Appendix D

Radiological Health and Safety Procedures

1.0 PURPOSE

This procedure is for training Energy Fuels Resources personnel and contractors who work at the Piñon Ridge Mill in Radiological Health and Safety.

2.0 APPLICABILITY

This training program provides procedures for instructing Mill employees, contractors, and visitors of their potential radiation doses and the health implications of those doses.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Current Colorado Radioactive Materials License

3.1.2 Colorado Radiation Control Regulations (6 CCR 1007-1)

3.1.3 NRC Regulatory Guide 8.31, "Information Relevant to Ensuring That Occupational Radiation Exposures at Uranium Recovery Facilities will be As Low As is Reasonably Achievable."

3.2 *APPENDICES*

Appendix A - Radiation Safety Briefing Checklist

Appendix B - Radiation Worker Training Syllabus

Appendix C - Sample Radiation Safety Test

4.0 EQUIPMENT AND MATERIALS

4.1 *MATERIALS*

4.1.1 Training materials will be updated periodically, as necessary, based on the current operating status of the milling facility and will address problems that have been experienced. A wide range of subjects, films, articles, new regulatory guides, personnel exposure incidents, and any other topic deemed appropriate by the Radiation Safety Officer (RSO) or designee will be addressed during radiation safety training/meetings.

4.1.1.1 A checklist is used for the Radiation Safety Briefing.

4.1.1.2 A syllabus is utilized for Basic Radiation Worker Training.

4.1.1.3 Topical information is used for Routine Periodic Radiation safety Retraining.

4.1.1.4 A written Radiation Work Permit work plan is generally used for Radiation Work Permit Training.

4.1.1.5 The specific Radiological Health and Safety Procedures contained in this plan are used for Training.

4.1.1.6 Competency checklists which mimic the Operating Procedures will be developed and will be used for the Competency Training for specific jobs and tasks.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.0 RESPONSIBILITY

5.1 The RSO or Assistant RSO is responsible for:

5.3.1 Determining the need for Radiation Safety Training or specialized training.

5.3.2 Training of all personnel, contractors, and visitors to the Mill.

5.3.3 Tracking the training provided to employees and contractors.

5.3.4 Grading the Radiation Safety Training Test. New employees are required to re-train and re-take test if they do not pass (70%).

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 Designated Training Personnel – An individual or group designated as being qualified to provide appropriate training to key individuals or to assess qualifications of key individuals.

6.2 *FREQUENCY*

6.2.1 Contractors, vendors, or visitors will receive a Radiation Safety Briefing unless they are accompanied by Energy Fuels personnel. Additional training will be provided as specified by the RSO or designee commensurate with their expected exposure potential and duration of time onsite.

6.2.2 New Energy Fuels employees and contractor employees designated by the RSO or designee must attend Basic Radiation Safety Training prior to starting their work assignments. Quarterly radiation safety/training meetings will be held for the purpose of accumulating an estimated 4 hours of yearly retraining of all mill employees.

6.2.3 Training associated with work to be performed under Radiation Work Permits will be completed prior to employees performing the work.

6.2.4 Training in execution of these radiological health and safety procedures will be completed prior to employees performing the work independently.

6.2.5 Competency checklists for Operating Procedure Training are designed to be completed within 90 days of an employee's assignment or reassignment and prior to performing the work independently.

7.0 PROCEDURES

7.1 The following sections describe the training categories, training documentation, and training testing required for Energy Fuels employees, contractors, and visitors. All unescorted personnel, contractors, and visitors who perform work will receive a briefing concerning potential radiological hazards. Depending on the work activity, duration and exposure potential, additional training will be provided. Listed below are the categories for training.

7.1.1 Training Categories

7.1.1.1 Radiation Safety Briefing – visitors, vendors, short-term contractors or contractors working in low radioactivity area.

7.1.1.2 Basic Radiation Worker Training – New employees, long-term contractors, or contractors working in restricted areas – test required for employees and contractors as determined by the RSO or designee.

7.1.1.3 Annual Radiation Safety Retraining – Energy Fuels employees receive approximately 4 hours of radiological refresher training per year. Contractors, as needed per RSO or designee.

7.1.1.4 Radiation Work Permit Training – As required by Radiation Work Permits.

7.1.1.5 Radiological Health and Safety Procedure Training – As determined by the RSO.

7.1.1.6 Operator, maintenance, and Utility Crew Competency Training – the radiological health and safety implications of individual operating assignments will be covered in the course of individual task or job-specific training programs. Employees will be required to read operating procedures related to their assignment and signify their understanding of these procedures.

7.1.2 Training Documentation – All Energy Fuels radiological training will be documented to show the course content, the attendees, the trainer, and the test results.

7.1.3 New employees existing employees, and contractors are required to pass the training exams with a score of 70% or more.

Radiological Health and Safety Training Procedure RH-010

Appendix A
Radiation Safety Briefing Checklist

Radiation Safety Briefing Checklist

1. Administrative
   a) Hazard Recognition
   b) Waiver Release
   c) Contractor Informational Form
   d) Prior Occupational Radiation Dosage Form
   e) Contractor Dose and Monitoring Evaluation Form
   f) Part 4 and 10 of Radiation Regulations

2. Baseline Monitoring Determination
   a) Bioassay
   b) TLD/pocket dosimeter
   c) Other

3. Site Rules
   a) Site entry and departure
   b) Eating, tobacco, and drinking policy
   c) Contamination control

4. Radiation Hazards and pathways (external and internal)
   a) Caution signs and posting requirements
   b) i) Radiation Symbol
      ii) Posting of Radiation
      iii) Airborne Radioactivity Area
      iv) Respiratory Protection Required

Radiological Health and Safety Training Procedure RH-010

Appendix B

Basic Radiation Safety Training Syllabus

Basic Radiation Safety Training Syllabus

- Nature of ionizing radiation, radioactivity and basic atomic structure
- Characteristics of alpha, beta and gamma radiation
- Principles of half life and radioactive decay
- Natural and man-made sources of Ionizing Radiation
- Units of exposure
- Radiation protection regulations – CDPHE Part 4 and 10, NRC, MSHA, OSHA
- Measurement of ionizing radiation
- Health effects of ionizing radiation as specific to uranium and progeny
- Principle of ALARA – methods of exposure control
- Contamination control (hygiene) and decontamination techniques
- Waste management
- Emergency procedures
- Use of respirators when appropriate per CDPHE requirements (CDPHE 2005)
- Safety designed features of process equipment
- Ventilation systems, effluent controls and inspection requirements
- Waste management systems and inspection requirements
- Standard operating procedures
- Security and access control to designated areas
- Measurements of airborne radioactive material
- Bioassay (urinalysis, fecal and in vivo counting)
- Surveys to detect contamination of personnel and equipment and decontamination methods
- Personnel dosimetry

Radiological Health and Safety Training Procedure RH-010

Appendix C

Sample Radiation Safety Training Test

Sample Radiation Safety Training Test

1. The basic philosophy of radiation protection includes three factors:
   a.. _____
   b. _____
   c. _____
2. ALARA means_____
_____
_____
3. The primary responsibility for radiation protection rests with the _____
4. Match the type of radiation listed with its characteristics. Each type of radiation may have more than one
   characteristic and each characteristic may fit more than one type of radiation.

_____ alpha	a. Identical to orbital electrons
	b. No mass, no charge
_____ beta	c. Electromagnetic radiation
	d. Very penetrating
_____ gamma	e. Identical to a helium nucleus
	f. Will not penetrate the layer of dead skin cells
_____ x-rays	g. Will penetrate up to about 1 cm in tissue
	h. Will be stopped by a piece of paper

5. Three ways of reducing exposure to external radiation are:
   a. _____
   b. _____
   c. _____
6. Three routes of entry of radionuclides into the body are:
   a. _____
   b. _____
   c. _____
7. For each of the three routes of entry listed above, name two ways of reducing or preventing intake:
   a.. _____
   _____
   b. _____
   _____
   c. _____
   _____

8. The unit of radioactivity is _____.

9. The unit of radiation dose is _____.

10. The most important risk of eposure to low-level radiation is _____.

11. The maximum allowable annual dose to a radiation worker is _____.

12. True or False

_____Radionuclides in the thorium decay series emit all three types of radiation.

_____Urnaium and thorium are naturally occurring elements in the earth's crust.

_____Radium is part of the decay series of uranium only.

_____Radon (Rn-222) is a noble gas.

_____Because uranium and thorium are naturally occurring, doses from exposure to them are not as hazardous as the same dose from man-made radionuclides would be.

_____Colorado is an "agreement state" so it does not have to follow the same regulations as the states that are controlled by the Nuclear Regulatory Commission.

_____The maximum allowable dose to a member of the general public from Energy Fuel operations, including radon decay products, is 100 mrem per year.

1.0 PURPOSE

This procedure describes decontamination methods used to remove radioactive materials from people or objects.

2.0 APPLICABILITY

This procedure applies to all personnel, their clothing, equipment, materials, and vehicles within the restricted area of the Mill prior to release to unrestricted areas.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Colorado Radiation Control Regulations (6 CCR 1007-1).

3.1.2 Procedure RH-070 Release of Equipment.

3.1.3 Procedure RH-200 Personnel Release Surveys.

3.1.4 Procedure RH-180 Respirator Inspection, Maintenance, Cleaning, and Storage.

3.2 *APPENDICES*

Appendix A – Table 1 Personnel Decontamination

Appendix B – Personal Release Survey Form RH-200A

Appendix C – Equipment and Materials Survey Form RH-070A

4.0 EQUIPMENT AND MATERIALS

4.1 *MATERIALS (Dependent on Decontamination Method)*

4.1.1 Soap and water

4.1.2 Lava soap

4.1.3 Soft brush

4.1.4 Detergent

4.1.5 Washing machine

4.1.6 Power washer

4.1.7 Steam cleaner

4.1.8 Hand tools (shovel and wire brush)

4.1.9 Sulfuric acid

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or Assistant RSO is responsible for providing guidance on which decontamination method to use.

5.2 The Radiation/Security Technician (RST) is responsible for directing the decontamination method or performing the decontamination.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

6.0 PREREQUISITE INFORMATION

6.1 *SAFETY*

6.1.1 Piñon Ridge Mill Health and Safety Plan.

6.2 *FREQUENCY*

6.2.1 Whenever contamination above background is encountered on personnel, their clothing, equipment or vehicles prior to leaving the restricted area of the Mill.

6.2.2 As directed by the RSO or Assistant RSO.

7.0 PROCEDURES

7.1 DECONTAMINATION OF PERSONNEL

7.1.1 The RST or his designee is to assist personnel in removing contamination from their bodies using the procedures listed in Appendix A, Table 1 prior to leaving the restricted area of the Mill. Begin with the first procedure and then proceed stepwise, with radiation surveys between steps, to the more aggressive procedures, as necessary. Continue decontamination procedures until the contaminated area has been removed to “background” or until advised to stop by the RSO or his designee.

7.1.2 See procedure RH-200 for the methods of monitoring personnel intending to leave the restricted area of the Mill.

7.1.3 Complete form RH-200A (Appendix B) or the equivalent to document the decontamination surveys.

7.1.4 File form RH-200A and retain for 3 years (6 CCR 1007-1 4.42.1).

7.2 *DECONTAMINATION OF CLOTHING*

7.2.1 Contamination on clothing can be removed prior to leaving the restricted area of the Mill using detergent and water and a brush. As an alternative, use detergent and water and a washing machine.

7.2.2 File form RH-200A and retain for 3 years (6 CCR 1007-1 4.42.1).

7.3 *DECONTAMINATION OF RESPIRATORS*

7.3.1 See Procedure RH-180 Respirator Inspection, Maintenance, Cleaning, and Storage.

7.4 *DECONTAMINATION OF EQUIPMENT AND MATERIALS*

7.4.1 Contamination on personal effects (cell phones, wallets, etc) above background must be removed prior to leaving the restricted area.

7.4.2 Begin by removing large deposits of ore, tailing, or yellowcake with a hand tools such as a shovel or wire brush. Use a respirator if dust is generated in the process.

7.4.3 Clean the item with a power washer or steam cleaner and monitor as specified in Procedure RH-070.

7.4.4 If the contamination is still above background, clean with a grinder using a respirator or clean with sulfuric acid.

7.4.5 Let the equipment or materials dry before monitoring.

7.4.6 BEFORE PAINTING monitor the levels of contamination on equipment and materials and document the results on Form RH-070A (Appendix C). Do not paint over contamination that is above the release limits.

7.4.7 File and retain for 3 years Form RH-070A (6 CCR 1007-1 4.42.1).

7.5 ***DECONTAMINATION OF VEHICLES***

7.5.1 Contamination on vehicles (trucks and cars) above the release limits must be removed prior to leaving the restricted area.

7.5.2 The release limits for vehicles are alpha or beta are equal 15,000 dpm/100 cm^2 max; 5,000 dpm/100 cm^2 average; or 1,000 dpm/100 cm^2 removable for uranium.

7.5.3 Vehicles are to be cleaned in the Truck Wash area by first removing large deposits of tailing, ore, or mud using high pressure sprays and/or hand tools (e.g. a shovel or wire brush). Use a respirator if dust is generated in the process.

7.5.4 Wash the vehicle in the Truck Wash. Monitor as specified in Procedure RH-070.

7.5.5 File and retain form RH-070A.

Table 1
Personnel Decontamination

Method	Surface	Action	Techniques	Advantages	Disadvantages
Soap and Water	Skin and Hands	Emulsifies and dissolves contaminant	Wash 2-3 minutes and monitor. Do not wash more than 3-4 times.	Readily available and effective for most radioactive contamination.	Continued washing will abrade the skin.
Soap and Water	Hair	Emulsifies and dissolves contaminant	Wash several times. If contamination is not lowered to acceptable levels, shave the head and apply skin decontamination methods.	Readily available and effective for most radioactive contamination.	Continued washing will abrade the scalp.
Lava soap, soft brush and water	Skin and Hands	Emulsifies and dissolves contaminant Use care	Use light pressure with heavy lather. Wash for 2 minutes, 3 times. Rinse and monitor. Do not scratch or erode skin. Apply lanolin or hand cream to prevent chapping.	Readily available and effective for most radioactive contamination.	Continued washing will abrade the skin.
Tide or other detergent	Skin and Hands	Emulsifies and dissolves contaminant Use care	Make into a paste. Use with additional water with a mild scrubbing action. Use care not to erode the skin.	Slightly more effective than washing with soap.	Will defat and abrade skin: must be used with care.

NOTE: Begin with the first listed method and then proceed step by step to the more severe methods, as necessary.

Decontamination Procedure RH-020

Appendix B

Personnel Contamination Form RH-200A

Personnel Contamination

Date of Survey: _____ Reviewed by _____ Location_____

Name of Surveyor: _____ Date _____

Alpha Instrument Type_____ Ser # _____ Alpha MDA_____ Alpha Correction Factor _____

Alpha Source _____cpm

Beta Instrument Type_____ Ser # _____ Beta MDA_____ Beta Correction Factor_____

Beta Source _____cpm Beta Bkg _____cpm

Person's name	CPM Alpha	Alpha Action Level	CPM Beta	Beta Action Level	Was Supervisor Notified?	Was worker decontam.?	Resurvey CPM Alpha or Beta

Alpha correction factor = $\frac{(\text{Source dpm})}{\text{meter cpm}} \frac{(100\ \text{cm}^2)}{\text{A cm}^2} = \frac{(\text{dpm})}{\text{cpm}} \frac{(100\ \text{cm}^2)}{\text{cm}^2} =$

Beta correction factor = $\frac{(\text{Source dpm})}{\text{meter cpm}} \frac{(100\ \text{cm}^2)}{} = \frac{(\text{dpm})}{\text{A cm}^2\ \text{cpm}} \frac{(100\ \text{cm}^2)}{\text{cm}^2} =$

Where "A cm^{2}" is, the open area of the probe and 100 cm^2 is the area specified in the license condition contamination limit.

Form RH-200A

Decontamination Procedure RH-020

Appendix A

Personnel Decontamination

Decontamination Procedure RH-020

Appendix C

Equipment and Materials Survey Form RH-070A

Equipment and Materials Survey

Location (Source of equipment)_____Surveyed by:_____ Date_____

Approval by RSO or Assistant RSO for release_____

Alpha check source _____dpm Type _____Gamma check source _____µCi Type _____

Beta check source _____dpm

Conversion factors

Equipment Description	Area Surveyed (or see map on back of form)	Total Alpha		Removable Alpha		Total Beta/Gamma		Removable Beta/Gamma		Permitted for Release	
		cpm	dpm/ 100cm^2	cpm	dpm/ 100cm^2	cpm	dpm/ 100cm^2	cpm	dpm/ 100cm^2	YES	NO
Instrument Serial No:											
Date Calibrated											
Correction Factor											
Background		0		0							
Limits – RH-070 Table A1		15000 dpm Max 100 cm^2 5000 dpm Ave 100 cm^2		1000 dpm Max 100 cm^2		15000 dpm Max 100 cm^2 5000 dpm Ave 100 cm^2		1000 dpm Max 100 cm^2			
LIMITS - ALTERNATIVE		See RH-070 Table 2		See RH-070 Table 2							

Release approved by: _____

1.0 PURPOSE

This procedure describes the posting of radiation warning signs to help maintain radiation exposure to levels that are As Low As Reasonably Achievable (ALARA) and caution signs for industrial safety.

2.0 APPLICABILITY

This procedure applies to all areas in and around the Mill.

3.0 OTHER DOCUMENTS

3.1 REFERENCES

3.1.1 Colorado Radiation Control Regulations (6 CCR 1007-1).

3.1.2 USNRC Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

4.0 EQUIPMENT AND MATERIALS

4.1 *MATERIALS (Where applicable)*

4.1.1 "Caution Radioactive Materials" signs.

4.1.2 "Caution Radiation Area" signs.

4.1.3 "Caution Airborne Radioactive Materials" signs.

4.1.4 "Any Area or Container on this Property May Contain Radioactive Materials" signs.

4.1.5 Industrial safety warning signs.

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or Assistant RSO is responsible for designating areas in and around the mill that require posting, and industrial safety posting.

5.2 The Radiation/Security Technician (RST) is responsible for posting the designated areas.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 Airborne Radioactivity Area - A room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations:

- In excess of the Derived Air Concentrations (DACs), or
- To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6 percent or the Annual Limit on Intake (ALI) or 12 DAC-hours.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

6.1.2 Radiation Area - An area accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

6.2 *SAFETY*

6.2.1 Piñon Ridge Mill Health and Safety Plan.

6.3 *FREQUENCY*

6.3.1 Continuous posting of areas specified in Section 7.0

7.0 PROCEDURES

7.1 *RADIATION SAFETY POSTING*

The RST or his designee is to conspicuously post the following signs:

7.1.1 Entrances to the property:

Any Area or Container on this Property May Contain Radioactive Material

7.1.2 Around areas in and around the mill that meet the definition of Radiation Areas:

Caution Radiation Area

7.1.3 Around areas in the mill that meet the definition of airborne radioactivity including the yellowcake drying and packaging room (Reg. Guide 8.30, page 20):

Caution Airborne Radioactivity Area

7.1.4 Areas in the mill specified by the RSO or Assistant RSO:

Respirator Use Required

7.1.5 Areas in the restricted area containing licensed material (CDPHE 6 CCR 1007-1, Part 4, Section 4.28.5) *Caution Radioactive Material*

7.2 *INDUSTRIAL SAFETY POSTING*

The RST or designee is to conspicuously post signs that may include:

7.2.1 Keep Out

7.2.2 Check in at the Administration Building

7.2.3 No Smoking in this Facility

7.2.4 Caution Flammable Material

7.2.5 Caution High Voltage

7.2.6 Eye Wash

7.2.7 Safety Shower

7.2.8 Hearing Protection Required

7.2.9 Tank names such as Ammonia Tank

7.2.10 Building names

1.0 PURPOSE

This procedure presents action levels and corrective actions in the radiation protection program at the Energy Fuels Piñon Ridge Mill.

2.0 APPLICABILITY

This procedure applies to all Mill personnel, contractors, and visitors who work with radioactive materials or are exposed to radiation in or around the Mill.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 NUREG-0874 Internal Dosimetry Model for Application to Bioassay at Uranium Mills, 1986.

3.1.2 Regulatory Guide 8.22 Bioassay at Uranium Mills, 1988.

3.1.3 Colorado Radiation Control Regulations (6 CCR 1007-1).

4.0 RESPONSIBILITY

4.1 The Radiation Safety Officer (RSO) or the Assistant RSO is responsible for:

4.1.1 Setting and enforcing action levels.

4.1.2 Investigating action levels that have been met or exceeded.

4.1.3 Informing affected worker(s) that action levels have been exceeded and directing Radiation Safety Technicians to take corrective actions.

4.2 The Radiation/Security Technician (RST) is responsible for:

4.2.1 Identifying action levels that have been met or exceeded and reporting the situation to the RSO or Assistant RSO.

4.2.2 Implementing with the mill staff correction actions for action levels that have been met or exceeded.

5.0 PREREQUISITE INFORMATION

5.1 *SAFETY*

5.1.1 Piñon Ridge Mill Health and Safety Plan

5.2 *FREQUENCY*

5.2.1 When an Action Level is met or exceeded implement the corresponding Corrective Action.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

6.0 PROCEDURE

6.1 *GAMMA RADIATION SURVEYS*

Action Level	Corrective Action
100% increase over average of previous 3 readings	Investigate cause and maintain exposures to levels that Are As Low As Reasonably Achievable (ALARA).
≥5 mR/hr	Investigate source of exposure, post as a "Radiation Area" and restrict access if warranted.
≥100 mR/hr	Investigate source of exposure, post as a "High Radiation Area" and restrict access.

6.2 *ALPHA AND BETA RADIATION SURVEYS*

Action Level	Corrective Action
≥1,000 dpm/100 cm^2 (removable) table surfaces	Contact responsible supervisor and decontaminate to below limit.
instrument background on personnel leaving restricted area	Contact RST and decontaminate to background before release.
≥1,000 dpm/100 cm^2 removable from non-table surfaces and equipment (Procedure RH-120)	Swipe to determine removable contamination level and decontaminate to below limit.
≥15,000 dpm/100 cm^2 total (Procedure RH-120)	Decontaminate to below limit and resurvey.

