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EPRI Alpha Monitoring Guidelines for Operating Nuclear Power Stations

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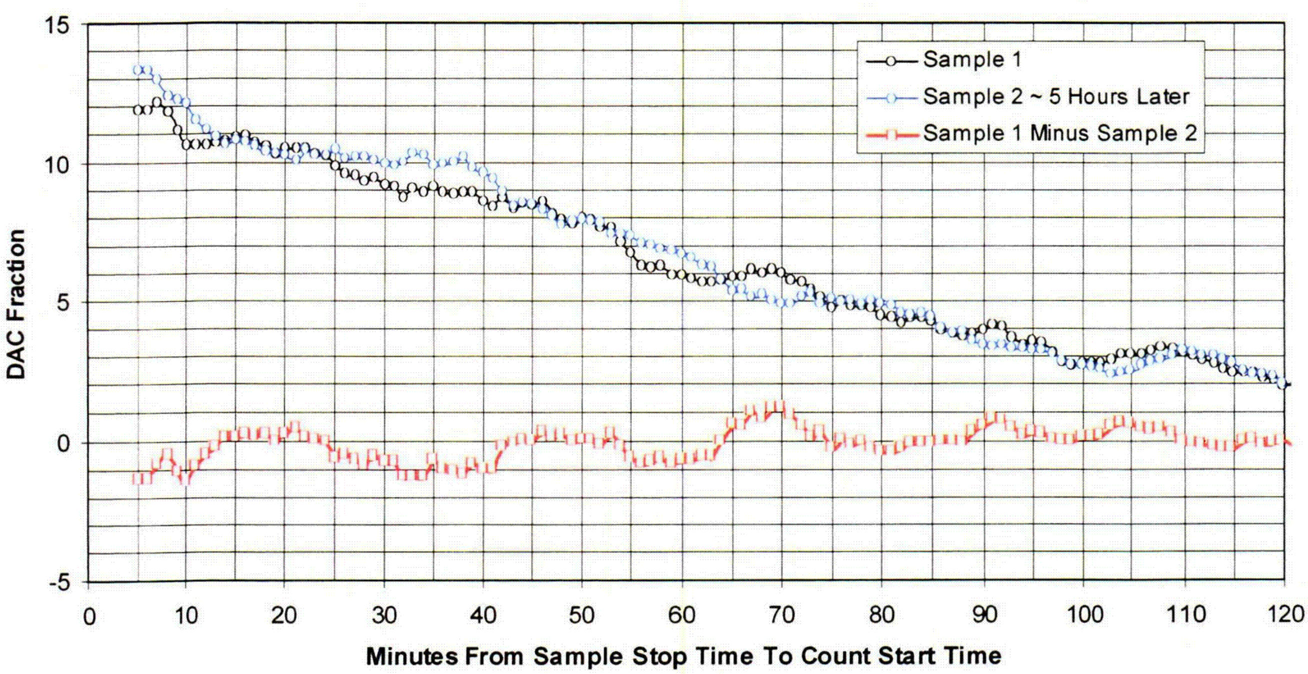
Technical Report



EPRI Alpha Monitoring Guidelines for Operating Nuclear Power Stations

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Final Report, November 2006

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REPORT SUMMARY

This report provides guidelines for monitoring alpha emitting radionuclides at PWR and BWR nuclear power plants developed by a group of radiation safety professionals. The project team intends the guidelines to demonstrate compliance with the regulatory requirements in 10 CFR Part 20, *Standards for Protection Against Radiation,* and to achieve excellence by following the recommendations in INPO 05-008, *Guidelines for Radiological Protection at Nuclear Power Stations.*

Background
Due to the restrictive annual intake limits for alpha emitting radionuclides, the nuclear power industry prudently decided to develop a standardized, graded approach to their monitoring. The approach described in these technical guidelines will allow the industry to maintain its excellence in radiation protection.

Objective
To provide guidelines for a standardized and graded approach to monitoring for alpha contamination based on predetermined workplace estimates of beta-gamma to alpha radioactivity.

Approach
A group of radiation safety professionals representing fifteen nuclear utility companies shared their collective experiences and expertise to develop, by consensus, these alpha-monitoring guidelines.

Results
This document provides a risk-informed, graded approach to monitoring alpha emitting radionuclides based on the relative abundance of alpha emitters compared to the beta-gamma emitters. The guidelines recommend workplace and personnel monitoring actions for work environments with Minimal (Level I Area), Significant (Level II Area), and Elevated (Level III Area) alpha to beta-gamma activity ratios. In Level I Areas, internal exposure from alpha radioactivity is unlikely to exceed 10% of the total internal dose, whereas in Level II Areas, alpha radioactivity is likely to contribute more than 10% of the total internal dose. In Level III Areas, internal exposure from alpha radioactivity is likely to exceed 90% of the internal dose based on the inhalation retention model.

