplanned Internal Contamination to Affected Personnel, March 22, 2011
IN 2011- 02 A 1	Technician Punctures Hand During TRU Waste Remediation Activities, June 14, 2010
OE32020/31303	High Alpha Airborne Levels Require Implementation of Bioassay (in-vitro) Sampling, May 12, 2010
SEN 286	Internal Exposures from Inadequate Alpha Contamination Control, December 2009
OE29433	Contamination on Partially Disassembled Smoke Detectors, July 15, 2009.
OE 29152	Airborne Area Created Inside Decontamination Tent, May 26, 2009
OE28693	Fuel Sipping Equipment Elevated Contamination Levels, April 15, 2009.
OE28239	Failure to Recognize Radiological Hazards During Fuel Reconstitution, December 29, 2008
OE27007	Alpha Uptake from Disassembling Smoke Detectors, May 16, 2008.
OE25842	Indication of Elevated Alpha Contamination After Decontamination Efforts, November 27, 2007.
OED 2007-15	Recent Industry Experiences Involving Alpha Contamination, July 2007.
OE22009	Alpha Contamination from Liquid Americium 241 Source, February 1, 2006.
OE21192	Foreign Plant Sends Highly Contaminated CRDM Test Equipment, August 8, 2005.
OE19696	Two Individuals Became Internally Contaminated During Transfer of Radioactive Filters, update of OE19263, December 12, 2004.
OE19263	Workers Contaminated While Preparing Legacy Filters for Shipment, October 8, 2004; updated by OE19696.

OE18151	Sandblasting Grit Material Found to Contain Naturally Occurring Radioactive Material, April 14, 2004.
SER 3-02 rev1	Radiation Protection and Dose Assessment Deficiencies Result in Ineffective Internal Dose Evaluations and Workers Exiting the Plant Site with Detectable External Radioactive Contamination, February 21, 2003.
JIT – 62	Reactor Cavity and Fuel Transfer Canal Work (Radiological Controls), September 2002.
JIT – 65	Radioactive Filter Handling, September 2002.
OE13889	Greater Than 10 Times Increase in Plant Gross Radioactivity Measurements, June 4, 2002.
OE12475	Operating with Failed Fuel, July 12, 2001.
EA-96-496	Fuel Transfer & Reactor Cavity Airborne Contamination Event of November 1996. NRC Enforcement Action Report Date, April 5, 1999.
IN 97-36	Unplanned Intakes by Worker of Transuranic Airborne Radioactive Material and External Exposure Due to Inadequate Control of Work, NRC Information Notice, August 11, 1997.
SER 3-93	Contamination Events Involving Alpha-Emitting Transuranic Elements, September 2, 1993.
LER 92-007-000	Manual ESF Actuation Initiated Due To Failure of Reactor Building HVAC Coincident With Alpha Contamination, September 30, 1992
OE 5620	Alpha (Americium) Contamination Event, September 22, 1992
SOER 90-02	Nuclear Fuel Defects, July 24, 1990.

Section 3: Alpha Monitoring

Alpha classification helps identify the areas of the plant where more concentrated alpha monitoring and controls are likely to be required. This should not be confused with the real time alpha contamination monitoring that is required in a work area to meet the intent of regulations that require that licensees perform surveys to evaluate the magnitude and extent of the potential radiological hazards in the workplace.

When the relative abundance of alpha activity is low (high $\beta\gamma/\alpha$ activity ratio), few smears are counted for alpha. Likewise, the alpha activity contribution to air samples will be low, and fewer air samples are counted for alpha. As the relative abundance of alpha activity increases (lower activity ratio), more smears are counted for alpha. Likewise, more air samples are taken and counted for alpha. W*here practical to do so, consider alpha counting simultaneous to beta-gamma counting.*

Each plant should ensure the availability and capability of alpha monitoring equipment on their site so that an adequate amount of workplace monitoring can be conducted to determine the alpha hazard and protect workers. [GS-18] This includes the use of counting equipment for smears and air samples and alpha frisking equipment to verify workplace and personal contamination in the field.

The monitoring recommended for area classifications are:

In Level I Areas where alpha contamination is expected to be minor, this should be verified by alpha counting representative[4] beta-gamma activity smears in areas or components with > 100,000 dpm/100cm^2. If any of these smears show alpha contamination levels >100 dpm/100 cm^2, additional smears need to be counted to determine the magnitude and extent of the alpha contamination in the area. Caution should also be taken if the level I classification was assigned simply because alpha smears showed < 20 dpm/100cm^2 to ensure that conditions have not changed, which may change the alpha contamination and therefore the classification. This condition might also warrant additional alpha monitoring. Air samples should be counted for alpha when beta-gamma concentrations exceed 1 DAC. [GS-19]

[4] Representative smears is defined as the number and location of smears that are sufficient to adequately characterize the hazard.

In Level II Areas, representative smears should be counted for alpha activity when the beta-gamma contamination exceeds 20,000 dpm/100 cm^2, when loose contamination levels may change and for work control where appropriate. If any of these smears show alpha contamination levels >100 dpm/100 cm^2, additional smears need to be counted to determine the magnitude and extent of the alpha contamination in the area. Air samples should be counted for alpha if the beta-gamma level on the air sample indicates that the total DAC value (alpha and beta gamma emitters) will be greater than 1. Figure 3-1 identifies when air samples should be counted for alpha i.e. the beta-gamma DAC fraction level, based on the ratio of beta-gamma to alpha activity. [GS-20] For reference, this is provided for Co-60, which is the basis for the guidelines, but also for Cs-137, which is more conservative.

Alternatively, in a Level II area, all air samples may be counted for alpha.

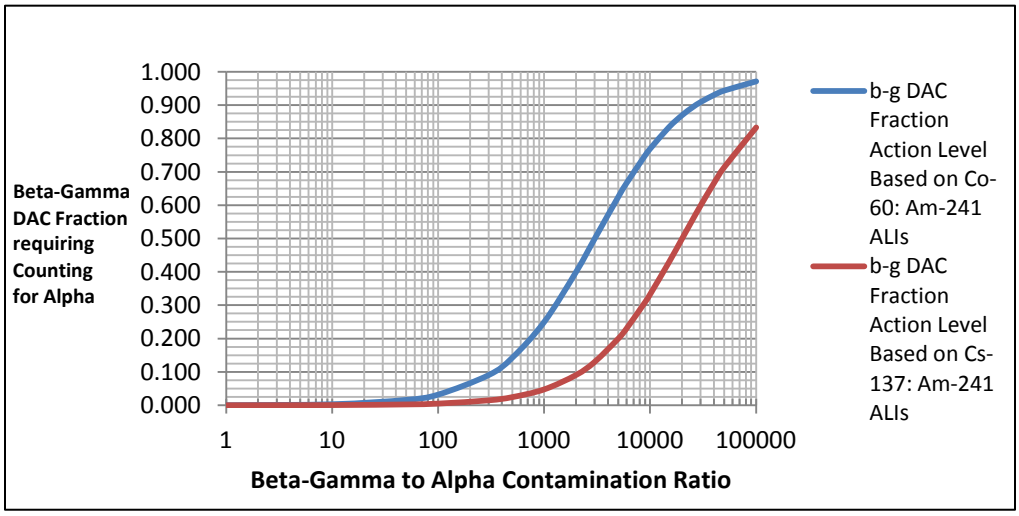

Figure 3-1
Beta-Gamma DAC Fraction versus Activity Ratio

In Level III Areas, a sufficient number of smears should be alpha counted to adequately evaluate the magnitude and extent of the alpha contamination. All air samples should be counted for alpha, or use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements at 0.3 DAC. [GS-21]

This is summarized in Table 3-1 below. Corresponding workplace and individual monitoring actions for each Alpha Area are summarized in the "Area Action Level Matrix" of Appendix A.

Table 3-1
Action Levels for Alpha Monitoring and Counting

Activity Ratio ($\beta\gamma/\alpha$)	**LEVEL I AREAS**[1] **(Minimal) >30,000**	**LEVEL II AREAS (Significant) 30,000 – 300**	**LEVEL III AREAS (Elevated) <300**
% of dose from alpha in inhaled material	**<10**	**10-90**	**>90**
DAC Fraction Ratio ($\alpha/\beta\gamma$)	<0.1	0.1 – 10	>10
Contamination Survey Action Levels	Count representative[1] smears for α in areas with > 100k dpm/100cm^2 $\beta\gamma$	Count representative[1] smears for α in areas >20K dpm/100 cm^2 $\beta\gamma$	Take smears and count specifically for α to adequately evaluate area
	If >100 dpm/100 cm^2 α, take smears and count specifically for α to adequately evaluate area		
Air Sampling Action Levels	If $\sum DAC_{\beta\gamma}$ >1, count air samples for α or use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements[2]	If > beta-gamma DAC Fraction shown in Figure 3-1 relative to the ratio, or > "beta-gamma DAC Fraction Action Level" count air samples for α or use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements[2]	Count all air samples for α or use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements[2]
	If beta-gamma to alpha contamination ratio or DAC Fraction Ratio ($\alpha/\beta\gamma$) is higher than expected for assigned Area Level, re-evaluate Area Level Assignment		

1. Representative smears are defined as the number and location of smears that are sufficient to adequately characterize the hazard.
2. When Continuous Air Monitors are used, they are to be capable of detecting 0.3 DAC alpha.

Table 3-1 is only a minimum guide to assist with determining the extent of alpha monitoring required based on the classification of the area. However, alpha contamination and airborne activity monitoring needs to be conducted as necessary to identify if conditions are changing.

The levels of loose surface contamination used to determine the classification, the type of work being performed and the physical nature of the contamination are used to predict potential airborne radioactivity levels and prescribe appropriate work controls. This is discussed further in Chapter 4.

Section 4: Work Controls

The purpose of this chapter is to provide a description of appropriate alpha work controls for worker protection at nuclear power stations. Work controls are developed and implemented so that each job can be completed efficiently with minimal overall radiological risk, consistent with keeping total effective dose equivalent ALARA.

A systematic approach is appropriate to identify and quantify the risk of generating alpha airborne contamination for each task based on the alpha contamination levels, the nature of the work, and the potential for the activity to re-suspend or to change the alpha contamination levels. *Consider strategies such as changes in job scope, varying work techniques, introduction of technologies, remote tooling, use of containment devices, etc., to avoid the risk from alpha emitters whenever possible. Where risk avoidance is not practical, consider the development of mitigation strategies involving the application of engineering controls, hold points, and/or respiratory protection.*

Although the characterization of the work area, system or component (addressed in Chapter 2) may assist in determining the baseline hazard, work controls should be based on a number of factors, not solely the classification of the work area (Level I, II or III). The actual levels of alpha activity found or suspected and the risk of worker exposure due to the work activity should also be considered. [GS-22] It is possible for an area to have lower levels of alpha contamination and pose a greater risk of worker exposure based on the characteristics of the contamination present and on the potential of the activity performed to create airborne contamination. For instance, an area with 100 dpm/100cm^2 alpha contained in dirt or dust may pose a greater threat to worker exposure than 3000 dpm/100cm^2 alpha contained in an oily film.

The specific work control requirements for entering an alpha Level III area with very low levels of alpha contamination present or very low risk of exposure to alpha hazards may be very minimal. However, the work controls for grinding a surface in an alpha Level II area with similar very low levels of alpha contamination may be more stringent (requiring respiratory protection, PAS), and the work controls for working in high levels of alpha contamination in a Level III area may be very stringent (requiring glove bags, respirators, PAS). The degree of work controls applied will depend on the conditions and hazards associated with the work rather than solely on the characterization. Work controls are graded according to the degree of risk, but implemented consistently for activities with comparable risks.

This chapter provides a step by step process to plan and control work where alpha contamination may be present.

4.1 Risk Assessment

A risk assessment to identify the radiological hazards of the work activity should be conducted prior to initiating the work. The degree of rigor used in the risk assessment should be increased where there is higher risk of generating airborne activity. An appropriate strategy to quantify the risk and determine the degree of rigor to be applied should include use of the most recent alpha characterization along with relevant job history files, a working knowledge of the task being performed, an understanding of the techniques being used to accomplish the task, and knowledge of the physical characteristics and limitations of the work area. [GS-23]

Items requiring special attention due to potentially lower beta-gamma to alpha ratios are identified in Chapter 2.4, Determining Contamination Levels.

4.1.1 Contamination and Airborne Activity Levels

The current characterization of the area should be reviewed and compared to recent survey data to ensure the job plan is based on current conditions. [GS-24]
If reports on previous surveys or work in the area of interest are not available, consider reviewing data from work performed in interconnected or related systems, areas or components, but take caution to ensure that related systems, areas or components are historically similar to those being evaluated.

If the work characterization is not recent or survey results are not representative of the current conditions, a pre-job survey or additional monitoring during the work should be performed. [GS-25]

If systems are suspected of having alpha contamination as indicated by the facility characterization and when aggressive surface destructive work is to be conducted, fixed alpha contamination should be assumed to be present. [GS-26]
Operating experience has also shown that alpha contamination might be masked by dirt, dust or corrosion, and activity levels could be higher below the surface. Aggressive surface destructive work includes all activities which can give rise to re-suspension of contamination, such as grinding, welding, decontamination, sanding, cutting, the use of volatile chemicals, removal of insulation, spraying cold water on hot piping and heating or drying.

4.2 Work Planning Approach

A graded approach towards the planning of the work should be used. Planning should be more rigorous for work where there is a higher potential for workers to be exposed to alpha activity during the conduct of the work based on the contamination levels and work to be performed. [GS-27] *Consider the following*:

- *Involving the work group in the planning process to assist in identification of the specific tasks and techniques to be used.*
- *Reviewing the potential for spreading alpha contamination, the level of activity that could be spread, and the risk this poses to the workers and others in the area.*
- *Identifying the potential for re-suspension of activity from the surface. Even very small amounts of alpha surface contamination can give rise to a high, localized airborne concentration. Most unplanned alpha exposures have resulted from unexpected airborne activity due to re-suspension.*
- *Implementing barriers, such as stop work criteria, to prevent deviation from the original work plan. Unplanned alpha exposures have also resulted when the work plan was changed in the field and this was not communicated to everyone involved with the original briefing.*
- *Making a qualitative or quantitative assessment of the risk of contamination levels and subsequent airborne activity. The risk could be minimal, for instance, non-abrasive work performed on a surface that is less than 20 dpm/100cm^2 requiring only a qualitative assessment, or more significant, requiring a quantitative assessment. Appendix F provides a quantitative example for predicting airborne activity based on contamination levels (note, however, that fixed contamination levels are often not measurable).*
- *Identifying work activities carried out in parallel by other work groups and evaluation of the potential for those activities to create airborne alpha contamination in the work area.*
- *Reviewing relevant operating experience. Information from other departments, such as chemistry, fuel, etc, may also be considered during work planning.*

4.3 Radiological Work Control Considerations

4.3.1 Radiation Work Permit

a) The ALARA planning tools and/or RWP should state if work is to be conducted in an Alpha Level II or III area. [GS-28]

b) The ALARA Plan and/or RWP should state the appropriate radiological hold points and stop work criteria. [GS-29]

c) Each site should establish criteria to define which alpha work plans require approval of the RPM or designee and work group supervision. [GS-30]

4.3.2 Minimizing surface and airborne contamination

Efforts should be made to minimize or eliminate the spread of alpha contamination or the generation of alpha airborne activity. [GS-31] Through dialogue with the work group, identify optimal work methods to reduce the potential for the spread of contamination and/or minimize re-suspension. For example, explore whether a less abrasive method could be used instead of a high-speed cutter to achieve the same result.

Consider decontamination of the area or system. If loose alpha contamination is present, it may be possible to adhere with fixatives. If it is not possible to fix contamination, it may be possible to dampen the surface during the conduct of work to mitigate re-suspension.

Because alpha contamination is difficult to detect and requires enhanced radiation protection controls, consider containing alpha activity at the source to prevent its spread. When avoidance through alternative work strategies is impractical, consider mitigation strategies to minimize the consequence of the alpha risk. Engineering controls are generally preferable, otherwise, the use of respiratory protection may be deemed prudent if it does not substantially increase the workers' overall TEDE.

Appendix F contains some work control examples, applying fixatives, engineering controls, etc.

If a TEDE ALARA evaluation is performed, then the review should include alpha as well as beta-gamma emitting radionuclides in the airborne inventory. [GS-32] Consider that a small change in airborne alpha contamination can result in significant changes in internal dose.

4.3.3 Prevention of cuts, scrapes and punctures

The consequences of intakes through cuts or wounds are more significant than those for intakes through inhalation. Operating experience has shown that there is potential for significant alpha exposure from cuts or wounds.[5]

Avoiding creation of and contact with sharp surfaces is extremely important. For example, experience has shown that cutting pipes using rotary cutters can leave sharp edges. **Appropriate personal protective equipment (e.g. cut or puncture resistant gloves) should be worn by the workers where there is a potential for cuts or scrapes in an alpha contaminated environment. [GS-33]**

See Chapter 4.10 on incident response and Section 5.4.3 for bioassay to assess dose from cuts and scrapes.