6.3 *GENERAL AIR SURVEY*

Action Level	Corrective Action
≥10% Derived Air Concentrations (DAC) ore dust or yellowcake (Procedure RH-130)	Investigate cause, determine if corrective measures are necessary, and sample as specified in Procedure RH-130 Table 1.

6.4 *OCCUPATIONAL BREATHING ZONE MONITORING*

Action Level	Corrective Action
≥10% DAC ore dust or yellowcake (Procedure RH-150)	Investigate cause, determine if corrective measures are necessary, and sample as specified in Procedure RH-150 Table 1.

6.5 *RADON DECAY PRODUCT SAMPLES*

Action Level	Corrective Action
≥10% DAC	Investigate cause and take weekly samples, check ventilation fans and ventilate if area is occupied.
≥25% DAC	Investigate cause, limit access, take daily samples until 4 consecutive samples are less than 25% DAC, check ventilation fans and ventilate if area is occupied.

6.6 *IN-VIVO LUNG SCANS (REG. GUIDE 8.22)*

Action Level	Corrective Actions
≥9 nCi U	1. Confirm results (repeat measurement within 6 months). Ensure that results are not caused by body surface activity. 2. Examine air sample data to determine source and concentrations of intake. If air sample results are anomalous, investigate air sampling procedures. Make corrections, if necessary. 3. Identify the cause of elevated activity and initiate additional uranium confinement control measures. 4. Determine whether other workers could have been exposed and perform special bioassay measurements for them. 5. Consider work assignment limitations that will permit the lung burden to be reduced through natural elimination; ensure that the lung burden does not exceed 16 nCi.

Action Level	Corrective Actions
≥16 nCi U	1. Within 90 days, take the actions listed above for 9 nCi. 2. Establish work restrictions for affected workers or increase uranium confinement control measures. (Normally, workers with a lung burden greater than 16 nCi are not allowed by their employer to resume work in airborne activity areas until the burden is reduced to less than 9 nCi. 3. Perform individual case studies (bioassays) for affected workers. 4. Continue operations only when it is virtually certain no additional workers will exceed 16 nCi.

6.7 *URINALYSIS (REG. GUIDE 8.22)*

Action Level	Corrective Actions
Exposure to ≥3E-11 µCi/ml soluble U for 40-hr work week (Procedure RH-050)	Implement uranium in urine bioassay program.
Exposure to ≥2E-12 µCi/ml ore dust or insoluble U for 1 calendar quarter	Implement uranium in urine bioassay program.
≥15 ug/l	1. Confirm results (repeat analysis). 2. Identify the cause of elevated urinary uranium and initiate control measures if the result is confirmed. 3. Examine air sample data to determine the source and concentration of intake. If air sample results are anomalous, investigate sampling procedures. Make corrections if necessary. 4. Determine whether other workers could have been exposed and perform bioassay measurements for them. 5. Consider work assignment limitations until the work's urinary uranium concentration falls below 15 ug/l. 6. Improve uranium confinement controls or respiratory protection programs as investigation indicates.

Action Level	Corrective Actions
≥35 ug/l	1. Take the actions given above. 2. Continue operations only if it is virtually certain that no other workers will exceed a urinary uranium concentration of 35 ug/l. 3. Establish work restrictions for affected employees or increase uranium confinement controls if ore dust or high-temperature dried yellowcake are involved. 4. Analyze bioassay samples weekly
Confirmed to be ≥35 ug/l for two consecutive specimens, confirmed to be ≥130 ug/l for any single specimen, or air sampling indication more than a quarterly limit of intake.	1. Take the actions give above. 2. Have urine specimen tested for albuminuria. 3. Obtain an in-vivo count if workers may have been exposed to Class Y material or ore dust. 4. Evaluate exposures. 5. Establish further uranium confinement controls or respiratory protection requirements as indicated. 6. Consider continued work restrictions on affected employees until urinary concentrations are below 15 ug/l and laboratory tests for albuminuria are negative.

6.8 *PERSONAL RADIATION MONITORING*

Level	Corrective Action
≥25% of the annual limit	Investigate and consider reassignment of employee.
100% of the annual limit	Report overexposure to CDPHE, investigate, and consider reassignment of employee.

6.9 *RELEASE OF EQUIPMENT AND MATERIALS TO UNRESTRICTED AREAS*

ACTION LEVEL FOR SURFACE CONTAMINATION				CORRECTIVE ACTIONS
NUCLIDE	AVERAGE	MAXIMUM	REMOVABLE	
Alpha emissions from U-nat, U-235, U-238, and associated decay products (Procedure RH-070)	5,000 dpm per 100 cm^2	15,000 dpm per 100 cm^2	1,000 dpm per 100 cm^2	Contact RST, decontaminate to below limit before release.
Beta emissions from U-nat, U-235, U-238, and associated decay products (Procedure RH-070)	5,000 dpm per 100 cm^2	15,000 dpm per 100 cm^2	1,000 dpm per 100 cm^2	Contact RST, decontaminate to below limit before release.
Alpha emissions from Ra-226, Ra-228, Th-230, Th-228, Ac-227 (Procedure RH-070)	100 dpm per 100 cm^2	300 dpm per 100 cm^2	20 dpm per 100 cm^2	Contact RST, decontaminate to below limit before release.
Alpha emissions from Th-232 and associated decay products	See Procedure RH-070	See Procedure RH-070	See Procedure RH-070	Contact RST, decontaminate to below limit before release.
Beta emissions from TH-232 and associated decay products	See Procedure RH-070	See Procedure RH-070	See Procedure RH-070	Contact RST, decontaminate to below limit before release.

6.10 *OTHER MONITORING*

ACTION LEVEL	CORRECTIVE ACTION
Physician's rejection of person for respirator use (Procedure AD-050).	Person not allowed to use respirator.
Spills	Clean up spill. Refer to Emergency Response Plan and Materials Containment Plan for further actions.
Leak test $\geq$ 0.005 µCi (Procedure RH-160)	Notify CDPHE, remove source from use, decontaminate.

ACTION LEVEL	CORRECTIVE ACTION
Monitoring inside respirator (Procedure RH-180) > Background removable alpha ≥100 dpm/100 cm^2 fixed alpha ≥5,000 dpm/100 cm^2 beta-gamma	Before using, decontaminate to below limit.
≥background on person leaving restricted area (Procedure RH-200)	Decontaminate to below limit prior to leaving restricted area.
≥0.1 rem to female of reproductive age (Procedure RH-300)	Present prenatal radiation exposure instruction.
≥0.1 rem to declared pregnant female (Procedure RH-300).	Dose to embryo-fetus not to exceed 0.5 rem.

6.11 ***CORRECTIVE ACTIONS***

6.11.1 When an Action Level is met or exceeded implement the corresponding Corrective Action.

6.11.2 Document the corrective action as specified in the associated procedure.

1.0 PURPOSE:

This procedure describes how to monitor the intakes of uranium in the bodies of workers at the Mill to keep radiation doses to levels that are As Low As Reasonably Achievable (ALARA).

2.0 APPLICABILITY

This procedure applies to:

2.1 Workers (1) routinely exposed to airborne yellowcake or directly involved in maintenance tasks in which yellowcake dust may be produced or (2) routinely exposed to airborne uranium ore dust.

2.2 Workers potentially exposed to a time-weighted exposure of 5E-11 μCi/mL of yellowcake for a 40-hour work week.

2.3 Workers potentially exposed to a time-weighted exposure of 2E-12 μCi/mL of ore dust for 1 calendar quarter.

2.4 Workers who wear a respirator to maintain inhalation exposures below the limits listed above.

2.5 Whenever required by the Radiation Safety Officer (RSO) or his designee.

3.0 OTHER DOCUMENTS:

3.1 *REFERENCES:*

3.1.1 American National Standard – Bioassay Programs for Uranium, HPS N13.22-1995

3.1.2 Nuclear Regulatory Commission Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program"

3.1.3 Nuclear Regulatory Commission Regulatory Guide 8.22,"Bioassay at Uranium Mills"

3.1.4 Nuclear Regulatory Commission Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus"

3.1.5 Nuclear Regulatory Commission NUREG – 0874, Internal Dosimetry Model for Applications of Bioassay at Uranium Mills

3.2 *APPENDICES:*

Appendix A – Uranium in Urine Report Form RH-050A

Appendix B – Bioassay Intake and Dose Estimates Form RH-050B

Appendix C – Table 1 Corrective Actions Based on Monthly Urinary Uranium Results [a]

Appendix D – Table 2 Corrective Actions Based on In vivo Results

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

4.0 EQUIPMENT AND MATERIALS:

4.1 *MATERIALS:*

Urine collection bottles.

5.0 RESPONSIBILITY:

- The RSO is responsible for both the urinalysis and in-vivo counting bioassay programs, and for the calculation of intakes and radiation doses.
- A Radiation/Security Technician (RST) is responsible for the collection of urine samples and the submission of the samples to the laboratory for analysis.
- The individual radiation worker participating in the bioassay program is responsible for the submission of urine samples and participation in the in-vivo counting when requested.

6.0 SAFETY:

Biohazard hepatitis shots may be advised by the RSO or the RST.

7.0 FREQUENCY:

Urinalysis will be conducted weekly/monthly, depending on worker exposures.
In-vivo lung (thorax) counts will be conducted based on results and exceedence of action levels in urinalysis program.

8.0 PROCEDURES:

8.1 *URINALYSES*

8.1.1 Baseline uranium urinalyses are to be conducted on each worker in the urinalysis program prior to their initial work assignments and upon termination of employment.

8.1.2 Weekly uranium in urine bioassay samples are to be collected from workers where potential exposures exceed a time weighted exposure of 3E-11 μCi/mL of soluble uranium for 40 hours. A week is 7 consecutive workdays or less. If the processing facility is operating on a 10-and-4 schedule, i.e., 10 days at work and 4 days off work, then the week is defined as 10 work days.

8.1.3 Monthly uranium in urine bioassays are to be performed on workers where exposures could exceed a time weighted exposure of 2E-12 μCi/mL (total alpha) of insoluble uranium or uranium ore dust for one calendar quarter.

8.1.4 In addition to the above schedule bioassays are to be preformed on the respirator users when the effectiveness of the respirator needs to be verified.

8.1.5 All uranium in urine bioassay samples are to be collected at least 36 hours after the most recent potential exposure in the processing facility.

8.1.6 On the morning urine samples are to be submitted each participant in the urinalysis program is to proceed to the change house to give the sample

before changing into their work clothes. PARTICIPANTS MUST WASH THEIR HANDS BEFORE SUBMITTING THE SAMPLE. Provide at least a 30 mL sample in the sample bottle with a label printed with the participant's name. Write the time and date of sampling, the last time and date worked, and their initials on the label to verify that the sample was submitted as required. Immediately return the sample to the RST, who is to check that the label contains the required information.

8.1.7 The RST is to deliver the urine samples to the laboratory with a chain of custody form to be completed by the laboratory personnel.

8.1.8 The RST and the Mill foremen are to coordinate to ensure the completion of the urine sampling as scheduled.

8.1.9 The measurement sensitivity for urine-analyses should be 5 µg of uranium per liter of urine or less.

8.1.10 Implement the corrective actions specified in Table 1 of the NRC Regulatory Guide 8.22 in response to the amount of uranium detected in the urine. That table is attached to this procedure in Form RH-050B.

8.1.11 Urinalysis results are to be available to the RSO or his designee within 7 days after sample collection. If an analysis exceeds 10 µg/L, the RSO is to be notified as soon as possible.

8.1.12 All laboratory analyses are to be performed in a laboratory that is essentially free of uranium contamination using contamination free equipment.

8.1.13 Each batch of specimens sent to the laboratory for analyses are to be accompanied by at least two control urine samples. Those control samples are to be taken from individuals who are not and have not been occupationally exposed to uranium; otherwise simulated controls known to contain uranium concentrations less than 1 µg/L may be used. Aliquots of each of these control urine samples should be taken; one should be a "blank," one should be spiked with uranium to obtain a concentration of 10-20 µg/L, and one should be spiked to 40-60 µg/L; the actual spike concentration being recorded confidentially and not available to the laboratory technicians. When results are received, the RSO or his designee should ensure that each reading is corrected for the reading of the corresponding blank, that the average of the percent deviation of the spiked sample net reported values from the "true" amount of spiked uranium sample is calculated. The percent deviation from the spiked samples accompanying each batch of urine samples should be within 30% of the spiked values. Likewise, the percentage deviation of the blank sample from the normal background count of the blank samples should be within 30% of the blank sample. If the spike or blank sample is not within the 30%, the most recent batch of affected samples will be rerun. Steps should be taken to correct the procedures for spiking, the procedures for

laboratory analyses, procedures for contamination control, or a combination of those procedures. The laboratory is to duplicate the analyses of 10-20% of the samples received, including the blanks and spikes. The results of the laboratory blanks and standard samples are to be available to the RSO or his designee.

8.1.14 Record the results of the urinalyses for all analyses greater than 10 µg/L on Form RH-050A or the equivalent. The form is to be routed to the personnel designated on the form within 4 days. Corrective actions and responses are to be specified on the form. Complete all portions of the form.

8.2 ***IN-VIVO LUNG (THORAX)***

8.2.1 When urinalysis results are confirmed to be greater than 35 µg/L for two consecutive specimens, are confirmed to be greater than 130 µg/L for any single specimen, or air sampling indicates more than a quarterly limit of intake, perform in-vivo counting a quickly as possible to determine if corrective actions are required.

8.2.2 When in-vivo measurements are to be conducted they should be performed as quickly as possible but no later than 3 months after such indications.

8.2.3 The lung counting procedure is to be capable of detecting 9 nCi or less of uranium.

8.2.4 Implement the corrective actions specified in Table 2 of the NRC Regulatory Guide 8.22 (see Appendix D) in response to the amount of uranium detected in the lungs of workers.

8.3 ***DECISION ANALYSIS FOR URINALYSES***

8.3.1 When an individual urine bioassay result indicates a potential uranium concentration greater than 10 µg per liter, the RSO is to initiate an investigation. Elevated urine bioassay uranium concentrations can result from several conditions:

1. Inhalation intake of uranium in airborne particulate matter
2. Ingestion intake of uranium
3. Contamination of sample during collection
4. Laboratory error

Investigation will consist of the following steps:

1. Immediate resample providing additional exposures have not occurred
2. Interview with the employee
3. Review the air sampling data for the period covered by the urine sample

4. Review of the employee's work record to determine if and when the assumed intake might have occurred.

8.3.2 If the investigation indicates that the elevated urine bioassay measurement may represent a confirmed intake, the RSO is to calculate the potential intake of uranium by the worker according to the attached Form RH-050B or the equivalent.

8.3.3 If the analysis indicates that the intake is less than 10 mg, a report is placed in the file.

8.3.4 If the calculated intake is greater than 10 mg, CDPHE IS TO BE NOTIFIED of the noncompliance with the 10 mg/week limit as specified in the Colorado Department of Public Health and Environment Radiation Control Division regulations (6 CCR 1007-1 4.5.2, 4.5.3, and 4.6.5).

Uranium Bioassay Procedure RH-050

Appendix A

Uranium in Urine Report Form RH-050A

Uranium In Urine Report Form RH-050A

Analytical results of >10 ug/l u must be reported on this form immediately*

SAMPLE SUB. FORM NO.	LABORATORY I.D. NO.	EMPLOYEE NAME	EMPLOYEE NUMBER	ORIGINAL RESULT	RERUN RESULT	SAMPLE DATE
				µg/L U	µg/L U	

ACKNOWLEDGEMENT OF SAMPLE ANALYSIS (initial and date) [all <4 days**]

Laboratory Analyst (same day**)	Laboratory Manager (1 day**)	Mill Manager (1 day**)	Radiation Safety Officer (1 day**)	General Foreman (1 day**)	RST	Other

SAMPLE INFORMATION

Was sample taken more than 36 hours after last work on site?	Original: [ ] yes [ ] no		Follow-up:[ ] yes [ ] no	
Previous sampling: [< 2 weeks**]	sample date:	analytical date:	result:	µg/L
Follow-up sampling: [< 2 weeks**]	sample date:	analytical date:	result:	µg/L
Was employee notified? [< 5 days**]	[ ] yes [ ] no	date:	by:	(initials)
Where did employee work? [< 5 days**] (circle suspect areas)				
Was respiratory protection used?	[ ] yes [ ] no	type:	duration:	
Other information:				

<u>RESPONSES AND ACTIONS PLANNED OR TAKEN (***)</u> (continue on back if necessary)

		[ ] continued on back
Final determination on potential exposure: [<21 days**]	[ ] exposure confirmed	[ ] contaminated sample
Justification for final determination:		
Final recommendations:		

ACKNOWLEDGEMENT OF FINAL DETERMINATION AND RECOMMENDATIONS (initial and date) [<30 days**]

Employee	Lab. Manager	Mill Manager	RSO	Gen. Foreman	RST	Other

Completed form added to employee's permanent file? [<45 days**] [ ] yes [ ] no

*Notes: * - Analyst must fill in sample data and results, initial, date and give the form to the Lab Manager.*
*** - target time to complete item following analysis of original sample.*
**** - see back of form for corrective action requirements.*

Uranium Bioassay Procedure RH-050

Appendix B

Bioassay Intake and Dose Estimates Form RH-050B

Bioassay Intake and Dose Estimates

The following procedure will be used to estimate the potential dose to a worker based on a urine bioassay concentration greater than 10 µg/L:

Employee Number_____

Table 1: Concentration of uranium in urine

Sample No.	Uranium concentration in urine (µg/L)	Sample Date	Days since previous sample less than 5 µg/L	Solubility class
1				
2				
3				
4				
5				

Table 2: **Uranium excretion functions** - fraction of intake by inhalation excreted in 24 hours (from *Bioassay Programs for Uranium*. HPS N13.22-1995)

Days since exposure	Retention Function (Class D)	Retention Function (Class W)	Retention Function (Class Y)
1	1.87E-01	4.13E-02	2.23E-03
2	7.27E-02	1.09E-02	5.49E-04
3	3.21E-02	4.72E-03	2.30E-04
4	1.82E-02	3.22E-03	1.57E-04
5	1.31E-02	2.69E-03	1.31E-04
6	1.09E-02	2.40E-03	1.17E-04
7	9.64E-03	2.19E-03	1.07E-04
8	8.71E-03	2.02E-03	9.81E-05
9	7.94E-03	1.88E-03	9.07E-05
10	7.26E-03	1.75E-03	8.42E-05
11	6.86E-03*	1.68E-03	8.04E-05
12	6.46E-03	1.61E-03	7.67E-05
13	6.06E-03	1.54E-03	7.30E-05
14	5.66E-03	1.47E-03	6.93E-05
15	5.26E-03	1.39E-03	6.56E-05
16	4.86E-03	1.32E-03	6.19E-05
17	4.46E-03	1.25E-03	5.82E-05
18	4.06E-03	1.18E-03	5.45E-05
19	3.66E-03	1.11E-03	5.07E-05
20	3.26E-03	1.03E-03	4.69E-05

*Excretion fractions for 11 to 20 days post intake are interpolated values.

Estimated intake = [U conc. in urine in µg/L][1.4 L/d][0.68 pCi/µg]/[F (d)]

Where F(d) = fraction of the initial intake excreted in urine per day at d days after an acute intake.

Uranium Bioassay Procedure RH-050

Appendix C

Table 1 Corrective Actions Based on Monthly Urinary Uranium Results

Table 1
Corrective Actions Based on Monthly Urinary Uranium Results [a]

Urinary Uranium Concentration	Interpretation	Actions
Less than 15 µg/L	Uranium confinement sampling programs are indicated to be adequate. [b]	None - Continue to review further bioassay results.
15 to 35 µg/L	Uranium confinement and air sampling may not provide an adequate margin of safety. [b]	1. Confirm results (repeat urinalysis). 2. Identify the cause of elevated urinary uranium and initiate additional control measure if the result is confirmed. 3. Examine air sampling data to determine the source and concentration of intake. If air sampling results are anomalous, investigate sampling procedures. Make corrections if necessary. 4. Determine whether other workers could have been exposed and perform bioassay measurements for them. 5. Consider work assignment limitations until the worker's worker's urinary uranium concentration falls below 15 µg/L. 6. Improve uranium confinement controls or respiratory protection program as investigation indicates.
Greater than 35 µg/L	Uranium confinement and perhaps air sampling programs are not acceptable. [c]	1. Take the actions given above. 2. Continue operations only if it is virtually certain that no other worker will exceed a urinary uranium concentration of 35 µg/L. 3. Establish work restrictions for affected employees or increase uranium confinement controls if ore dust or high-temperature dried yellowcake are involved. 4. Analyze bioassay samples weekly.
Confirmed to be greater than 35 µg/L for two consecutive specimens, confirmed to be Greater than 130 µg/L for any single specimen, or air sampling indication of more than a quarterly Limit on Intake.	Worker may have exceeded regulatory limit on intake.	1. Take the actions given above. 2. Have urine specimen tested for albuminuria. 3. Obtain an in vivo count if worker may have been exposed to Class Y material or ore dust. 4. Evaluate exposures. 5. Establish further uranium confinement controls or respiratory protection requirements as indicated. 6. Consider continued work restrictions on affected Employees until urinary concentrations are below 15 µg/L and laboratory test for albuminuria are negative.

[a] Use Figures 1-3 to adjust action levels for other frequencies of bioassay sampling. The model used in NUREG-0874 (Ref.1) employs fractional composition values (F1, F_2, F_3) for Class D, Class W, and Class Y components of yellowcake compounds. The assigned values in NUREG-0874 are based on data from invaliable literature. The use of alternative values of F1, F_2, F_3 specific for a particular operation are acceptable provided (1) details regarding their determination are described and mentioned in employee exposure records (use paragraph 20.401(e) (1) of 10 CFR Part 20) and (2) the model as published in NUREG-0874 is then used in the determination of alternative urinalysis frequencies and action levels.