Appendix A of the report outlines the entire Area Action Level Matrix program on one page to facilitate field-implementation. To support guideline implementation, the document includes information on characterization of a facility's alpha source-term, radon compensation, internal dose assessment, and instrumentation.

EPRI Perspective

A previous EPRI report, *Program Considerations for Addressing Alpha Emitting Radionuclides at Nuclear Power Plants* (1003126), provided a wide range of technical considerations in managing radiation protection programs where alpha emitting radionuclides are present in sufficient quantities to consider changes in various aspects of these programs. This document focuses on an industry need to provide specific technical guidelines for addressing alpha contamination at nuclear power plants. Utility experts, technical consultants, NEI, INPO and ANI provided significant input in development of these guidelines. EPRI remains committed to the implementation of these guidelines through communication of its content during industry meetings and document updates as determined necessary by our utility advisors. In addition, EPRI plans to make a spreadsheet for performing internal dose assessments available for download from the EPRI website.

Keywords

Transuranics
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Air sampling
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Activity ratios
Derived air concentration
Alpha

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1 INTRODUCTION

These guidelines for monitoring alpha emitting radionuclides at PWR and BWR nuclear power plants are intended to demonstrate compliance with the regulatory requirements in 10 CFR Part 20, "Standards for Protection Against Radiation" and to achieve excellence by following the recommendations in INPO 05-008, "Guidelines for Radiological Protection at Nuclear Power Stations" [Ref. 1].

The action levels and practices described in these guidelines are not a substitution for regulations, but are recommended good practices approved by industry consensus. Different practices may be acceptable, provided there is a basis for concluding they comply with the regulations.

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. The previously published EPRI Technical Report, "Program Considerations for Addressing Alpha Emitting Radionuclides at Nuclear Power Plants" [Ref. 2], may be used as an additional reference.

Radiation safety barriers are measures that provide assurance that radiation exposure to workers performing tasks in an area are monitored and controlled. A single occurrence of a degraded or failed safety barrier that results in an unintended occupational exposure greater than 100 mrem (TEDE) is one of the screening criteria for an "unintended exposure occurrence". If an internal exposure is not anticipated or specifically included in job planning or controls, the full amount of the dose (CEDE) should be considered unintended. See NEI 99-02, "Regulatory Assessment Performance Indicator Guideline" [Ref. 3].

Given their restrictive annual limits on intake, these guidelines recognize the importance and the potential impact of alpha emitting radionuclides with respect to the unintended exposure dose criteria.

2 APPROACH

This guideline recommends that workplaces be classified according to the relative abundance of alpha contamination as follows:

- Level I Areas: Minimal
- Level II Areas: Significant
- Level III Areas: Elevated

In Level I Areas, the relative abundance of alpha contamination is minimal. Internal exposure from the alpha emitters is not likely to exceed 10% of the total internal dose. Action Levels are recommended to verify the low abundance of alpha emitters when high contamination or high airborne radioactivity is present.

In Level II Areas, the relative abundance of alpha contamination is significant. Alpha emitters are likely to contribute more than 10% of the internal dose and airborne radioactivity levels expressed as DAC fractions. Contamination survey action levels are intended to alert radiation safety personnel of the presence of alpha emitters. When significant levels of alpha contamination are identified, additional smears are counted to adequately evaluate the magnitude and extent of alpha contamination in the area. Air sampling action levels are intended to demonstrate compliance with the requirement for posting airborne radioactivity areas. Air samples are counted for alpha whenever the total airborne radioactive material is likely to exceed the derived air concentrations specified in Appendix B to 10 CFR Part 20, or when an individual may receive an intake in excess of 0.6% ALI or 12 DAC hours in a week.

In Level III Areas, the relative abundance of alpha contamination is elevated. Internal exposure from the alpha emitters is likely to exceed 90% of the total internal dose based on the inhalation retention model. Most smears and all air samples should be counted for alpha contamination. Use of Personal Air Samplers as internal dosimeters is recommended. Also, alpha frisking of personnel is recommended when the beta-gamma to alpha activity ratio is below 50-to-1.

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

3
FACILITY CHARACTERIZATION OF THE ALPHA SOURCE TERM

Each facility should characterize the transuranic radionuclide distribution and alpha activity levels in areas of the plant where alpha contamination may be present in order to determine which Area Action Levels (I, II, or III) should initially be assigned. The primary source of alpha emitting transuranic nuclides is irradiated fuel leaking from fuel pin cladding defects. Transuranic nuclides in solid nuclear fuel are in the form of highly insoluble oxides. The principal transuranic nuclides are formed by neutron activation and decay predominately by alpha emission in the energy range of 4 to 6 MeV. Alpha contamination is most commonly associated with systems and components associated with fuel, and radioactive waste systems associated with the reactor coolant system and spent fuel pool.