4.3.4 Field Monitoring Considerations

Alpha contamination monitoring requirements and airborne alpha concentration level limits are identified in Chapter 3 and Table 3-1. The following chapters describe additional field monitoring. All survey data should be recorded in sufficient detail and in the appropriate units. Beta-gamma to alpha ratios can be compared with expected ratios. *Consider noting on survey documentation, the physical surface condition, e.g. wet, dusty, clean, etc.*

4.3.4.1 Contamination Monitoring of Work Area

Alpha contamination monitoring should be conducted with an appropriate frequency commensurate with the risk of changing radiological conditions. For example, conduct more frequent monitoring during grinding, which could release undetected transuranics from lower oxide layers, as compared to building scaffolding. [GS-34]

Portable alpha contamination monitoring equipment is minimally affected by surrounding beta-gamma dose rates. This allows placement of the alpha instruments close to and even inside the work area. It is important to ensure that an adequate number of portable alpha contamination monitors are available. Information on contamination monitoring and counting methods is provided in Appendix D.

4.3.4.2 Airborne Concentration Monitoring of the Work Area

The low ALI values for alpha emitters demands careful monitoring of the work area. To ensure adequate alpha monitoring of the area:

[5] Type B Accident Investigation: Technician Punctures Hand during TRU Waste remediation activities, June 14, 2010, Savannah River Site

- **General area air samples should be of a sufficient volume and counted for sufficient time using sensitive detection equipment to detect 0.3 DAC alpha. [GS-35]**
- **Caution should be taken in dusty or dirty environments to minimize filter loading since filter loading may mask the quantity of alpha contamination present. [GS-36]** For example, air filters could be changed more frequently, minimize run time.
- **Air samplers should be placed properly to provide the most representative assessment of general airborne activity in the work area. [GS-37]**
- *Consider additional air samplers to verify the integrity of engineering controls, if used.*
- *Where elevated alpha airborne activity is anticipated or where there is a concern for alpha airborne activity to spread beyond the immediate work area, consider the use of alarming air samplers or portable alpha continuous air monitors to provide early warning to personnel inside and around the work area of an unanticipated airborne hazard.* Appendix D provides information on alarming air samplers.
- *Consider changing and counting air samples periodically so that results parallel changes in alpha risk; for example, if the job is long, one sample might be taken during the breeching of a large bore valve in an Alpha Level III system and then a new sample taken during lapping of the valve seat. Consider including air sampling at the boundary of the work area where appropriate to assess exposures to workers outside the immediate area. Where positive air samples are obtained, also consider collecting air samples following work completion to ensure conditions were controlled as anticipated, and any residual airborne contamination is acceptable.*
- Alpha activity on air samples from the decay products from naturally occurring radon gases can interfere with the initial evaluation of alpha activity from the long lived alpha emitters of interest. Appendix C provides methods to assist with the discrimination of long lived activity from radon decay products. Appendix D provides information on alarming air samplers, which may be used to discriminate against radon daughters where high levels of airborne alpha are anticipated. Care needs to be taken to neither under-estimate the presence of long lived alpha emitters, by assuming the presence of naturally occurring decay products, nor to over-estimate the significance of air sampling results.
- **Counting of air samples should be prioritized to ensure high risk air samples are counted in a timely fashion. [GS-38]**
- *Consider retaining air samples with high levels of alpha for further analysis (for example for intake investigation or to support characterization).*
- The use of PAS is not a substitute for general area alpha airborne monitoring.

4.4 Personal Dosimetry

Where there is potential for workers to be exposed to alpha airborne concentrations that indicate a potential dose that exceeds the verification level (100 mrem or 1 mSv committed[6] effective dose) as in ANSI N13.39 (2011), there should be a programmatic approach to confirm that the verification level has not been exceeded, whether or not respiratory protection was used. Dosimetry used for this purpose should include use of:

- **PAS as internal dosimeters, and/or**
- **A periodic (non-incident based) alpha excreta sampling program to confirm that the committed effective dose was not exceeded.**

The programmatic approach to personal dosimetry should be documented in station procedures. [GS-39]

Personal Air Samplers (PAS) should be issued to assess intake of activity for work in Level III areas and in alpha Level II areas where alpha exposure is a concern, for example where aggressive work is being conducted and/or the ratio of beta-gamma to alpha indicates that alpha may be a significant contributor to the airborne hazard. [GS-40]

Exceptions may be made where any of the following are in place:

- When air supplied suits are worn if the industrial hazards associated with the use of PAS outweigh the benefits.
- Where a periodic (non-incident based) alpha excreta sampling program is in place.
- During a specific task evolution where it is concluded that a PAS is not necessary, because the potential for airborne alpha had been evaluated and is considered improbable, and appropriate stop work controls are in place and communicated to the workers.
- Where engineering controls, (e.g. a glove box) adequately contain the source term.

The use of PAS is not a substitute for general area alpha airborne monitoring. **PAS should not be used for posting purposes. [GS-41]**

Chapter 5.1 also provides information on personal air sampling of breathing zone air.

[6] Term "Committed Effective Dose" can be used interchangeably with "Committed Effective Dose Equivalent" with the same meaning – the term reflects the system of dose limitation provided in ICRP 60 as opposed to earlier dose systems.

4.5 Stop Work Controls, Limits and Authorities

Appropriate stop work criteria should be established based on alpha hazards. [GS-42]

Consider the following criteria:

- *Unplanned intake as identified by personnel monitoring*
- *Suspected uptake based on contaminated wound*
- *Unexpected air sample results in the work area*
- *Change in job scope or alpha levels not covered in RWP/ALARA planning documents or change in work activities not discussed during the ALARA and pre-job briefings*
- *Beta-gamma to alpha activity ratio based on in-progress surveys (i.e. contamination swipes or air samples) changes the initial alpha level to a higher level from a Level I to II or III, or from Level II to III*
- *A single contamination smear indicates the beta-gamma to alpha ratio is ≤ 50:1 and this was unexpected and not covered by the pre-job or ALARA/RWP briefing*
- *Loss of or change in the type of any required engineering controls: HEPA ventilation, containment device integrity, respiratory protection equipment failure, failure of other engineering controls, (e.g. failure to maintain area, equipment, component wet or covered with sealant or gel).*

Where the stop work activity indicates that personnel may have been exposed to alpha emitting radionuclides, refer to Chapter 4.10, which provides information on potential responses to inadvertent exposures.

4.6 Monitoring of Equipment, Materials and Personnel

The following controls should be applied:

Efforts should be made wherever possible to reduce the alpha contamination levels on equipment, tools or materials. [GS-43] *Consider conducting this in the work location prior to removing them from the area.*

- **Equipment and materials exiting Level III areas should be properly labeled to adequately inform the workers of the alpha risk. [GS-44]** (Caution: Level II contamination could decay into Level III with long term storage of equipment and materials).
- **Equipment and materials exposed to a beta-gamma to alpha activity ratio ≤ 50:1 should be segregated until surveys/assessments have been conducted prior to release of the item from alpha controls. [GS-45]**
- **Since alpha contamination cannot be detected in crevices or internal surfaces of equipment, particular care should be taken to ensure items that may have been in areas of beta-gamma to alpha activity ratios of ≤ 50:1, are adequately**

evaluated to confirm they are free of detectable alpha contamination before unrestricted release. [GS-46]

- **Personnel should be evaluated and physically surveyed for alpha contamination when exposed to beta-gamma to alpha activity ratios of ≤ 50:1 according to the job work plan. [GS-47]** Note that alpha monitoring equipment cannot easily detect very low levels of alpha contamination during frisking. Therefore, alpha frisking needs to be conducted carefully and slowly in order to properly detect the contamination at the lower levels of detection of the equipment (see Appendix D).

4.7 Selection and Use of Protective Clothing

In Level II or III areas, consider:

- *Using disposable protective clothing instead of re-useable clothing*
- *Wiping down or fixing contamination on protective clothing prior to removal to prevent re-suspension of alpha activity*
- *Providing assistance with removal of protective clothing*
- *Using downdraft step off pads*

4.8 Job Coverage

Consider the appropriate RP coverage. **For aggressive work in known Level III areas or potentially changing conditions, continuous RP coverage should be provided. [GS-48]** *When technicians are assigned to perform surveys or cover work in alpha Level II or Level III areas, consider the following:*

- *Preparation for the conditions to be encountered,*
- *The radiological response necessary to control the work activity, and*
- *How the work activity could significantly alter the current alpha ratio/hazard*

Information that can be used by technicians for preparation of work and job coverage is provided in Appendix G.

4.9 Communications

4.9.1 Radiological Postings

Alpha Level III areas shall be clearly posted to inform workers and radiation protection technicians of this condition. Posting of areas with a beta-gamma to alpha ratio of ≤ 50:1 shall contain similar words that "alpha frisking/monitoring is required upon exit". [GS-49] *Alpha Level II or alpha Level I areas may be posted at the discretion of the plant.*

4.9.2 Briefing of Workers

Worker briefings should discuss the unique aspect of the alpha hazards and controls for the specific task/work activity as described in the ALARA plan, RWP or work instructions for alpha Level II and III areas. This should include communicating to workers the hold points and stop work expectations. [GS-50]

4.10 Investigating Potential Alpha Exposures from Incidents

Where radiological conditions indicate that a worker may have been exposed to unexpected airborne alpha concentrations or to an unplanned intake of alpha emitting radionuclides, an investigation into the extent of exposure should be initiated.

Examples of such conditions meriting further investigation should be identified and documented in station procedures. [GS-51] Examples could include:

- Facial beta-gamma contamination or a positive nasal swipe are detected when beta-gamma to alpha ratios indicate there may be alpha contamination present
- Personnel beta-gamma contamination monitor alarms in the torso region without the confirmed presence of external contamination when beta-gamma to alpha ratios indicate there may be alpha contamination present
- Alpha contamination monitoring results in a work area are higher than expected
- Personnel contamination surveys indicate the presence of alpha contamination on the hands or face
- Personal air sampling results indicate alpha airborne activity
- General air sampling results, either when beta-gamma to alpha ratios indicate the presence of alpha contamination or indicate alpha airborne activity directly
- A wound sustained in an area or on an item where beta-gamma to alpha ratios or alpha monitoring indicates the presence or possible presence of alpha contamination

The investigation should include the following steps:

- **Gather all relevant data concerning the event.** For example, obtain personal air sample results from all personnel involved; investigate personnel contamination levels, contamination in the work area, air sample results from the area, beta-gamma to alpha ratios in the area, alpha or gamma spectrometry data or other information on radionuclide distribution, etc.
- **Estimate the potential dose to the worker from the event**. For example, estimate as indicated by personal air sample results, fixed air sampler results, or from contamination levels in the area using re-suspension factors. Fixed

air samplers can underestimate personal exposures[7] by factors that range from 100 to 1,000, so ensure that the potential dose is not under-estimated by the use of fixed air samplers that are not representative of the worker's breathing zone.

- **For wounds sustained in an area or on an item that is potentially alpha contaminated, monitoring should be conducted on the item that caused the wound as well as on the wound itself.** Alpha monitoring may not be practical or may not be sensitive enough to detect alpha contamination at the level of concern for an intake. In such a case, beta-gamma contamination monitoring of the item, area and the wound itself can be conducted and scaling factors used to estimate the presence of alpha activity based on the characterization of the radionuclide mix from the alpha characterization of the area. This can then be used to estimate the potential magnitude of the intake. Also see Chapter 5.4.3.
- **Initiate further individual monitoring using a graded approach, dependent on the potential dose to the worker as in Table 4-1. Record dose in the individual's dose record:**

Table 4-1
Individual Monitoring Requirements Based on Potential Dose

Potential Dose [1]	Definition[8]	Action	Techniques which can be used
> 10 mrem CEDE	Screening Level	Confirm dose by other means	Whole body counting, PAS, or excreta measurements
> 100 mrem CEDE	Verification level	Validity of dose assignment to be confirmed by individual monitoring[2]	Excreta measurements are preferred technique[3]
> 500 mrem CEDE	Investigation level	Individual measurements must be taken to define the dose more accurately	Extensive excreta sampling should be conducted[3].

[7] NCRP Report No. 127, Chapter 7.4.1

[8] ANSI N13.39 (2011)

Notes:

1. **When the potential dose to the individual cannot properly be determined or remains uncertain, then excreta sampling should be used whenever possible to confirm the magnitude of the intake.**
2. **While a conservative assumption can be used with regard to the assessment of airborne activity, accurate radionuclide distributions should be used in the assignment of dose > 100 mrem.**
3. **For contaminated wounds, urinalysis should be used. [GS-52]**

Individual monitoring techniques are discussed further in Chapter 5.

Communication concerning the amount of internal contamination, an estimate of the effective dose, the dose in perspective and special instructions for sampling should be made with the affected worker(s) during the investigation and especially during follow up excreta sampling since alpha intakes can cause considerable worker concern. [GS-53] *Also consider removing the worker(s) from further exposure to radiation commensurate with the magnitude of the potential intake.*

On completion of the investigation, dose assessments should be prepared as appropriate to the level of intake. [GS-54] Appendix H contains some methods for calculating internal dose.

Section 5: Individual Monitoring

The following techniques can be used to indicate the potential presence of alpha emitters in individuals.

- Personal/Lapel Air Samples
- Measurement of radioactivity in the body
- Measurement of radioactivity in excreta

Each of these techniques has strengths, limitations and uses as discussed in the following chapters.

Any calculated dose exceeding 10 mrem Committed Effective Dose should be recorded in the worker`s exposure history. [GS-55]

5.1 Personal/Lapel Air Samplers

Air sampling from the breathing zone provides reasonable indications that the worker has been exposed to an inhalation hazard from TRU. A breathing zone air sample is one taken within a 25 cm radius of the workers' nose and mouth, usually with air sampling filters attached to the collar or lapel[9]. The location of air samples is important for the evaluation of potential exposure to airborne radionuclides[10]. Fixed air samplers can under or overestimate personal exposures by factors that range from 100 to 1,000[11].

Personal air samplers that are worn by the worker are therefore the method of choice for monitoring workers in areas of airborne alpha activity. Assigning the results of breathing zone air samples from one worker to be representative of co-workers is NOT recommended.

To reduce uncertainties associated with the use of personal air samplers (PAS) for the detection of alpha intakes, care should be taken to ensure that air samples are not contaminated due to improper handling. [GS-56] Efforts should also be made to obtain a lower limit of detection (LLD) of 10 mrem committed effective dose [GS-57] (see Appendix D).

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[9] US Department of Labor, Occupational Safety and Health Administration (OSHA) definitions

[10] NUREG-1400

[11] NCRP Report No. 127, Chapter 7.4.1

The results from PAS can be used to determine individual intake and dose from routine work activities. **Whenever a PAS indicates that a potential exposure may unexpectedly exceed the screening level of 10 mrem committed effective dose[12] as in ANSI N13.39 (2011), action should be initiated to confirm the extent of exposure. [GS-58]** Chapter 4.10 provides information on incident investigation. **Where PAS results indicate potential exposures that exceed the verification level of 100 mrem committed effective dose as in ANSI N13.39 (2011), excreta measurements should be used to investigate and determine the intake of alpha emitting radionuclides. [GS-59]**

5.2 In-Vivo Counts

5.2.1 Lung Counting

Specialized lung counting equipment distinct from whole body counting equipment is available for direct measurement of alpha activity in the lung, although this type of equipment is not typically located at nuclear power stations. It detects the low energy gamma and X rays emitted from Am-241 and Pu-239 among others. Models have been developed to facilitate intake computations from lung burdens found using lung counting. Lung counting can provide useful information for several weeks after an intake, though this timeframe is dependent on the magnitude of the intake.

5.2.2 Whole Body Counting

Whole body counting (WBC) is a technique used for estimating a worker's intake from *gamma emitting radionuclides.* However, most alpha emitting radionuclides are not accompanied by gamma photon emissions with sufficient energy to be detected by whole body counting, especially at the levels of intake of concern. Less than detectable WBC results cannot therefore be used to state no intake of alpha materials occurred because the WBC cannot detect alpha emitting nuclides. Americium-241 (Am-241) emits a low energy photon (59.5 keV) that may be detected by the whole body counter, but whole body counting is of limited use because of low detection efficiency and difficulties in distinguishing Am-241 from background radiation.

Whole body counting may be used to investigate a low level intake from alpha emitting nuclides (for example, above the screening level of 10 mrem committed effective dose[13]), using scaling factors determined from representative characterization samples from the work area at times soon after exposure. However, as the relative abundance of beta-gamma contamination decreases from Level I towards Level II ratios, the ability of whole body counters to detect a low level of internal exposure also decreases.

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[12] Term "Committed Effective Dose" can be used interchangeably with "Committed Effective Dose Equivalent" with the same meaning – the term reflects the system of dose limitation provided in ICRP 60 as opposed to earlier dose systems.

[13] ANSI N13.39 (2011)

Air samples or smears from the work area can be used to determine the scaling factors. **WBC results should not be used as the sole method to assign alpha doses above the verification level (100 mrem Committed Effective Dose). [GS-60]** If an intake of alpha emitting nuclides >100 mrem is considered likely, excreta sampling or lung counting are the preferred methods to assign dose.