[b] However, if a person is exposed to uranium ore dust or other material of Class W or Y alone, refer to Section 6 of NURGEG-0474 about the possibility of the need for conducting in vivo lung counts on selected personnel or about using alternative urine sampling times and associated action levels computed using NUREG-0874.

[c] Unless the result was anticipated and caused by conditions already corrected.

Uranium Bioassay Procedure RH-050

Appendix D

Table 2 Corrective Actions Based on In-Vivo Results

Table 2
Corrective Actions Based on In-Vivo Results [a]

Amount of Uranium Detected	Interpretation	Actions
Below 9 nCi (330 Bq)	May be below detection limit. This result does not necessarily indicate that uranium confinement and air sampling programs are validated.	Rely on urinalysis results to determine corrective actions (unless air sampling indicates quarterly intake limits are exceeded for ore dust).
9 to 16 nCi (330 to 590 Bq)	Confinement and air sampling programs should be examined.[b] Uranium activity in lungs could be too high.	1. Confirm result (repeat measurement within 6 months). Ensure that results are not caused by body surface activity. 2. Examine air sampling data to determine source and concentrations of intake. If air sampling results are anomalous, investigate air sampling procedures. Make corrections, if necessary. 3. Identify the cause of elevated activity and initiate additional uranium confinement control measures. 4. Determine whether other workers could have been exposed and perform special bioassay measurements for them. 5. Consider work assignment limitations that will permit the lung burden to be reduced through natural elimination; ensure that the lung burden does not exceed 16 nCi (590 Bq).
More than 16 nCi (590 Bq)	Uranium confinement and air sampling probably are not acceptable.[b] Uranium activity in the lungs should be reduced by increased protection measures for the workers involved.	1. Within 90 days, take the actions listed above for 9 to 16 nCi (330 to 590 Bq). 2. Establish work restrictions for workers or increased uranium confinement control measures. (Normally workers with a lung burden greater than 16 nCi (590Bq) are not allowed by their employee to resume work in airborne activity areas until the burden is reduced to less than 9 nCi (330Bq). 3. Perform individual case studies (bioassays) for affected workers. 4. Continue operations only when it is virtually certain no additional workers will exceed16 nCi (590 Bq).

[a] The model used in NUREG-0874 (Ref.1) employs fractional composition values (F1, F_2, F_3) for Class D, Class W, and Class Y components of yellowcake compounds. The assigned values in NUREG-0874 are based on data from invaliable literature. The use of alternative values of F1, F_2, F_3 specific for a particular operation are acceptable provided (1) details regarding their determination are described and mentioned in employee exposure records (use paragraph 20.401(e) (1) of 10 CFR Part 20) and (2) the model as published in NUREG-0874 is then used in the determination of alternative urinalysis frequencies and action levels.

[b] Unless the result was anticipated and caused by conditions already corrected.

1.0 PURPOSE

This procedure describes the issuance and use of Radiation Work Permits (RWPs) to maintain worker radiation doses to levels that are As Low As Reasonably Achievable (ALARA).

2.0 APPLICABILITY

An RWP is required for any work with radioactive materials which has not been described in a written operating or maintenance procedure, for work that is performed under conditions that deviate from assumptions in existing procedures, and for work that may involve exposure above action levels. RWPs may include:

2.1 Work on equipment (pumps, piping etc) in the near vicinity of any unshielded (shutter open) nuclear source.

2.2 Work on the all bag houses or pollution control equipment which may contain radioactive material.

2.3 Work involving processing equipment, e.g. work in a tank containing or suspected to contain concentrated uranium (yellowcake or pregnant solution) in a dry form or in such a form that contamination by ingestion or inhalation of radioactive materials could occur.

2.4 Work offsite where occupational or environmental monitoring is necessary.

2.5 Work as directed by the Radiation Safety Officer (RSO) or the Assistant RSO.

3.0 OTHER DOCUMENTS

3.1 REFERENCES

3.1.1 Colorado Radiation Control Regulation, Part 4, Standards for Protection Against Radiation (6 CCR 1007-1).

3.2 APPENDICES

Appendix A – Radiation Work Permit Form RWP-1

4.0 RESPONSIBILITY

4.1 The Maintenance or Mill Foreman is responsible for:

4.1.1 Initiating the RWP by completing, signing and dating the "Work Description" area of Form RWP-1 and presenting it to the RSO, Assistant RSO or Radiation/Safety Technician (RST) for approval.

4.1.2 DAY SHIFT - Review and issuance of an RWP will first be by the RSO, Assistant RSO, or RST.

4.1.3 WEEKEND, SWING or GRAVEYARD SHIFTS - Review and issuance of RWP will be by the RST on shift. If there is no RST on shift, the RSO, Assistant RSO, or RST will be contacted at home or by cell phone. The RSO, Assistant RSO or RST will return to the Mill if necessary, to review and issue the RWP or may direct the Maintenance or Mill Foreman to

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

issue the RWP. The RSO, Assistant RSO and/or the RST should be available during these hours. Home phone numbers and cell phone numbers are posted at the Mill Safety Office. The cell phone number for the Safety Department is posted throughout the Mill area. The RSO, Assistant RSO or the RST should be contacted as far as possible in advance in order to ensure adequate review of the work task prior to commencement of work.

4.1.4 Conferring with RSO, Assistant RSO and/or the RST to determine the need for a pre-planning conference and the degree of involvement in the conference by personnel involved in or affected by performing the RWP.

4.1.5 Obtaining written approval on the RWP by the RSO, Assistant RSO or RST prior to commencement of work. NOTE: If appropriate, verbal approval can be granted by the RSO, Assistant RSO or RST during off-duty hours. However, the on-duty Mill Foreman must sign the completed RWP form. The Maintenance or Mill Foreman should acknowledge verbal approval as a note on the RWP form.

4.1.6 Retaining one copy of the current RWP. The RSO, Assistant RSO or RST will retain remaining copies.

4.1.7 Insuring all personnel involved in the work have read and understand the RWP prior to commencement of work. Their understanding of the RWP requirements is documented by the workers' signing a training sheet for the work to be performed.

4.1.8 Posting the current RWP at the maintenance or work location at the entrance of the RWP restricted zone.

4.1.9 Insuring that the RWP zone is clearly posted to prevent inadvertent entry to the affected RWP area.

4.1.10 Insuring that all requirements of the RWP are met by employees or contractors performing the work or in the immediate vicinity of the affected area.

4.1.11 Completing the post task information required on the "Work Description" section of his copy of the RWP after completion of the work.

4.1.12 Notifying the RSO, Assistant RSO or the RST of the completion of work under the RWP.

4.1.13 Returning the completed RWP used at the work location to the Radiation Safety Department.

4.1.14 Insuring appropriate notification of oncoming personnel at shift change of the RWP requirements for RWP operations still in progress at the time.

4.1.15 Authorizing the RWP with his signature, only during the RSO, Assistant RSO's or RST's absence during their off duty hours. This is done after verbal communication and approval from the RSO, Assistant RSO or RST.

4.2 The RSO, Assistant RSO, or RST is responsible for:

4.2.1 Receiving the request for the RWP and review the work description section obtaining any additional information from the Maintenance or Mill Foreman to evaluate the work task.

4.2.2 Completing the "Radiation Safety Evaluation and Requirements" section.

4.2.3 Conferring with the Maintenance or Mill Foreman and make a determination of the need for a pre-planning conference and the degree of involvement in the conference by personnel performing the work under the RWP and others affected by it.

4.2.4 Granting permission by signing the RWP to proceed with the work.

NOTE: If appropriate, verbal approval can be granted by the RSO, Assistant RSO or RST during off-duty hours. Verbal approval will be given to the Maintenance or Mill Foreman.

4.2.5 Distributing copies of the RWP as follows: one copy to the Maintenance or Mill Foreman; and one copy for the RSO, Assistant RSO or RST.

4.2.6 Determining if collection of bioassay samples or special showering and personnel monitoring is necessary for employees or contractors in tasks requiring an RWP.

4.2.7 Noting appropriate information concerning the operation in the "Task Evaluation and Results" section of the RWP after completion of work under the RWP.

4.2.8 Conducting, as necessary, a final inspection of the work area affected by the RWP to insure the integrity of the system has been maintained, this will be noted in the "Task Evaluation and Results" section as a follow-up inspection.

4.2.9 Reviewing all survey results, monitoring results, sample results, comments and/or deficiencies involving the RWP.

4.2.10 Sign and date the final evaluation section of the RWP after all necessary information and results have been received.

4.3 Employees and Contractors are responsible for:

4.3.1 Reading, asking questions, and understanding the tasks and stipulations of the RWP.

4.3.2 Initialing next to their names that they understand all aspects of work to be conducted under the RWP.

5.0 PREREQUISITE INFORMATION

5.1 *SAFETY*

5.1.1 Safety concerns are addressed in the individual RWPs.

5.2 *FREQUENCY*

5.2.1 Whenever work with radioactive materials is required that has not been addressed in written operating and maintenance procedures.

6.0 PROCEDURE

6.1 The Maintenance or Mill Foreman, initiating a work order, work plan or any other requestor requiring an RWP as described above will complete, sign and date the "Work Description" section of the RWP form and submit it to the RSO, Assistant RSO or the RST for approval.

6.2 Upon receipt of the RWP request, the RSO, Assistant RSO or the RST will review the work description section and obtain any additional information necessary to evaluate the work task. The evaluation will include the need for respiratory protection, bioassay sampling, personal protective equipment, and work restrictions to maintain radiation exposures to levels that are ALARA. After evaluation, the RSO, Assistant RSO or RST will complete the "Radiation Safety Evaluation and Requirements" section of Form RWP-1 and will review the requirements with the Maintenance or Mill Foreman. Following this review, the Maintenance or Mill Foreman will sign and date this section of the RWP form indicating his responsibility to insure compliance with the RWP requirements. The RSO, Assistant RSO or RST will make a determination of the need for a pre-planning conference and the degree of involvement in the conference by personnel performing the RWP. In all cases, written approval by the RSO, Assistant RSO or RST must be obtained prior to commencement of the work task. All involved personnel shall acknowledge that they have read and understood the requirements of the RWP by signing the training form prior to commencement of the work task.

NOTE: If appropriate, verbal approval can be granted by the RSO, Assistant RSO or RST during off-duty hours. However, the Maintenance or Mill Foreman must sign Form RWP-1 for the RSO, Assistant RSO or RST in the case of verbal approval and note that verbal approval was obtained from the RSO, Assistant RSO, or RST.

Copies of the RWP form will be distributed in the following manner:

- 1 copy for the Maintenance or Mill Foreman
- 1 copy for the RSO, Assistant RSO or RST

6.3 If it is determined that collection of bioassay samples or special showering and personnel monitoring is necessary for employees involved in an operation requiring an RWP, the employee/contractor will be notified.

6.4 If immediate review and issuance of a RWP is necessary for activities during weekend, swing or graveyard shifts, the RST on shift will be contacted. If there is no RST on shift, the RSO, Assistant RSO or the RST will be contacted at home or by the cell phone number and will return to the Mill, if necessary, to review and issue an RWP, or they may direct the Maintenance or Mill Foreman to issue the RWP. In all cases, only the RSO, Assistant RSO or the RST can approve commencement of work to be done on an RWP.

6.4.1 The RSO, Assistant RSO or the RST should be available during these hours. Their home phone numbers are posted at the Mill Safety Office. Also the cell phone for the Safety Department is posted through out the Mill area.

6.4.2 The RSO, Assistant RSO or the RST shall be contacted as far in advance as possible in order to insure adequate review of the work task prior to commencement of work.

6.5 The Safety Department must be notified at the completion of any RWP. Foremen will be responsible for notifying their relief on the status of the RWP prior to shift changes and for insuring appropriate notification of oncoming personnel at shift change of the RWP requirements for RWP operations still in progress at that time.

6.6 A copy of the current RWP will be posted at the entrance of the work area requiring a RWP. The Mill Foreman will be responsible for assuring that all requirements of the RWP are met by employees performing the operation or in the immediate vicinity of the affected area. The RWP area will be clearly posted to preclude inadvertent entry and will be described on the RWP form.

6.7 Contractors or individuals performing work tasks who are not employed by Energy Fuels will not be exempt from RWP requirements. It will be the responsibility of Energy Fuels personnel authorizing work performed by contractors to determine whether an RWP is necessary by contacting the RSO, Assistant RSO or RST.

6.8 The RSO, Assistant RSO or RST will be responsible for noting appropriate information concerning the operation in the “Task Evaluation and Results” section of the RWP. The Maintenance or Mill Foreman will complete the post task information required in the “Work Description” section of his copy of the RWP. All copies of the RWP will be returned to the Safety Department.

6.9 The RSO, Assistant RSO or RST will conduct, as necessary, an inspection of the work area following completion of the task to insure that the integrity of the system has been maintained. The results of this inspection will be noted in the “Task Evaluation and Results” section of Form RWP-1 as a follow-up inspection.

6.10 The RSO, Assistant RSO or RST will review all survey results, monitoring results, sample results, comments and/or deficiencies involving the RWP. When all necessary information and results have been received, the RSO, Assistant RSO or RST will sign and date the final evaluation section of the RWP form.

Radiation Work Permits Procedure RH-060

Appendix A

Radiation Work Permit Form RWP-1

Radiation Work Permit

Work Description:

Date of Request: _____ Rad. Safety Rep. Contacted: _____

Date of Operation: _____ Requesting Maint or Opr. Supr.: _____

Anticipated Time Required: _____ Actual Time Required: _____

Description of Work Task and Process Equipment Involved:

_____

_____

Employees Performing Work Task or in Affected Area (List)				Anticipated Tools Needed (List)	Actual Tools Used (List)
Name	Job Category	Performing Work Task	In Affected Area		
		( )	( )		
		( )	( )		
		( )	( )		
		( )	( )		
		( )	( )		
Maintenance or Operations Supervisor:				Date:	
Comments:					

Radiation Safety Evaluation and Requirements:

Personnel Safety Requirements:

Head: Cloth Hat ( ) Hard Hat ( ) Safety Glasses ( ) Safety Goggles ( )

Body: Work Clothes ( ) Paper Suit ( ) Washable Coveralls ( )

Hands: Cloth Gloves ( ) Surgical Gloves ( ) Heavy Rubber Gloves ( )

Feet: Work Shoes ( ) Shoe Covers ( ) Cloth Boots ( ) Rubber Boots ( )

Respiratory: None ( ) Half Mask ( ) Full Mask ( ) Full Mask w/air ( )

Radiation Safety Representation:

At Start ( ) Intermittent ( ) Continuous ( )

Monitoring Requirements:

Survey	Pre	Dur	Post
Alpha	( )	( )	( )
Beta	( )	( )	( )
Gamma	( )	( )	( )
Air Samples			
General	( )	( )	( )
B.Z.	( )	( )	( )
O.C.C.	( )	( )	( )
Dosimetry			
TLD	( )	( )	( )
Pocket	( )	( )	( )
Other (Specify)	( )	( )	( )

Cleanup: At Task Completion ( ) Continuous ( ) Washdown ( ) Vacuums ( ) Other:

Special Disposal Requirements (Specify): _____

Other Requirements or Comments: _____

_____

Radiation Safety Representative Signature: _____ Date: _____

Maintenance or Operations Supervisor Signature: Date:

Task Evaluation and Results:

Deviations from above work description or Radiation Safety Requirements (List and give reason for deviation):

_____

_____

Survey and Monitoring Results (List): _____

_____

Follow-up Inspections, Surveys or Monitoring Results (List): _____

_____

Special Disposal Method and Location (When Required): _____

Operation Deficiencies, Recommendations and General Comments: _____

_____

_____

Final Evaluation Date: _____

Radiation Safety Representative Signature: _____

1.0 PURPOSE

This contamination control procedure describes how to survey equipment and materials before they are released from the restricted area of the Mill for unrestricted use. The purpose of the survey is to assure that contaminated equipment and materials are not released with contamination above the regulatory guidance.

2.0 APPLICABILITY

This procedure applies to all equipment and materials that leave the restricted area of the facility.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Radioactive Materials License

3.1.2 NUREG 1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions"

3.1.3 Draft NUREG 1761, "Radiological Surveys for Controlling Release of Solid Materials," July, 2002, pages 31-32, 49

3.1.4 Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills."

3.1.5 Colorado Radiation Control Regulations, Part 4 (6 CCR 1007-1).

3.2 *APPENDICES*

Appendix A – Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use (NRC, 1987)

Appendix B – Equipment and Materials Survey Form RH-070A or its equivalent

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 Ludlum model 43 series alpha detectors (scintillation, proportional or phoswich) or equivalent

4.1.2 Ludlum model 2241 series scaler/ratemeter or equivalent

4.1.3 Ludlum model 2929 alpha/beta sample counter or equivalent

4.1.4 Ludlum 44-9 pancake GM probe or equivalent

4.1.5 Ludlum model 12s micro R meter or equivalent

4.1.6 Ludlum model 44 -10 gamma scintillation probe, 2 x 2 or equivalent

4.1.7 Swipes.

4.1.8 Form RH-070A or the equivalent.

4.1.9 NIST alpha and beta check sources for determining instrument and detector efficiencies.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.0 RESPONSIBILITY:

5.1 The Radiation Safety Officer (RSO) or the Assistant RSO is responsible for:

5.1.1 Reviewing the forms for accuracy and completeness and to evaluate if any patterns of contamination may be developing over time.

5.1.2 Approving the release of material from the restricted area of the Mill.

5.2 The Radiation/Security Technician (RST) or his designee is responsible for:

5.2.1 Conducting the alpha and beta-gamma surveys of equipment and materials.

5.2.2 Calculating the MDAs and efficiencies.

5.2.3 Filing form RH-070A or its equivalent and retaining.

5.3 The Quality Assurance Officer or designee is responsible for:

5.3.1 Assuring that personnel considered as key individuals relevant to this procedure are documented as properly trained and/or qualified to perform the required duties.

5.3.2 Assuring that the quality principles of this procedure are followed and maintained.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 Surface Contamination - Radionuclide contamination on the surface of equipment of materials.

6.1.2 Class 1 Solid Materials - Equipment or materials that have known contamination or the potential for contamination above the release criteria based on process knowledge, usage location, previous surveys, and that have been in direct contact with radioactive materials.

6.1.3 Class 2 Solid Materials - Equipment or materials that have the potential for contamination and may have come in contact with radioactive materials but are not expected to be above the release criteria.

6.1.4 Class 3 Solid Materials - Equipment or materials that are not expected to be contaminated, or are contaminated at only a small fraction of the release criteria.

6.1.5 Secular Equilibrium - Equal activity of each of the decay products in a decay series such as natural uranium or Th-232.

6.2 *SAFETY*

6.2.1 Piñon Ridge Mill Health and Safety Plan

6.3 *FREQUENCY*

6.3.1 Prior to the release of potentially contaminated equipment or materials from the restricted area of the Mill, conduct alpha and beta contamination surveys.

7.0 PROCEDURES

7.1 *LOCATION OF RADIOACTIVE MATERIAL*

7.1.1 Equipment and materials to be surveyed for potential release are to meet the following criteria:

7.1.1.1 Equipment and material shall have been washed with a detergent wash or the equivalent as necessary to remove contamination.

7.1.1.2 Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Appendix A, Table A1 prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

7.1.1.3 The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork.

7.1.1.4 Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

7.2 *CLASSIFICATION OF EQUIPMENT AND MATERIALS*

7.2.1 Equipment, materials, items or a batch of items (e.g. a box of hand tools) to be released for unrestricted use are evaluated for potential contamination and classified based on contamination potential into one of three survey classes. Table 1 presents guidance on the percentage of items to be surveyed and the percentage of the surface area of each item to be surveyed (NUREG-1761 as revised). See Section 6.1 for the definition of Class 1 Solid Materials, Class 2 Solid Materials and Class 3 Solid Materials. Determine the class for the equipment and/or material to be surveyed. Survey the percentage of items and the percentage of the area shown in Table 1.

7.2.2 The classifications provide guidance to the surveyor on the percentage of items to be surveyed and the percentage of surface area of each item to be surveyed. For items or equipment classified as Class 2 Materials or Class 3 Materials, survey the surface areas which have the highest potential for contamination.

7.2.3 If it can not be readily identified by process knowledge, usage location etc., then the equipment or material will be initially surveyed at a minimum using the requirements specified for Class 2 Materials (50%-100% of the items and 50%-100% of the surface area). For example if a less than 100% survey is performed and 50% of the survey measurements indicate contamination

levels at less than the release limit but greater than twice background, increase the percentage or number of items surveyed and the percentage of surface area measured. Examples of the survey classes are listed in Table 1 below.

Table 1
Minimum Survey Release Requirement for Release of Equipment and Materials

Equipment and Material Classification	Example	Percent of items to be surveyed	Percent of surface area of each item to be surveyed [(a)]
• **CLASS 1** • Equipment or material from the process area of the mill with known contamination or potentially contaminated above the release criteria	Electric motors from the packaging area	100	100
• **CLASS 2** • Equipment or materials that have the potential for contamination that is not expected to be above the release criteria	Instrumentation from a control room in the mill	50-100	50-100
• **CLASS 3** • Equipment or materials that are not expected to be contaminated or contaminated at only a small fraction of the release criteria	Office furniture from the mill office	10-50	10-50

(a) Recommended values from Draft NUREG 1761.

7.3 *RELEASE CONTAMINATION LIMIT CRITERIA*

7.3.1 The alpha activity limits specified in Appendix A, Table A1, line 1 are to be used as release criteria for materials that contain U-nat, U-235, U-238 and associated decay products.

7.4 *TOTAL ALPHA SURVEYS OF EQUIPMENT AND MATERIALS*

7.4.1 The RST is to turn on the alpha survey meter. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.4.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-070A or the equivalent (See Appendix B).