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

- Assessing historical and current fuel cladding defects
- Determining the alpha radionuclide distribution
- Determining alpha contamination levels in the workplace

3.1 Historical and Current Fuel Cladding Defects

Significant fuel cladding defects can produce persistent alpha contamination even when these defects occurred many fuel cycles earlier in the plant's history. An assessment of fuel performance during the plant's operating life can provide insights into alpha contamination levels.

Significant fuel cladding defects during the current fuel cycle can result in significantly higher radiation, contamination, and airborne radioactivity levels, higher beta radiation energies, increased fuel fragment contamination, and a higher relative abundance of alpha activity. Radiological work planning for a refueling outage and plant maintenance activities should take into account current fuel cycle cladding defects. See INPO Significant Operating Experience Report 90-02, "Nuclear Fuel Defects" along with other related industry experience reports referenced in Section 3.4 below.

3.2 Alpha Radionuclide Distribution

The nuclide distribution of irradiated fuel has been thoroughly studied. Appendix B, Table B-2 lists the distribution of transuranic nuclides in spent fuel one-year and ten-years after discharge from the core. 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. 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 and annual limit on intake.

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
- Determining the alpha nuclide distribution for specific areas of the plant

The easiest and most conservative approach is to assume all alpha emitters are americium-241; whose derived air concentration is 3E-12 μCi/cc.

Alternatively, a single site-specific nuclide distribution can be established. A reasonable approach would be to collect a few samples of loose surface contamination from plant systems expected to have significant levels of alpha contamination (see Appendix B). The alpha nuclide distribution should not be based solely on samples collected for waste classification of dry activated waste. Based on alpha isotopic analysis of these samples, a conservative alpha nuclide distribution can be determined and applied to all gross alpha activity measurements throughout the site.

Alternatively, a facility may choose to establish area-specific nuclide distributions. Area-specific nuclide distributions may be appropriate for areas where significant levels of "old/aged" alpha contamination are present. Examples include waste storage buildings containing legacy waste, contaminated insulation that had been exposed to primary water leakage during a historical fuel defect cycle, pressurizer heater sleeves where old contamination can "hide-out", and contaminated components from another facility with an unknown history.

Appendix B, "Source Term Assessment" provides an example of how a nuclide distribution can be used to calculate an effective derived air concentration, inhalation stochastic annual limit on intake, and ingestion stochastic annual limit on intake.

After initial facility alpha characterization, follow-up assessments are recommended: 1) per fuel cycle (during the refueling outage), 2) if a fuel failure event occurs, and 3) when entering an unknown area, especially if there is a history of failed fuel.

3.3 Assigning Alpha Contamination Levels in the Workplace

Based on routine, pre-job, and job coverage contamination surveys, each workplace is determined to be a Level I, II, or III Area according to the relative abundance of gross alpha activity. The relative abundance of alpha activity associated with each "Area Level" is defined in Table 3-1 below. Corresponding workplace and individual monitoring actions for each "Area Level" are provided in Section 4, and are summarized in the "Area Action Level Matrix" of Appendix A.

Table 3-1
Area Levels

	LEVEL I	LEVEL II	LEVEL III
Activity Ratio ($\beta\gamma/\alpha$)	>30,000	30,000 – 300	<300

It is recognized that $\beta\gamma/\alpha$ ratios can vary substantially within a particular site and during a specific operating cycle due to a large number of variables (type of fuel failures or defects, age of contamination, half-lives of the radionuclides, and specific system operating history). However, these ranges should be viewed as trigger levels that initiate recommended actions for those areas where personnel may be occupationally exposed to airborne and loose surface alpha contamination.

Caution should be exercised in performing work where "old/aged" alpha contamination may be present, but inaccessible. As stated above, some examples of such areas include radioactive waste storage buildings containing legacy waste, contamination beneath primary system component heat insulation that may have experienced primary fluid contamination during a fuel defect cycle, pressurizer heater sleeves, and Rx Head CRD Thermo Sleeves. Another potential source of alpha contamination can be equipment from another facility with an unknown history. In these cases, consideration should be given to determining a separate area-specific distribution, assigning a higher Area Level, or assuming the most restrictive nuclide in the mix.

3.4 Operating Experience Reports

Listed below are some useful INPO reports (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.

OE22009 Alpha Contamination from Liquid Americium 241 Source (ANO), February 1, 2006.

OE21192 Foreign Plant Sends Highly Contaminated CRDM Test Equipment to Salem-2, August 8, 2005.