5.3 In-Vitro Counts

5.3.1 Excreta

Excreta samples may be used to determine an intake from alpha emitting nuclides following a suspected exposure. This may be prompted by a high result on an air sampler or PAS, high alpha contamination monitoring results or from a contaminated wound. **Where such results indicate potential exposures that exceed the verification level (100 mrem Committed Effective Dose) as in ANSI N13.39 (2011), excreta measurements should be used to investigate and determine the intake. [GS-61]**

To be adequately prepared to respond to potential alpha events, stations should be ready to conduct excreta sampling and perform internal dosimetry assessments. ICRP 54[14], ICRP 78[15], NUREG 4884, and NCRP 87[16] provide guidance on the development of excreta collection procedures. **Adequate excreta collection kits should be available and maintained in a clean area for prompt use if needed along with instructions for their use and adequate storage capability (refrigerated storage is recommended). Vendor arrangements to ensure suitable means for analyzing the samples should be in place to facilitate prompt turnaround of samples. Procedures should be prepared to collect timely excreta samples and to perform internal dosimetry assessments. [GS-62]**

Excreta sampling provides the most accurate measurement of a worker's intake from alpha emitting radionuclides. **A sample lower level of detection (LLD) should be low enough to yield a dose LLD of <100 mrem for the intake under investigation. [GS-63]**

Dosimetry models have been developed which determine absorption, transfer, and excretion of most nuclides of concern. Thus for a given intake either by ingestion, inhalation or injection, models can predict the location and excretion rates of nuclides based on a known intake. Conversely, the same models can be used to estimate an intake, and therefore a dose, based on the activity found in excreta. As such, excreta sampling provides the best means to estimate dose of alpha emitting nuclides.

Use of data from both urine and fecal samples may improve the dose assessment for more significant inhalation scenarios. **The excreta sampling process should**

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[14] ICRP 54, Chapters 4.2 & 4.3

[15] ICRP 78, Chapter 4.2

[16] NCRP Report 87, Chapters 5.3.1, 5.3.2, & 5.3.6

begin as soon as an exposure event is suspected. [GS-64] The magnitude of the exposure should drive the sampling regime, i.e., higher exposures would demand more sampling for longer periods. **For exposures expected to exceed the ANSI verification level (100 mrem), excreta sampling should be initiated as soon as possible following detection of the exposure, and continue for a 24 hour period or until at least one sample is collected (following the first void for urine).[17] Following an exposure that is expected to exceed the investigation level (500 mrem) of ANSI N13.39 (2011), excreta sampling should be collected immediately, and continued for a minimum of 5 days[18]. [GS-65]**

Excretion rates change rapidly for several days after intake. Caution must be exercised in estimating intakes and doses using early results because of the large uncertainties involved with rapidly changing excretion rates. However, early samples give the best detection capability (i.e. detection of intake) due to the much higher activity levels and are easily observable by most detection technologies. The instantaneous excretion rate changes rapidly for the first 48 hours after intake. In addition, the excreta output rate is subject to physiologically related variability. Cumulative sampling helps average out the variations in the early phases after an exposure.

When multiple 24 hour samples taken within the first few weeks after intake are obtained, the excretion pattern can provide information about the relative magnitude of an intake and on the primary route of intake (ingestion versus inhalation). Conservative assumptions can be used on these early bioassay samples to perform preliminary dose estimates. **For intakes resulting in doses exceeding an investigation level (500 mrem) based on initial estimates, further sampling, sometimes extending months after the intake, should be collected to determine the long term components of the biokinetic models. Sampling is not expected to be continuous for this period; rather, samples obtained at appropriate intervals are recommended once the time after exposure exceeds 10 days. [GS-66]**

The excreta rates in urine and feces are dependent on particle size of the inhaled material and also on lung clearance classification. Alpha emitters from power reactors are associated with mostly uranium based fuel, and the alpha contaminants may behave similarly to the fuel matrix as opposed to the physical-chemical form of the alpha emitter. **Since internal dosimetry is a complex subject requiring expert interpretation, when there is limited information on particle size and lung clearance classification, care should be taken in assuming whether urine or fecal analyses are more sensitive [GS-67].** For example, an assumption on lung clearance classification for selection of the most conservative DAC may not be the most conservative assumption in the interpretation of urine or fecal sample results.

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[17] Reg. Guide 8.9, Chapter 3, page 8.9-4

[18] NUREG/CR-4884 Chapters 2.5.2.1, page 17

5.3.1.1 Urine

Urine bioassay detects the nuclides which have been transferred into the bloodstream by absorption in the body fluids. The urine excretion activities of alpha emitters are very low unless the intake was very large.

Alpha spectrometry detection limits for urine samples will only give observable results for a few weeks after an intake unless a very large uptake has occurred. Thermal Ionization Mass Spectrometry (TIMS) has much lower detection limits and will facilitate observable results many years after intake of recordable doses. The disadvantage of TIMS is that, unlike alpha spectrometry, which can detect many nuclides, TIMS can only detect Pu-239 and Pu-240. TIMS also is very expensive, can take a long time to be carried out, and there is very limited commercial availability for such instruments.

Alpha emitting activity in parts of the body other than the lungs or GI tract can only be removed by dissolution into the bloodstream and excretion through the kidneys. Therefore, there will be relatively more activity found in urine bioassay making it the primary tool for alpha contaminated wound events.

Evaluating spot urine measurements as instantaneous rates of excretion can give erroneously high estimates of the intake and associated dose equivalent. [19] In general, cumulative urine samples are preferable to 24-hour urine samples, and both are preferable to spot samples. Note that a series of 24- hour samples can be mathematically combined to create a cumulative sample. For some cases, adequate analytical sensitivity can be achieved only by analysis of several days' excreta.[20] Timed samples are also useful in the days and weeks after the intake to provide information on daily excretion rates as used in most model predictions.

5.3.1.2 Feces

The majority of the inhaled or ingested material is excreted through feces. Fecal bioassay detects nuclides which have entered the upper respiratory passages or GI tract.

Alpha spectrometry of fecal samples may provide useful information for several months after an intake. The fecal compartment will contain a substantial fraction of accumulated intake activity following an inhalation of 1 μm AMAD particles and a greater amount following the inhalation of larger particles.

Cumulative samples are less sensitive to model assumptions and provide a better estimate of the respiratory tract deposition cleared via the fecal excretion pathway. Samples accumulated over a 1-3 day period are useful in the days and weeks after an intake to provide information on daily excretion rates as used in most model predictions.

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[19] NUREG/CR- 4884, pp. 17

[20] ICRP 54, Chapter 4.2, paragraph 59

5.4 Potential Pathways/Routes of Intake

5.4.1 Inhalation

Inhalation is the primary route of intake of TRU in nuclear power plants. Tramp uranium and the various processes that create transuranic particles from the fuel elements produce particles of respirable size[21]. Crud layers containing TRU particulates are converted into an airborne inhalation hazard by cutting, grinding, welding etc. The smaller particles can be deposited deep in the lungs and remain there for considerable periods of time and constitute the inhalation retention model. Smaller particles can remain airborne for sufficient time to be carried by ventilation currents and present an inhalation hazard to workers well away from the actual worksite. Though it is possible to measure the particle sizes generated during work activities, this requires specialized equipment and is rarely done. Assuming that an intake follows the inhalation retention model produces conservative calculated dose estimates. Large particles are cleared from the upper respiratory region and subsequently swallowed; therefore, the excretion pattern may appear to be closer to ingestion excretion patterns than an inhalation excretion pattern.

5.4.2 Ingestion

It is unlikely that TRU ingestion events would occur independently of an airborne hazard; however, ingestions could happen through transfer of activity from the hands to the mouth when an individual is not wearing respiratory protection or during doffing of protective clothing. An individual knowledgeable in alpha dosimetry may be able to differentiate an ingestion intake from an inhalation intake using early and extended excreta sampling.

5.4.3 Contaminated Wounds

More than 90% of contaminated wounds occur on the arms and hands (primarily fingers) and most of the wounds involve punctures; chemical burns account for the bulk of the remainder.[22]

The uptake of TRU into the systemic circulation from a wound is variable and depends on the physical and chemical form of the nuclide, depth of the wound and extent of injury, treatment administered, and time elapsed between injury and treatment.[23]

In order to assess the dosimetric and medical consequences of a contaminated wound, it is necessary to identify, and quantify the radionuclides present.

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[21] M. D. Dorrian and M. R. Baily. Particle Size Distributions of Radioactive Aerosols Measured in Workplaces. Radiation Protection Dosimetry, Vol. 60, No2. pp 119-133; 1995.

[22] NCRP 156, page iii

[23] NCRP 156, page iii

Typically, direct measurements at the wound site with an external detector and excreta sampling and analysis are performed.

Methods for the direct monitoring of the wound are described in NCRP 156.[24] Smears of the work area and structure or device that caused the wound can assist in the determination of the nuclides involved and their chemical form.

Urinalysis should be used to determine if TRU has been absorbed into the systemic circulation from the wound site[25] when exposures are expected to exceed the ANSI verification level (100 mrem). [GS-68]

NCRP 156 provides additional guidance in the event that medical intervention is necessary and guidance on assessing the local dose to tissue from the radioactive material in the wound.

Intake Retention Fractions for contaminated wounds are developed as described in NCRP 156.[26] Ishiqure[27] describes the practical implementation of the NCRP wound model for the prediction of systemic behavior (including the retention and excretion rate) of some important nuclides encountered in the nuclear industry. Toohey, *et al*,[28,29] published an extensive list of dose coefficients for determining the dose from contaminated wounds.

A comprehensive treatment of the process for assessment of contaminated wounds is beyond the scope of these Guidelines. Collectively, the work of the NCRP, Ishiqure, and Toohey, *et al*, provide a comprehensive description for the monitoring and assessment of contaminated wounds, bioassay and biokinetic modeling of systemic intakes from contaminated wounds, assessment of dose, and decision levels for potential medical intervention.

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[24] NCRP 156, Chapter 5.1

[25] NCRP 156, Chapter 5.3.1

[26] NCRP 156, Chapter 4

[27] Ishiqure, N., "Implementation of the NCRP wound Modle for Interpretation of Bioassay Data for Intake of Radionuclides Through Contaminated Wounds", *j. Radiat Res.* 2009 may;50(3):267-76

[28] Toohey, R.E., *et al*, "Dose Coefficients for Intakes of Radionuclides Via Contaminated Wounds", *Health Phys.*2011 May;100(5):508-14.

[29] Toohey, R.E., *et al*, "Dose Coefficients for Intakes of Radionuclides Via Contaminated Wounds", available from the Oak Ridge Institute for Science and Education and http://orise.orau.gov/reacts/resources/retention-intake-publication.aspx

Section 6: Training

The presence of transuranics radionuclides in the workplace has an impact on virtually every aspect of the radiation protection program. The degree of impact is dependent on the extent of contamination of the facility. The presence of transuranics can pose potential and/or actual hazards in the facility. In order to manage these hazards, effective communications and training within all levels of the organization are important to ensure that the proper perspective and resources are used to manage the transuranic source term.

To meet regulatory training objectives, the nuclear industry uses a multi-tiered instructional approach including:

- Radiation Protection Personnel Training (including RP supervisors)
- Management Training (managers and supervisors of nuclear fuels, chemistry, maintenance, plant management, etc.)
- General Employee Training / Radiation Worker Training

6.1 Radiation Protection Personnel Training

Radiation protection personnel should be trained to understand the internal exposure potential of transuranics and the measurement challenges posed by alpha radiation. [GS-69] *Information on the following elements can be found in the referenced parts of this guideline and may be provided in the training program for radiation protection personnel.* Training will need to be supplemented with technical health physics and facility specific information:

- Alpha contamination
  - Why alpha is important and needs to be controlled (Chapter 1)
  - Where alpha is found and detection challenges (Section 2.1)
  - Operating experience on events involving alpha activity (Section 2.6)
  - Fundamentals of alpha radiation
  - Comparisons of ALI and DAC Values (Section 2.2, Appendix B)
- Classification and characterization of the workplace
  - Approach to alpha classification (Chapter 1, Section 2.5)
  - Results of characterization of the workplace (facility specific)
  - Areas not characterized or ongoing characterization plans (facility specific)

- Typical and historic alpha nuclide distributions and plant distributions (Section 2.2. Appendix B)
- Postings (Section 4.9.1)

- Planning work controls
  - The relationship between work controls and characterization (Chapter 4)
  - Conditions that might be of concern, of previous concern or that have given rise to incidents (Section 4.1 and 4.2)
  - Assessment of hazards from work activities (Sections 4.2, 4.3.1, 4.3.2)
  - Minimizing the alpha hazard (Section 4.3.1)
  - Conservative decision making (Section 4.3.1)
  - Wounds, cuts and scrapes (Section 4.3.3)
- Monitoring for alpha
  - When monitoring is required (Chapter 3, Section 4.3.4)
  - Instrumentation used for alpha monitoring (Appendix D)
- Alpha Frisking Instrumentation and Techniques
- Alpha Counting Instrumentation and Sensitivities
  - Practical Air sampling considerations (Section 4.3.4.2)
- Radon Decay Chain and impact on assessment of air samples (Appendices C and D)
  - Use of Personal Air Samplers (Appendix H, Sections 4.4, 5.1)
- Field Work Controls
  - Stop work controls (Section 4.5)
  - Monitoring of personnel and materials (Section 4.6)
  - Use of PPE (Section 4.7)
  - RWPs and radiological briefings (Section 4.9)
  - Job Coverage considerations (Section 4.8, Appendix G)
- Individual monitoring techniques
  - Breathing zone air samples (Section 5.1)
  - Whole Body Counting (Section 5.2)
  - Excreta (Section 5.3.1)
- Overview of Assessment of Intakes
  - Routes of Intake of alpha (Section 5.4)
  - Routine exposures (Section 4.4)
  - Incident Exposures (Section 4.10)
- Scenario based studies for specific work examples
  - Classification of areas
  - Air sample interpretation
  - Work control examples
  - Work coverage considerations

Consider the use of hands-on practical training (also known as dynamic learning activities) to keep radiation technicians current on alpha monitoring techniques when not frequently employed and to reinforce proper practices in alpha areas. Consider the use of mock-up training to demonstrate the use of engineering controls when employed for alpha control if different from controls employed for beta-gamma engineering controls.

6.2 Management Training

Management should be informed of the impact of transuranics, including the risk of alpha in the workplace, the tools needed to provide adequate worker protection while maintaining a productive working environment and the potential consequences of alpha in the workplace. [GS-70] *Discussions or activities on the following elements may be included in information provided to management:*

- Alpha contamination
  - Why alpha is important and needs to be controlled
  - Where alpha is found and detection challenges
  - Operating experience on events involving alpha activity
  - Internal exposure potential of TRU (comparison of ALI and DAC Values)
  - Consequences of alpha events
- Characterization of the Workplace
  - Approach
  - Results of characterization of the workplace, including historic information
- Effective Monitoring Instrumentation
  - Types and numbers required
- Impact of alpha in the workplace
  - Work activities that can give rise to alpha
  - Impact on work planning
  - Work-place Impact (e.g., alpha frisking, personal air sampling, etc.)
  - Engineering controls
  - Respirator use
  - Potential for increase in Whole Body Counts and bioassay sampling
  - Dosimetry arrangements in the event of a suspected intake
  - Documentation of internal dose
  - Radiation protection staffing and training

6.3 General Employee/Radiation Worker Training

The level of Radiation Worker Training provided should be commensurate with the level of radiological risk. [GS-71] For facilities where all areas are characterized as Level I, the course content described in INPO ACAD-00-007, "Guidelines for Radiation Worker and Radiological Respiratory Protection

Training" provides sufficient training. **For facilities where numerous areas are characterized as Level II or Level III, training should be enhanced to include additional topics. [GS-72]** *The following elements may be provided in the training for radiation workers:*

- Alpha Contamination
  - Why alpha is important and needs to be controlled
  - Where alpha is found and detection challenges
  - Operating experience on events involving alpha activity
- Classification of the Workplace
  - Approach
  - Results of classification of the workplace, including historic information
- Impact of alpha in the workplace
  - Work activities that can give rise to alpha
  - Impact on work planning
  - Work place impact (alpha frisking, contamination control, personal air sampling, bioassay, etc)

Hands-on practical and mock-up training may also be appropriate for radiation workers who are working in Level III areas or with high levels of alpha contamination.

Appendix A: Area Action Level Matrix

Table A-1
Area Action Level Matrix

	Level I Areas (Minimal)	**Level II Areas (Significant)**	**Level III Areas (Elevated)**
Activity Ratio[1,2] ($\beta\gamma/\alpha$)	>30,000	30,000 – 300	<300
Contamination Survey Action Levels	Count representative smears for α in areas with > 100kdpm/100cm^2 $\beta\gamma$[3]	Count representative smears for α in areas >20K dpm/100 cm^2 $\beta\gamma$	Take smears and count specifically for α to adequately evaluate area
	If >100 dpm/100 cm^2 α, take smears and count specifically for α to adequately evaluate area		
DAC-Fraction Ratio ($\alpha/\beta\gamma$)	<0.1	0.1 – 10	>10
Air Sampling Action Levels	If fDAC$_{\beta\gamma}$ >1, count air sample for αor use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements[4]	If > beta-gamma DAC Fraction shown in Figure 3-1 relative to the ratio, or > "beta-gamma DAC Fraction Action Level" count air samples for α or use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements[4]	Count all air samples for α or use Continuous Air Monitors (CAMs) capable of direct alpha activity measurements[4]
	If beta-gamma to alpha contamination ratio or DAC-Fraction Ratio ($\alpha/\beta\gamma$) is higher than expected for assigned Area Level, re-evaluate Area Level Assignment		
WBC[5]	Consider where alpha CEDE is >10 mrem		
Personal Air Samplers	In Level II and III issue Personal Air Samplers as per Section 4.4[6]		
Alpha Frisk			Personnel Alpha Frisk each person at ≤50:1 $\beta\gamma/\alpha$

Table A-1 (continued)
Area Action Level Matrix

	Level I Areas (Minimal)	Level II Areas (Significant)	Level III Areas (Elevated)
Alpha Internal Dose	Internal dose from alpha emitters exceeding 10 mrem (100μSv) Committed Effective Dose should be recorded		
Bioassay	Urine and/or fecal analysis should be performed when alpha intake exceeds 100 mrem (1 mSv) Committed Effective Dose alpha		

[1]See Appendix E for technical basis
[2] Properly characterized areas with low alpha activity levels, such as less than 20 dpm/100 cm^2, may be assigned Level I Areas.