7.4.3 Inspect the survey meter for damage. If an alpha scintillation probe is to be used in the survey, hold the probe towards a light source. If the count rate suddenly increases, look for tiny pinholes in the mylar surface of the probe face. Repairs can be made with "whiteout." If two or more whiteout spots are present, use another instrument. Repair other damage or use a functional instrument.

7.4.4 Preferably survey equipment and materials in a low background area. Observe the alpha background count rate of the instrument. If the meter reads more than five to ten counts per minute (cpm) background, use a different survey meter and report the potential contamination and/or malfunction to the RSO or the Assistant RSO.

7.4.5 If the item to be monitored is wet, dry the item prior to conducting the alpha survey.

7.4.6 Position the probe face of the survey meter within 1.0 cm of the Th-230 check source or the equivalent alpha check source. Use the end of one's finger under the edge of the alpha probe as a guide for the 1.0 cm spacing. Use a source-to-detector distance for the calibration that is similar to the expected surface-to-detector spacing used in the field. Record the reading on Form RH-070A. If the reading is within 20% (Reg. Guide 8.30, page 21) of the historic reference readings conclude that the instrument is functioning properly. If not, use an instrument that is functioning properly.

7.4.7 Determine the instrument MDA and record on calibration form.

7.4.7.1 For an integrated measurement over a preset time, the MDA can be approximated by:

$$MDA = \frac{2.71 + 4.65 \sqrt{[(B_R)(t)]}}{(t)(E)(A/100)}$$

7.4.7.2 For a ratemeter instrument the MDA can be approximated by taking twice the time constant of the meter as the counting time:

$$MDA = \frac{4.65 \sqrt{[(B_R)/(2t_c)]}}{(E)(A/100)}$$

where

MDA = activity in dpm/100 cm^2

B_R = background cpm

t = counting time in minutes

t_c = meter time constant in minutes

E = instrument detector efficiency

A = probe area in cm^2

7.4.8 Position the probe face of the survey meter within 1.0 cm of the equipment or material being monitored. Use the end of one's finger under the edge of the alpha probe as a guide to the 1.0 cm spacing. Slowly (~1 cm/sec) move the probe over the area to be surveyed stopping at areas of suspect contamination to obtain an accurate determination of the level of contamination. Averaged surface contamination measurements are for 100 cm^2 areas averaged over 1 m^2. Maximum contamination measurements are averaged over 100 cm^2. Survey several different areas to obtain an average count rate and a maximum count rate.

7.4.9 Calculate the portable survey instrument correction factor by:

$$\text{Alpha Correction factor} = \left(\frac{\text{Source dpm}}{\text{meter cpm}}\right)\left(\frac{100\ cm^2}{X\ cm^2}\right)$$

Where X is the detection surface area of the probe with no corrections made for the protective screen and 100 cm^2 is the area specified in the release limits. The "source dpm/meter cpm" measurement corrects for the screen covering part of the detection area of the probe (NUREG-1507, page 5-7).

7.4.10 Record the observed cpm on Form RH-070A under the Total Alpha column. Multiply the highest and average survey result by the correction factor and record on Form RH-070A under the Total Alpha dpm/100 cm^2 column.

7.4.11 Enter the following data on Form RH-070A or its equivalent:

- Equipment description and area surveyed
- Technician conducting the survey
- Date
- Survey equipment serial number for both the alpha and beta-gamma surveys
- Date calibrated

7.5 *REMOVABLE ALPHA SURVEY OF EQUIPMENT AND MATERIALS*

7.5.1 Measure for removable alpha activity only if the Total Alpha activity exceeds the appropriate removable activity limit listed in Appendix A, Table A1.

7.5.2 Number the swipes for identification.

7.5.3 Using a swipe rub lightly an area of 100 cm^2 of potential contamination. Place the swipe on a clean flat surface and count the alpha emission rate using the portable alpha meter and multiply the observed reading by the calculated instrument efficiency factor, or count the swipe in a Ludlum

model 2241 2000 scaler/43 series alpha swipe counter or similar instrument.

7.5.4 Document all survey results on Form RH-070A under the Removable Alpha dpm/100 cm^2 column.

7.6 *BETA-GAMMA EXPOSURE RATE SURVEY OF EQUIPMENT AND MATERIALS*

7.6.1 Beta-gamma exposure rate measurements are not part of the routine surveys for release of equipment to unrestricted areas because "none of the uranium or thorium decay series isotopes will provide any significant dose rate *at the surface release limits* and certainly should not approach 200 μR/h." (CDPHE Docket #5205, December 9, 2003). However beta-gamma exposure rate surveys in the Mill are included in procedure RH-110, Beta and/or Gamma Exposure Rate Surveys.

7.7 *BETA-GAMMA CONTAMINATION SURVEY OF EQUIPMENT AND MATERIALS*

7.7.1 Turn on the instrument equipped with a GM pancake probe. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.7.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-070A or the equivalent.

7.7.3 Inspect the meter and probe for damage. If damaged, use another instrument.

7.7.4 Observe the instrument background count rate and record on Form RH-070A.

7.7.5 Using a Cl-36 calibration standard or the equivalent and a source-to-detector distance for the calibration that is similar to the expected surface-to-detector spacing used in the field, e.g., 1 cm, measure the source activity and background with a pancake probe. Proceed if the measured activity is within 20% of the historic calibration measurements.

7.7.6 Preferably survey equipment and materials in a low background area. Calculate the pancake probe beta correction factor by:

$$\text{Pancake beta correction factor} = \frac{(\text{source dpm})(100\ cm^2)}{(\text{meter cpm-bkg cpm})(X\ cm^2)}$$

where:

X = The pancake probe detection surface area (NUREG-1507, page 4-7)

100 cm^2 = The surface area specified in the CDPHE regulations.

7.7.7 Calculate the efficiency of the pancake probe as:

$$E = \frac{\text{meter cpm} - \text{bkg cpm}}{\text{Source dpm}}$$

7.7.8 Determine the MDA, if needed, using E calculated above.

7.7.8.1 For an integrated measurement over a preset time, the MDA can be approximated by:

$$MDA = \frac{2.71 + 4.65 \sqrt{[(B_R)(t)]}}{(t)(E)(A/100)}$$

7.7.8.2 For a ratemeter instrument the MDA can be approximated by taking twice the time constant of the meter as the counting time:

$$MDA = \frac{4.65 \sqrt{[(B_R)/(2t_c)]}}{(E)(A/100)}$$

where

MDA = minimal detectable activity in dpm/100 cm^2

B_R = background cpm

t = counting time in minutes

t_c = meter time constant in minutes

E = instrument detector efficiency

A = probe area in cm^2

7.7.9 Position the pancake probe of the survey meter using a contaminated equipment-to-detector distance similar to the calibration source-to-detector distance used above. (If alpha contamination is expected, a piece of paper can be placed between the probe and the surface being monitored to shield out the alpha particles). Slowly move the probe over the area to be surveyed stopping at areas of suspect contamination to obtain an accurate determination of the level of contamination. Survey several different areas, to obtain an average count rate and a maximum count rate.

7.7.10 Subtract the background reading from the reading observed during the survey and multiply by the pancake correction factor. Record the readings obtained in the total beta/gamma column of Form RH-070A or the equivalent.

7.7.11 Measure for removable activity only if the total beta-gamma activity exceeds the Removable release limit specified in Appendix A, Table A1.

7.7.12 Number the swipes for identification.

7.7.13 Using a swipe rub lightly an area of 100 cm^2 of potential contamination. Place the swipe in the pancake probe holder or on a clean flat surface and count the beta-gamma emission rate. Multiply the observed reading less the background reading by the pancake probe efficiency. Alternately count the swipe in an Eberline model BC-4 beta swipe counter or similar instrument. Record the measurements in the removable beta/gamma column of Form RH-070A or the equivalent.

7.8 ***AUTOMOBILE AND TRUCK SURVEYS***

7.8.1 Conduct the alpha survey and a beta-gamma contamination survey of cars, trucks and other vehicles leaving the restricted area of the facility. The survey should include a survey of at least the driver's door handle, the driver's seat, one tire, the floor of the inside of the vehicle, and any dirty or suspect areas. The RSO, Assistant RSO, or a designee can authorize the release of cars and trucks leaving the restricted area, providing contamination levels have not exceeded the limits in Table A1.

7.9 ***RELEASE OF EQUIPMENT OR MATERIALS***

7.9.1 Compare the results recorded on Form RH-070A or the equivalent with the limits listed on Table A1. If any of the limits in Table A1 are exceeded, check NO in the release column of Form RH-070A or its equivalent and **DO NOT** release the material from the restricted area of the mill. Contact the RSO or Assistant RSO for further instructions.

7.9.2 If the values are less than the limits, check YES in the Release column and give Form RH-070A to the RSO, the Assistant RSO, or a designee for his signature and approval to release the equipment, materials, or vehicles from the restricted area. A copy of the signed form serves as documentation of the survey and approval to release the items from the restricted area to the general public. Equipment and materials that are to be sent to another licensee, that is permitted under their license to accept the items, need only comply with the Department of Transportation limits and not the limits specified on Form RH-070A. See Procedure RH-080, Shipment of Yellowcake, Ore, or Contaminated Equipment by Truck.

Release of Equipment to Unrestricted Areas Procedure RH-070

Appendix A

Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use Or Termination Of Licenses For Byproduct, Source Or Special Nuclear Material

Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material

The instructions in this guide in conjunction with Table A1 specify the radioactivity and radiation exposure rate limits that should be used in accomplishing the decontamination and survey of surfaces of premises and equipment prior to abandonment or release for unrestricted use.

1.) The Mill shall make a reasonable effort to eliminate residual contamination.

2.) Paint, plating, or other covering material shall not cover radioactivity on equipment or surfaces unless contamination levels, as determined by a survey and documented, are below the limits specified in Table A1 prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3.) The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap, which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement, are presumed to be contaminated in excess of the limits.

Table A1
Acceptable Surface Contamination Levels

Nuclides [a]	Average [b c f] (dpm per 100 cm^2)	Maximum [b d f] (dpm per 100 cm^2)	Removable [b e f] (dpm per 100 cm^2)
Alpha Emissions from U-nat, U-235, U-238 and associated decay products	5,000	15,000	1,000
Alpha emissions from Ra-226, Ra-228, Th-230, Th-228, Ac-227	100	300	20
Alpha Emissions from Th-nat, Th-232, Ra-223, Ra-224, U-232	1,000	3,000	200
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except others noted above	5,000	15,000	1,000

Notes:

[a] Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average containment should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filters or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Release of Equipment to Unrestricted Areas Procedure RH-070

Appendix B

Equipment and Materials Survey Form RH-070A

Equipment and Materials Survey

Location (Source of equipment)_____Surveyed by:_____ Date_____

Approval by RSO or Assistant RSO for release_____

Alpha check source _____dpm Type _____Gamma check source _____µCi Type _____

Beta check source _____dpm

Conversion factors

Equipment Description	Area Surveyed (or see map on back of form)	Total Alpha		Removable Alpha		Total Beta/Gamma		Removable Beta/Gamma		Permitted for Release	
		cpm	dpm/ 100cm^2	cpm	dpm/ 100cm^2	cpm	dpm/ 100cm^2	cpm	dpm/ 100cm^2	YES	NO
Instrument Serial No:											
Date Calibrated											
Correction Factor											
Background											
Limits – RH-070 Table A1		15000 dpm Max 100 cm^2 5000 dpm Ave 100 cm^2		1000 dpm Max 100 cm^2		15000 dpm Max 100 cm^2 5000 dpm Ave 100 cm^2		1000 dpm Max 100 cm^2			

Release approved by: _____

1.0 PURPOSE

This procedure outlines the Department of Transportation requirements for the shipment of yellowcake (uranium oxide) or surface contaminated equipment from the Piñon Ridge Mill by truck. The requirements for the transportation of ore to the processing facility are also included in this procedure to help ensure that incoming shipments meet the DOT requirements. 49 CFR 100-399 presents the complete regulations for the shipment of all types of radioactive material by all methods of transportation. This procedure addresses:

- What to include on shipping papers
- Package specifications and markings
- Radiation and contamination limitations
- Placarding of trucks

2.0 APPLICABILITY

This procedure applies to the Radiation Safety Officer (RSO), the Assistant RSO, the Mill Foreman, Radiation/Security Technician (RST), and the Mill Operators.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Department of Transportations Regulations 49 CFR Parts 100-399

3.2 *APPENDICES*

Appendix A - Form RH-100A Shipping Papers

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 Ludlum model 43 series alpha detectors (scintillation, proportional or phoswich) or equivalent

4.1.2 Ludlum model 2241 series scaler/ratemeter or equivalent

4.1.3 Ludlum model 2929 alpha/beta sample counter or equivalent

4.1.4 Ludlum 44-9 pancake GM probe or equivalent

4.1.5 Ludlum model 12s micro R meter or equivalent

4.1.6 Ludlum model 44 -10 gamma scintillation probe, 2 x 2 or equivalent

4.1.7 NIST alpha and beta check sources for determining instrument and detector efficiencies.

4.2 *MATERIALS*

4.2.1 Swipes

4.2.2 “Low Specific Activity” stencils and spray paint

4.2.3 “Radioactive 7” DOT placard

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

4.2.4 Form RH-100A or the equivalent

5.0 RESPONSIBILITY

5.1 The Mill Foreman or his designee is responsible for:

5.1.1 The preparation of the shipping papers.

5.2 The RSO is responsible for:

5.2.1 The shipment of yellowcake and other radioactive materials from the uranium processing facility.

5.3 The Assistant RSO or his designee is responsible for:

5.3.1 The placarding of the trucks.

5.4 The RST is responsible for:

5.4.1 The monitoring of individual packages of radioactive materials and transport vehicles.

5.5 The Mill Operators are responsible for:

5.5.1 The proper loading and bracing of packages and/or barrels in the trucks and for the labeling of packages or barrels.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 Exclusive Use – Sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee.

6.1.2 LSA - Low Specific Activity.

6.1.3 RQ - Reportable Quantity.

6.1.4 SCO - Surface Contaminated Objects.

6.1.5 Transport Index - mrem/hr at one meter from the external surface of the package, barrel, piece of equipment or transport vehicle.

6.2 *SAFETY*

6.2.1 Piñon Ridge Mill Health and Safety Plan

6.3 *FREQUENCY*

6.3.1 Every time uranium oxide (yellowcake), ore, or contaminated equipment is offered for shipment from the Mill.

7.0 PROCEDURES

7.1 *PREPARATION OF SHIPPING PAPERS*

7.1.1 The Mill Foreman or his designee is to prepare the shipping papers for yellowcake, ore, equipment, and/or materials contaminated with uranium and its progeny using Form RH-100A or its equivalent. The Department of Transportations (DOT) regulation in 40 CFR 172.201(a)(2) requires that the shipping papers be legible, and printed manually in English. The

material to be shipped is specified in the middle column of the following table. The information to be included on the form is in quotes in the right-hand column. The left-hand column presents the applicable section of the DOT regulations.

DOT Regulation in 40 CFR	Material To Be Shipped	Information For Shipping Papers
§172.202(a)(1) §172.101 Hazardous Materials Table	Yellowcake, ore	Proper Shipping Name: Radioactive Material, low specific activity, LSA-1, UN2912
§172.202(d) §172.101 Hazardous Materials Table § 173.403 *LSA material*	Equipment or materials contaminated by naturally occurring uranium or thorium ores or concentrates	Proper Shipping Name: Radioactive Material, surface contaminated object or SCO, UN2913
§171.8, §172.203(c)(2)	Yellowcake, ore, SCO equipment	Reportable Quantity: "RQ" before or after the shipping name
§172.202(a)(2)	Yellowcake, ore, SCO equipment	Hazard Class: "7"
§172.202(a)(3) + §172.101 Hazardous Materials Table	Yellowcake, ore SCO equipment	I.D. Number: "UN2912" "UN2913"
§172.203(d)(1)(i)	Yellowcake, ore, SCO equipment	Radionuclide(s): "U (Natural)" plus decay products
§172.203(d)(1)(ii)	Yellowcake, SCO equipment	Physical and Chemical form: "Solid Oxide"
§172.203(d)(1)(ii)	Ore	Physical and Chemical form: "Solid Ore"
§172.202(a)(5)	Yellowcake	Total Quantity of Material: Enter number, "#" lbs, "#" barrels.
§172.202(a)(5)	SCO contaminated equipment	Total Quantity of Material: Enter number "#" (e.g. 3 electric motors)
§172.202(a)(5)	Ore	Total Quantity of Material: "#" lbs

DOT Regulation in 40 CFR	Material To Be Shipped	Information For Shipping Papers
§172.203(d)(4)	Yellowcake or SCO contaminated equipment	Activity of Radioactive Material: "#"TBq calculated as (1): (# lbs U_3O_8)(1.01E-5) "#"Ci calculated as (2): (#TBq)(27 Ci/TBq)
§172.203(d)(4)	Ore	Activity of Radioactive Material: "#"TBq calculated as (1): (%U_3O_8 content)(# lbs Ore)(4.01E-5) "#"Ci calculated as (2): (#TBq)(27 Ci/TBq)
§172.203(d)(10)(ii)	Exclusive use shipment only	Type of Shipment: "Exclusive Use Shipment"
§173.403 *Exclusive use*	Yellowcake, ore, SCO contaminated equipment	Exclusive Use Instructions: "Only transport material specified on this shipping paper."
§172.204(a)(1)	Yellowcake, ore, or SCO contaminated equipment	Shipper's certification: "This is to certify that the above-named materials are properly classified, described, packaged, marked and labeled, and are in proper condition for transportation according to the applicable regulations of the Department of Transportation."
Signature §172.204(d)(1+2)	Yellowcake, ore, or SCO contaminated equipment	Print Name, Signature, Title: "_____" Print Name "_____" Signature of principal, officer, partner, or agent "_____" Title

NOTES: A packaging group Roman numeral is not required by §172.101 Hazardous Materials Table for Radioactive Materials, low specific activity, UN2912.

For domestic use only a radioactive materials transportation label is not required on each container by §173.427(a)(6)(vi).

Transportation index is not required because a transportation label is not required as per §172.203(d)(6).
(1)To convert pounds of uranium oxide (yellowcake) to TBq multiply the following:
(#U_3O_8)(7.1E-7 Ci/g)(453.6 g/#)(0.85 U_3/ U_3O_8)(3.7E10 Bq/Ci)(E-12 TBq/Bq) = (#U_3O_8)(1.01E-5).
To convert pounds of ore to TBq multiply the following:
(%U_3O_8 content in ore)(#ore)(0.85 #U/# U_3O_8)(453.6 g/#)(2.6E-8 TBq/g)(4 to account for chain)= (%U_3O_8 content)(#pounds)(4.01E-5)
(2)To convert TBq to Ci multiply the following: (TBq)(27 Ci/TBq)

7.1.2 Barrel Specifications And Marking

7.1.2.1 Each offeror of a Specification 7A package must maintain on file, for at least one year after the latest shipment, complete documentation of tests and an engineering evaluation of comparative data showing that the construction complies with the 7A specification §173.415(a). Label the shipments as specified below:

DOT Regulation in 49 CFR	Material To Be Shipped	Marking On Each Barrel Or Requirement
§173.427(c)(l)	Yellowcake shipped in strong tight packages as exclusive use.	"Radioactive-LSA, RQ" on exterior of each barrel.
§173.427(c)(l)	SCO Contaminated equipment	"Radioactive-SCO-1, RQ" on equipment
§173.427(6)(ii) §173.427(b)(3)	Yellowcake	None. Barrels must be strong and tight to prevent leakage of contents under normal conditions of transport.
§173.427(6)(ii)	Ore, SCO contaminated equipment	None. Leakage of radioactive material from the equipment or package is not permitted.
§173.427(a)(6)(i)	Yellowcake, ore, or SCO contaminated equipment	None. Shipments must be loaded by the consignor and unloaded by the consignee from the conveyance in which originally loaded.

7.1.3 Radiation and Contamination Limitations

7.1.3.1 Monitor the 7A barrels, ore trucks, and/or equipment and materials contaminated with uranium and its progeny as specified below: See procedure RH-110, Beta and/or Gamma Surveys and RH-120, Alpha, Beta Contamination Radiation Survey

DOT Regulation in 49 CFR	Material To Be Shipped	Limitation
§173.441(a)(1)	Yellowcake or SCO contaminated equipment in exclusive use shipments	200 mrem/hr at any point on the external surface of the each barrel or piece of equipment and 10 mrem/hr at one meter from the external surface of each barrel or piece of equipment
§173.443(b)	Yellowcake, Ore or SCO contaminated equipment at beginning of shipment, EACH barrel for exclusive use shipment	6,600 dpm/300 cm^2 beta/gamma removable 660 dpm/300 cm^2 alpha removable
§173.443(b)	Yellowcake, Ore or SCO contaminated equipment anytime during shipment, EACH barrel for exclusive use shipment	66,000 dpm/300 cm^2 beta/gamma removable 6,600 dpm/300 cm^2 alpha removable
§173.443 & §177.843(b)	Contamination on any exclusive use transport vehicle	10 mrem/hr at external surfaces or 2 mrem/hr at 1 meter from any interior surface

NOTE: Empty bulk shipment containers can be released per the requirements listed in the DOT regulations §173.443(d)(1),(2),(3) pertaining to external radiation and contamination control.
NOTE: Exclusive use vehicles used to transport radioactive materials must be stenciled with the words “For Radioactive Materials Use Only.” In letters at least 7.6 cm (3 inches) high in a conspicuous place on both sides of the exterior of the vehicle (§173.443 & §177.843(b)).

7.1.4 General Transportation Requirements

7.1.4.1 Load the 7A barrels or contaminated equipment as described below:

DOT Regulation in 49 CFR	Material To Be Shipped	Limitation
§173.448(a)	Yellowcake or SCO contaminated equipment	Each shipment must be secured to prevent shifting during normal conditions.
§173.448(b)	Yellowcake	Barrels of yellowcake or contaminated equipment may not be carried in compartments occupied by passengers, except in those compartment exclusively reserved for couriers accompanying the barrels.