OE19696	Two Individuals Became Internally Contaminated During Transfer of Radioactive Filters (Millstone), update of OE19263, December 12, 2004.
OE19263	Workers Contaminated While Preparing Legacy Filters for Shipment (Millstone), October 8, 2004; updated by OE19696.
OE18151	Sandblasting Grit Material Found to Contain Naturally Occurring Radioactive Material (Browns Ferry), 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 (Davis Besse), 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 (Vermont Yankee), June 4, 2002.
OE12475	Operating with Failed Fuel (Fort Calhoun), July 12, 2001.
EA-96-496	CY 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 (Brunswick, Fort Calhoun), September 2, 1993.
SOER 90-02	Nuclear Fuel Defects (several sites referenced), July 24, 1990.

4
GUIDELINES

4.1 Workplace Monitoring

10 CFR 20 requires that licensees evaluate the magnitude and extent of the radiological hazards in the workplace. Contamination and airborne radioactivity surveys are performed for this purpose.

4.1.1 Contamination Surveys

The levels of loose surface contamination, the type of work being performed, and the physical nature of the contamination are used to predict potential airborne radioactivity levels and prescribe appropriate controls.

The importance of alpha contamination in this prospective assessment depends of 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

*As determined with a frisker or ion chamber equivalent

When the relative abundance of alpha activity is low (high activity ratio), few smears need to be counted for alpha. As the relative abundance of alpha activity increases (lower activity ratio), more smears are counted for alpha.

Table 4-1
Action Levels for Alpha Counting Smears

<table>
<tr><td>Activity Ratio
($\beta\gamma/\alpha$)</td><td>LEVEL I AREAS
(Minimal)
>30,000</td><td>LEVEL II AREAS
(Significant)
30,000 – 300</td><td>LEVEL III AREAS
(Elevated)
<300</td></tr>
<tr><td rowspan="2">Contamination Survey Action Levels</td><td>Count highest smear for α in HCAs</td><td>Count highest smears for α in areas >20K dpm/100 cm^2 $\beta\gamma$</td><td rowspan="2">Count sufficient number of smears for α to adequately evaluate area</td></tr>
<tr><td colspan="2">If >100 dpm/100 cm^2 α, count sufficient number of smears for α to adequately evaluate area</td></tr>
</table>

In Level I Areas where alpha contamination is expected to be minor, this condition should be verified by alpha counting the highest beta-gamma activity smear in high contamination areas (example: >100,000 dpm/100 cm^2).

In Level II Areas, the highest smears should be counted for alpha activity when the beta-gamma contamination exceeds 20,000 dpm/100 cm^2 or when alpha contamination is expected. If these smears show alpha contamination levels >100 dpm/100 cm^2, additional smears may need to be counted to determine the magnitude and extent of the alpha contamination in the area.

In Level III Areas, a sufficient number of smears should be alpha counted to adequately evaluate the magnitude and extent of the alpha contamination.

Alpha activity on smears may be determined with a one-minute count (or longer) on a laboratory instrument such as a zinc sulfide or gas flow proportional counting system. When the smears have high levels of beta-gamma activity, cross contamination of laboratory instrumentation could occur; in this case consider the use of a portable field survey instrument to measure gross alpha activity. See Appendix E for information on the limitations of alpha instrumentation such as "cross talk" and MDA considerations.

4.1.2 Airborne Radioactivity Surveys

Levels of airborne radioactivity determine area posting and may be used to assess internal exposure. Alpha airborne radioactivity is considered when it represents a significant fraction of its derived air concentration (DAC) value listed in 10 CFR 20, Appendix B.

The importance of alpha airborne activity depends of its relative abundance compared to beta-gamma airborne radioactivity. It is therefore convenient to define the term "DAC-Fraction Ratio".

$$DAC - Fraction\ Ratio = \frac{fDAC_{\alpha}}{fDAC_{\beta\gamma}}$$

where:

$f\mathrm{DAC}_{\alpha}$ - is the sum of each alpha emitting nuclide's concentration divided by its corresponding 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 with no contribution from natural sources. See Appendix C for information on radon interference.

When the relative abundance of alpha activity is low, few air samples need to be counted for alpha. As the relative abundance of alpha activity increases, more air samples are counted for alpha activity.