[3]Caution should also be taken if the level I classification was assigned simply because alpha smears showed < 20 dpm/100cm^2 to ensure that conditions have not changed, which may change the alpha contamination and therefore the classification. This condition might also warrant additional alpha monitoring.

[4]When Continuous Air Monitors are used, these are to be capable of detecting 0.3 DAC alpha.

[5]See Chapter 5, Internal Dosimetry, for the limitations of whole body counting

[6]In alpha Level II areas where alpha exposure is a concern, for example where aggressive work is being conducted and/or the ratio of beta-gamma to alpha indicates that alpha may be a significant contributor to the airborne hazard.

Appendix B: Source Term Assessment

Table B-1
List of Common Systems/Components Associated With Alpha Contamination

System / Area	Component / Activity
Reactor Cavity (PWR)	Fuel Transfer System Flanges and Gates
	Fuel Handling Components and Tools
	Reactor Vessel Studs / Stud Holes
Primary Heat Transport System Components (PHWR)	Primary and auxilliary system components
	D2O storage tanks, transfer valves, sampling points
Steam Generator	Diaphragm/Inspection Doors
	Nozzle Dam Installation and Removal
Pressurizer (PWR)	Heaters / Heater Sleeves/Manway
Fuel Handling Components (PHWR)	Fuelling machine heads, trolley package filters, IX columns, port maintenance areas
Dry-Well (BWR)	Control Rod Drives
	Recirculation / Jet Pumps
	MSIV Internals
	SRVs
	RWCU valves
Spent Fuel Pool	Sludge / Filters
	Pumps
	Heat Exchanger
	Fuel Handling Tools and Equipment
Reactor Head	Interior, seal leaks, flange
Reactor Coolant Pumps	Seals
Primary Circuit Clean-up Systems	Pumps, valves and associated equipment
Emergency Coolant Injection System	D2O loop valves
Radwaste Processing Systems	Drums, tools, equipment, waste storage tanks, pumps and associated equipment
Abandoned systems	Any associated equipment
Primary System Components in storage	When opening containers, when access is available, when putting into service, etc

Note: This list is not exhaustive and is intended only to provide typical system examples in water cooled reactors.

Table B-2
Principal Transuranic Nuclides in Spent Fuel from a LWR

Nuclide	Half-Life (yr)	Decay Mode	@ 1 yr	@ 10 yr
Pu-238	88	Alpha	18%	38%
Pu-239/240	24131	Alpha	6%	14%
Am-241	432	Alpha	2%	30%
Cm-242	0.45	Alpha, feeds Pu-238	62%	0%
Cm-243/244	18	Alpha, feeds Pu-240	12%	18%
TOTAL			100%	100%
Pu-241	14.4	Beta, feeds Am-241	924% of Alpha	1390% of Alpha

Reference: Light Water Reactor Nuclear Fuel Cycle, CRC Press, 1981

Radionuclide Distribution

A site-specific radionuclide distribution can be established using laboratory analysis of several representative smears or air samples. Analytical laboratories need samples with sufficient gross alpha activity to obtain reliable results. This should be confirmed with the laboratory. Typically, this gross alpha activity ranges from several hundred to several thousand disintegrations per minute.

The example below, which arbitrarily assumes a 20% distribution between each of the principal alpha-emitting nuclides, illustrates how a representative site-specific nuclide distribution can be used to calculate an "effective derived air concentration", "effective inhalation annual limit on intake", and an "effective ingestion annual limit on intake".

Table B-3
Effective Derived Air Concentration (EFF DAC)[30]

Nuclide	Fraction of Total Activity (*f*ACT)	DAC (μCi/cc)	*f*ACT/DAC Ratio	Percentage Total DAC
Pu-238	0.2	8E-12	2.5E+10	15%
Pu-239/240	0.2	7E-12	2.9E+10	18%
Am-241	0.2	3E-12	6.7E+10	41%
Cm-242	0.2	1E-10	2.0E+09	1%
Cm-244	0.2	5E-12	4.0E+10	25%
Total Alpha	1.0		1.6E+11	100%
EFF DAC *		6E-12		

* Effective DAC = 1 / Σ *f*ACT/DAC Ratio

Another important transuranic nuclide is plutonium-241. This pure beta emitter can be taken into account by including it in the effective alpha DAC calculation to reduce the effective alpha DAC since Pu-241 cannot be detected by alpha counting. In this example, plutonium-241 activity is assumed to be 10 times the total of all alpha emitting transuranic radionuclides.

Table B-3 (continued to show addition of Pu-241)
Effective Derived Air Concentration (EFF DAC)

Nuclide	Fraction of Total Activity (*f*ACT)	DAC (uCi/cc)	*f*ACT/DAC Ratio	Percentage Total DAC
Pu-241	10	3E-10	3.3E+10	21%
Total Alpha & Beta				121%
EFF DAC*		5E-12		

* The Effective DAC is lowered by including the Pu-241 fACT/DAC Ratio in the Effective DAC calculation (1 / Σ fACT/DAC Ratio) for alpha emitters

[30] Oxide values from ICRP Publication 30

Table B-4
Effective Inhalation Stochastic Annual Limit on Intake (EFF HALI)[1]

Nuclide	Fraction of Total Activity (fACT)	HALI (μCi)	fACT / HALI Ratio	Percentage Total HALI
Pu-238	0.2	0.02	1.0E+01	20%
Pu-239/240	0.2	0.02	1.0E+01	20%
Am-241	0.2	0.01	2.0E+01	39%
Cm-242	0.2	0.3	6.7E-01	1%
Cm-244	0.2	0.02	1.0E+01	20%
Total Alpha	1.0		5.1E+01	100%
EFF HALI [2]		0.02		

[1]Based on ICRP 30

[2]Effective HALI = 1 / Σ *f*ACT/HALI Ratio

Table B-4 (continued to show addition of Pu-241)
Effective Inhalation Stochastic Annual Limit on Intake (EFF HALI)[1]

Nuclide	Fraction of Total Activity (*f*ACT)	HALI (uCi)	*f*ACT / HALI Ratio	Percentage Total HALI
Pu-241	10	1	1.0E+01	20%
Total Alpha & Beta				120%
EFF HALI *		0.016		

* The Effective HALI is lowered by including the Pu-241 *f*ACT/HALI ratio in the EFF HALI calculation (1 / Σ *f*ACT/HALI Ratio).

Table B-5
Effective Ingestion Stochastic Annual Limit on Intake (EFF GALI)[1]

Nuclide	Fraction of Total Activity (*f*ACT)	GALI (µCi)	*f*ACT / GALI Ratio	Percentage Total GALI
Pu-238	0.2	2	1.0E-01	18%
Pu-239/240	0.2	1	2.0E-01	35%
Am-241	0.2	1	2.0E-01	35%
Cm-242	0.2	50	4.0E-03	1%
Cm-244	0.2	3	6.7E-02	12%
Total Alpha	1.0		5.7E-01	100%
EFF GALI [2]		1.8		

[1]Based on ICRP 30
[2]Effective GALI = 1 / Σ *f*ACT/GALI Ratio

Table B-5 (continued to show addition of Pu-241)
Effective Ingestion Stochastic Annual Limit on Intake (EFF GALI)[1]

Nuclide	Fraction of Total Activity (*f*ACT)	GALI (µCi)	*f*ACT / GALI Ratio	Percentage Total GALI
Pu-241	10	70	1.4E-01	25%
Total Alpha & Beta				125%
EFF GALI *		1.4		

*The Effective GALI is lowered by including the Pu-241 *f*ACT/GALI ratio in the EFF GALI calculation (1 / Σ *f*ACT/GALI Ratio)

Appendix C: Radon Compensation

Air sampling is used at nuclear power plants to measure airborne radioactivity in the workplace so that the area can be properly posted and to verify that controls intended to limit airborne radioactivity are effective.

Timely assessment of the alpha airborne radioactivity is made difficult because of the presence of natural radioactivity – radon & radon decay daughters. Delaying the alpha analysis of air samples for 4-hours is sufficient to allow for a significant fraction of the natural radioactivity to decay. Longer delay times are needed to allow for complete decay. This appendix describes methods and instrumentation that can be used to improve the timeliness of analyzing air samples for licensed alpha-emitting radioactive material. If high levels of alpha airborne activity are anticipated, consider other techniques which can discriminate against radon daughters, such as the use of real time alarming air samplers, instead of the decay technique. These instruments are described further in Appendix D.

Additionally, the field check methods described in this appendix over short time periods are not a substitute for gross alpha counts of air samples after the samples have been decayed from natural radioactivity for longer than 24 hours.

Sources and Decay of Natural Airborne Radioactivity

Natural airborne radioactivity arises from the gaseous decay products in the ^{238}U and ^{232}Th decay series. Since ^{238}U and ^{232}Th are ubiquitous in the earth's crust, their gaseous decay products – ^{222}Rn (radon) and ^{220}Rn (thoron), emanate from soil and concrete. While the relative concentration of uranium and thorium in soil are often similar, airborne ^{220}Rn activity is only about 2 percent of the radon activity. This is due to the shorter half-life of ^{220}Rn (55 seconds) as compared to ^{222}Rn (3.8 days), permitting radon to become airborne from greater depth of material (Ref. C.1).

Gaseous ^{222}Rn and ^{220}Rn decay into particulate radioactive daughters that are collected on particulate air sample filters. ^{222}Rn and ^{220}Rn gases pass through particulate filters and are not collected.

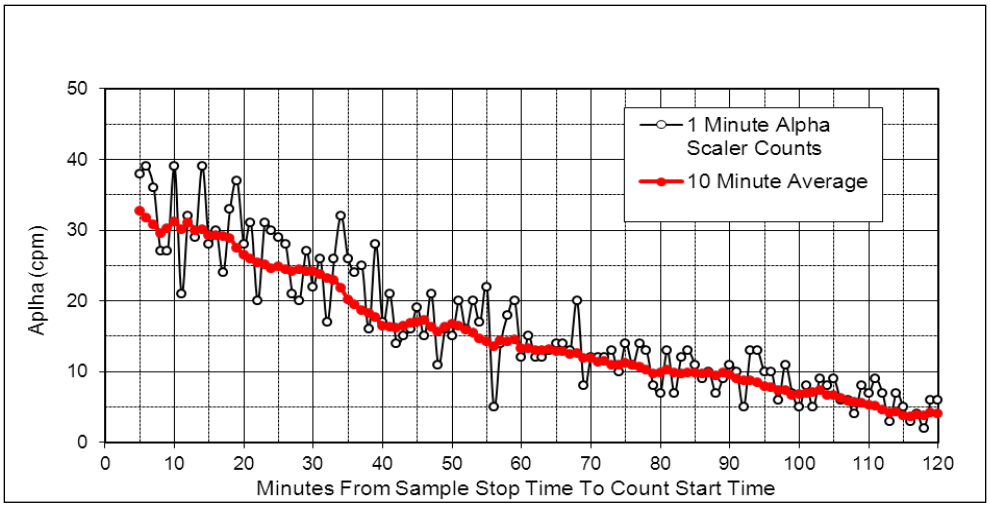

Figure C-1
Decay of Natural Radioactivity

Figure C-1 shows the decay of an air sample taken inside a typical concrete building over the first 2-hours. The solid line shows the improvement in accuracy of a 10-minute scintillation-based alpha scaler gross alpha count compared with a 1-minute count.

Note: The radon decay effective half-life will depend on the relative abundances of the ^{222}Rn and ^{220}Rn which in turn depend on the local geology.[31]

After the air sampler has been turned off, the decay of the ^{222}Rn and ^{220}Rn daughters is fairly complicated. The effective half-life of alpha activity collected on a particulate air sample filter changes over time and is also dependent upon several factors including the air sample collection time, and the rate at which room air exchanges with outdoor air.

The natural alpha activity from this air sample is well over 10 times the alpha DAC for licensee alpha activity. Even after 2-hours decay, it's about two times the alpha DAC. Without compensating for natural activity, field checks of alpha activity produce false positive results.

Compensation Methods

Background Method

The "Background Method" compensates for radon by subtracting a background air sample taken in the general vicinity but unaffected by any work-in-progress. The background air sample can be taken at the work location before the job

[31] The effective half-life will vary from 40 minutes for ^{222}Rn decay products and 10.6 hours for ^{220}Rn decay products.

begins. Both the background and job coverage air samples are taken with the same type of air sampler and have about the same volume (+/-50%).

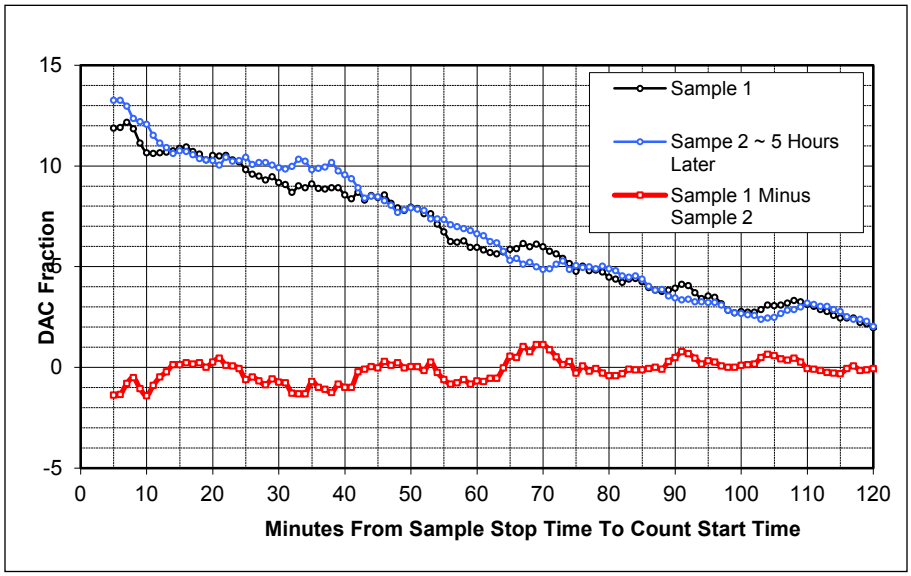

Figure C-2
Net Result of Subtracting Background from Job Coverage

Figure C-2 shows the decay of the background (sample 1) and job coverage (sample 2) air samples taken several hours apart (at the top). The net result of the "background" sample subtracted from the "job coverage" sample is shown at the bottom of the chart.

This method requires that the decay times for the background and job coverage air samples be the same. Decay time is the time between the end of sample collection and the start of an alpha scaler count.

Half-Life Method

The "Half-life Method" compensates for radon by counting a single job coverage air sample twice. The first alpha scaler count starts > 4 hours after the end of sample collection to ensure the contribution from ^{222}Rn is negligible. The second alpha scaler count is performed approximately 18 hours after the first count[32]. This method requires that the time between the first and second count be accurately measured.

The two counts are used to estimate the long-lived alpha radioactivity using the following equation (Ref. C.2, section 6.4, Equation 10)[33]:

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[32] Earlier timed decay periods can be used, but the contribution from long lived alpha emitters may be masked.

[33] Assumes ^{220}Rn progeny ^{212}Pb (10.64 hr half-life) is in secular equilibrium with its alpha-emitting daughters ^{212}Bi and ^{212}Po.

$$A_{LL} = \frac{(A_{t_2} - A_{t_1} e^{-\lambda \Delta t})}{\left(1 - e^{-\lambda \Delta t}\right)} \qquad \textit{Eq. C-1}$$

where:

A_{LL} - long-lived alpha activity

A_{t2} - Activity at time 2

A_{t1} - Activity at time 1

Δt - time between the 1st and 2nd count (same units as the inverse of λ)

λ - radiological decay constant for ^{212}Pb = ln(2)/10.64 hr = 0.0651 hr^{-1}

Both the "background" and "half-life" methods are accurate to within a "DAC Fraction" of about 1 to 2 for a sample volume of 1,000 liters under ideal conditions. On-the-job performance is likely to have less precision.

While these methods can validate that radon daughters is present, they may not be adequate to validate if there is (or is not) long lived radioactivity present. Appendix D describes instrumentation that can be used to discriminate long lived radioactivity from radon daughters.

Annular Kinetic Impactors (AKI)

Since natural airborne radioactivity arises from the decay of radon gases, the resulting particles tend to be much smaller than normal airborne particulate material. The AKI air sampler takes advantage of this fact.

An AKI uses an inertial collector-head. Sampled air enters the collection head at the rear and makes a 180-degree turn near a greased planchet at the front of the head before exiting out the center tube.

Only very small radon and radon daughter particles can navigate the turn and not be collected on the planchet. The size of the particles that are collected is determined by adjusting the slit width and air flow velocity.

The AKI sample head avoids collecting about 95% of the very small radon daughter particles, while effectively collecting particles containing licensed radioactive material (1-micron and larger). The disadvantage of the AKI is significant self-absorption of the alpha component of the sample in the collection media (i.e., oil/grease). See Reference C.3 for more details.