7.1.5 Placarding Of Trucks

7.1.5.1 Place placards on the trucks as described below:

DOT Regulation in 49 CFR	Material To Be Shipped	Requirement
§172.504(a) + §172.507 + §172.556(a)	Yellowcake or SCO contaminated equipment	Each motor vehicle used to transport barrels of yellowcake or contaminated equipment must have a "RADIOACTIVE" warning placard placed on a square background on each end and on each side of the transport vehicle. The required placarding of the front of a motor vehicle may be on the front of a truck-tractor instead of or in addition to the placarding on the front of the cargo body to which the truck-tractor is attached.

DOT Regulation in 49 CFR	Material To Be Shipped	Requirement
§172.506(a)	Yellowcake or SCO contaminated equipment	Shipper to supply placards unless truck is already placarded.
§173.427(6)(v)	Ore	No placards required on trucks. Note requirements listed in §173.443(d)(2))

Shipment of Yellowcake, Ore or Contaminated Equipment by Truck
Procedure RH-100

Appendix A

Shipping Papers Form RH-100A

Shipping Papers

SHIPPER:
Energy Fuels Resources
Montrose County, Colorado

EMERGENCY CONTACTS
Shipper:
xxxxx (Radiation Safety Officer) Tel #xxxxxxx
xxxxx (Plant Manager) Tel #xxxxxxx
24 Hour Emergency Numbers(Cell): Tel #xxxxxx
Transporter:

TRANSPORTER:
Name and address: _____
_____
_____
Tractor #_____
Trailer #_____
Time In _____
Time Out _____
Tailgate Check _____

SHIPPER INFORMATION

PROPER SHIPPING NAME:	Radioactive Material LSA-1, RQ	PHYSICAL AND CHEMICAL FORM:	__Solid Oxide__Ore
HAZARD CLASS:	7	TOTAL QUANTITY OF MATERIAL:	____lbs., ___barrels
I.D. NUMBER:	UN2912	ACTIVITY OF RADIOACTIVE MATERIAL:	_____(___) TBq(Ci)
RADIONUCLIDE(S):	U(Natural)		
TYPE OF SHIPMENT:	Exclusive Use		
EXCLUSIVE USE INSTRUCTIONS:	Only transport material specified on this shipping paper.		
DATE OF SHIPMENT:	_____		

DESTINATION:
Name and Address: _____
_____

SHIPPER CERTIFICATION

This is to certify that the above mentioned material are properly classified, described, packaged, and labeled, and are in proper condition for transportation according to the applicable regulations of the Department Of Transportation.

_____ _____ _____
Print Name Signature Title

RECEIVER VERIFICATION

Receiver:
Location:
Date Received:
Comments:

Form RH-100A

1.0 PURPOSE

This procedure describes how to conduct a beta and/or a gamma survey within the Mill. The purpose of the survey is to identify areas of elevated beta or gamma exposures so that the radiation doses may be kept to levels that are As Low As Reasonably Achievable (ALARA). Gamma monitoring of personnel using dosimeters is addressed in Procedure RH-210, "Personnel Radiation Monitors."

2.0 APPLICABILITY

This procedure provides guidance to the radiation protection staff or others conducting beta and/or gamma exposure rate surveys of the facilities. All personnel and visitors present inside the uranium processing facility are to maintain their radiation doses at levels that are ALARA.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills."

3.2 *APPENDICES*

Appendix A - Correction Factor for Beta Survey Instruments

Appendix B - Beta/Gamma Survey Form RH-110A

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 Ludlum model 2241 series scaler/ratemeter or equivalent

4.1.2 Ludlum model 2929 alpha/beta sample counter or equivalent

4.1.3 Ludlum 44-9 pancake GM probe or equivalent

4.1.4 Ludlum model 12s micro R meter or equivalent

4.1.5 Ludlum model 44 -10 gamma scintillation probe, 2 x 2 or equivalent

4.1.6 NIST beta and gamma check sources for determining instrument and detector efficiencies.

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or the Assistant RSO is responsible for:

5.1.1 Reviewing Form RH-110A for accuracy and completeness

5.1.2 Evaluating any trends of increasing or decreasing exposures rates over time.

5.2 The Radiation/Security Technician (RST) is responsible for:

5.2.1 Conducting gamma surveys at representative work locations.

5.2.2 Performing beta exposure surveys of areas or equipment at the Mill.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.3 The Quality Assurance Officer or designee is responsible for:

5.3.1 Assuring that personnel considered as key individuals relevant to this procedure are documented as properly trained and/or qualified to perform the required duties.

5.3.2 Assuring that the quality principles of this procedure are followed and maintained.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

Radiation Area – An area accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in 1 hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

6.2 *SAFETY*

6.2.1 Beta calibration of instruments using yellowcake or other processing material samples subject to dusting will be performed under a dust collection hood.

6.3 *FREQUENCY*

6.3.1 Monthly gamma radiation surveys in Radiation Areas and where specified by the RSO.

6.3.2 Quarterly gamma radiation surveys.

6.3.3 Annual beta radiation surveys at locations where beta dose rates may have changed.

7.0 PROCEDURES

7.1 *GAMMA SURVEYS*

7.1.1 The RST is to turn on the gamma survey meter. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.1.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or as specified by the manufacturer. If not, use an instrument that is in calibration. Record the instrument and probe serial numbers on Form RH-110A or the equivalent.

7.1.3 Inspect the survey meter, probe and probe cable for damage etc. If damaged, replace cable etc. or select another instrument for the survey.

7.1.4 Note the instrument reading without the check source present and record on calibration check form. This is the background count rate.

7.1.5 Place the gamma check source on contact with the instrument detector and observe the reading, subtract the background measurement and record the net reading on Form RH-110A. Use the same check source that was used to determine the reference reading when the instrument was calibrated. If the reading with the check source varies more than 20% (Reg. Guide 8.30,

page 21) from the historic reference readings, select another instrument for the survey and submit the defective instrument for repair.

7.1.6 The RST is to survey at least 20 locations semiannually to characterize the whole-body exposures to workers in the facility. Survey areas where workers might stand or work or approximately 12 inches from the equipment. Record the results on Form RH-110A.

7.1.7 If the dose rate in areas accessible to workers exceeds 5 mrem/hr at 12 inches from the radiation source, post the area as a "Radiation Area" and resurvey quarterly.

7.2 ***BETA-GAMMA EXPOSURE RATE SURVEY OF EQUIPMENT AND MATERIALS*** *(See also Procedure RH-070)*

7.2.1 Turn on the instrument equipped with a GM pancake probe or GM probe with a sliding shield. Use battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.2.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-110A or the equivalent.

7.2.3 Inspect the meter and probe for damage. If damaged, use another instrument.

7.2.4 Observe the instrument background count rate and record on the calibration form.

7.2.5 Measure the Cs-137 check source activity. Proceed if the measured activity is within 20% of the historic reference readings. Record the check source activity on Form RH-110A or the equivalent.

7.2.6 Annual beta radiation surveys are to be performed at representative locations where workers are exposed, when equipment or operating procedures are modified in a way that may have changed the beta dose that would be received by the worker, and specifically at locations that involve direct handling of large quantities of aged yellowcake. Additional surveys will be performed at locations where other materials for processing are stored and processed to evaluate the exposure levels.

7.2.7 Perform a measurement with the pancake probe without a probe shield and subtract the background reading. This net reading is the beta + gamma exposure rate.

7.2.8 Perform a measurement using the pancake probe with the plastic shield in place. Subtract the background reading recorded in 7.2.4 above. This net reading is the gamma only exposure rate from the item surface.

7.2.9 Determine the difference between the two measurements and multiply the reading by the beta correction factor. See Appendix A for the determination of the beta correction factor. The correction factor is noted on a label attached to the instrument.

7.2.10 Determine the beta-gamma exposure rate by summing the gamma only measurement as determined in 7.2.8 and the corrected beta reading calculated in 7.2.9. Record the average and maximum beta-gamma exposure rates on Form RH-110A.

Beta and/or Gamma Exposure Rate Surveys Procedure RH-110

Appendix A

Correction Factors

Correction Factor for Beta Survey Instruments

As noted in the USNRC Regulatory Guide 8.30, "Health Physics Surveys of Uranium Mills," Appendix C, a correction factor needs to be generated for the beta survey instrument. The beta survey correction factor is calculated as follows:

$$CF = \frac{YC}{R_u - R_s}$$

Where:

CF = Correction factor

YC = Actual beta dose rate from an old yellowcake source

= 75 mrem/hr at 2 cm from the surface or 150 mrem/hr at the surface (Reg. Guide 8.30, Appendix C)

R_u = Unshielded exposure rate

R_s = Shielded exposure rate

Correction Factor for Probes with a Sliding Shield

For survey instruments with a GM probe and a sliding shield the beta survey correction factor was calculated as shown in the following example:

Instruments:

Eberline Model E 140 Survey Meter with an HP 270 GM probe with a sliding shield, Serial # 1760, calibrated 2-02-02. Eberline Model E 140 Survey Meter with an HP 270 GM probe with a sliding shield, Serial # 2263, calibrated 2-02-02.

Source: "Black yellowcake, 9-25-85, Lot 8406-60, Uncovered lab sample from drum. Data courtesy of Cotter Corporation.

Measurements at 2 cm from source on 7/17/02:	Meter # 1760	Meter # 2263
• Background	0.04 mR/hr	0.04 mR/hr
• Probe shield closed	10. mR/hr	12. mR/hr
• Probe shield open	40. mR/hr	36. mR/hr

CF at 2 cm for instrument # 1760 $= \frac{YC}{R_u - R_s} = \frac{75}{40-10} = 2.5$

$$\text{CF at 2 cm for Instrument \# 2263} = \frac{YC}{R_u - R_s} = \frac{75}{36-12} = 3.1$$

Different correction factors may be determined when processing high thorium content ores.

Correction Factor for a Pancake Probe

For survey instruments with a pancake probe, a beta survey correction factor can be calculated as follows:

Instruments: Eberline E-140 with a HP-210 probe or the equivalent.

Material Source:

Processing material sample e.g. Black aged yellowcake, Uncovered lab sample from drum.

Measurements:

Take two measurements with the meter:

- At 2 cm above the surface of the yellowcake or other processing material <u>without the plastic detector shield</u>,
- At 2 cm above the surface of the yellowcake or other processing material <u>with the plastic detector shield in place.</u>

A 1.0 cm thick plexiglass or Lucite plate may be used for the plastic detector shield.

Calculate the correction factor as shown above.

Application of the Correction Factor

The actual beta dose rate = $(S_u – S_s)(CF)$

where:

S_u = Unshielded exposure rate from a source such as a contaminated tool.
S_s = Shielded exposure rate from a source such as a contaminated tool. The shield can be a sliding shield on a GM detector or a piece of 1.0 cm thick plexiglass used with a pancake probe.

Beta-gamma dose rate = Actual beta dose rate + Gamma (shielded) dose rate

Beta and/or Gamma Exposure Rate Surveys Procedure RH-110

Appendix B

Beta/Gamma Survey Form RH-110A

Beta/Gamma Survey Form

Instrument_____ SN_____

Detector/Probe_____ SN_____ Calibration Date_____

Net Check Source Reading_____ Historical check source reading_____

Date	Survey Location	Type of Measurement	Gamma plus Beta Reading	Gamma Only Reading

1.0 PURPOSE

This procedure describes methods for measuring alpha and beta surface contamination.

2.0 APPLICABILITY

This procedure applies to all equipment and materials potentially contaminated in and around the Piñon Ridge Mill.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 NRC Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

3.1.1 NUREG 1507 "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminates and Field Conditions."

3.1.1 Colorado Radiation Control Regulations, Part 4 (6 CCR 1007-1).

3.2 *APPENDICES*

Appendix A – Alpha, Beta Contamination Survey Report Form RH-120A.

Appendix B – Acceptable Surface Contamination Levels

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 Ludlum model 43 series alpha detectors (scintillation, proportional or phoswich) or equivalent

4.1.2 Ludlum model 2241 series scaler/ratemeter or equivalent

4.1.3 Ludlum model 2929 alpha/beta sample counter or equivalent

4.1.4 Ludlum 44-9 pancake GM probe or equivalent

4.1.5 Ludlum model 12s micro R meter or equivalent

4.1.6 Ludlum model 44 -10 gamma scintillation probe, 2 x 2 or equivalent

4.1.7 Swipes.

4.1.5 Form RH-120A or the equivalent.

4.1.6 Th-230 National Institute of Standards and Technology (NIST)-traceable alpha check source or equivalent.

4.1.7 NIST alpha and beta check sources for determining instrument and detector efficiencies.

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or Assistant RSO is responsible for:

5.1.1 Assuring that all personnel performing the surveys are properly trained.

5.1.2 Determining the locations throughout the facility where surveys should be

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

performed.

5.1.3 Determining the frequency of sampling at a given location based on previous scans or by administrative controls.

5.1.4 Determining any corrective actions that should be put into affect to alleviate areas of higher radioactivity in the Mill.

5.1.5 Reviewing Contamination Survey Reports.

5.1.6 Reporting any trends to management and documenting the trends in the annual report.

5.2 The Radiation/Security Technician (RST) is responsible for:

5.2.1 Assuring that data are collected according to this procedure.

5.2.2 Filing and retaining the Alpha Contamination Survey Report Form RH-120A or the equivalent.

5.3 The Quality Assurance Officer or designee is responsible for:

5.3.1 Assuring that personnel considered as key individuals relevant to this procedure are documented as properly trained and/or qualified to perform the required duties.

5.3.2 Assuring that the quality principles of this procedure are followed and maintained.

6.0 PREREQUISITE INFORMATION

6.1 *SAFETY*

See Piñon Ridge Mill Health and Safety Plan.

6.2 *FREQUENCY*

6.2.1 Alpha and beta radiation surveys for total and removable contamination will be conducted weekly throughout restricted areas.

6.2.2 Surveys at other locations for alpha contamination will be conducted in facilities throughout the project site as specified by license conditions and/or the RSO or the Assistant RSO.

7.0 PROCEDURES

7.1 *TOTAL ALPHA SURVEYS OF EQUIPMENT AND MATERIALS*

7.1.1 The RST is to turn on the alpha survey meter. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.1.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-120A or the equivalent.

7.1.3 Inspect the survey meter for damage. If an alpha scintillation probe is to be used in the survey, hold the probe towards a light source. If the count rate suddenly increases, look for tiny pinholes in the mylar surface of the probe face. Repairs can be made with “whiteout.” If two or more whiteout spots are present, use another instrument and report the damage to the

RSO or Assistant RSO.

7.1.4 Preferably survey equipment and materials in a low background area. Observe the alpha background count rate of the instrument. If more than five counts per minute, use a different survey meter and report the potential contamination and/or malfunction to the RSO or the Assistant RSO.

7.1.5 If the item to be monitored is wet, dry the item prior to conducting the alpha survey.

7.1.6 Position the probe face of the survey meter within 1.0 cm of the Th-230 check source or the equivalent alpha check source. Use the end of one's finger under the edge of the alpha probe as a guide for the 1.0 cm spacing. Use a source-to-detector distance for the calibration that is similar to the expected surface-to-detector spacing used in the field. Record the reading on Form RH-120A. If the reading is within 20% (Reg.Guide 8.30, page 21) of the historic reference readings conclude that the instrument is functioning properly. If not, use an instrument that is functioning properly.

7.1.7 Determine the instrument minimum detectable activity (MDA) and record on the calibration form.

7.1.7.1 For an integrated measurement over a preset time, the MDA can be approximated by:

$$MDA = \frac{2.71 + 4.65 \sqrt{[(B_R)(t)]}}{(t)(E)(A/100)}$$

7.1.7.2 For a ratemeter instrument the MDA can be approximated by taking twice the time constant of the meter as the counting time:

$$MDA = \frac{4.65 \sqrt{[(B_R)/(2t_c)]}}{(E)(A/100)}$$

where

MDA = activity in dpm/100 cm^2

B_R = background cpm

t = counting time in minutes

t_c = meter time constant in minutes

E = instrument detector efficiency

A = probe area

7.1.8 Position the probe face of the survey meter within 1.0 cm of the equipment or material being monitored. Use the end of one's finger under the edge of the alpha probe as a guide to the 1.0 cm spacing. Slowly (~1 cm/sec) move the probe over the area to be surveyed stopping at areas of suspect contamination to obtain an accurate determination of the level of contamination. Average surface contamination measurements are for 100 cm^2 areas averaged over 1 m^2. Maximum contamination measurements are averaged over 100 cm^2. Survey several different areas to obtain an average

count rate and a maximum count rate.

7.1.9 Calculate the portable survey instrument correction factor by:

$$\text{Alpha Correction factor} = \left(\frac{\text{Source dpm}}{\text{meter cpm}}\right)\left(\frac{100\ \text{cm}^2}{X\ \text{cm}^2}\right)(E_s)$$

Where X is the detection surface area of the probe with no corrections made for the protective screen and 100 cm^2 is the area specified in the release limits. The "source dpm/meter cpm" measurement corrects for the screen covering part of the detection area of the probe (NUREG-1507, pages 5-7). E_s = Efficiency of the source = 0.5 for paper wipes (NUREG-1509, pages 5-20)

7.1.10 Record the observed cpm on Form RH-120A under the Total Alpha column. Multiply the highest and average survey result by the correction factor and record on Form RH-120A under the Total Alpha dpm/100 cm^2 column.

7.1.11 Enter the following data on Form RH-120A or its equivalent:

- Equipment description and area surveyed
- Technician conducting the survey
- Date
- Survey equipment serial number for both the alpha and beta-gamma surveys
- Date calibrated

7.2 *REMOVABLE ALPHA SURVEY OF EQUIPMENT AND MATERIALS*

7.2.1 Measure for removable alpha activity only if the Total Alpha Activity exceeds the appropriate removable activity limit listed in Appendix B.

7.2.2 Number the swipes for identification.

7.2.3 Using a swipe rub lightly an area of 100 cm^2 of potential contamination. Place the swipe on a clean flat surface and count the alpha emission rate using the portable alpha meter and multiply the observed reading by the calculated instrument efficiency factor, or count the swipe in a Ludlum model 2000 scaler/43-10 alpha swipe counter or similar instrument.

7.2.4 Document all survey results on Form RH-120A under the removable alpha dpm/100 cm^2 column.

7.3 *BETA CONTAMINATION SURVEY OF EQUIPMENT AND MATERIALS*

7.3.1 Turn on the instrument equipped with a GM pancake probe. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.3.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-120A or the equivalent.

7.3.3 Inspect the meter and probe for damage. If damaged, use another

instrument.

7.3.4 Observe the instrument background count rate and record on Form RH-120A.

7.3.5 Using a Cl-36 calibration standard or the equivalent and a source-to-detector distance for the calibration that is similar to the expected surface-to-detector spacing used in the field (e.g. 1 cm) measure the source activity and background with a pancake probe. Proceed if the measured activity is within 20% of the historic calibration measurements.

7.3.6 Preferably survey equipment and materials in a low background area. Calculate the pancake probe beta correction factor by:

$$\text{Pancake beta correction factor} = \frac{(\text{source dpm})(100\ \text{cm}^2)}{(\text{meter cpm-bkg cpm})(X\ \text{cm}^2)}$$

where:

X = The pancake probe detection surface area (NUREG-1507, page 4-7)

100 cm^2 = The surface area specified in the CDPHE regulations.

7.3.7 Calculate the efficiency of the pancake probe as:

$$E = \frac{\text{meter cpm} - \text{bkg cpm}}{\text{Source dpm}}$$

7.3.8 Determine the pancake minimum detectable activity (MDA), if needed, using E calculated above.

7.3.8.1 For an integrated measurement over a preset time, the MDA can be approximated by:

$$MDA = \frac{2.71 + 4.65\sqrt{[(B_R)(t)]}}{(t)(E)(A/100)}$$

7.3.8.2 For a ratemeter instrument the MDA can be approximated by taking twice the time constant of the meter as the counting time:

$$MDA = \frac{4.65\sqrt{[(B_R)/(2t_c)]}}{(E)(A/100)}$$

where

MDA = minimal detectable activity in dpm/100 cm^2

B_R = background cpm

t = counting time in minutes

t_c = meter time constant in minutes

E = instrument detector efficiency

A = probe area in cm^2

7.3.9 Position the pancake probe of the survey meter using a contaminated equipment-to-detector distance similar to the calibration source-to-detector distance used above. (If alpha contamination is expected, a piece of paper can be placed between the probe and the surface being monitored to shield out the alpha particles). Slowly move the probe over the area to be surveyed stopping at areas of suspect contamination to obtain an accurate determination of the level of contamination. Survey several

different areas, to obtain an average count rate and a maximum count rate.

7.3.10 Subtract the background reading from the reading observed during the survey and multiply by the pancake correction factor. Record the readings obtained in the total beta/gamma column of Form RH-120A or the equivalent.

7.3.11 Measure for removable activity only if the Total Beta-Gamma Activity exceeds the Removable Release Limit specified in Appendix B.

7.3.12 Number the swipes for identification.

7.3.13 Using a swipe rub lightly an area of 100 cm^2 of potential contamination. Place the swipe in the pancake probe holder or on a clean flat surface and count the beta-gamma emission rate. Multiply the observed reading less the background reading by the pancake probe efficiency. Alternately count the swipe in an Eberline model BC-4 beta swipe counter or similar instrument. Record the measurements in the removable beta/gamma column of Form RH-120A or the equivalent.