Table 4-2
Action Levels for Alpha Counting Air Samples

<table>
<tr><td>Activity Ratio
($\beta\gamma/\alpha$)</td><td>LEVEL I AREAS
(Minimal)
>30,000</td><td>LEVEL II AREAS
(Significant)
30,000 – 300</td><td>LEVEL III AREAS
(Elevated)
<300</td></tr>
<tr><td>DAC-Fraction Ratio
($\alpha/\beta\gamma$)</td><td>LEVEL I AREAS
(<0.1)</td><td>LEVEL II AREAS
(0.1 – 10)</td><td>LEVEL III AREAS
(>10)</td></tr>
<tr><td rowspan="2">Air Sampling Action Levels</td><td>If $f\mathrm{DAC}_{\beta\gamma}>1$, count A/S for α</td><td>If > "beta-gamma DAC-Fraction Action Level", count A/S for α</td><td rowspan="2">Count all A/S for α</td></tr>
<tr><td colspan="2">If DAC-Fraction Ratio ($\alpha/\beta\gamma$) is higher than expected for assigned Area Level, re-evaluate Area Level Assignment</td></tr>
</table>

In Level I Areas where alpha airborne is not expected to be significant, this 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, consideration should be given to reclassifying the area as Level II.

In Level II Areas where alpha activity is expected to be significant, air samples greater than the "beta-gamma DAC-fraction action level" should be counted for alpha activity. If the alpha DAC-fraction ratio exceeds 10, consideration should be given to reclassifying the area as Level III.

Each facility may develop a site-specific "beta-gamma DAC-fraction action level" for counting air samples for alpha activity. That action level should ensure that air samples are counted for alpha whenever the total airborne radioactive material is likely to exceed the derived air concentrations specified in Appendix B to 10 CFR Part 20, or when an individual may receive an intake in excess of 0.6% ALI or 12 DAC hours in a week.

In lieu of a site-specific action level, the following section of this report, "Technical Basis for Work Place Monitoring", recommends default action levels based on cobalt-60 and americium-241.

Alpha activity on air samples is normally determined with a ten-minute count (or longer) on a laboratory instrument such as a zinc sulfide or gas flow proportional counting system. Count times and air sample volumes should be sufficient to detect 0.3 of the effective alpha DAC.

4.1.3 Technical Basis for Workplace Monitoring

Table 4-3
Activity and DAC-Fraction Ratios for Co-60 and Am-241

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	3000	99.97	1E-8	47
	Am-241	1	0.03	3E-12	53
III	Co-60	300	99.7	1E-8	8
	Am-241	1	0.3	3E-12	92

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

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 need be counted to verify this condition.

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.

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 specified in Appendix B to10 CFR Part 20, or when an individual may receive an intake in excess of 0.6% ALI or 12 DAC hours in a week. This action level ensures compliance with the airborne radioactivity posting requirement in 10 CFR 20.1902.

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.

Alternatively, an action level can be based on the most restrictive gamma emitter (cobalt-60) and most restrictive alpha emitter (americium-241). Two examples follow:

- When the alpha is about 50% of the total airborne radioactivity DAC fraction (activity ratio ~3000), count an air sample for alpha when its beta-gamma DAC-fraction is greater than 0.3.
- When the 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 with beta-gamma activity should be counted to adequately characterize the levels and extent of alpha contamination.

4.2 Individual Monitoring

10 CFR 20 requires that licensees monitor exposure to radiation and radioactive material at levels sufficient to demonstrate compliance with the occupational dose limits. Personnel contamination surveys, measurement of radioactivity in the body, and air sampling can be used to monitor intakes of radioactive material.

4.2.1 Beta-Gamma Monitoring

When personnel contamination surveys identify a potential internal exposure, whole body counts are used to investigate and quantify the amount of internal contamination. When this whole body count exceeds a "whole body count investigation level", an internal dose assessment is performed to assign occupational dose. This standard approach to beta-gamma monitoring is outlined below:

- Personnel contamination monitoring when exiting the radiologically controlled area

  Whole body contamination monitors, plus portal monitors are used in the pause mode set to reliably detect contamination and identify potential internal exposures.

- Whole body counts are used to investigate when internal exposure is suspected, such as when:
  - Facial contamination in the respirator seal area and/or a positive nasal swipe are detected
  - Personnel contamination monitor alarms without the confirmed presence of external contamination
- An internal dose assessment is performed when a positive whole body count exceeds the "whole body count investigation level", indicating that an internal dose is likely to exceed 10 millirem committed effective dose equivalent (See section 4.3.4).

4.2.2 Alpha Monitoring

To account for alpha emitters, these additional monitoring practices are recommended:

- A whole body count should be considered if an individual is exposed to airborne concentrations >1 DAC alpha or an intake >4 DAC-hours alpha (plus βγ per 7 consecutive days).

  (The DAC value is reduced by the assigned protection factor when respiratory protection is used.)
- In Level III Areas, the use of Personal Air Samplers as internal dosimeters is recommended.
- In Level III Areas, where the βγ/α activity ratio is < 50, alpha frisking of personnel is recommended.
- When an internal dose assessment is performed, the committed effective dose equivalent from alpha emitters should be included if it is likely to exceed 10 mrem.
- Consideration should be given to collecting excreta samples when an intake from alpha emitters exceeds 5% of an Annual Limit on Intake (ALI).