References

C.1 Eisenbud, M. and Gesell, T., "Environmental Radioactivity from Natural, Industrial & Military Sources", Academic Press, 1997.

C.2 ANSI/HPS N13.56-2012, “Sampling and Monitoring Releases of Airborne Radioactivity in the Workplace”, American National Standards Institute, October 2012.

C.3 IAEA Safety Series No. 49, “Radiological Surveillance of Airborne Contaminants in the Working Environment”, International Atomic Energy Agency, Vienna, 1979.

Appendix D: The Detection and Analysis of Alpha-Emitting Radionuclides

A number of different types of equipment are used for the detection and analysis of alpha emitting radionuclides:

- Survey instruments are used in the field to measure alpha contamination levels on equipment, in areas and on personnel and to determine beta-gamma to alpha ratios on higher activity smears.
- Counting equipment is used to count lower levels of alpha contamination and to determine beta-gamma to alpha ratios on lower activity smears.
- Air sampling equipment is used to monitor airborne alpha activity.
- Personal air sampling equipment is used to monitor intakes of personnel.
- Laboratory equipment is used to determine radionuclide distributions in contamination.

Each of these types of instrumentation is described, along with its strengths and limitations and any other relevant information.

Survey Instruments

Proportional Counter Friskers

Handheld proportional counter friskers are capable of measuring beta and alpha radiation simultaneously. Some instruments display both measurements and provide a separate audible indication for each pulse type.

These instruments generally have good sensitivity for alpha TRU. To achieve the required sensitivity for alpha contamination of 100 dpm/100 cm^2 (1.7 Bq/100 cm^2) (Reference D.1), very slow probe speeds (~1 inch/sec), close probe to surface distances (~ ¼ inch) and larger probe areas (≥ 100 cm^2) are needed. Typically a detector 100 cm^2 or larger is required to obtain the needed sensitivity.

The expected count rate, corresponding to 100 dpm/100 cm^2 (1.7 Bq/100 cm^2), could be approximately 10 to 20 cpm, or one count every three to six seconds. At this low count rate, frisking should be performed at slow speeds (~ 1 inch/sec or

approximately 2.5 cm/sec depending on the probe dimensions). If any counts are detected, the surveyor should then stop and perform a static count. This speed is substantially slower than traditional beta gamma frisking and may require training for both workers and the radiation protection staff.

Scintillation Counter Friskers

Various large area portable scintillation detectors may be used to facilitate alpha frisking for both personnel and items discharged from the Radiologically Controlled Area (RCA). These include standard ZnS, as well as dual scintillation detection where a ZnS film is layered on beta scintillation plastic. Some dual-scintillation detectors are similar to the proportional counter detectors discussed above, where simultaneous indication of both beta and alpha radiation is available.

Many of the dual alpha/beta instruments employ a single discriminator to differentiate alpha from beta pulses. This can result in significant cross-talk between the channels and should be taken in to consideration when using this type of instrument. Where possible and practical, counting instruments with dual discriminators should be selected. Efficiencies for ^{241}Am and ^{239}Pu are in the range of 15% to 20% for ZnS type plastic scintillators.

Scintillation detectors are more convenient to use since proportional counting gas is not required and counting efficiencies are only slightly lower. These instruments are also sensitive to light leaks and should be checked for this condition periodically. However, the portability of these instruments makes them an excellent tool for the early detection of alpha contamination on personnel in the field.

GM Friskers

A handheld frisker is a common frisking instrument for beta gamma contamination, which uses a 15 cm^2 pancake GM detector. Although sensitive to alpha radiation, their relatively high background (40 to 80 cpm) results in poor sensitivity for alpha TRU. Given the required sensitivity for alpha contamination of 100 dpm/100 cm^2 (1.7 Bq/100 cm^2) (Reference D.1) and a nominal detection efficiency of 10%, the corresponding count rate would be only 10 cpm. Observation of such a low count rate becomes challenging in a field environment, even with very low background.

Ion Chambers

Most facilities use ion chamber survey meters to measure high levels of beta/gamma contamination collected on smears (swipes) since GM based pancake friskers generally have a maximum range between 50,000 and 500,000 cpm (with dead time correction), depending on the manufacturer and model.

With a maximum range of 50,000 cpm for GM based pancake friskers and a counting efficiency of 10%, the corresponding beta/gamma activity is 500,000

dpm (8,300 Bq). This detection capability may not be sufficient for quantifying beta-gamma to alpha ratios on the order of 3000:1. Thus, higher activity samples may be needed where the beta activity is estimated using an ion chamber survey meter. In order to use this instrument to assess beta-gamma activity, the relationship between beta-gamma activity and instrument response must be evaluated.

Ion chamber and GM based instruments are not recommended for use in directly measuring and quantifying alpha radioactivity due to the high energy dependence and low ability to discriminate between alpha and other radiations.

Scanning using survey instruments

There are two Stages of Scanning[34] (reference D.2). The framework for determining the scan MDC is based on the premise that there are two stages of scanning. That is, technicians do not make decisions on the basis of a single indication, rather, upon noting an increased number of counts, they pause briefly and then decide whether to move on or take further measurements. Thus, scanning consists of two components: continuous monitoring and stationary sampling. In the first component, characterized by continuous movement of the probe, the surveyor has only a brief "look" at potential sources, determined by the scan speed. The surveyor's willingness to decide that a signal is present at this stage is likely to be liberal, in that the surveyor should respond positively on scant evidence, since the only "cost" of a false positive is a little time. The second component occurs only after a positive response was made at the first stage. This response is marked by the surveyor interrupting his scanning and holding the probe stationary for a period of time, while comparing the instrument output signal during that time to the background counting rate. Owing to the longer observation interval, sensitivity is relatively high. For this decision, the criterion should be stricter, since the cost of a "yes" decision is to spend considerably more time taking a static measurement or a sample

Additional information can be found in NUREG-1507 "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions" (reference D.3).

Gross Counting

Gross counting instruments capable of identifying alpha and beta radiation represent an important tool in measuring the relative and absolute hazard from TRU contamination. Instrument selection depends on the number of samples to be analyzed, the type of radioactivity to be measured, the required sensitivity, and the training and qualification of the staff.

For facilities where a large fraction of smears and air samples are analyzed for alpha radioactivity, consider using a high-throughput gas-flow proportional

[34] MARSSIM

counter. For other facilities, a plastic scintillation detector, in conjunction with a counter/scaler will provide valuable information and is easily used by technicians.

To quantify high beta-gamma to alpha ratios, a sample with high beta activity is recommended. High beta-gamma to alpha samples should be counted on an instrument capable of accurately discriminating between alpha and beta activity. A gas-proportional counter that performs the alpha portion of the measurement on the alpha counting plateau will provide the greatest accuracy while minimizing any beta interference or "cross-talk" in the alpha channel. Additionally, counting a sample with very high a beta activity could result in erroneously high reported alpha activity if the instrument used counts the alpha and beta simultaneously. For instance, to maintain random counting errors less than approximately 10%, approximately 100 counts gross alpha would be needed. Using an alpha counting efficiency of 30% (^{241}Am on a shielded gas-proportional counter) and a 15-minute count time results in approximately 22 dpm (0.4 Bq) alpha activity. At this alpha activity, with a facility beta to alpha ratio of 3000:1, the beta activity would be approximately 67,000 dpm (1.1 kBq). Counting and analysis of samples with this level of beta-gamma activity requires care to minimize contamination of personnel and instrumentation.

The analysis of smears and air samples using gross beta and alpha detection should account for self-absorption correction of the beta and alpha particles within the sample media, although beta particle self absorption is usually small compared to alpha particles. This correction is reasonable to apply for air and lapel samples, however for smears substantial variability in the self-absorption correction factor is expected depending on smear techniques and surface conditions.

Gas Proportional Counters

Gas-proportional counters utilize electronic discrimination controls to allow alpha radiation to be detected in the presence of beta-gamma radiation, either simultaneously or by individual counting. Careful setup of the instrument will minimize (but not eliminate) beta-to-alpha cross-talk when counting alpha and beta simultaneously. The degree of cross talk should be tested and understood when counting alpha and beta simultaneously, since cross-talk may be interpreted as alpha activity.

Where there is a need for a low alpha MDA or there is a high beta to alpha ratio, it is more precise to use an alpha-then-beta count mode. In this count mode, the instrument performs two sequential counts of the sample –one on the alpha voltage plateau and one on the beta voltage plateau. This counting mode has the advantage of eliminating beta-to-alpha cross-talk and many of the energy-dependent instrument setup parameters (e.g., discriminators).

These instruments also allow counting in an "Alpha Only" mode. This mode normally performs a single sample count on the alpha voltage plateau. Like the alpha-then-beta mode, this count mode nearly eliminates beta-to-alpha cross-talk. It has the disadvantage of not providing any beta activity data.

These devices generally have low background and high detection efficiencies, such that MDAs of 1 dpm (17 mBq) alpha and 30 dpm (0.5 Bq) beta are readily achievable. Alpha and beta detection efficiencies are generally in the range of 20% to 50% depending on the specific geometries and reference radionuclides used. The advantage in using these devices lies in the ability to analyze a large number of smears and air samples for the presence of alpha with relative ease.

There are three common configurations for gas-proportional counters. All typically have very thin Mylar windows allowing for detection of beta radiation ≥20 keV and significantly improved alpha detection.

1. Single sample counter. This type of instrument requires the user to manually change samples between counts.
2. Sample changer: This type of instrument incorporates a mechanism to sequentially count a number of samples with no user intervention once the machine is loaded. One sample is counted at a time. Sample capacity for typical devices range between 25 and 100 samples.
3. Drawer type: This type of instrument consists of one or more sample/detector drawers. Each drawer typically consists of several detectors. In situations where longer count times are needed and there is a high sample volume, this type of instrument has the advantage of counting multiple samples simultaneously.

Most of the gas-proportional instruments available have the option for a guard detector. The guard detector is used to automatically compensate for most of the effect of background gamma radiation on the sample count. Guard detector equipped instruments have the disadvantage of severely limiting the ability of the instrument to detect and quantify gamma emitting radionuclides – particularly those that do not also emit beta or alpha radiation.

When selecting a gas-proportional counting instrument, it is important to consider the expected number of samples to be counted and what the needed throughput is. For example, counting thirty air samples with a 15-minute count time will take at least 7-½ hours on a single or sample changer type instrument. Drawer type instruments are available with up to 16 detectors (four per drawer). This type of system could perform the same thirty 15-minute counts in about ½ hour.

Zinc-Sulfide

Zinc Sulfide (ZnS) detectors are an important tool in quantifying alpha contamination and determining alpha TRU air concentrations. These detectors can be used with portable and semi-portable counter/scalers and do not require shielding. Therefore, they can be used in the work environment to support counting or screening of smears and air samples for alpha TRU.

Used in the laboratory for counting air samples, typical background count times are 20 minutes (longer times may be needed to adequately characterize alpha background for instruments with low alpha background count rates), sample

count times are 10 to15 minutes, alpha background count rates are ≤0.3 cpm, minimum detectable count rates are ≤1 cpm, and minimum detectable activity is ≤4 dpm. For counting smears, a typical background count time is 20 minutes and sample count time is 1 minute.

This type of detector has several advantages: relatively little "cross talk" from beta/gamma radiation, portability, lower costs and simplicity.

The disadvantages include low-throughput (for single sample instruments) and lower sensitivity relative to gas-proportional counters.

Air Sampling

General Area (GA) air sampling may be performed with fixed location, continuous air monitors (CAMs), or portable air sampling in locations associated with specific operational or work activities. General area air samplers collect airborne particulates through a filter and usually have good sensitivity to airborne alpha TRU, provided sufficient total air volume is collected.

These air samplers may be divided into two types: low volume, and high volume. Low volume samplers generally operate in the range of 1 to 10 CFM (28 to 280 lpm), whereas high volume samplers operate in the range 10 to 30 CFM (280 to 840 lpm). Both types generally have the needed capability for quantification of alpha TRU given the appropriate analysis sensitivity. When upgrading or replacing equipment, it is recommended that samplers with digital flow/volume meters are preferred in order to maximize the accuracy of the sample volume used in calculations.

Continuous Air Monitors

Three types of CAMs are generally available; GM-based, proportional counter, and silicon based surface barrier detectors.

- **GM-based CAMs**

  Since GM detectors are not capable of discrimination between alpha and beta/gamma radiation, a GM based CAM will not provide a direct indication of airborne TRU. However, if the source term assessment indicates relatively high beta to alpha ratios, these devices may be adequate using a surrogate relationship to identify airborne TRU radioactivity.

  For facilities where the beta to alpha ratios are low, or quite variable, sole reliance on a GM based CAM may not provide adequate assurance airborne radioactivity is detected at the appropriate sensitivity.

- **Proportional Counter CAMs**

  Proportional counter CAMs allow discrimination of alpha from beta using electronic pulse height discrimination. For facilities or locations where interference from radon progeny is low, this device should be effective in providing real-time monitoring for TRU in the presence of fission and activation products.

- **Passively Implanted Planar Silicon CAMs**

  Some CAMs use silicon based surface barrier detectors for real-time alpha spectroscopy of air concentrations. Some silicon based surface barrier detectors devices have the capability of monitoring beta, radon, TRU, and gamma exposure rate simultaneously, while providing electronic storage of historical alpha and beta spectrums.

Commonly available alpha CAMs utilize multi-channel analyzers and spectroscopy algorithms to provide radon (and radon daughter) interference correction. The ability of the various systems to compensate for the presence of radon depends greatly on the radon levels, physical characteristics of the CAM, and capabilities of the computing software & hardware. The equipment typically also has the ability compensate in real-time for changes in background radiation levels and to connect to remote monitoring and control systems (i.e., telemetry).

Minimum Detectable Activity (MDA)

In general, the minimum detectable activity is used to determine the capabilities of the counting system used and to ensure that the action levels appropriate for the analysis are statistically distinguishable from background. Under most circumstances a 95% confidence (5% type I and 5% type II errors) is used. The example formulae in this appendix are based on a 95% confidence factor. Use of any other factor will require recalculating the constants in the provided MDA formulae.

This section will use a standard MDA formula derived from the equations in reference D.4 (equations 25 and 26) as its basis. Other accepted MDA formula like the Stapleton Approximation (reference D.5, figure 20.54) may be used as desired by the individual facility. To ensure consistency, it is recommended that an organization adopt a single MDA formula/methodology.

$$MDA = \frac{2.71 + 3.29\sqrt{R_b t_s\left(1 + \frac{t_s}{t_b}\right)}}{t_s \varepsilon} \qquad \textit{Eq. D-1}$$

For the case where sample and background count times are equal:

$$MDA = \frac{2.71 + 4.65\sqrt{R_b t_s}}{t_s \varepsilon} \qquad \textit{Eq. D-2}$$

Where:

MDA	=	Minimum Detectable Activity (dpm)
R_b	=	background count rate (cpm)
t_s	=	sample count time (minutes)
t_b	=	background count time (minutes)

ε = counting efficiency

NOTE: When utilizing low-background counting equipment for counting for alpha activity (e.g., shielded gas-proportional detectors), is may require that the background count time to be significantly longer than the sample count time in order to obtain sufficient counts to characterize the alpha background.

Sample Volume Limitations

The volume of a general area air sample will have a direct, linear effect on the minimum detectable concentration (MDC) of the sample. The purpose of this chapter is to provide guidance on determining the minimum sample volume needed to meet the desired MDC while maintaining practical sample count times.

This section does not apply to lapel samples as the sample volume is not part of the lapel sample result calculations.

Due to the number of variables involved and the mathematical limitations of solving for a variable under the square-root radical (e.g., sample count time), it is recommended that a computerized spreadsheet be used in the calculations.

For the purposes of this calculation, the maximum allowed background for the measurement quality control and the minimum acceptable instrument efficiency should be used when determining the MDA.

$$Air\ Sample\ Concentration\left(\mu Ci/cc\right): C\left(\frac{\mu Ci}{cc}\right) = \frac{(A)\left(\frac{1\mu Ci}{2.22E6dpm}\right)\left(\frac{1ft^3}{28{,}317cc}\right)}{(V)(S)}$$

Eq. D-3

Where:

C = sample concentration (µCi/cc)

A = sample activity (dpm)

V = sample volume (ft^3)

S = self-absorption of activity on filter media (unitless)

Substituting the target limiting or effective DAC value for air sample concentration and MDA for the sample activity in the Air Sample Concentration formulae then solving for sample volume yields:

$$Minimum\ sample\ volume\ V_{min} = \frac{(MDA)\left(\frac{1\mu Ci}{2.22E6dpm}\right)\left(\frac{1ft^3}{28{,}317cc}\right)}{(D)(S)}$$

Eq. D-4

Where:

V_{min} = minimum sample volume (ft^3)

MDA = Minimum Detectable Activity (dpm)

S = self-absorption of activity on filter media (unitless)

D = controlling DAC value (μCi/cc)

Example:

R_b = 1 cpm

t_s = 15 minutes

t_b = 60 minutes

S = 0.8

D = 1.26E-12 μCi/cc (0.3*4.2E-12 μCi/cc)

ε = 0.30

Substituting the above values into Equation D-2:

$$MDA = \frac{(2.71) + 4.65\sqrt{(1)(15)}}{(0.30)(15)} = 4.60 dpfi$$

Substituting the above values into Equation D-4:

$$V_{min} = \frac{(4.60 dpm)\left(\frac{1\mu Ci}{2.22E6 dpm}\right)\left(\frac{1 ft^3}{28{,}317 cc}\right)}{(1.26E-12)(0.8)} = 72.6 ft^3$$

In this example, if the field technician obtains an air sample of at least 72.6 ft^3 would be sufficient to ensure that the needed minimum detectable concentration for the target radionuclide(s) can be met.