7.4 *RECORDS*

7.4.1 The RST is to file Form RH-120A or its equivalent and retain those records.

Alpha Beta Contamination Surveys Procedure RH 120

Appendix A

Alpha/Beta Contamination Survey Report Form RH 120A

Instrument Calibration Date:_____

Alpha/Beta Contamination Survey Report Form

Date of Survey: _____ Reviewed by _____ Location Piñon Ridge Mill

Name of Surveyor: _____ Date _____

Alpha Instrument Type _____ SN _____ Alpha MDA_____ Alpha Correction Factor _____

Beta Instrument Type _____ SN _____ Beta MDA_____ Beta Correction Factor_____

Potential Location	Total Alpha Activity (dpm/100cm^2)	Removable Alpha Activity (dpm/100cm^2)	Total Beta Activity (dpm/100cm^2)	Removable Beta Activity (dpm/100cm^2)	Was Supervisor Notified?	Was Area Cleaned?	Final Reading (dpm/100cm^2)

Form RH-120A

Instrument Calibration Date:_____

Potential Location	Total Alpha Activity (dpm/100cm^2)	Removable Alpha Activity (dpm/100cm^2)	Total Beta Activity (dpm/100cm^2)	Removable Beta Activity (dpm/100cm^2)	Was Supervisor Notified?	Was Area Cleaned?	Final Reading (dpm/100cm^2)

Alpha, Beta Contamination Surveys Procedure RH-120

Appendix B

Acceptable Surface Contamination Levels Table 1

Acceptable Surface Contamination Levels

NUCLIDES	AVERAGE (per 100 cm^2)	MAXIMUM (per 100 cm^2)	REMOVABLE (per 100 cm^2)
Alpha emissions from U-nat, U-235, U-238 and associated decay products	5,000 dpm	15,000 dpm	1,000 dpm
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except others noted above	5,000 dpm	15,000 dpm	1,000 dpm

1.0 PURPOSE

This procedure is for the measurement of radioactive airborne particulate concentrations at the Piñon Ridge Mill to establish:

- Working locations where airborne concentrations are As Low As Reasonably Achievable (ALARA),
- The need for respiratory protection, and
- Trends in workers' radiation doses from long-lived airborne radionuclides.

2.0 APPLICABILITY

General air particulate sampling will be used in areas of the Mill where potential exists that employees could be exposed to air concentrations at or above 10% of the applicable Derived Air Concentration (DAC) for both long-lived radionuclides. Radon progeny monitoring is addressed in Procedure RH-140. Breathing zone air sampling is addressed in Procedure RH-150.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 U.S. Nuclear Regulatory Commission Regulatory Guide 8.25, Air Sampling in the Workplace.

3.1.2 U.S. Nuclear Regulatory Commission Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

3.1.3 Procedure RH-302, Radiological Concentrations in Air Samples Based on Gross Alpha Measurements and Laboratory Analyses.

3.1.4 Procedure RH-040, Radiation Exposure Action Limits

3.1.5 Colorado Radiation Control Regulations Part 4 (6 CCR 1007-1)

3.2 *APPENDICES*

Appendix A - General Air Sample Data Form RH-130A

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 Eberline RAS-1 air sampler**,** or the equivalent sampler capable of sampling at approximately 30 liters per minute (lpm).

4.1.2 Breathing zone air sampler may be used in areas where electrical power is not available or as specified by the Radiation Safety Officer (RSO) or Assistant RSO.

4.2 *MATERIALS*

4.1.1 47 mm, glass fiber filter sized to fit the Eberline RAS-1 air sampler, or the equivalent.

4.1.2 37 mm membrane filter for the breathing zone type sampler, if used.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.0 RESPONSIBILITY

5.1 The RSO or the Assistant RSO is responsible for:

5.1.1 Identifying and assigning the locations for sampler placement and frequency of sampling to the Radiation/Security Technicians (RSTs).

5.1.2 Training the RSTs on occupational air sampling.

5.1.3 Reviewing data from the general air samples, determining the needs for respiratory protection, and if necessary, taking any corrective actions.

5.2 The Radiation/Security Technician (RST) is responsible for:

5.2.1 Collecting the general air samples according to the schedule and frequency provided by the RSO.

5.2.2 Insuring that the equipment used for this sampling is properly calibrated and maintained.

5.2.3 Following the procedures contained herein to accomplish the sampling.

5.2.4 Informing the RSO or Assistant RSO of any deviations or problems that were encountered during sampling activities.

5.2.5 Retaining forms for a minimum of 3 years (6 CCR 1007-1 4.42.1).

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 General Air Samples - Air samples from the area of the Mill where personnel are working with radioactive materials. The general air samples are collected in the airflow pathway downstream of the sources of airborne radioactive material and between the source and the worker. "These samples do not have to be placed near the worker's breathing zone (within about 1 foot of the worker's head) and thus concentrations (measured by general air samples) might be considerably different from the concentrations in the breathing zone." (Reg. Guide 8.25, pages 4-5).

6.1.2 Mobile Air Particulate Sampler –Air sampler which can be moved to various locations to measure airborne concentrations.

6.1.3 Breathing Zone Air Particulate Sampler – A portable, battery operated air pump and filter cassette worn by the worker to measure concentrations of particulates in the vicinity of the "breathing zone" since the filter cassette is typically attached to a shirt lapel.

6.2 *SAFETY*

6.2.1 Care should be taken to avoid electrical shock. Samplers should be placed where water cannot come in contact with the pump or power cords.

6.2.2 Place power cords in positions where they are not tripping hazards.

6.2.3 Follow safety procedures in the Health and Safety Plan.

6.3 *FREQUENCY*

6.3.1 Weekly, monthly, or quarterly air particulate samples based on percent of applicable Derived Air Concentration (DAC).

- Concentration >25% DAC = Daily

- Concentration >10% and <25% DAC = Weekly
- Concentration <10% DAC = Monthly

6.3.2 The RSO or Assistant RSO is to determine the locations, sampling frequencies and parameters to be analyzed. He/she is to convey that information to the RST. Criteria for the types and frequency of general air particulate sampling are provided in Table 1.

6.3.3 Criteria for the RSO to establish the sampling locations are:

- General air samples should be collected in airflow pathways downstream from sources of airborne radioactive material and between the source and the worker.
- Sampling locations should be as close to the frequently occupied work locations as practical without interfering with work.
- Air samples intended to measure workplace concentrations should not be located in or near exhaust ducts because of the usual diluting effect of the exhausts.

6.3.4 For materials not listed in Table 1, a work plan and/or RWP may specify material-specific DACs, sampling locations and frequencies.

Table 1
Summary of Air Particulate Survey Frequencies and Type

Type of Survey	Type of Area	DAC[(a)]	Frequency and Type of Survey[(b)]	LLD[(c)]
Uranium ore dust	Airborne Radioactivity Area	>DAC	Daily 30-minute grab samples at ~30 lpm in worker occupied areas	2E -12 µCi/ml (uranium)
Uranium ore dust	Ore handling area indoors or Ore pad	>10% and <25%	Weekly 30-minute grab sample at ~ 30 lpm in worker occupied areas	2E -12 µCi/ml (uranium)
Uranium ore dust	Ore handling area indoors or Ore pad	<10%	Monthly 30-minute grab sample at ~30 lpm in worker occupied areas	2E -12 µCi/ml (uranium)
Uranium ore dust	Ore pad	>10%	Breathing zone grab sample if front-end loader generates visible ore dust	2E -12 µCi/ml (uranium)
Yellowcake	Special maintenance involving high airborne concentrations of yellowcake	≥DAC	Continuous breathing zone air monitoring (See Procedure RH-150) (Analyze samples before end of next shift).	3E -11 µCi/ml

Type of Survey	Type of Area	DAC[a]	Frequency and Type of Survey[b]	LLD[c]
Yellowcake	Special maintenance involving high airborne concentrations of yellowcake	>12 DAC-hrs per week	Breathing zone grab samples (See Procedure RH-150)	3E -11 µCi/ml
Yellowcake	Airborne Radioactivity Area or soluble uranium area	>10%	Daily 30-minute grab samples at ~30 lpm in worker occupied areas	3E -11 µCi/ml
Yellowcake	Other indoor process areas	>10% and <25%	Weekly 30-minute grab sample at ~ 30 lpm in worker occupied areas	3E -11 µCi/ml
Yellowcake	Other indoor process areas	<10%	Monthly 30-minute grab sample at ~30 lpm in worker occupied areas	3E -11 µCi/ml

(a) Derived air concentration (DAC). See Reg. Guide 8.25. See Procedure RH-300 for material specific DACs.
(b) Type and frequencies of sampling recommended in Reg. Guide 8.25 and 8.30. Alternatively, a grab sample may consist of a 5-minute high-volume sample at ~850 liters per minute (lpm), which is equivalent to 30 cubic feet per minute (cfm). Breathing zone air sampling is included in Table 1 to demonstrate when general air samples are not sufficient.
(c) LLD = Lower limit of detection, from Reg. Guide 8.30.

7.0 PROCEDURE

7.1 The RST is to collect the samples at locations and at the frequencies specified by the RSO or Assistant RSO. Sample the work area for a minimum of 30 minutes at approximately 30 liters per minute or equivalent volume using a breathing zone type air sampler.

7.2 The RST is to check the sampling equipment for visible damage and operational status before each use.

7.3 Calibration of the air samplers flow rate is required at least once per quarter. Verify that the calibration is current. A tag attached to the sampler notes when the latest calibration was performed.

7.3.1 If the calibration is out of date, remove the sampler from service and tag as out of use until recalibrated.

7.3.2 If calibration is current the instrument should be visually inspected for loose fittings and electrical connections. If this inspection reveals any obvious problems the sampler should be removed from service and repaired.

7.4 At the assigned sampling location install a 47 mm glass fiber filter or equivalent in the sampler filter holder. Assure that the filter is installed correctly to provide a good seal on the filter media.

7.5 Turn on the sampler and check for airflow being drawn in through the sample holder. Adjust the air sampler flow rate to the desired set point as indicated on the sampler rotometer. The set point value is noted on the calibration tag attached to the air sampler.

7.6 At the completion of sampling, recheck the sampler flow rate as indicated on the rotometer and record any difference from the initial setting. Turn off the sampler, remove the sampler filter, and place the filter in a petri dish or envelope.

7.7 The RST is to mark the top of the sample collection petri dish or envelope with:

- Location ID Number
- Date of collection
- Sample Submittal and Tracking Form (SSTF) number.

7.8 The RST is to fill in the following information on the General Air Sample Data Form:

- Sample Identification number
- Location
- Date and time of collection
- Sampling instrument number
- Sampler flow rate
- Sample collection minutes
- Total liters sampled
- Area operational status
- Status of emission control equipment
- Ventilation conditions at time of collection
- Sample Submittal and Tracking Form Number (SSTF).

7.9 After completing the day's sampling, the RST is to turn in samples to the Mill or contract laboratory for analysis of gross alpha and/or total U, Th-230, Ra-226, and Th-232 based on the requirements in Procedure RH-300.

7.10 Sample analysis procedures are specified in the Laboratory Procedures Manual.

7.11 The RSO or Assistant RSO is to review the measured concentrations, compare them to the DAC for specific areas and type of ore being processed (See Procedure RH-302), and assess the need for additional engineering controls to maintain radiation exposures to As Low As Reasonably Achievable (ALARA). This assessment will determine:

- The need for respiratory protection,
- The type of respirator for the area,
- The needs for breathing zone samplers, and
- Compliance with Radiation Work Permits.

7.12 The RST is to retain the results for a minimum of three years (6 CCR 1007-1 4.42.1).

Occupational General Air Particulate Survey Procedure RH-130

Appendix A

General Air Sampled Data Form

General Air Sample Data Form

RWP-_____ Sampler Name_____ Sampler ID_____

Employee Name_____ Employee ID_____

ID NO.	LAB	LOCATION	SAMPLE DATE	START TIME	FINISH TIME	TOTAL TIME (minutes)	AIR FLOW (lpm)	INST. USED	TOTAL VOL. (liters)	P	V	E
COMMENTS:												

P = Production area, V = Visible dust, E = Emission control equipment in operation

Form RH-130A

1.0 PURPOSE

This procedure is for the determination of airborne concentrations of Radon Decay Products (RDP) in work areas at the Piñon Ridge Mill to establish:

- Concentrations of radon progeny in air to which workers are potentially exposed,
- The need for respiratory protection and/or corrective action, and
- Trends in workers doses from RDP.

2.0 APPLICABILITY

This procedure applies to employees and contractors to Energy Fuels Resources at the Mill.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 NRC Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

3.1.2 Colorado Radiation Control Regulations Part 4 (6 CCR 1007-1)

3.1.3 Health Physics Vol. 50, No.3 1986. A simple Two-Count Method for Routine Monitoring of Rn222 and Rn220 Progeny Working Levels in U Mines.

3.1.4 Kunsetz, HL. Radon daughters in mine atmospheres: a field method for determining concentrations. Amer. Ind Hyg Assoc 17:85. 1956.

3.2 *APPENDICES*

Appendix A - Sampling Locations, Monthly & Quarterly

Appendix B - Time Factors for Radon Progeny Determination

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 Eberline RAS -1 air sampler or equivalent.

4.1.2 SKC Model PCXR4 or Gillian GilAir 5 air pumps or equivalent

4.1.3 Ludlum model 2929 alpha/beta sample counter or equivalent.

4.2 *MATERIALS*

4.2.1 RDP concentration worksheet.

4.2.2 Gelman Type A/E 25mm fiberglass filter for portable sampler or the equivalent.

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or the Assistant RSO is responsible for:

5.1.1 Determining the locations throughout the facility where samples should be

collected.

5.1.2 Determining the frequency of sampling at a given location based on previous results or by administrative controls.

5.1.3 Reviewing data from the RDP surveys, determining the needs for respiratory protection, and if necessary, taking any corrective actions.

5.2 The Radiation/Security Technician (RST) is responsible for:

5.2.1 Collecting the RDP air samples according to the schedule and frequency provided by the RSO or Assistant RSO.

5.2.2 Ensuring that the equipment used for this sampling is properly calibrated and maintained.

5.2.3 Following the procedures contained herein to accomplish the sampling.

5.2.4 Recording information from sampling to include:

- The operational status of the area being sampled.
- Location of sampling
- Sampling rate
- Sampling time
- Sample volume
- Sampling activity

5.2.5 Analyzing the samples for RDP concentrations and calculating the Working Level (WL).

5.2.6 Informing the RSO or Assistant RSO of any deviations or problems that were encountered during sampling activities.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 WL - Working Level

6.1.2 DAC - Derived Air Concentration

Rn-222 = 3.0 E-8 μCi/ml (0.33 WL)

Rn-220 = 9.0 E-9 μCi/ml (1.0 WL)

6.2 *FREQUENCY*

6.2.1 Weekly, monthly, or quarterly RDP surveys based on percent of the applicable derived air concentration (DAC).

- Concentration >25% DAC = Daily
- Concentration >10% and <25% DAC = Weekly
- Concentration <10% DAC = Monthly

6.2.2 Routine samples are collected at least monthly from locations where RDP concentrations have been most likely to occur (See Appendix A).

7.0 PROCEDURES

7.1 *RADON DECAY PRODUCT (RDP) SURVEYS*

7.1.1 Airborne concentrations are determined using the following methods:

7.1.1.1 The modified Kusnetz Method for determination of Rn-222,

7.1.1.2 The method specified in Health Physics Vol. 50, No.3 1986. "A simple Two-Count Method for Routine Monitoring of Rn222 and Rn220 Progeny Working Levels in U Mines" for the determination of Rn-222 or Rn-220,

7.1.2 The following applies to either method utilized.

7.1.2.1 The RSO or Assistant RSO is to determine the sampling locations, sampling frequencies and convey that information to the RST.

7.1.2.2 The RST is to check the sampling equipment for visible damage and operational status prior to use.

7.1.2.3 Verify that air sampling equipment and alpha counting instruments are properly calibrated and ready for use.

7.2 *MODIFIED KUSNETZ METHOD*

7.2.1 At the sampling area or location attach a sampling cassette with a 25 mm glass fiber filter onto the hose attached to the portable air sampling pump. Assure that the filter is installed correctly to provide a good seal on the filter media.

7.2.2 Remove the sampling cassette cover, and turn on the sampling pump.

7.2.3 Collect the air sample for a period of five minutes at a sample rate of 2 liters per minute (lpm).

7.2.4 Turn off the sampling pump and replace the cassette cover.

7.2.5 Record the following information:

- Sampling location or area
- Operational status at sampling location
- Start/Stop time
- Air sampling equipment used and sampling flow rate

7.2.6 When ready to count the sample filter, remove the sample filter from the sampling cassette and place the filter into the counting chamber of the alpha instrument detector.

7.2.7 The filters must be counted between 40 and 90 minutes after sample collection.

7.2.8 Count the sample filter for one minute and determine the alpha activity.

7.2.9 Calculate the disintegrations per minute per liter (dpm).

7.2.10 Select the appropriate time factor (see Appendix B) based on the elapsed time from the end of sample collection and calculate the WL concentration.

7.2.11 Record the following on the RDP survey form:

- Name of surveyor
- Sampling location or area
- Operational status at sampling location
- Start/Stop time
- Sampling minutes
- Sampler flow rate in lpm
- Air sampling equipment used
- Alpha counting instrument used
- Elapsed time from the end of sample collection to the count time.
- Alpha activity determined from sample in dpm

7.2.12 Calculated working level (WL) by:

$$\text{Working Levels} = \frac{(\text{CPM})(\text{efficiency factor})}{(\text{volume sampled in liters})(\text{time factor})}$$

$$\text{Efficiency factor} = \frac{\text{Calibration source DPM}}{\text{Calibration source CPM}}$$

CPM – counts per minute

Time factors are listed in Appendix B.

7.2.13 Calculate the sample's percentage of the DAC by dividing the sample concentration by the appropriate DAC and multiplying by 100.

7.3 *TWO-COUNT METHOD*

7.3.1 Collect the sample the same as specified in section 7.1 above, except with the following changes:

7.3.2 Set the air sampler to a flow rate of 3 lpm

7.3.3 Collect the sample for a period of 10 minutes.

7.3.4 For Rn-222 determination, alpha count the sample from 2 to 7 minutes (5 minutes total) after sample collection and apply a weighting factor of 235.

7.3.5 For Rn-220 determination, alpha count the sample from 340 – 350 minutes (10 minutes total) after sample collection and apply a weighting factor of 13.

Radon-222/Radon-220 Decay Product Surveys Procedure RH-140

Appendix A

Sampling Locations, Monthly and Quarterly

Insert Sampling Location Map here when available

Radon-222/Radon-220 Decay Product Surveys Procedure RH-140

Appendix B

Time Factors for Radon Progeny Determinations

Time Factors For Radon Progeny Determinations

(5 Min. Sample and 1 Min. Count)

Elapsed Time (Minutes)	Time Factor	Elapsed Time (Minutes)	Time Factor
40	150	66	98
41	148	67	96
42	146	68	94
43	144	69	92
44	142	70	90
45	140	71	89
46	138	72	87
47	136	73	85
48	134	74	84
49	132	75	83
50	130	76	82
51	128	77	81
52	126	78	78
53	124	79	76
54	122	80	75
55	120	81	74
56	118	82	73
57	116	83	71
58	114	84	69
59	112	85	68
60	110	86	66
61	108	87	65
62	106	88	63
63	104	89	61
64	102	90	60
65	100		

1.0 PURPOSE

This procedure is for the measurement of radioactive airborne particulates concentrations in the breathing zone of employees and contractors at the Piñon Ridge Mill to establish:

- Concentrations of radon progeny in air to which workers are potentially exposed,
- The need for respiratory protection, and
- Trends in workers' radiation doses from airborne radionuclides.

2.0 APPLICABILITY

Breathing zone air particulate samplers are used in worker occupied areas where an employee is potentially exposed to airborne concentrations of radon progeny in excess of 10% of the DAC or 0.03 working levels.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 U.S. Nuclear Regulatory Commission Regulatory Guide 8.25, Air Sampling in the Workplace.

3.1.2 U.S. Nuclear Regulatory Commission Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

3.1.3 Colorado Radiation Control Regulations Part 4 (6 CCR 1007-1)

3.2 *APPENDICES*

Appendix A - Breathing Zone Issue/Collection Form RH-300A

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 SKC Model PCXR4 or Gillian GilAir 5 air pumps or equivalent.

4.1.2 Ludlum model 2929 alpha/beta sample counter or equivalent.

4.1.3 Cleaning containers, brushes.

4.2 *MATERIALS*

4.2.1 37 mm AA membrane filter paper/support pads, or the equivalent.

4.2.2 Breathing Zone Issue/Collection Form, or the equivalent.

4.2.3 Breathing Zone Cassette label, or the equivalent.

4.2.4 Soap solution, acidic cleaning solution (e.g. EDTA), or the equivalent.

4.2.5 Dilute HCl solution.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.0 RESPONSIBILITIES

5.1 The Radiation Safety Officer (RSO) or the Assistant RSO is responsible for:

5.1.1 Identifying and assigning the locations or job categories for sampler use and frequency of sampling to the Radiation Security Technicians (RST).

5.1.1 Training the RSTs on occupational air sampling.

5.1.3 Reviewing data from the breathing zone samples, determining the needs for respiratory protection, and if necessary, taking corrective actions.

5.1.4 Insuring that non-routine monitoring, such as during maintenance repair operations, are accomplished under a Radiation Work Permit (RWP).

5.2 The Radiation Security Technician (RST) or designee is responsible for:

5.2.1 Collecting the breathing zone samples according to the schedule and frequency provided by the RSO or Assistant RSO.

5.2.2 Insuring that the equipment used for sampling is properly calibrated and maintained.

5.2.3 Following this procedure.

5.2.4 Informing the RSO or Assistant RSO of any deviations or problems that were encountered during sampling activities.

5.2.5 Laboratory analyses of the breathing zone samples.

5.2.6 Retaining forms for 3 years.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 **Breathing Zone Air Survey** - Monitoring of airborne radioactive particles generally within 1 foot of the individual's head (Regulation Guide 8.25, pages 4-5) using a belt worn, battery operated air pump connected to a filter and filter head usually clipped on the workers' lapel.

6.1.2 **DAC** - Derived Air Concentration.

6.2 *SAFETY*

6.2.1 Care should be taken not to catch the tygon tube, from the belt-worn air pump to the lapel filter holder, on process equipment.

6.3 *FREQUENCY*

6.3.1 Continuous breathing zone air samplers will be required to be worn when exposure to radon progeny is or could be ≥ 10% of the DAC or 0.03 WL or as directed by the RSO or Assistant RSO.