  (Urine samples contain soluble material that has been absorbed into the blood from the lungs or the GI tract. Fecal samples contain insoluble material that has been rapidly eliminated through the GI tract.)

4.2.3 Technical Basis for Individual Monitoring

Personnel contamination monitors are very effective at identifying individuals exposed to high airborne radioactivity. Taken soon after an exposure, whole body counts provide the most accurate measurement of a worker's internal contamination from gamma emitters. Internal contamination from alpha emitters can be determined using scaling factors based on a representative air sample, or in the absence of a representative air sample, smear or radioactivity samples from the work area.

As the relative abundance of alpha contamination increases, the ability of personnel contamination monitors to detect a low level of internal exposure decreases. To compensate, personal air samplers and additional whole body counts should be considered to ensure that individuals with internal contamination are identified.

The action level for considering alpha frisking is based on guidance in IE Circular No. 81-07 [Ref. 4] regarding the detection of removable contamination (1000 dpm/100 cm^2 beta-gamma and 20 dpm/100 cm^2 alpha).

The 5% ALI action level for excreta sampling is half the 10% ALI "investigation level" recommended in Regulatory Guide 8.9 [Ref. 5].

4.3 Internal Dose Assessment

10 CFR 20 requires that licensees assess the committed effective dose equivalent (CEDE) to adults likely to exceed 10% of the Annual Limits on Intake. INPO recommends that workers suspected of receiving 10 mrem CEDE should be whole body counted to accurately assess the dose.

Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program" [Ref. 5] describes methods acceptable to the NRC staff for estimating the intake of radionuclides using bioassay measurements.

Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses" [Ref. 6] provides methods acceptable to the NRC staff for calculating occupational doses when the intake is known.

Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Dose Data [Ref. 7], provides guidance on when internal monitoring is required. When monitoring is required, instructions are provided on recording and reporting internal dose.

While the internal dose assessment method described in this section and detailed in Appendix D is consistent with NRC guidance, other methods of assessing internal dose described in the NRC regulatory guides are equally acceptable.

The ALIs used in this section and in Appendix D are all stochastic ALIs. Non-stochastic ALI's associated with organ doses need not be calculated if the Committed Effective Dose Equivalent (CEDE) does not exceed 1 rem, and there are no overexposures (Ref. 7, Section 2.2).

4.3.1 Estimates of Intake and Assessment of Dose

Investigative whole body counts provide a timely and direct measure of the deposition of gamma emitting radionuclides in the body. An estimate of the intake of gamma emitting radioactivity is determined from these whole body count measurements and applicable intake retention fractions. Caution should be exercised to avoid including external skin contamination in the assessment of the intake.

The CEDE from gamma emitters can be calculated using the estimates of the intake and the applicable 10CFR20 Appendix B, Stochastic Annual Limits on Intake. The CEDE from alpha emitters is calculated based on the CEDE for gamma emitters times the alpha-to-gamma scaling factor (See Appendix D) or from scaling the alpha-emitter intakes from the beta-gamma emitter intakes and calculating the CEDE from the intake of each radionuclide.

4.3.2 Scaling Factors

Since the ratio of alpha-to-gamma activity is variable, a job specific scaling factor should be determined (See Appendix D). A "representative" air sample provides the most accurate measurement of the relative abundance of alpha and gamma emitters. When specific job coverage air samples are not available, other air samples that are related to the same area and type of work may be used. If no representative air samples are available, job specific loose surface contamination smears may be used.

Typically, transuranic radionuclides are the only dose significant non-gamma emitters. While the levels of activity for other non-gamma emitters such as iron-55 may be high, expressed as a fraction of its corresponding ALI, its contribution to dose is small.

4.3.3 Retention Models

Inhalation and ingestion ALIs are based on biological retention models. The inhalation model predicts the retention of radioactive material assuming a particle size distribution having an activity aerodynamic diameter (AMAD) of 1-micron. However, the particle size distribution is typically much larger.

In practice, the larger inhaled particles are rapidly cleared from the upper respiratory track and cleared through the GI tract. As a result, the ingestion model more accurately describes the actual retention of radioactive material.

When the retention of radioactive material closely follows the inhalation model, the inhalation ALIs should be used to calculate internal dose. When the retention closely follows the ingestion model, the ingestion ALIs should be used.

Appendix D provides details on how initial and follow-up whole body counts are used to determine the "retention fractions" and how these retention fractions are compared to the ingestion and inhalation "intake retention fractions" in NUREG/CR-4884 [Ref. 8].

Transuranic alpha emitting nuclides are highly insoluble. Their retention is similar to other insoluble gamma emitters. Cobalt-58 and cobalt-60 are well suited for determining retention fractions for alpha emitters because of their relative high abundance and solubility class.