Sample Counting Considerations

The collection of particulate aerosols on membrane or cellulose filters inevitably results in the deposition of these particulates deep within the filter matrix or within a dust layer on the filter surface. Likewise, the mechanical action and pressure applied in obtaining smear samples of removable contamination from surfaces results in deposition of particulates within the sample media. In either case, this deposition results in absorption of alpha particles within the material.

Correction for "self-absorption" when performing alpha particle counting is extremely important. Literature indicates self-absorption correction factors as

much as 50% are not uncommon (Reference D.6, D.7, and D.8) and depend on several factors, including:

- Characteristics of the sample media,
- Impact velocity of particulate aerosols for air samples,
- Effective atomic number and density of the particulate or absorbing material, and
- Particle diameter (AMAD)
- Dust Loading

Self-absorption can be minimized by utilizing hard-surface filter media (e.g., Teflon, Polytetrafluoroethylene (PTFE), etc.), lower flow rates, and the minimum air volume needed to meet the desired MDC. Hard-surface types of media have significantly better alpha performance due to the lower depth of penetration of the sample particles. Lower flow rates will reduce the depth of deposition of particulate and minimizing the air volume will reduce the amount of natural activity deposited on the filter. Due to a higher resistance to flow, hard-surface types of media are typically limited to lower flow rate (≤ 3 cfm) applications. Self-absorption factors in the range of 10% to 20% have been documented for PTFE filter media. Reference D.9 recommends no self-absorption factor for the Teflon based filter.

These variables are difficult to accurately characterize for each environment within a nuclear power facility. One conservative approach is to calculate a correction factor using the technique from NCRP (Reference D.7), which assumes that aerosols are uniformly deposited throughout filter media. This factor can then be used as a multiplier in determining the filter's activity from the count rate.

Lower Limit of Detection

To determine the lower limit of detection for an air sample, the minimum detectable activity (MDA) for the sample counting instrument and counting protocol that will be used to quantify the activity on the sample filter must be known.

For the purposes of this document, the MDA formula provided in Equation D-2 is used.

Once the MDA is determined, substitute the MDA into the formula for determining the DAC associated with an air sample filter. The result is the lower limit of detection in DAC for the sample.

Starting with the air sample concentration formula (Equation D-3) and substituting the counting system MDA for sample activity and inserting a conversion from sample concentration in μCi/cc to DAC yields:

Air Sample LLD:

$$LLD\,(DAC) = \frac{(MDA)\left(\frac{1\mu Ci}{2.22E6dpm}\right)\left(\frac{1ft^3}{28{,}317cc}\right)}{(V)(D\frac{\mu Ci/cc}{DAC})(S)} = \frac{MDA(1.6E-11)}{(V)(D)(S)} \qquad \textit{Eq. D-5}$$

Where:

MDA = MDA of the counting system/protocol in dpm

V = Sample volume in ft^3

D = DAC value in µCi/cc

S = self-absorption factor

1.6E-11 = Unit conversion factor

In general, the easiest method for lowering the LLD of an air sample is to increase the volume of the air sample because the change is linear with the change in volume.

Personal Air Sampling

Personal, or lapel, samplers are a relatively small battery operated air pump connected to a sample media holder via a flexible tube. For compatibility with common counting systems and their calibrations, it is recommended that a 47 mm or 2-inch sample media be used as consistent with the local standard for air sample filters and smears.

Other issues that should be considered when using Personal Air Samplers are:

- For alpha emitting nuclides, a very small number of particles may correspond to a significant intake. The statistics of sampling small numbers of events becomes the critical factor in determining sampling accuracy.
- Lapel sample pumps vary as to flow rate and battery life. In general, the higher the flow rate, the better the net sensitivity of the resulting sample. The battery can have a significant effect on the cost of the unit and the sample run time. If a battery with a shorter run time is selected, consider recharging time and/or spare batteries.
- The sample media should be protected from cross-contamination while in the work area. A partial enclosure to protect the sample media can provide significant benefit while not affecting the validity of the sample.

Lower Limit of Detection

To determine the lower limit of detection for a personal air sample, the minimum detectable activity (MDA) for the sample counting instrument and

counting protocol that will be used to quantify the activity on the sample filter must be known.

For the purposes of this calculation, the MDA formula provided in Equation D-2 is used:

Once the MDA is determined, substitute the MDA into the formula for determining the DAC–hr associated with a lapel sample filter. The result is the lower limit of detection in DAC-hr for the sample.

Lapel Sample LLD:

$$LLD\,(DAC-hr)=\frac{(\boldsymbol{MDA})(\frac{2000hr}{year}\left(\frac{1\mu Ci}{2.22E6dpm}\right)}{(\frac{\boldsymbol{Fl}/\min}{20l/\min})(\boldsymbol{D}\frac{\mu Ci/ml}{DAC})(\boldsymbol{2}E9\frac{ml}{year})(\boldsymbol{S})}=\frac{MDA(9.0E-12)}{(F)(D)(S)}$$

Eq. D-6

Where:

MDA	=	MDA of the counting system/protocol in dpm
F	=	Sample flow in liters/min
D	=	DAC value in µCi/cc
S	=	self-absorption factor
2E9 ml/year	=	volume breathed/year
20 l/min	=	standard man breathing rate in liters/min
9.0E-12	=	Unit conversion factor

OR if a LLD in mrem is desired:

$$LLD\,(mrem)=\frac{(MDA)(\frac{5{,}000mrem}{ALI}\left(\frac{1\mu Ci}{2.22E6dpm}\right)}{(\frac{Fl/\min}{20l/\min})(S)}=\frac{MDA(0.045)}{(ALI)(S)(F)}$$

Eq. D-7

Where:

MDA	=	MDA of the counting system/protocol in dpm
F	=	Sample flow in liters/min
ALI	=	ALI value in µCi
S	=	self-absorption factor

20 l/min = standard man breathing rate in liters/min

0.045 = Unit conversion factor (at 5 rem/ALI)

The 5,000 mrem is based on the 5 rem per ALI in 10CFR20. Other conversion from ALI to dose may be used as appropriate to the regulatory basis.

In general, the easiest method for lowering the LLD of a personal air sample is to increase the personal air sample flow rate because the change is linear with the change in flow rate. For example, increasing the flow rate from 3 LPM to 5 LPM will result in a 40% reduction of the LLD.

Laboratory Instruments

Gamma Spectroscopy

Gamma spectroscopy has substantial limitations and cannot be the sole methods of monitoring for TRU radionuclides since most TRU radionuclides emit only low energy gamma rays or X-rays. The presence of these low-energy X-rays can be an indication of the presence of TRUs, but they cannot be reliably used to quantify TRUs because most TRUs emit them. However, ^{241}Am emits a 59.5 keV photon that is relatively easy to detect using HPGe based gamma spectroscopy. Identification of this radionuclide may be a good indicator of the presence of other TRUs, provided the relative abundance of ^{241}Am to other TRUs is relatively constant throughout the facility. The lack of identified ^{241}Am in a gamma spectrometry analysis does not preclude the presence of alpha emitters, but its presence does confirm the presence of TRU.

Note: The ^{241}Am to other TRU ratio will change over time, for example due to decay of ^{241}Pu (β- decay to ^{241}Am).

In relying on gamma spectroscopy to identify the presence of ^{241}Am, the sensitivity of the system to ^{241}Am in the presence of other fission and activation products must be evaluated. This sensitivity may be quite variable depending on the type of detector used and the amount of Compton background (i.e. continuum) created from the presence of other gamma-emitting radionuclides.

High-purity Germanium (HPGe) Detectors

Most commercial nuclear power facilities use high-purity germanium (HPGe) gamma spectrometers for quantitative sample analysis. These can be used to quantify the activity of ^{241}Am. However, as the relative activity of higher energy gamma-emitters increases, the sensitivity to ^{241}Am may be relatively poor such that beta/gamma to alpha ratios of less than approximately 200:1 would be needed to quantify this TRU nuclide.

Another type of HPGe spectrometer has improved sensitivity to ^{241}Am in the presence of other beta/gamma-emitters, called a reversed-electrode high-purity germanium (REGe) system. REGe and similar detector based spectrometers

offer an increase in sensitivity to ^{241}Am of approximately a factor of four compared to a standard HPGe.

Sodium-Iodide (NaI) Based Spectrometers

NaI based detectors can provide useful gamma spectroscopy results at a considerably lower cost than HPGe based systems. However, NaI detectors have a relatively poor peak resolution to the point of not being able to accurately resolve adjacent gamma peaks common in an operating nuclear power facility.

Lanthanum-Bromide ($LaBr_3$) Spectrometers

$LaBr_3$ detectors offer improved energy resolution, good linearity characteristics over a standard NaI based detector. Unlike HPGe detectors, they can operate at room temperature while still providing good energy resolution.

Table D-1
Gamma Detector Comparison Summary

Detector Type	Resolution @662 keV (%)
HPGe	0.2 (~1.3 keV)
$LaBr_3$	2.8 - 4
NaI	7

Alpha Spectroscopy

Alpha spectroscopy can be used for identification and quantification of alpha TRU using either laboratory based or portable alpha spectroscopy technology. Solid state detectors, such as silicon based surface barrier detectors, are the most widely used for this application.

For air samples containing substantial radon progeny, these devices can be a useful tool to confirm the presence of this progeny for the initial analysis. Depending on the detection efficiency and the amount of radon and radon daughters present, the TRU MDC may be greater than 0.3 DAC. Therefore, air samples may still need to be analyzed following a decay period to accurately quantify the long-lived alpha activity.

Laboratory Based

Laboratory based silicon based surface barrier detectors are typically mounted within a vacuum chamber that also serves as a sample holder. The standard process for analyzing samples is to remove air within the chamber using a vacuum pump. Chemical separations are performed to extract the species of interest and create a thin sample with minimal self-absorption. Once prepared, these samples can be accurately measured with excellent detection sensitivity.

The equipment and personnel expertise required for alpha spectroscopy lead to substantial resources being necessary to maintain this capability. To obtain the best results, chemical processing for each sample is needed depending on the analysis methods used. However, it is possible to obtain nuclide identification and relative ratios without chemical processing of the sample media. Combining this information with the gross alpha data obtained using a gas-proportional counter can provide a reasonably accurate and complete alpha characterization of the sample.

Commercial laboratories are available for more detailed analyses and detection of low concentrations of individual actinides in fecal bioassay and other samples. In this case, actinide separation is accomplished by chemical treatment of the samples and sequential separation of the actinides by anion-extraction chromatography using various resins. Inductively Coupled Plasma Mass Spectrometry (ICP-MS) or Alpha Spectrometry is used to detect the individual actinides. Thermal Ionization Mass Spectrometry (TIMS) can also be used to determine very low levels of Pu-239 and Pu-240. It is important to ensure that the commercial laboratory has in place a recognized quality assurance program, is accredited by an appropriate national organization, and is recognized by the regulator where required.

Semi-Portable

Most of the alpha spectroscopy capable semi-portable units utilize silicon surface barrier detectors and have alpha efficiencies typically in the 20% to 25% range for ^{241}Am or ^{239}Pu. In order to quantify air samples to 0.3 DAC, air sample volumes in the range of 100 ft^3 to 150 ft^3 are usually sufficient.

Some of the currently available semi-portable alpha spectrometers use methods to discriminate against radon and its progeny from TRU activity. These spectrometers employ multi-channel analyzers and automated spectrum analysis to estimate the TRU activity in the presence of radon and radon decay daughters. Some of these devices are estimating the TRU activity based on known radioactive decay chains and an assumed radon (long-lived to short-lived) ratio. Other devices also employ proprietary additional spectral stripping techniques to remove any radon interferences.

The presence of high levels of radon will increase the uncertainty of the analysis, may preclude an accurate estimate of low-level TRU activity, and/or require very long count times. The reported uncertainty should be considered when determining the usefulness of any radon compensated result when using these semi-portable units.

Calibration Considerations

Ensure that the correct geometry is used to calibrate the instrument as well as the appropriate nuclides/energies for the expected radionuclides in the sample. The following need to be taken in to account:

- sample size relative to the detector

- sample size relative to the calibration source
- sample to detector distance
- energy response of the detector/instrument

It is important that the reference source geometry and its location relative to the detector be as close as possible to those of the sample. Very small differences can result in significant errors in reported results. This is particularly true for alpha measurements.

References

D.1 IE Circular No. 81-07, "Control of Radioactively Contaminated Material", May 14, 1981; USNRC SSINS: 6830.

D.2 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," August 2000.

D.3 NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions", June 1998.

D.4 NUREG/CR4007, "Lower Limit of Detection: Definition and Elaboration or a Proposed Position for Radiological Effluent and Environmental Measurements," August 1984

D.5 NUREG-1576, "Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP)", July 2004.

D.6 BNWL-SA-2084; "Effectiveness of Filter Media for Surface Collection of Airborne Radioactive Particulates", Dale H. Denham, Pacific Northwest Laboratory, Battelle Memorial Institute, Richland, Washington, January 1969.

D.7 NCRP Report No. 58, "A Handbook of Radioactivity Measurements Procedures," November 1, 1978.

D.8 PNL-5278, "Effects of Particle Size and Velocity on Burial Depth of Airborne Particles", D.P. Higby, November 1984.

D.9 PNNL-SA-62011, "Results of Self Absorption Study on the Versapor 3000 Filters for Radioactive Air Sampling", JM Barnett, August 2008.

Appendix E: Technical Basis for Guidelines

The importance of alpha airborne activity depends on its relative abundance compared to beta-gamma airborne radioactivity. Airborne radioactivity is a hazard when present in concentrations that approach the DAC values.

It is convenient to define the term "DAC-Fraction Ratio".

$$DAC_{FractionRatio} = \frac{fDAC_{\alpha}}{fDAC_{\beta\gamma}} \qquad \textit{Eq. E-1}$$

Where:

$fDAC_{\alpha}$ - is the total alpha activity divided by its DAC value

$fDAC_{\beta\gamma}$ - is the sum of each beta-gamma emitting nuclide's concentration divided by its corresponding DAC value

Alpha DAC-fractions refer to licensed radioactivity (long lived) with no contribution from natural sources.

The table below shows the relative importance of beta-gamma and alpha emitters based on the gamma emitter (cobalt-60) and alpha emitter (americium-241) present at operating nuclear power plants.

Table E-1
Activity and DAC-Fractions for Co-60 and Am-241[1]

LEVEL		Activity Relative to Am-241	% Activity Fraction	DAC (μCi/cc)	% DAC-Fraction
I	Co-60	30,000	99.997	1E-8	90
	Am-241	1	0.003	3E-12	10
II	Co-60	3,000	99.970	1E-8	47
	Am-241	1	0.030	3E-12	53
III	Co-60	300	99.700	1E-8	8
	Am-241	1	0.300	3E-12	92

[1]Based on ICRP 30. The technical basis is slightly different when ICRP 60 or later values are applied, and will result in different beta-gamma to alpha activity ratios corresponding to a given alpha dose fraction (e.g., 15,000 and 150 instead of 30,000 and 300, respectively. Co-60 may also not be the beta-gamma emitter with the most representative DAC and this will change the technical basis.

In Level I Areas where the relative abundance of alpha activity is low, the alpha DAC-fraction is not likely to be significant (≤10%), so only high activity smears and air samples should be counted to verify this condition. The condition should be verified by alpha counting all air samples with a beta-gamma DAC-fraction greater than 1. If the alpha DAC-fraction ratio exceeds 0.1, consider reclassifying the area as Level II.

In Level II Areas, the alpha DAC-fraction is likely to be 10% to 90% of the total airborne radioactivity. A reasonable number of smears should be counted to characterize the levels and extent of alpha contamination. All air samples greater than the "beta-gamma DAC-fraction action level" should be counted for alpha activity. If the DAC-fraction ratio exceeds 10, consider reclassifying the area as Level III.

The "beta-gamma DAC-fraction action level" ensures that air samples are counted for alpha whenever the total airborne radioactive material is likely to exceed the derived air concentrations This action level ensures compliance with regulatory airborne radioactivity posting requirements.

A site-specific "beta-gamma DAC-fraction action level" may be developed based on the alpha nuclide distribution and relative abundance of alpha activity at a particular facility. The beta-gamma DAC –fraction level is calculated as follows:

$$DAC_{\beta actionlevel} = \frac{1}{1 + DAC_{FractionRatio}} \qquad \text{Eq. E-2}$$

Alternatively, an action level can be based on the most restrictive gamma emitter and most restrictive alpha emitter. Two examples follow using Am-241 and Co-60:

- When alpha is about 50% of the total airborne radioactivity DAC fraction (activity ratio ~3,000), count an air sample for alpha when its beta-gamma DAC-fraction is greater than 0.5.
- When alpha is about 90% of the total airborne radioactivity DAC fraction (activity ratio ~ 300), count an air sample for alpha when its beta-gamma DAC-fraction is greater than 0.03.

In Level III Areas where the activity ratio is <300, the alpha DAC-fraction is elevated (>90%). A sufficient number of smears and all air samples should be counted to adequately characterize the levels and extent of alpha contamination.