7.0 PROCEDURE

7.1 The RSO or Assistant RSO is to determine the locations, sampling frequencies and parameters to be analyzed. He is to convey that information to the RSTs.

7.2 The RST is to check the sampling equipment for visible damage and operational status prior to use. Before issuing a breathing zone sampler, the RST is to perform the following operational checks:

7.2.1 Calibration of the breathing zone sampler flow rate is required at least

once per quarter. Verify that the calibration of the breathing zone sampler is current. A tag attached to the sampler notes when the latest calibration was performed.

7.2.2 If the calibration date of the sampler is not current, remove the sampler from service for calibration and notify the Safety Department.

7.2.3 Select a different breathing zone sampler if the sampler calibration is not current.

7.3.4 Check the battery pack of the pump unit for charge level.

7.2.5 Briefly turn on the sampler to verify operation. If the sampler fails to run correctly, select a different sampler for issue and notify the Safety Department.

7.2.6 Check the sample line from the pump to filter holder for splits or cracks, replacing if warranted.

7.2.7 Check filter gasket, replacing if worn or visible defects are observed.

7.2.8 Prior to installing the filter, check for sample line leaks by plugging the sample line at the filter end. Air flow through the pump should be nil.

7.2.9 When the above operational checks are completed and the sampler is deemed operational for use, it is ready for issue.

7.3 The RST is to install an assembled filter cassette. If monitoring for more than one day or shift, reinstall the appropriate filter cassette for that person.

7.3.1 New cassettes are received from the supplier preloaded.

7.3.2 Previously used cassettes are cleaned and decontaminated, allowed to dry and then reloaded with a new support pad, filter and then reassembled for use as follows:

7.3.2.1 Remove any remaining labels, tape, etc. from the cassettes and check for damage.

7.3.2.2 Discard cassettes that have cracks, splits etc.

7.3.2.3 Separate the inlet and outlet sections of the cassettes and place in to cleaning containers.

7.3.2.4 Add tap water, cleaning soap such as Alconox, and EDTA. Make sure all cassettes are covered with solution, and stir occasionally. Soak for approximately 8 hours.

7.3.2.5 Scrub cassettes using a stiff bristle brush to remove any remaining residue, labels, tape etc.

7.3.2.6 Place the soap-cleaned and scrubbed cassettes into a container and rinse twice with tap water and drain.

7.3.2.7 Refill containers with tap water and add 10-20 ml of HCl acid. Add HCl to water not water to HCL.

7.3.2.8 Soak cassettes in HCL solution for approximately 4 hours.

7.3.2.9 Drain HCL solution from containers and thoroughly rinse twice with tap water. Drain.

7.3.2.10 Perform final rinse of cassettes using reagent grade water and

drain.

7.3.2.11 Remove cassettes from containers and place on clean tray lined with paper towels making sure to place the cassettes open end face down.

7.3.2.12 Place paper towels between layers of cassette as needed.

7.3.2.13 After the cassettes have thoroughly dried, install a new filter support pad and membrane filter on top of pad using tweezers or forceps.

7.3.2.14 Reassemble cassettes and seal the inlet and outlet opening of the cassettes with plugs or electrical tape.

7.3.2.15 Check to make sure cassette sections are seated together. A small hammer and socket can be used to seat the two sections.

7.3.2.16 Try twisting the two sections. If they are loose, seal the two section halves using electrical tape.

7.3.2.17 Place reassembled cassette into storage box for later use.

7.4 The RST is to attach the breathing zone filter cassette on to the sampling hose attached to the pump.

7.4.1 Verify that the sampler filter cassette is installed correctly.

7.4.1.1 The breathing zone cassette is two or three pieces with one side marked as the inlet. The side of the cassette marked inlet is the particulate collection side. Attach the pump sampling line to the opposite side of the cassette, which looks like a wagon wheel.

7.5 The RST is to remove the cassette inlet plug.

7.6 The RST is to turn on the sampler.

7.7 The RST is to check and adjust the sampler flow rate to the value noted on the calibration sticker attached to the sampler.

7.8 The employee is to be issued the breathing zone sampler prior to reporting to their assigned work area.

7.9 The RST is to attach the sampling line with cassette installed on to the employee's shirt collar etc. or other locations near the workers breathing zone (1 foot from worker's head).

7.9.1 When coveralls, wet/chemical suits or coats, etc., are worn make certain the breathing zone cassette is attached to the outside of the outer clothing.

7.9.2 The cassette should be attached to the wearer so the particulate inlet to the cassette is facing downward.

7.10 The RST is to record data needed on the Breathing Zone Issue/Collection Form Sheet prior to each time a sampler is issued. The items are to include.

7.10.1 Breathing Zone Sample Identification Number

7.10.2 Start date of sampler

7.10.3 Start time of sampler

7.10.4 Air Sampler used

7.10.5 Name of employee being sampled

7.10.6 Employee number

7.10.7 Routine or special sample

7.10.8 When the cassette is to be worn for more than one day, the information is to be recorded on the same sheet for that person for that sample cassette

7.11 The employee is to wear the sampler for the intended amount of time with it in operation.

7.11.1 If the sampler shuts down or flow faults during the wearing period, the employee is to contact the RST who is to:

7.11.1.1 Ask the wearer the estimated time the sampler faulted.

7.11.1.2 Record all information on the Breathing Zone Issue/ Collection Form.

7.11.1.3 Restart the breathing zone sampler and document restart time on the Breathing Zone Issue/Collection Form.

7.12 The employee, after the intended amount of time or end of shift with the breathing zone sampler in operation, is to return to the location where the sampler was initially issued to have the sampler collected by the RST.

7.13 The person collecting the samplers is to record the following information on the Breathing Zone Issue/Collection Form.

7.13.1 If the indicated flow rate on the sampler is different than that noted on the calibration sticker, record the rate at which the sampler is presently operating.

7.13.2 Record stop date and time.

7.13.3 Record sampler elapsed timer reading.

7.13.4 Turn off the sampler.

7.13.5 Reinstall the cassette inlet plug and carefully remove the breathing zone cassette from the pump sample line if the sampler is not going to be reissued to the same employee for use during their next work shift.

7.13.6 Ask the wearer of the sampler the following questions and record their answer on the Breathing Zone Issue/Collection Form.

7.13.6.1 Was respiratory protection worn? If so how long was the respiratory protection worn? Record what type of mask was used.

7.13.6.2 Were other workers performing the same task that did not have a breathing zone sampler?

7.13.6.3 Were there any unusual circumstances encountered when wearing the sampler?

7.14 The person who was issued and wearing the breathing zone sampler must sign the Breathing Zone Issue/Collection Form verifying the information provided was accurate.

7.15 The RST receiving the breathing zone sampler is to proceed with the following:

7.15.1 Visually inspect the filter inside the breathing zone cassette. Note on the Breathing Zone Issue/Collection form the appearance of the filter (clean, discolored, etc).

7.15.2 Store the cassette so the particulate inlet side of the sample cassette is facing upwards.

7.15.3 Inspect the sample pump and hose for any visible damage. If damage is noted, document it on the Breathing Zone Issue/Collection Form, determine how the damage occurred, and replace the damaged part.

7.15.4 Place the breathing zone sampling pump on to the battery charger or replace the discharged battery with a charged one.

7.16 The RST retrieving the breathing zone sample filters that need to be analyzed is to:

7.16.1 Maintain custody of the filters that are being sent to the laboratory for analysis.

7.16.2 Label a filter-transfer envelope to identify the sample. Place the samples in the filter-transfer envelop and transfer to the laboratory for analysis.

7.17 The RSO is to review the results of the laboratory analyses for accuracy and completeness. The RSO is to sign the reports to signify his review and understanding of the report and file the report.

Occupational Breathing Zone Monitoring Procedure RH-150

Appendix A

Breathing Zone Sample Issue/Return Report Form

Breathing Zone Sample Issue/Return Report Form

Name_____ Employee #_____ Dept./Company_____ Routine_____ Special_____

Pump #	Issue Date	Time On	Return Date	Time Off	Timer Reading	Minutes Per On/Off Time	Minutes used for Calculations	Filter Inspection Comments	Respirator Type and Total Time Worn	Work Area Code or Task	Unusual Circumstances	Number of Other Workers Without Breathing Zone Monitor Performing Same Task	Wearer Initials

RADIATION SAFETY DEPARTMENT USE ONLY

BZ#	Sample Submittal #	Lab #	Flow Rate (lpm)	Total Minutes	Total Volume liters	Remarks	Radiation Tech.'s Initials

Respirator Type:
HF: Half Face FF: Full Face AL: AirLine RH: Racal Helmet RFF: Racal Full Face

Reviewed by:_____ Date:_____

Form RH-300A

1.0 PURPOSE

This procedure provides guidance and instruction for the bubble calibration of personal air samplers at the Piñon Ridge Mill.

2.0 APPLICABILITY

This procedure applies to breathing zone personal air samplers (PAS).

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Breathing Zone Particulate Air Sampler (PAS) Manufacturer's Operations Manual.

3.1.2 Colorado Radiation Control Regulations (6 CCR 1007-1).

3.2 *APPENDICES*

Appendix A - Air Sampler Calibration Form RH-151A

4.0 EQUIPMENT AND MATERIALS

4.1 *EQUIPMENT*

4.1.1 1 liter glass burette

4.1.2 Support base and clamps for burette

4.1.3 Sealable jar with two connections through the lid for tubing

4.1.4 Beaker or other container that will fit over the larger end of the burette

4.1.5 Stopwatch

4.1.6 Calculator

4.1.7 Small screw driver

4.1.8 Sample pump – battery fully charged

4.1.9 Cassette with filter installed

4.1.10 Tubing

4.1.11 Calibration jar with lid

4.2 *MATERIALS*

4.2.1 Soap solution

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or Assistant RSO is responsible for:

5.1.1 Assuring that all PAS's are calibrated.

5.2 The Radiation/Security Technician (RST) is responsible for:

5.2.1 Calibrating and recording calibration data according to this procedure.

5.2.2 Retaining Form RH-151A on record.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 PAS - Personal (breathing zone) Air Sampler.

6.2 *SAFETY*

6.2.1 Care should be taken to avoid breaking glass.

6.3 *FREQUENCY*

6.3.1 PAS's that are in use will be calibrated at least semiannually and after repairs.

7.0 PROCEDURE

7.1 *BUBBLE CALIBRATION METHOD* - Assembly calibration equipment.

7.1.1 Make a soap solution by adding a few drops of liquid soap in 200 ml of tap water in a 250 ml beaker or similar container and stir.

7.1.2 Wet the inside of the burette with the soap solution and secure to the support base with clamps, placing the larger opening of the burette towards the bottom of the support.

7.1.3 Using tubing connect:

- The sampling pump to one of two tube fittings on the calibration jar lid.
- The other tube fitting on the jar lid to the small opening in the burette.

7.1.4 Remove the calibration jar lid and attach a filter cassette with filter to the tube fitting that is connected to the sampling pump. Secure the lid to the jar.

7.1.5 Make sure that the burette is perpendicular to the table.

7.1.6 Turn on sampling pump and check for leakage. Correct leakage if present.

7.1.7 Raise the container containing the soap solution and briefly submerge the larger opening of the burette to generate a bubble inside the burette.

7.1.8 Repeat several times thoroughly wetting the inside walls of the burette. Continue until a bubble travels the entire length of the burette without breaking.

7.1.9 Adjust the sampling pump flow to the rotameter set point for 2 liters per minute (lpm) or other sampling rate designated by the RSO or Assistant RSO.

7.1.10 Observe the bubble moving up the burette.

7.1.11 Start the stopwatch when the bubble reaches the zero mark on the burette.

7.1.12 Stop the stopwatch when the same bubble reaches the one-liter mark on the burette.

7.1.13 Measure the elapsed time the bubble traveled one liter.

7.1.14 Calculate flow rate from the sample pump using the following equation:

$$FlowRate\,(lpm) = \frac{BuretteVolume(liters)}{Elapsedtime(\sec)}(x60)$$

7.1.15 Record the flow rate on Form RH-151A or the equivalent. Repeat steps 7.1.9 thru 7.1.14.

7.1.16 Average the two calibration runs.

7.1.17 Compare the average to 2 lpm or other set point.

7.1.18 If the average is within 2 lpm±10% (1.8 lpm – 2.2 lpm), no further calibration checks are needed. Complete the top portion of the Air Sampler Calibration Form RH-151A.

- Apply a new calibration sticker to the pump specifying the calibration date, the date next calibration is due, the rotameter reading corresponding to 2 lpm, the sampler serial number and/or the sampler number, and the calibration technician's initials.
- Mark on the pump with tape or grease pencil the rotameter reading corresponding to 2 lpm. Return sampling pump for use.

7.1.19 If the calibration average does not fall within 2 lpm ± 10%, adjust the sampler flow volume adjustment screw using the small screwdriver to more accurately pump 2 lpm.

7.1.20 Repeat steps 7.1.1 thru 7.1.19. until sample pump falls within the acceptable range.

- Document rotameter reading on the new calibration form.
- Apply a new calibration sticker to the sample pump.
- Return sampler for use.

7.1.21 If the sampling pump cannot be calibrated to within 2 lpm ± 10% or other set point ± 10%, repair or remove pump from service.

7.1.22 The Radiation Safety Technician is to file and retain for at least 3 years Form RH-151A.

Calibration of Air Samplers Using the Bubble Method RH-151

Appendix A

Air Sampler Calibration Form

Air Sampler Calibration Form

Date: _____ Signature:_____ Type of air sampler:_____

Sampler #	**BZ or GA Sampler**	**As Found Reading Using Previous Set Point**				**New Set Point**			
ID #	Serial #	Indicated Flow (from Rotometer)	Run 1 Flow Rate (lpm)	Run 2 Flow Rate (lpm)	Average of calibration runs (lpm)	Indicated Flow (from Rotometer)	Run 1 Flow Rate (lpm)	Run 2 Flow Rate (lpm)	Average of calibration runs (lpm)

Date	Temp. °F	BP (in. Hg)	Flow rate correction factor formula: F= √((P2/P3)*(T3)(T2)) =
			Actual lpm flow rate = (Std. lpm flow rate)(F) =
			P2= Std. BP, 29.92 in. Hg; P3= Average BP (in. Hg); T2= Std. Temp 298 ^{o} K; T3= Average Temp. =^{o} C + 273. = [(^{o}F-32)/1.8] + 273.

The calibration method used: ( ) 1 liter bubble tube method ( ) KURZ 544 SN# 518 ( ) IHIP300 # 1516

Indicated readings are based on the location of the middle of the flow ball unless noted otherwise. Readings are liters per minute (lpm).
Note: A flow rate correction from lpm to Actual lpm is required when using the KURZ 544 or IHIP 300 calibrator units.
BP – Barometric Pressure

1.0 PURPOSE

This procedure describes how to leak test, shutter test and inventory nuclear density and nuclear level gauges.

2.0 APPLICABILITY

This procedure applies to nuclear density gauges and nuclear level indicators used at the Piñon Ridge Mill.

3.0 OTHER DOCUMENTS

3.1 REFERENCES

3.1.1 Colorado Radiation Control Regulations Part 4 (6 CCR 1007-1).

3.2 APPENDICES

Appendix A - Leak Test Form RH-160A

4.0 EQUIPMENT AND MATERIALS

4.1 EQUIPMENT

4.1.1 Ludlum 44-9 pancake GM probe or equivalent

4.1.2 Ludlum model 2241 series scaler/ratemeter or equivalent

4.1.3 Cs -137 check source or equivalent

4.2 MATERIALS

4.1.1 Leak Test Survey Form

4.1.2 Survey swipes

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or the Assistant RSO is responsible for:

5.1.1 Scheduling the leak tests and shutter tests.

5.1.2 If the source is leaking:

5.1.2.1 Contacting the Colorado Department of Public Health and Environment (CDPHE).

5.1.2.2 Contacting the manufacturer, vendor, or the equivalent for repairs, removal or replacement.

5.2 The Radiation/Security Technician (RST) or his designee is responsible for:

5.2.1 Conducting the leak tests and shutter tests according to this procedure.

5.2.2 Documenting the results of the test.

5.2.3 Keeping an inventory of the sealed sources.

5.2.4 Retaining the records of the tests for a minimum of three years. (6 CCR 1007-1 4.42.1)

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

6.0 PREREQUISITE INFORMATION

6.1 *SAFETY*

6.1.1 When accessing the source on a ladder take precautions not to fall (see Procedure HS-100 Ladders and Scaffolding).

6.2 *FREQUENCY*

6.2.1 Every three years, conduct a shutter mechanism test on each source that is in use.

6.2.2 Every three years, make a physical inventory of each fixed source.

6.2.3 Every three years, leak test each fixed source.

7.0 PROCEDURES

7.1 If the source remains in the location where it was previously used (e.g. on the same pipe or tank) and has the shutter mechanism locked in the closed position, the source is considered "not in use" and a shutter test is not required until the source is placed back in service. Otherwise open and close the shutter. If the shutter opens and closes without sticking, document on the leak test form that the shutter is operating properly.

7.2 Conduct a physical inventory of the fixed gauges by documenting on the leak test form the precise location of the device, serial number, and the condition of the source.

7.3 Conduct a leak test on each sealed source whether in use or not.

7.3.1 MAKE SURE THE SOURCE IS OFF (SHUTTER CLOSED).

7.3.2 Record the serial number of the source on the leak test form and number each wipe.

7.3.3 Wearing latex rubber gloves or the equivalent use a piece of filter paper or the equivalent to wipe the area of each source where activity from a leaking source may be suspected. Wipe the areas with moderate pressure.

7.3.4 Fold the filter paper wipe and place it in an envelope for transport to the counting room.

7.3.5 Place a calibrated Cs-137 source of approximately 0.005 μCi in the Harshaw shielded sample changer or the equivalent and measure the activity with a gamma scintillation detector. Measure the background with an unused wipe in the sample chamber for 1 minute. Calculate the efficiency by:

$$E = \frac{\text{Observed CPM} - \text{background CPM}}{\text{DPM of source}}$$

7.3.6 Place the sealed source wipe in the sample chamber, count the activity for 1 minute and calculate the activity on the wipe as:

$$\text{Activity on Wipe in } \mu\text{Ci} = \frac{\text{Wipe CPM-background CPM}}{(E)(2.22E6 \text{ DPM/ } \mu\text{Ci})}$$

7.3.7 Record the activity of the wipe in μCi on the Leak Test Form (Appendix A Form RH-160A or the equivalent).

7.3.8 ***IF THE ACTIVITY ON THE WIPE IS 0.005 µCi OR MORE, THE SOURCE IS LEAKING:***

7.3.8.1 **IMMEDIATELY WITHDRAW THE SOURCE FROM USE.**

7.3.8.2 Secure immediate area to limit access to prevent spread of contamination from the source.

7.3.8.3 Decontaminate the area around the source as needed to prevent employee exposures.

7.3.8.4 Contact the manufacturer, vendor or service organization authorized by the CDPHE to repair, replace, or remove the source. Energy Fuels personnel are not authorized to perform these functions.

7.3.8.5 Within 5 days file a report with the CDPHE describing the equipment involved, the test results and the corrective action taken.

7.3.9 If the activity on the wipe is less than 0.005 µCi, no action is needed.

7.3.10 The RST is to retain the leak test form.

Source Leak Test, Shutter Test and Inventory Procedure RH-160

Appendix A

Leak Test Form 160A

Leak Test Form

Source Location:_____ Source SN:_____

Radionuclide _____Model #_____Activity: _____mCi Wipe #._____

Leak Test Survey Instrument:

Scaler Manufacturer_____Model_____,Ser. # _____

Probe Manufacturer_____Model_____,Ser. # _____

Calibration By_____Date_____

Cs-137 Ser. # _____µCi_____efficiency_____%

Test Results: (_____cpm Wipe) - (_____cpm Bkg) (% eff)(___µCi____) =_____µCi
2.22E6 dpm

[ ] Wipe is less than the 1465 Net cpm action level.
Less than 0.005 µCi removable activity.
Source is not leaking. No Action Required.

[ ] Wipe is greater than the 1465 Net cpm action level.

(_____Net cpm)((____µCi____)) = _____µCi removable activity
2.930 E5 cpm

Source is leaking. Immediately withdraw the source from use.
Within five (5) days after obtaining the results of the leak test, file a report with the CDPHE to describe the equipment involved, the test results, and the corrective action taken. If personnel have been contaminated, contact CDPHE. Contact the manufacturer of the source for instructions on repairs and replacement.

Source shutter mechanism turned source on and off successfully __Yes __ No

Next test due: _____

Leak test performed by _____

1.0 PURPOSE

This procedure specifies how to install nuclear density and nuclear level gauges.

2.0 APPLICABILITY

This procedure applies to individuals who have received specialty training in the removal and installation of nuclear level gauges and have been approved by the Colorado Department of Public Health and Environment (CDPHE) to install the nuclear devices.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 NUREG 1556 Vol. 4.

4.0 RESPONSIBILITY

4.1 The Radiation Safety Officer (RSO) is responsible for:

4.1.1 Making sure the nuclear density gauges are only installed or removed by personnel that have received training in their installation and removal and have been approved by the CDPHE.

5.0 PROCEDURES

5.1 Nuclear density gauges or nuclear level gauges are to be installed on and/or removed from Mill process equipment ONLY by persons specifically trained and authorized by the CDPHE to perform those operations pursuant to the current Colorado Radioactive Materials License.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

1.0 PURPOSE

This procedure describes how to conduct a contamination survey of personnel working in and around the Piñon Ridge Mill. The purpose of the survey is to minimize the potential for ingestion of radioactive materials while personnel eat and drink at work and to minimize contamination on individuals leaving the Mill to limits that are as low as reasonably achievable (ALARA). Guidelines on how to decontaminate personnel and clothing are listed in Procedure RH-020.

2.0 APPLICABILITY

This procedure applies to all personnel leaving the restricted area of the Mill.