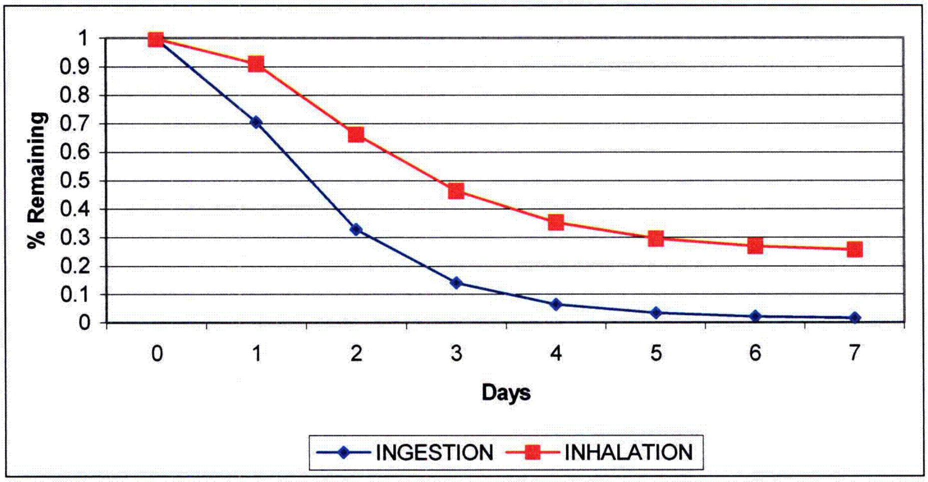

Figure 4-1
ICRP 30 Ingestion and Inhalation Retention Models for Cobalt-60

4.3.4 Whole Body Count Investigation Level

An "investigation level" is defined as the intake of radioactive material at which an internal dose assessment should be performed. That level should roughly equate to about 10 mrem CEDE.

After taking into account the expected retention of internal contamination and the expected fraction of dose attributable to alpha emitters, a whole body count investigation level corresponding to a nominal value of 0.1% ALI (as measured by whole body count) is recommended for Level I and II Areas (see example in Appendix D). Lower values should be considered for Level III Areas.

4.3.5 Personal Air Sampling

When personal air samplers are used as internal dosimeters, the worker's intake of beta-gamma and alpha contamination is determined directly from the air sample, modified by the ratio of the worker's breathing rate (20 liters per minute) and the sampler's flow rate. This tool provides the measured intake for any duration of time that it is used by the worker. However, separate filters shall be used when workers change respiratory protection methods to appropriately account for respirator protection factors. Personnel air samplers are not intended to be used for assessing airborne concentrations for posting purposes.

$$I = 20\frac{A_f}{F}$$

where:

I	intake of airborne radioactivity (same activity units as A_f)
A_f	activity measured on the personal air sampler (same activity units as I)
20	light work breathing rate for reference man, lpm
F	air sampler flow rate, lpm

5
TRAINING AND COMMUNICATIONS

The presence of TRU 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 TRU can pose perceived 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, respect, and resources are used to manage the TRU source term.

10CFR19.12, Instructions to Workers, states "all individuals who in the course of employment are likely to receive in a year an occupational dose in excess of 100 mrem shall be instructed in the health protection problems associated with exposure to radiation and/or radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed". To meet these objectives, the nuclear industry uses a multi-tiered instructional approach including:

- Management Training
- Radiation Protection Personnel Training
- General Employee Training / Radiation Worker Training (GET/RWT)
- The Radiation Work Permit (RWP)
- Radiological Postings in the Workplace

5.1 Management Training

Management training regarding the impact of TRU may focus on the tools needed to provide adequate worker protection while maintaining a productive working environment. Discussions or training activities with management may include the following elements:

- Alpha Nuclide Distribution
- Internal Exposure Potential of TRU (comparison of ALI and DAC Values)
- Effective Monitoring Instrumentation
- Work Place Impact (e.g., alpha frisking, personal air sampling, etc.)
- Increase Use of Respirators and Engineering Controls
- Increase in Whole Body Counts and Bioassay Sampling
- Documentation of Internal Dose
- Radiation Protection Staffing

5.2 Radiation Protection Personnel Training

It is suggested that radiation protection personnel be trained to fully understand the internal exposure potential of TRU and the measurement challenges posed by alpha radiation. The training program may contain the following elements:

- Alpha Nuclide Distribution
- Comparisons of ALI and DAC Values
- Use of Personal Air Samplers
- Air Sampling Capabilities
- Radon Decay Chain
- Radon Progeny Interference
- Free Release of Materials
- Contamination Control
- Alpha Frisking Instrumentation and Techniques
- Alpha Counting Instrumentation and Sensitivities
- Whole Body Count Action Levels
- Bioassay Action Levels
- Internal Dose Assessment – Inhalation vs. Ingestion Intakes

5.3 General Employee/Radiation Worker Training

The level of Radiation Worker Training provided should be commensurate with the level of the potential radiological safety issue. 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 those topics listed below.