Appendix F: Work Control Examples

Example 1: Predicting airborne activity levels from surface contamination

Methodology:

Based on NUREG-1400, the airborne radioactivity DAC Value can be represented by:

$$\text{DAC Value} = \frac{S \times R \times C}{(2.2E6) \times (100) \times DAC} \qquad \textit{Eq. F-1}$$

Where:

S = Measured gross activity of the smear (dpm/100cm^2)

R = (1) Resuspension factor based upon type of work activity (cm^{-1})

C = (2) Confinement factor for the expected activity

2.2 E 6 = Conversion factor from dpm to µCi

100 = Converts dpm/100 cm^2 to dpm/cm^2

DAC = DAC used by the station for the isotope representative of the contamination (µCi/cm^3).

1. Resuspension factor, R, assumes an initial 1 E -07 cm^{-1} resuspension factor for dry contamination of a non-volatile, powdery nature and adjusts this value for the type of work to be performed.

Type of Work to Be Performed	Resuspension Factor (R) (cm^{-1})
Surveillance or Light Mechanical (Operator Rounds, Taking Readings, Valve Manipulation, Electrical Work, Instrumentation Work, Testing)	1 E –07
Heavy Mechanical or Close Contact Work (Heavy Assembly, Pounding, Filing, Crawling, Work in Tight Quarters, Brushing)	1 E –06
High Energy (Welding, Grinding, Heating, High Pressure)	1 E - 05

2. Confinement factor, C, takes into consideration whether the material is separated and confined while a worker is present, or whether it is actually handled in the open.

Level of Confinement/ Ventilation	Confinement Factor (C)
Contaminated Items Handled in Glovebox	0.01
Contaminated Surface Area Maintained Wet or Covered with a Gel or Similar Confining Preparation	0.1
Contaminated Items Handled in Well-Ventilated Hood or Use of HEPA Unit	0.1
Open Area with Normal, Good Ventilation	1
Poor or Unknown Ventilation	10

Example:

In an area with poor ventilation, welding is to be conducted on a flat surface on which direct alpha monitoring has identified a surface contamination level of 1000 dpm/100cm^2 alpha contamination. What is the predicted airborne activity ?

S = 1000 dpm/100cm^2

R = 1E-5

C = 10

$$\text{DAC Value} = \frac{\text{S} \times \text{R} \times \text{C}}{(2.2\text{E}6) \times (100) \times \text{DAC}}$$

$$\text{DAC Value} = \frac{1000 \times 1E-5 \times 10}{(2.2\text{E}6) \times (100) \times 3\text{E}\text{-}12}$$

= 147 DACs

Airborne activity of 147 DACs is predicted.

Note: It is more common that fixed contamination is not directly measurable. Do not use only measurable loose contamination information in this calculation to calculate airborne activity where aggressive work is being performed and could release fixed alpha activity, as this will under-estimate the predicted airborne activity.

Example 2: Methods to Contain Alpha Contamination and Airborne Activity

- Glycerin based coatings can be used to soak into sludge, dirt and the oxide layer to trap particles (the glycerin can then be wiped off). This can be followed by a chemically compatible latex paint coat which bonds with the glycerin and then dries.
- Contamination within pipes or tubing can be contained by laying flat plastic as a sleeve around the item or by sealing the exposed ends.
- Ventilated glove bags may be used to contain contamination at the source, and the work can be conducted while the contaminant is isolated.
- In dusty environments, engineering controls should be used to contain contamination.
- HEPA ventilation may be used to direct airflow into containment away from the worker and surrounding areas. HEPA filtered ventilation can be deployed in conjunction with fixatives and containment or may be used as a stand-alone method in certain applications. The adequacy of ventilation is typically verified, for example by smoke testing or other physical indicators such as flags. Where ventilation is used without a containment device, respiratory protection is typically considered.
- If the contaminant cannot be contained at the source, or if further containment is required to prevent the spread of contamination or airborne activity from the area, then a containment tent structure may need to be built around the work area. The tented structure should be adequately designed, constructed and ventilated to ensure airborne contamination is contained within it and respiratory protection should be used within the tented area.

Appendix G: Job Preparation and Coverage

Technician Preparation and Validation of Conditions in Pre Job Survey

When technicians are assigned to perform surveys or cover work in alpha Level II or Level III areas, special emphasis should be placed on preparation for the conditions to be encountered and the radiological response necessary to control the work activity and how the work activity could significantly alter the current alpha ratio/hazard.

Typical items include:

- Review of the Radiation Work Permit
- Previous survey data
- Expected beta-gamma to alpha ratio
- TEDE ALARA evaluation
- Prescribed engineering controls
- HP Hold points during the work activity
- Stop work criteria
- Preparation of the necessary equipment
  - Personal air samplers are available for worker use if required
  - Alpha monitoring equipment availability to ensure timely counting of smears and air samples
  - Alarming CAMs if required for the work
  - Engineering controls (HEPA Ventilation / Glove Bags, etc.) if required for the work

Job Coverage Considerations

Based on the controls identified in the radiological work permit, the following actions would typically be performed by the technician during job coverage:

- Perform initial survey to verify current conditions as listed in pre-job survey or job history files. Examine alpha and beta-gamma contamination data to confirm that the classification of the area is as expected or as found

previously. If the data identifies that the classification of the area is not as expected, then notify RP supervision. Follow stop work criteria as determined by the RWP.

- Conduct additional surveys based on the potential to alter the radiological conditions of the work area or as per the RWP. Aggressive work activities (such as grinding, welding, and lapping) can change the radiological conditions by exposing hidden oxide layers or releasing fixed contamination and potentially change the work area classification. Place special emphasis on assessing changes in work area conditions as work progresses.
- Take an adequate number of air samples to assess the alpha hazard associated with each work task – typically both inside the work area and in adjacent areas.
- Check smears and air samples to aid in timely identification and assessment of changes in work area conditions. Communicate any unexpected changes (outside those covered in the RWP) to RP Supervision. Early detection and mitigation (including stop work) is critical to minimize the potential for significant exposures.
- If stop work criteria is reached, have workers place area in a safe condition and exit the work area. Control access to the area until follow-up surveys are performed as needed to assess the radiological condition of the work area and the extent of the contamination spread.
- Report to the RP supervisor any indication that personnel may have been inadvertently internally exposed to alpha emitting radionuclides.

Additional Considerations during Job Coverage

- If engineering controls are utilized, ensure they are placed to maximize effectiveness.
- Ask individuals who sustain a cut or scrape in an alpha contaminated environment to immediately leave the Radiologically Controlled Area and report the event promptly to RP supervision for evaluation. If necessary, perform rigorous follow up to ensure internal dose is properly assessed.
- If work is performed in Alpha Level III areas, ensure proper alpha surveys are conducted of all tools and equipment used in the work activity. Decontaminate items if necessary prior to their removal from the work area.
- Properly control and label all equipment and material, including waste, removed from an alpha Level III area. Segregate equipment and materials exposed to 50:1 beta-gamma to alpha activity ratio until release surveys have been conducted
- Monitor personnel for alpha contamination when exposed to 50:1 beta-gamma to alpha activity ratio. Alpha monitoring needs to be conducted carefully, slowly and as close to a flat surface as possible to detect contamination at the lower level of detection of the equipment. Use static checks where appropriate.

Recording of in-field survey data

- All survey data should be recorded in sufficient detail and appropriate units so that beta-gamma to alpha ratios can be compared with expected ratios. *Consider noting on survey documentation the physical surface condition, e.g. wet, dusty, clean, etc.*

Appendix H: Internal Dose Assessment

Intake assessments are performed consistent with applicable document(s), for example, USNRC Regulatory Guide 8.9, ANSI HPS N13.39, NUREG-4884, ICRP 30, ICRP 78, and RS-G 1.2. 2, or country specific regulations. For USNRC licensees, the methodology for calculating dose from the intake must be consistent with the bases for the ALIs.

Reference Levels

The magnitude of an intake should be estimated for each bioassay measurement that indicates internally deposited material from licensed activities. The scope of the evaluation should be commensurate with the potential magnitude of the intake.[35] Reference Levels should be established below which no action is required and above which there exists a graded system of additional measurements and other investigative actions.[36] The Reference Levels should be established in accordance with ANSI HPS N13.39-2011[37], GD-150[38], RSG-1.2 or other approved guidance, as appropriate.

Estimation of Intake from Personal Air Sampler

The intake of alpha emitting nuclides can be estimated based on the activity of alpha emitting nuclides on the filter, the breathing rate, and the flow rate of the sampler:

$$I_i = \frac{A_i \times BR}{F} \qquad \textit{Eq. H-1}$$

Where:

- I_i - the intake of radionuclide, i;
- A_i - the activity of radionuclide, i, on the air filter;
- BR - the appropriate breathing rate from the table below, in L/min; and
- F - flow-rate of the sampler, in L/min

[35] Regulatory Guide 8.9, Regulatory Position 2.3

[36] NUREG/CR-4884, pp. 6

[37] ANSI HPS N13.39-2001, Chapter 5, page 9

[38] GD-150, Chapter 4, page 12

Estimation of Intake from Air Sampler

In the absence of other data, the intake of alpha emitting nuclides can be estimated based on the concentration of alpha emitting nuclides in the air as measured by air samples, the breathing rate, and the duration of exposure:

$$I_i = C_i \times BR \times t \qquad \textit{Eq. H-2}$$

Where:

I_i - the intake of radionuclide, i;

C_i - the concentration of radionuclide, i, in the air;

BR - the appropriate breathing rate from the table below; and

t - duration of exposure

Breathing Rate for Reference Man		
	Male	**Female**
ICRP 23[39]	20 L/ min	19 L/ min
ICRP 89[40]	1.5 m^3/ hr	1.3 m^3/ hr

Estimation of Alpha Intake from Whole Body Counts

The intake of gamma emitting radionuclides can be determined from whole body count measurements and applicable intake retention fractions. The intake of alpha emitting radionuclides can be estimated based on scaling to the gamma emitting radionuclide intakes as determined from the whole body count result.

$$I_{\gamma,i} = \frac{A_{wbc,i,t}}{IRF_{\gamma,i},t} \qquad \textit{Eq. H-3}$$

Where:

$I_{\gamma,i}$ - Intake of gamma emitting radionuclide, i;

$A_{wbc,i,t}$ - Activity gamma emitting radionuclide, i, in the whole body count at time, t, post intake; and

$IRF_{\gamma,i,t}$ - Intake Retention Fraction of gamma emitting radionuclide, i, at time, t, post intake

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[39] ICRP 23, page 346

[40] ICRP 89, page 99

The ratio of the gamma nuclide activity to the alpha emitting nuclide activity in the intake can be determined by scaling the activity of the alpha emitting nuclides based on laboratory analysis of air samples and/or smears representative of the work area.

$$I_{\alpha,i} = I_{\gamma} \cdot \frac{A_{\alpha,i}}{A_{\gamma}} \quad 、 \qquad \textit{Eq. H-4}$$

Where:

$I_{\alpha,i}$ - Intake of alpha emitting nuclide, i, as determined by scaling to the intake of the appropriate gamma emitting nuclide;

I_{γ} - Intake of the gamma emitting nuclide used for scaling;

$A_{\alpha,i}$ - Activity of alpha emitting nuclide, i, in the sample used for scaling; and

A_{γ} - Activity of the gamma emitting nuclide in the sample used for scaling.

Estimation of Alpha Intake by Urine or Fecal Sampling

The intake can be estimated based on the total activity excreted in a 24 hour or accumulated urine or fecal sample according to[41] :

$$I_i = \frac{A_i}{IRF_{i,t}} \qquad \textit{Eq. H-5}$$

Where:

I_i - intake of radionuclide, i;

A_i - the activity of the i^{th} radionuclide in the sample;

$IRF_{i,t}$ - the IRF of the i^{th} radionuclide at time, t, post intake, corresponding to the 24 hour sample or the accumulated sample, as appropriate.

The term, "24 hour sample" means the total urine or feces output collected over a 24 hour period, and the term "accumulated sample" means the total urine or feces output since the time of the exposure.[42]

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[41] Reg Guide 8.9, Chapter 4.3, page 8.9-5

[42] Reg Guide 8.9, Chapter 4.3, footnote 7, page 8.9-5

For a single intake, the computation of intake and therefore dose using default assumptions, is relatively straight forward. If multiple bioassay samples have been obtained, modeling needs to occur to ensure that all bioassay samples are in agreement as to the intake mode as much as possible.

Assessment of Dose from Intake

Once an estimate of the intake of each alpha emitting nuclide has been determined, the dose for each alpha emitting nuclide may be determined based on the ALI values:

$$Dose_{\alpha,i} = \frac{I_{\alpha,i}}{ALI_{\alpha,i}} \times 5 \qquad \textit{Eq. H-6}$$

Where:

5 Rem is the annual dose limit (this may be changed to 2 Rem depending on the applicable regulatory dose limit)

$Dose_{\alpha,i}$ - Committed Effective Dose for alpha emitting nuclide, i;

$I_{\alpha,i}$ - Intake of alpha emitting nuclide, i;

$ALI_{\alpha,i}$ - Stochastic Annual Limit on Intake for alpha emitting nuclide, i; from 10 CFR Part 20 Appendix B, Table 1, Column 2, or derived from ICRP 68[43], or other approved guidance as appropriate.

Alternatively, appropriate dose conversion factors can be used to calculate the dose:

$$Dose_{\alpha,i} = I_{\alpha,i} \cdot DCF_{\alpha,i} \qquad \textit{Eq. H-7}$$

Where:

$Dose_{\alpha,i,}$ - the Committed Effective Dose (CEDE) or the Committed Effective Dose (CED) as appropriate, for alpha emitting nuclide, i;

$I_{\alpha,i}$ - Intake of alpha emitting nuclide, i; and

$DCF_{\alpha,i}$ - Dose Conversion Factor for alpha emitting nuclide, i, from Federal Guidance Report No. 11, Table 2.1,[44] , ICRP 68[45], or other approved guidance as appropriate.

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[43] ICRP 68, page 17

[44] Federal Guidance Report No. 11, Table 2.1, "Inhalation"

[45] ICRP 68, Annexe B, Table B.1, "Ingestion and inhalation of particulates"

Scaling Factors

Since the ratio of beta-gamma to alpha activity is variable, a job specific scaling factor should be determined. A breathing zone air sample provides the most accurate measurement of the relative abundance of alpha and gamma emitters a worker could breathe. When breathing zone air samples and specific job coverage air samples are not available or not of sufficient total alpha activity to determine the nuclide mix which may involve multiple analyses and fractionating the sample, other air samples that are related to the same area and type of work may be used. If no representative air samples are available, representative loose surface contamination smears may be used.

Retention Models

Inhalation ALIs are based on biological retention models described in ICRP 30 and ICRP 78, as appropriate. The inhalation model predicts the retention of radioactive material based on the Activity Median Aerodynamic Diameter (AMAD).

Example 1 – Estimating a Dose from an Intake of Alpha Emitting Transuranics Using ICRP 30, NUREG/CR 4884 and Federal Guidance Report No. 11

At a station that is Alpha Level II with Alpha Level III areas and systems due to historical fuel failures, work was being performed in containment to remove a hotspot from the Reactor Coolant Storage System piping. The initial cut out of the hotspot and welding of the new pipe resulted in no airborne radioactivity levels.

However, during welding of the new pipe section water was encountered, which spoiled the weld. Additional cutting and grinding was required to remove the defective new weld. This disturbed Alpha Level III pipe sections. The additional cutting and grinding was not covered during the pre-job brief. The radiological controls in place for the originally planned job were inadequate for the new work performed.

During this additional cutting and grinding general area air samples indicated elevated airborne radioactivity. Four workers were involved in the additional cutting work. The original gamma spectroscopy of the general area air sample filter identified Am-241 at 59 KeV. Although Am-241 is rarely identified in a gamma spectroscopy samples, the count room Radiation Protection (RP) technician recognized that Am-241 was an alpha emitter. The technician failed to recognize the significance of the alpha airborne hazard.

An investigation was initiated to determine if the filter paper had been cross contaminated, and in the process the air sample filter paper was destroyed. It was suspected that a hot particle or small drop of water had contaminated the air sample filter and the RP technician was instructed to try to remove it. Using the sticky back of a Nucon smear, the technician dabbed the air sample filter surface above the suspected hot particle, which separated the top layer of the filter from the back layer. This action made it impossible to perform a gross alpha count on the filter to quantify the actual airborne level.

Backup air samples were obtained and indicated <0.25 total DAC. There were no facial contaminations or personnel contamination events. The unexpected high airborne activity was believed to be the result of disturbing the magnetite layer within the piping (which has a history of Alpha Level III levels) with the cutting wheel used to cut the piping.

Because the air sample filter was destroyed, no quantitative analysis could be performed. An information only gross alpha count was performed, but could not be used for internal dose calculations. Using the activity of Am-241 identified by gamma spectroscopy (which had 50% error) and the alpha nuclide distribution from primary resin 10CFR61 results, the total alpha DAC of the air sample was estimated at 55 DAC with a beta-gamma particulate DAC of 0.05. This equated to a DAC Ratio of 1100.

Normally, alpha exposures are assessed by scaling whole body (in-vivo) counts based on the ratio of alpha to beta gamma activity found on air samples. This

method was applied but due to the very high DAC Ratio estimated for this event, this method was not adequate to demonstrate that the workers had not exceeded NRC Performance Indicator levels for unanticipated exposure in excess of 100 mrem (1 mSv).