- Alpha Nuclide Distribution (e.g., plutonium, americium, curium)
- TRU Internal Exposure Hazards
- Use of Personal Air Samplers
- Contamination Control
- Alpha Frisking
- Urine and Fecal Sampling

5.4 Radiation Work Permit

For those plants with a limited number of Level II or Level III areas that are not routinely entered by the normal work force, the plant may opt not to modify General Employee Training / Radiation Worker Training. Alternatively, other methods of providing instruction to workers entering Level II or Level III areas are acceptable, such as job-specific training or pre-job briefings provided through the RWP (or equivalent) process. Job-specific training should be documented and retained.

5.5 Radiological Postings

Radiological postings are used to caution workers of the potential radiological safety hazards in the workplace. Consideration should be given to posting areas as High Alpha Contamination Areas as a method of cautioning workers of the presence of high alpha contamination in the workplace.

6 REFERENCES

1. INPO 05-008, "Guidelines for Radiological Protection at Nuclear Power Plants", Atlanta, GA, December 2005.
2. EPRI Technical Report - 1003126, "Program Considerations for Addressing Alpha Emitting Radionuclides at Nuclear Power Plants", Final Report, November 2001.
3. NEI 99-02, Revision 4, "Regulatory Assessment Performance Indicator Guideline", April 2006.
4. IE Circular No. 81-07, "Control of Radioactively Contaminated Material", U.S. Nuclear Regulatory Commission, Washington DC, May 14, 1981.
5. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program", U.S. Nuclear Regulatory Commission, Washington DC, July 1993.
6. Regulatory Guide 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses", U.S. Nuclear Regulatory Commission, Washington DC, July 1992.
7. Regulatory Guide 8.7, Revision 2, "Instructions for Recording and Reporting Occupational Radiation Dose Data", U.S. Nuclear Regulatory Commission, Washington DC, November 2005.
8. NUREG/CR-4884, "Interpretation of Bioassay Measurements", U.S. Nuclear Regulatory Commission, Washington DC, July 1987.
9. INPO ACAD-00-007, "Guidelines for Radiation Worker and Radiological Respiratory Protection Training", Atlanta, GA (April, 2000).

A
APPENDIX: AREA ACTION LEVEL MATRIX

Table A-1

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B
APPENDIX: SOURCE TERM ASSESSMENT

Table B-1
Partial List Associated With Alpha Contamination

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Table B-2
Principal Transuranic Nuclides in Spent Fuel

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Radionuclide Distribution

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

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(Table B-3 Continued)

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

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(Table B-4 Continued)

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

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(Table B-5 Continued)

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C
APPENDIX: RADON COMPENSATION

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Sources and Decay of Natural Airborne Radioactivity

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Figure C-1
Decay of Natural Radioactivity

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Compensation Methods

Background Method

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

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Half-Life Method

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Figure C-3
Half-life Method

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Annualar Kinetic Impactors (AKI)

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References

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D

APPENDIX: INTERNAL DOSE ASSESSMENT

Introduction

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Gamma Dose Assessment

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Alpha Dose Assessment

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Scaling Factors

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Example

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Table D-1
Event and WBC Date and Times

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Table D-2
WBC Data

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Table D-3
WBC Investigation Level

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Determining the Retention Model

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Table D-4
Comparison of Measured and Predicted Activity Fraction Remaining for Inhalation and Ingestion

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Determining the Alpha Scaling Factor

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Table D-5
Representative Air Sample Data

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Table D-6
Alpha Scaling Factors

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Determining Internal Dose

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Table D-7
Example Dose Calculation Using the Alpha Scaling Factors

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E

APPENDIX: THE DETECTION AND ANALYSIS OF ALPHA-EMITTING RADIONUCLIDES

Gross Counting

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High-Throughput Gas-flow Proportional Counters

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Zinc-Sulfide

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Ion Chamber Survey Instruments

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Gamma Spectroscopy

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High-purity Germanium (HPGe) Detectors

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REGe

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Alpha Spectroscopy

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Laboratory Based

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Portable

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Figure E-1
Spectrometer Collimator

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General Area Air Sampling

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GM-based CAMs

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Proportional Counter CAMs

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Passively Implanted Planar Silicon (PIPS) CAMs

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Sample Counting Considerations

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Handheld Friskers

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Proportional Counter Friskers

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Scintillation Counter Friskers

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References

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