Pipefitter #1 was wearing a full-face respirator, while RP, the welder and pipefitter #2 were not in respiratory protection. With only one air sample, it is assumed that all individuals involved could have been exposed to significant alpha airborne radioactivity levels.

To ensure an accurate assessment of internal dose, collection of fecal samples began approximately 24 hours after the event, collecting 4 samples total from each individual. (Note that this incident occurred BEFORE the current revision to the guideline. If this incident were to occur today, Station RP would have requested that sampling start immediately and continue until five days had elapsed). Workers were given fecal specimen containers and instructions for collecting the samples. They were instructed to write the date and time of the sample on each container. Each worker was given a small ice chest to transport the samples back to the plant. The samples were stored in a freezer in a building outside the protected area until ready for shipment to a contract laboratory for alpha spectroscopy. Each sample was analyzed separately.

Doses to the exposed workers are calculated as follows:

- For each worker, the activity per sample is plotted versus sample time. The resulting curve show that the samples collected contained the majority of alpha activity passing directly through the GI tract. The plots start near zero, rise to a peak and then fall to near zero. The bioassay sample results are then summed in order to obtain a total activity for each nuclide for all fecal samples combined.
- Fecal sample collection was started approximately 24 hours after the event. Examination of the activity per sample versus time confirms that very little activity was missed prior to the start of sample collection.
- If a radionuclide is not detected, its activity is assumed zero.
- NUREG/CR- 4884 Internal Retention Fractions (IRFs) for the accumulated feces are interpolated to the last sample collection time.
- Interpolation of IRFs is performed in accordance with guidance from Regulatory guide 8.9.
- Federal Guidance Report No. 11 Dose Conversion Factors (DCFs) are used to determine Committed Effective Dose Equivalent (CEDE).
- IRFs and DRFs use lung clearance classifications based on ICRP-30 Metabolic Data:
    - Am-241 (W)
    - Cm-243/244 (W)
    - Pu-238 (Y)
    - Pu-239/240 (Y)

- NUREG/CR- 4884 Internal Retention Fractions for Cm-243/244 (W) are assumed the same as Cm-242 (W).
- As a check, dose is also calculated assuming ingestion rather than inhalation. The resulting dose is lower by a factor of 400. Therefore, inhalation is conservatively assumed for the final dose calculation.

Final calculated doses (CDE) to workers are:

RP 9.2 mrem (92 μSv)

Pipefitter #1 5.4 mrem (54 μSv)

Pipefitter #2 60 mrem (600 μSv)

Welder 41 mrem (410 μSv)

Example dose calculation for Welder:

Table H-1
Fecal Sample Results in pCi

Sample Date/Time	Days Post Event	Am-241	Cm-242	Cm-243/244	Pu-238	Pu-239/240
5/14/2010 8:20	1.93	0.172	0	0.0424	0.0638	0.0238
5/14/2010 22:20	2.51	15.1	0.0412	2.16	11.6	6.24
5/15/2010 9:00	2.96	2.2	0	0.377	1.4	0.778
5/16/2010 7:00	3.88	0.555	0	0.0871	0.456	0.351
	Accumulated	18.027	0.0412	2.6665	13.5198	7.3928

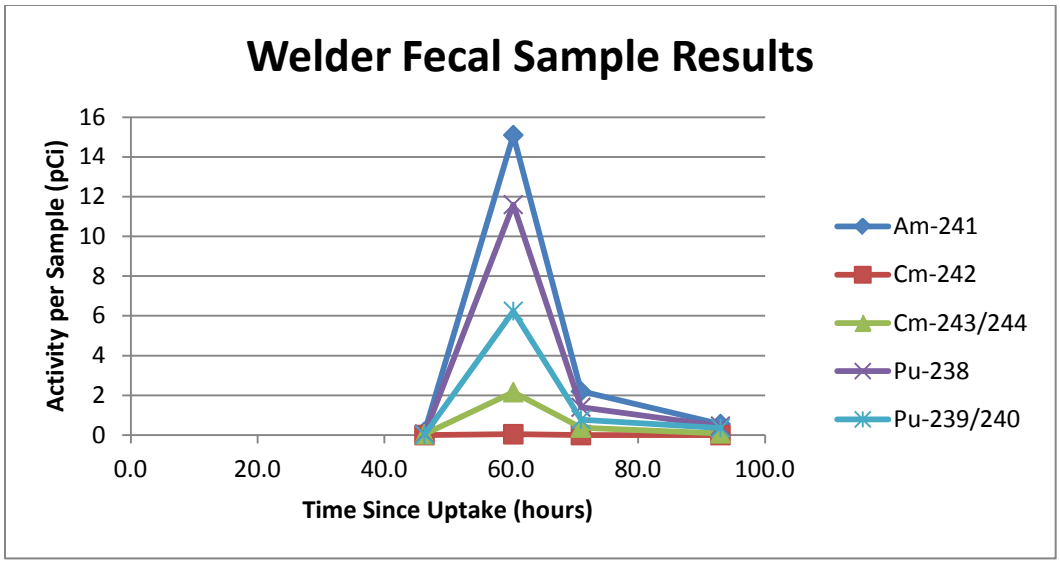

Figure H-1
Welder Fecal Sample Results

Table H-2
Interpolated IRF for Accumulated Feces at 1.93, 2.51, 2.96, 3.88 Days Post Event

Time (days)	Am-241 (W)	Cm (W)	Pu (Y)
1	0.0421	0.042	0.0521
1.93	0.160	0.159	0.193
2	0.177	0.176	0.213
2.51	0.228	0.226	0.272
2.96	0.286	0.282	0.337
3	0.292	0.288	0.344
3.88	0.349	0.344	0.407
4	0.358	0.352	0.417

Table H-3
Estimated Intake Reported to Three Significant Figures

	Total Fecal Activity (pCi)	Last Sample Retention Fraction	Intake (pCi)
Pu-238	13.52	0.407	33.2
Pu-239/240	7.3928	0.407	18.2
Am-241	18.027	0.349	51.6
Cm-242	0.0412	0.344	0.12
Cm-243/244	2.6665	0.344	7.75

Conservative estimate of intake

Table H-4
Dose Calculation for Welder to Two Significant Figures

Radionuclide	Intake (pCi)	Conversion (Bq/pCi)	Dose Conversion Factor (Sv/Bq)	Conversion (mrem/Sv)	CEDE (mrem)
Pu-238	33.2	0.037	7.79E-05	1E+5	9.6
Pu-239/240	18.2		8.33E-05		5.6
Am-241	51.6		1.20E-04		23
Cm-242	0.12		4.67E-04		0.21
Cm-243/244	7.75		8.30E-05		2.4
				Total	41mrem

Example 2 – Estimating a Dose from an Intake of Transuranics (using ICRP 68)

A Dosimetry Lab of a nuclear power plant received a Personal Air Sampler (PAS) filter to be measured for transuranic alpha activity. The filter was received on a Tuesday, after a long weekend. The filter was measured with a gas flow proportional counter, and the gross alpha transuranic activity result was $A = 14.35\ Bq$. The Dosimetry Health Physicist (DHP) was immediately informed of the result. The beta channel count rate for the filter was above 100 cpm, so according to procedure the Dosimetry Lab also initiated a gamma spectrometry measurement of the filter using a high purity Ge detector.

The DHP performed an initial assessment of the dose from transuranics. The Committed Effective Dose (CED) was calculated in accordance to the dosimetry procedure, using default parameter values for the effective dose coefficient, breathing rate and PAS flow rate:

$$CED = A \cdot \frac{B}{F} \cdot e_{inh}(50)$$

In the above equation, A is the measured gross alpha activity on the PAS, $A = 14.35\ Bq$. The value of the default parameters were as follows:

- The effective dose coefficient, $\boldsymbol{e_{inh}(50)}$, was conservatively taken to correspond to inhalation of Am-241, type M solubility, 5 μm AMAD, as given in ICRP 68, i.e.,

$$e_{inh}(50) = 2.7 \cdot 10^{-5}\ Sv\ Bq^{-1} = 2.7\ \ mrem/Bq$$

- $\boldsymbol{B}$, the breathing rate for the reference male was $\boldsymbol{B = 1.5\ m^3/hr = 25\ l/min}$ (from ICRP 89)
- $\boldsymbol{F}$, the PAS flow rate was $\boldsymbol{F = 2\ l/min}$

The default particle size for inhaled material at this nuclear plant was 5 μm AMAD, as recommended by ICRP 68 (P.3, Chapter 2.1 (5)), and as confirmed by several workplace sample measurements.

With these values, the initial dose calculation yielded $a\ CED \cong 484.3\ mrem$.

Since this value was above the dose verification level (100 mrem) and close to the dose investigation level (500 mrem), the DHP initiated additional actions as follows:

- He contacted the worker to obtain details about the exposure and to provide requirements for additional dosimetry;
- He confirmed with the Dosimetry Lab that they should provide the results of the gamma spectrometry measurements when available and will quarantine the filter for alpha spectrometry measurements at an external lab.

The worker met the DHP on Wednesday morning and provided the following information:

- He was working on Friday afternoon on disassembling a Heat Transport Pump (alpha Level III work).
- He was using protective equipment and dosimetry as specified in the Radiation Exposure Permit (REP), i.e., air supplied plastic suit and PAS worn inside the plastic suit.
- An alarming alpha CAM was set up in the room, but it was far away from the work site, and there was no alarm.
- He finished the job late Friday afternoon, doffed and bagged his plastic suit at the exit of the room (with the PAS still running), and performed the required contamination surveys when leaving the room (using a whole body contamination monitor and performing alpha frisking of the face and hands). There was no indication that the worker was contaminated.
- He proceeded to the change room and went through several other inter-zonal whole body contamination monitors and gamma sensitive portal monitors with no alarm.
- He did not perform a Whole Body Count (whole body gamma spectrometry measurement), as it was late Friday afternoon and he rushed home.
- He returned to work for the night shift on Tuesday night and was planning to perform a Whole Body Count (WBC) on Wednesday morning, before going home.

The DHP assisted the worker to perform a WBC, however, no activity was found above the MDA (MDA values for a 5 minute whole body count were ~ 120 Bq Co-60 and ~ 150 Bq Zr+Nb-95).

The results of the gamma spectrometry measurements on the filter were submitted to the DHP:

Radionuclide	Activity on PAS (Bq)
Ce-144	18
Co-60	140
Cs-137	1.6
Fe-59	22
Nb-95	200
Zr-95	110

The DHP calculated the potential intake for Co-60 and for Zr+Nb-95 based on the PAS measurements as follows:

$$I_{Co-60} = A_{Co-60} \cdot \frac{B}{F} \cdot \cong 1750\ Bq$$

$$I_{Zr+Nb} = A_{Zr+Nb} \cdot \frac{B}{F} \cdot \cong 3875\ Bq$$

(A_{Co-60} is the measured Co-60 activity on the PAS, 140 Bq, and A_{Zr+Nb} is the total activity of Zr-95 and Nb-95 on the PAS, 310 Bq. The DHP used total Zr-95 + Nb-95 activity, because the WBC had NaI detectors and could not easily resolve the gamma peaks of Zr-95 and Nb-95).

The DHP found that the gamma activity in the body corresponding to this intake in the day of the event was ~4500 Bq Co-60 + Zr/Nb-95. This was obtained by multiplying 0.8, the whole body retention fraction for Co-60 and Zr/Nb-95 at day 1, by the calculated intake for Co-60 + Zr/Nb-95, i.e., 5625 Bq. However, at this value (4500 Bq Co-60 + Zr/Nb-95) of activity in the body, the gamma sensitive portal monitors should have alarmed.

The DHP also calculated the Co-60 and Zr/Nb-95 activity remaining in the body on day 5 (i.e., on Wednesday) based on the assumed intake, and using a whole body retention fraction of ~0.08 at day 5 for both Co-60 and Zr/Nb-95. (In fact the WB retention fraction has different values for Co-60, Zr-95 and Nb-95, and also depends on the solubility type, but at day 5 these values are sufficiently close and can be approximated by 0.08):

$$M_{Co-60} = 1750 \cdot 0.08 = 140\ Bq$$

$$M_{Zr+Nb} = 3875 \cdot 0.08 = 310\ Bq$$

These values were well above the MDA values for WBC, and should have produced measurable results in the WBC at day 5.

Based on the fact that the calculated gamma activity intake was not consistent with the WBC results (and also there were no portal monitor alarms), the DHP suspected that the real intake was probably different (in fact smaller) than the intake calculated using the PAS.

The DHP requested the worker to submit fecal and urine samples collected over 24 h. The samples were submitted at day 7 after the assumed intake.

The samples were measured by alpha spectrometry at an external laboratory and the following results were obtained:

Nuclide	Activity in Fecal Sample (Bq)	Activity in Urine Sample (Bq)
Am-241	0.0640	0.000021
Pu-239/240	0.0520	<DL
Pu-238	0.0199	<DL
Cm-243/244	0.0145	<DL
Cm-242	0.0463	<DL

With the alpha spectrometry method used by the external Lab, the measurement Decision Level (Critical Level) for both fecal and urine measurements for all

radionuclides was approximately the same, 0.2 mBq (0.0002 Bq). The minimum detectable activity (MDA) was approximately 0.5 mBq.

To obtain information on the solubility type of the inhaled material, the DHP looked-up the bioassay excretion fractions for type S and M material for Am-241, 5 µm AMAD:

Am-241	Type S	Type M
Fecal Excretion Fraction, Day 7	2.473E-03	2.311E-03
Urine Excretion Fraction, Day 7	1.145E-06	5.767E-05
Excretion Fraction Ratio (Fecal to Urine)	2160	40

Since the urine measurements were at or below the measurement critical level (0.2 mBq), the data indicated that the transuranic material had type S solubility (type M would have given activities in urine at ~1.6 mBq for Am-241, well above the measurement MDA). The DHP decided to use parameters corresponding to type S solubility for dose calculations.

The DHP calculated the intake for each transuranic, dividing the fecal measurement value by the appropriate excretion fraction for type S material (the DHP used a software program such as IMBA to obtain the fecal excretion fractions):

Nuclide	Activity in Fecal Sample (Bq)	Fecal Excretion Fraction (type S solubility, 5 µm AMAD)	Intake (Bq)
Am-241	0.0640	2.47E-03	25.9
Pu-239/240	0.0520	2.47E-03	21.0
Pu-238	0.0199	2.47E-03	8.0
Cm-243/244	0.0145	2.47E-03	5.9
Cm-242	0.0463	2.40E-03	19.3

The results of the alpha spectrometry of the PAS filter (performed at an external lab) were also made available to the DHP:

Nuclide	PAS activity (Bq)	Fraction of total activity
Am-241	4.7	0.35
Pu-239/240	3.41	0.25
Pu-238	1.28	0.09
Cm-243/244	1.12	0.08
Cm-242	3.02	0.22
Total transuranic activity	13.53	1

The fractions of activities for the calculated intake (based on fecal sample measurements) were similar to the ones measured on the PAS filter:

Nuclide	Intake, (Bq) (based on fecal measurements)	Fraction of total activity
Am-241	25.9	0.32
Pu-239/240	21.0	0.26
Pu-238	8.0	0.10
Cm-243/244	5.9	0.07
Cm-242	19.3	0.24
Total transuranic activity	80.1	1

The intakes calculated from fecal measurements were smaller by a factor of approximately 2 than the intakes calculated using the PAS measurement:

Nuclide	Intake, (Bq) (based on fecal measurements)	Intake, (Bq) (based on PAS measurements)
Am-241	25.9	58.8
Pu-239/240	21.0	42.6
Pu-238	8.0	16.0
Cm-243/244	5.9	14.0
Cm-242	19.3	37.8
Total transuranic activity	80	169.1

The DHP decided that several factors were likely to account for these differences, mainly the actual breathing rate of the worker, the actual PAS flow rate, uncertainties in the biokinetic model, and the distance between the PAS air inlet and the breathing zone. The worker recalled that while bagging the plastic suit, the inlet tube of the PAS detached and was hanging for a short while inside the bag used for containing the plastic suit.

The DHP performed a new dose calculation based on the bioassay sample results. Using effective dose coefficients for type S solubility, 5 μm AMAD, he obtained the following results:

Nuclide	Intake, (Bq) (based on fecal measurements)	Effective Dose Coefficient, $e_{inh}(50)$ (mrem/Bq)	Dose (mrem)
Am-241	25.9	0.86	22
Pu-239/240	21.0	0.83	17
Pu-238	8.0	1.1	9
Cm-243/244	5.9	1.01	6
Cm-242	19.3	0.4	8
Total Committed Effective Dose			62

In the table above, the dose coefficients for Am-241, Cm-243/244 and Cm-242 were obtained for type S solubility using the internal dosimetry software IMBA.

Since all the information about the exposure was now consistent with the measurement data, and the dose was relatively small (below 100 mrem), the DHP decided that no further bioassay sampling was required, and assigned a dose of 62 mrem from transuranics for this intake. (If the calculated dose at this stage would have exceeded 100 mrem, the DHP would have requested additional fecal and urine samples, depending on the magnitude of the exposure – e.g., one more fecal/urine pair at day 15 for doses below 500 mrem, or several fecal/urine samples for doses above 500 mrem).

Note: Of course it also remains to calculate the dose from beta-gamma activity (including Pu-241 if applicable), but this is outside the scope of this example.

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