3.0 OTHER DOCUMENTS

3.1 REFERENCES

3.1.1 NRC Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

3.1.2 NUREG 1507 Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminates and Field Conditions.

3.2 APPENDICES

Appendix A – Self-Survey Form RH-200A

Appendix B – Quarterly Random Spot Check Form RH-200B

4.0 EQUIPMENT AND MATERIALS

4.1 EQUIPMENT

4.1.1 Ludlum model 43 series alpha detectors (scintillation, proportional or phoswich) or equivalent

4.1.2 Ludlum model 2241 series scaler/ratemeter or equivalent

4.1.3 Ludlum model 2929 alpha/beta sample counter or equivalent

4.1.4 Ludlum 44-9 pancake GM probe or equivalent

4.1.5 Ludlum model 12s micro R meter or equivalent

4.1.6 Ludlum model 44 -10 gamma scintillation probe, 2 x 2 or equivalent

4.1.7 NIST alpha and beta check sources for determining instrument and detector efficiencies.

4.2 MATERIALS

4.2.1 NIST alpha and/or beta check sources for determining the instrument detector efficiency

4.2.2 Contamination swipes

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or Assistant RSO is responsible for:

5.1.1 Assuring that all personnel performing the surveys are properly trained.

5.1.2 Determining the frequency of the personnel surveys.

5.1.3 Determining the locations where surveys are to be conducted.

5.1.4 Reviewing the survey results, investigating contamination results that exceed the appropriate action limits or release criteria, and determining any corrective actions that should be put into affect.

5.2 The Radiation/Security Technician (RST) or designee is responsible for:

5.2.1 Following the requirements of this procedure.

5.2.2 Assuring that contamination survey instruments are functional, properly calibrated and available for use at the self-survey locations designated by the RSO.

5.2.3 Conducting or assisting with the contamination surveys of personnel and assisting with any decontamination that may be needed.

5.2.4 Filing and retaining the personnel contamination survey forms.

5.3 Each Individual is responsible for:

5.3.1 Surveying themselves and/or requesting assistance for a survey, comparing their results to the instrument background reading or other release action limits specified by the RSO, performing decontamination if the action levels or release limits are exceeded, and documenting the results of the survey.

5.3.2 Notifying the RSO, Assistant RSO, RST, or designee if survey assistance is needed and if any problems are encountered.

6.0 PREREQUISITE INFORMATION

6.1 DEFINITIONS

6.1.1 dpm - Disintegrations per minute.

6.1.2 cpm - Counts per minute observed on the survey meter.

6.2 FREQUENCY

6.2.1 Each time a person exits the restricted area at the Mill.

6.2.2 Additional survey frequencies at specified by RSO. These locations may include:

- Before entry to a designated lunchroom.
- As required by a Radiation Work Permit (RWP) or work plan.

6.2.3 Quarterly spot surveys for contamination by the RST or his designee on 10% of the workers leaving the facility.

7.0 PROCEDURE

7.1 TOTAL ALPHA CONTAMINATION SURVEYS

7.1.1 The RST or designee will check the operation of the instrument(s) selected for monitoring. Refer to the instruments operational manual for specific information.

7.1.2 Turn on the alpha survey meter. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.1.3 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-200A or B, or the equivalent, as applicable.

7.1.4 Inspect the survey meter for damage. If an alpha scintillation probe is to be used in the survey, hold the probe towards a light source. If the count rate suddenly increases, look for tiny pinholes in the mylar surface of the probe face. Repairs can be made with "whiteout." If two or more whiteout spots are present, consider using another instrument. Repair damage or use a functional instrument.

7.1.5 Observe the instrument alpha background count rate:

7.1.5.1 For a portable instrument, place the detector face down on a clean uncontaminated surface and observe the meter response. If more than five to ten counts per minute, use a different survey meter and report the potential contamination and/or malfunction to the RSO or the Assistant RSO.

7.1.5.2 Record the background rate observed on the instrument(s) on Form RH-200A, B or C.

7.1.6 Position the probe face of the survey meter within 1.0 cm of the Th-230 check source or the equivalent alpha check source. Use the calibration jig or the end of one's finger under the edge of the alpha probe as a guide for the 1.0 cm spacing. Use a source-to-detector distance for the calibration that is similar to the expected surface-to-detector spacing used in the field. Record the reading on Form RH-200A, B or C. If the reading is within 20% (Reg. Guide 8.30, page 21) of the historic reference readings, conclude that the instrument is functioning properly. If not, use an instrument that is functioning properly.

7.1.7 Determine the instrument Minimum Detectable Activity (MDA) and record on calibration form.

7.1.7.1 For an integrated measurement over a preset time, the MDA can be approximated by:

$$MDA = \frac{2.71 + 4.65 \sqrt{[(B_R)(t)]}}{(t)(E)(A/100)}$$

7.1.7.2 For a ratemeter instrument the MDA can be approximated by taking twice the time constant of the meter as the counting time:

$$MDA = \frac{4.65 \sqrt{[(B_R)/(2t_c)]}}{(E)(A/100)}$$

Where:

MDA = activity in dpm/100 cm^2

B_R = background cpm

t = counting time in minutes

t_c = meter time constant in minutes

E = instrument detector efficiency

A = probe area in cm^2

7.1.8 Calculate the portable survey instrument correction factor by:

$$\text{Instrument Correction factor} = \frac{(\text{Source dpm})(100\ cm^2)}{(\text{meter cpm} - \text{bkg cpm})(X\ cm^2)}$$

Where X is the detection surface area of the probe with no corrections made for the protective screen and 100 cm^2 is the area specified in the release limits. The "source dpm/meter cpm" measurement corrects for the screen covering part of the detection area of the probe (NUREG-1507, page 5-7).

7.1.9 If the surface to be monitored is wet, let the surface dry or dry the surface prior to conducting the alpha survey.

7.1.10 Position the probe face of the survey meter within 1.0 cm of the surface being monitored. Use the end of one's finger under the edge of the alpha probe as a guide to the 1.0 cm spacing. Slowly (~1 cm/second) move the probe over each hand, each foot and over other potentially contaminated areas of clothing, stopping at areas of suspect contamination to obtain an accurate determination of the level of contamination.

- Minimize touching the probe face with the surface being surveyed to avoid the possibility contaminating the probe face.
- Note increases in the count rates as indicted by the instruments audible output.

7.1.11 Subtract the background cpm reading from the cpm reading observed during the survey and multiply by the instrument correction factor. Record the readings obtained on Form RH-200A, B or C or the equivalent and compare to the release limits or action levels.

7.1.12 If the survey measurements exceed the instruments background reading or other appropriate release limits specified by the RSO, perform decontamination following the guidelines listed in Procedure RH-020, Table 1.

7.1.13 After decontamination efforts, resurvey and document results on Form RH-200A, B or C.

7.1.14 In the event that decontamination is not successful, notify the RSO or Assistant RSO for assistance.

7.1.15 Document the following data on Form RH-200A, B or C or its equivalent:

- Name of person being surveyed
- Technician conducting the survey (if not self-survey)
- Location where survey is conducted
- Date and Time of survey
- Results of survey
- Survey instrument serial number for both the alpha and beta-gamma surveys

7.2 TOTAL BETA-GAMMA CONTAMINATION SURVEYS

7.2.1 Turn on the instrument equipped with a GM pancake probe or an equivalent instrument. Use the battery scale on the meter to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.2.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-200A, B or C or the equivalent.

7.2.3 Inspect the meter and probe for damage. If damaged, use another instrument.

7.2.4 Observe the instrument background count rate and record on Form RH-200A, B or C.

7.2.5 Position the probe face of the survey meter within 1.0 cm of the Cl-36 check source or the equivalent check source. Use a calibration jig or the end of one's finger under the edge of the probe as a guide for the 1.0 cm spacing. Use a source-to-detector distance for the calibration that is similar to the expected surface-to-detector spacing used in the field. Record the reading on Form RH-200A. If the reading is within 20% (Reg. Guide 8.30, page 21) of the historic reference readings conclude that the instrument is functioning properly. If not, use an instrument that is functioning properly.

7.2.6 Determine the instrument minimum detectable activity (MDA) and record on calibration form.

7.2.6.1 For an integrated measurement over a preset time, the MDA can be approximated by:

$$MDA = \frac{2.71 + 4.65 \sqrt{[(B_R)(t)]}}{(t)(E)(A/100)}$$

7.2.6.2 For a ratemeter instrument the MDA can be approximated by taking twice the time constant of the meter as the counting time:

$$MDA = \frac{4.65 \sqrt{[(B_R)/(2t_c)]}}{(E)(A/100)}$$

Where:

MDA = activity in dpm/100 cm^2

B_R = background cpm

t = counting time in minutes

t_c = meter time constant in minutes

E = instrument detector efficiency

A = probe area in cm^2

7.2.7 Calculate the portable survey instrument correction factor by:

$$\text{Instrument Correction factor} = \frac{(\text{Source dpm})(100\ \text{cm2})}{(\text{meter cpm} - \text{bkg cpm})(X\ \text{cm2})}$$

Where X is the detection surface area of the probe with no corrections made for the protective screen and 100 cm^2 is the area specified in the release limits. The "source dpm/meter cpm" measurement corrects for the screen covering part of the detection area of the probe (NUREG-1507, page 5-7).

7.2.8 Position the probe face of the survey meter within 1.0 cm of the surface being monitored. (If alpha contamination is expected, a piece of paper can be placed between the probe and the surface being monitored to shield out the alpha particles). Use the end of one's finger under the edge of the probe as a guide to the 1.0 cm spacing. Slowly (~1 cm/second) move the probe over each hand, each foot and over other potentially contaminated areas of clothing, stopping at areas of suspect contamination to obtain an accurate determination of the level of contamination.

- Minimize touching the probe face with the surface being surveyed to avoid damaging the probe and possibility contaminating the probe face.
- Note increases in the count rates as indicted by the instruments audible output.

7.2.9 Subtract the background cpm reading from the cpm reading observed during the survey and multiply by the instrument correction factor. Record the readings obtained on Form RH-200A, B or C or the equivalent and

compare to the release limits or action levels.

7.2.10 If the survey measurements exceed the instrument background reading or other appropriate release limits specified by the RSO perform decontamination following the guidelines listed in Procedure RH-020, Table 1.

7.2.11 After decontamination efforts, resurvey and document results on Form RH-200A, B or C.

7.2.12 In the event that decontamination is not successful, notify the RSO or Assistant RSO for assistance.

7.2.13 Document the following data on Form RH-200A, B or C or its equivalent:

- Name of person being surveyed
- Technician conducting the survey (if not self-survey)
- Location where survey is conducted
- Date and Time of survey
- Results of survey
- Survey instrument serial number for both the alpha and beta-gamma surveys

7.3 REMOVABLE ALPHA CONTAMINATION SURVEYS

7.3.1 The RST is to check the operation of the instrument(s) selected for monitoring. Refer to the instruments operational manual for specific information.

7.3.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-200B or C or the equivalent.

7.3.3 For portable instruments press the battery button to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.3.4 For line powered instruments, verify the cord is plugged in and there is power to the outlet.

7.3.5 Determine and record the instrument alpha background count rate on Form RH-200B or C or the equivalent.

7.3.5.1 For a laboratory-type counting instrument the background rate should be less than 5 cpm. If not, decontaminate the counting chamber using an alcohol wipe or the equivalent and the recount the background.

7.3.6 Determine the MDA and record on the calibration form.

7.3.6.1 For a laboratory-type instrument the MDA can be approximated by:

$$MDA = \frac{2.71 + 3.45 \sqrt{[(B_R)(t)]}}{(t)(E)}$$

7.3.7 Calculate the survey instrument correction factor by:

$$\text{Alpha Correction factor} = \frac{(\text{Source dpm})}{(\text{meter cpm} - \text{bkg cpm})}$$

7.3.8 Number the swipes for identification.

7.3.9 Using the swipe rub lightly an area of 100 cm^2 (4 in. X 4 in.) of potential contamination.

7.3.10 If the surface monitored was wet, let the swipe dry or dry the swipe in a microwave oven before counting.

7.3.11 Place the swipe and count in a Ludlum model 2000 scaler/43-10 alpha swipe counter or similar instrument or place the swipe on a clean flat uncontaminated surface and count the alpha emission rate using the portable alpha meter. Subtract the cpm background from the observed cpm reading and multiply by the instrument correction factor.

7.3.12 Document all survey results on Form RH-200B.

7.3.13 If the survey measurements exceed the instruments background reading or other release level specified by the RSO, decontaminate following the guidelines listed in Procedure RH-020, Table 1.

7.3.14 After decontamination, resurvey, and record results on Form RH-200B or C.

7.3.15 In the event that decontamination is not successful, notify the RSO or Assistant RSO for assistance.

7.3.16 Document the following data on Form RH-200B or C or its equivalent:

- Name of person being surveyed
- Technician conducting the survey (if not a self-survey)
- Date and Time of survey
- Location where survey is conducted
- Results of survey
- Survey instrument serial number for both the alpha and beta-gamma surveys

7.4 REMOVABLE BETA-GAMMA CONTAMINATION SURVEYS

7.4.1 The RST is to check the operation of the instrument(s) selected for monitoring. Refer to the instruments operational manual for specific information.

7.4.2 Read the calibration sticker on the instrument to verify that the instrument has been calibrated within the last year or at a frequency specified by the manufacturer. If not, use an instrument that has a current calibration. Record the instrument serial number on Form RH-200B or C or the equivalent.

7.4.3 For portable instruments, press the battery button to check the condition of the batteries. Change the batteries if low battery voltage is indicated.

7.4.4 For line powered instruments, verify the cord is plugged in and there is power to the outlet.

7.4.5 Determine and record the instrument beta-gamma background count rate on Form RH-200B or C or the equivalent.

7.4.5.1 For a laboratory-type counting instrument, the background rate should normally be less than 100 cpm. If not, decontaminate the counting chamber using an alcohol wipe or the equivalent and then recount the background.

7.4.6 Determine the instrument MDA and record on the calibration form.

7.4.6.1 For a laboratory-type instrument the MDA can be approximated by:

$$MDA = \frac{2.71 + 3.45 \sqrt{[(B_R)(t)]}}{(t)(E)}$$

7.4.7 Calculate the survey instrument correction factor by:

$$\text{Instrument Correction factor} = \frac{(\text{Source dpm})}{(\text{meter cpm} - \text{bkg cpm})}$$

7.4.8 Number the swipes for identification.

7.4.9 Using the swipe rub lightly an area of 100 cm^2 (4 in. X 4 in.) of potential contamination.

7.4.10 If the surface monitored was wet, let the swipe dry or dry the swipe in a microwave oven before counting.

7.4.11 Place the swipe in an Eberline model BC-4 swipe counter or an equivalent instrument or by using the pancake probe with holder and count the beta-gamma emission rate. (Note if alpha contamination is expected, a thin paper disk can be placed between the detector and the swipe to shield out the alpha particles). Subtract the cpm background from the observed cpm reading and multiply by the instrument correction factor. Record the readings on Form RH-200B or C or the equivalent.

7.4.12 If the survey measurements exceed the instruments background reading or other release level specified by the RSO, decontaminate following the guidelines listed in Procedure RH-020, Table 1.

7.4.13 After decontamination, resurvey, and record results on Form RH-200B or C.

7.4.14 In the event that decontamination is not successful, notify the RSO or Assistant RSO for assistance.

7.4.15 Document the following data on Form RH-200B or C or its equivalent:

- Name of person being surveyed
- Technician conducting the survey
- Date and Time of survey
- Location where survey is conducted
- Results of survey
- Survey instrument serial number for both the alpha and beta-gamma surveys

8.0 RELEASE OF PERSONNEL

8.1 If the observed readings for alpha and/or beta-gamma on the person's body (hands, face, etc.) exceed the survey instrument(s) average background or other release action level as specified by the RSO, follow the decontamination procedures listed in procedure RH-020.

8.1.1 Notify the RST or designee for guidance on the decontamination procedures. Usually washing with soap and water with the use of a soft brush is sufficient to remove the contamination to background levels or below the release criteria.

8.2 Removal of the contamination from the person must be attempted before he/she can leave the restricted area.

8.3 Wash and resurvey until the contamination is at the background levels or other release level specified by the RSO. Notify the RSO or Assistant RSO for assistance.

8.4 If alpha or beta-gamma contamination on the person's clothing or shoes etc. exceeds the instruments background reading or other release level specified by the RSO, then the clothing or other item must be decontaminated, the contamination removed to background levels or other release criteria specified by the RSO or the item removed.

8.5 Only after the surveys indicate that both the person and his/her clothing are at the survey instruments background level or other release level specified by the RSO are they allowed to leave the restricted area of the Mill.

8.6 The RST is to file the survey Form RH-200A, B or C or its equivalent and retain the records.

Personnel Release Surveys Procedure RH-200

Appendix A

Self-Survey Form RH-200A

Self-Survey Form

Survey Date: _____

Area Location: _____

Gross Meter Readings Not to Exceed Action Limit
(Limit Includes Background and correction to 100 cm^2)

Instrument Action Limits	
Alpha	Beta/Gamma
cpm	cpm

NOTE: If Action Limit for Alpha or Beta/Gamma is exceeded, then decontamination and resurvey is required before entry.

NOTE: The recorded cpm readings include background

Empl. #	Name	Total Alpha (100 cm^2)		Total Beta/Gamma (100 cm^2)		Survey Time	Survey After DeCon (if required)				
							Total Alpha (100 cm^2)		Total Beta/Gamma (100 cm^2)		Survey Time
		cpm	dpm	cpm	dpm		cpm	dpm	cpm	dpm	

Safety Department Use Only

Alpha Efficiency x Probe Factor: _____ Beta/Gamma Efficiency x Probe Factor: _____

Alpha Background: _____ cpm Meter Efficiency: _____ Beta/Gamma Background: _____ cpm Meter Efficiency: _____

Alpha Meter SN: _____ Beta/Gamma Meter SN: _____

Form RH-200A

Personnel Release Surveys Procedure RH-200

Appendix B

Quarterly Random Spot Check Form RH-200B

Quarterly Random Spot Check

Release Limit Used_____ Date:_____

		Initial				Removable				Other					
Employee Name	Employee Number	Right Hand	Left Hand	Right Foot	Left Foot	Right Hand	Left Hand	Right Foot	Left Foot	Item	Reading	Instrument Used	Mantle/ Source Check	Initial	Remark

Material	Code #	Total Average	Total Maximum	Removable
Ore	1	4,000	12,000	800
Tailings	2	3,000	9,000	600

Surveyed by:_____ Date:_____

Reviewed by:_____ Date:_____

Form RH-200B

1.0 PURPOSE

This procedure is for the use of personnel dosimeters by workers to provide an exposure to ionizing radiation at the Piñon Ridge Mill.

2.0 APPLICABILITY

Personal radiation dosimeters are issued to individuals at the Mill to provide an exposure estimate from external gamma radiation. As part of the As Low As Reasonably Achievable (ALARA) program, employees and contractors at the facility are issued Thermoluminescent Dosimeters (TLD) or Optically Stimulated Luminescence Dosimeters (OSL) if there is any chance that their dose from gamma rays could exceed 10% or more of the applicable radiation dose limit, i.e. 500 mrem/year.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 NRC Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

3.1.2 Colorado Radiation Control Regulations, Part 4 (6 CCR 1007-1)

3.2 *APPENDICES*

Appendix A - Lost or Broken TLD/OSL Badge Report Form RH-210A

4.0 EQUIPMENT AND MATERIALS

4.2 *EQUIPMENT*

4.2.1 Thermoluminescent Dosimeter (TLD) or Optically Stimulated Luminescent Dosimeter (OSL)

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) is responsible for:

5.1.1 Assuring that TLD/OSL badge vender is NVLAP certified.

5.1.2 Reviewing the TLD/OSL Occupational Radiation Exposure Report and the Occupational Exposure Record for a Monitoring Period.

5.1.3 Investigating any exposure results exceeding the action level guidelines.

5.1.4 Determining any corrective actions that should be put into affect based on the investigation.

5.1.5 Reviewing the Lost or Broken TLD/OSL Badge Report (as needed) and the Lost Badge Report (yr) Spreadsheet (quarterly).

5.2 The Assistant RSO is responsible for:

5.2.1 Verifying shipment of received and shipped TLD/OSL badges.

5.2.2 Issuing TLD/OSL badges to the Safety/Security Technicians at the beginning of each new quarter.

5.2.3 Tracking lost and/or broken TLD/OSL badges.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

5.2.4 Distributing the Occupational Exposure Record for a Monitoring Period.

5.2.5 Notifying the dosimetry vender of additions to and deletions from the list of personnel designated to receive TLD's/OSL's.

5.2.6 Issuing to or receiving badges from personnel and storing the TLD/OSL badges at the guard shack.

5.2.7 Exchanging the badges once per quarter or when specified by the RSO.

5.4.2 Completing the Lost or Broken TLD/OSL Badge Report when a badge is reported.

5.5 Each Individual issued a radiation monitor is responsible for:

5.5.1 Wearing the badge while at work at the Mill.

5.5.2 Reporting lost or damaged badges to the Safety/Security Technician.

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 TLD - Thermoluminescent Dosimeter

6.1.2 OSL - Optically Stimulated Luminescent Dosimeter

6.1.3 NVLAP - National Voluntary Laboratory Accreditation Program

6.1.4 DDE - Deep Dose Equivalent

6.1.5 SDE - Shallow Dose Equivalent

6.1.6 LDE - Lens Dose Equivalent

6.1.7 EDE - Extremity Dose Equivalent

6.2 *FREQUENCY*

6.2.2 TLD/OSL badges are exchanged on a quarterly basis or as specified by the RSO.

7.0 PROCEDURES

7.1 *ISSUANCE OF RADIATION MONITORS*

7.1.1 Individuals are issued personal radiation monitors at and return them to the Assistant RSO or Safety/Security Technician at the guard house.

7.1.2 When not worn, TLD/OSL badges are stored on a badge board in the guard house.

7.1.2.1 Each TLD/OSL badge is identified by a unique badge number assigned by the vender along with the name of the employee.

7.1.3 TLD/OSL badges or dosimeters should be worn on the trunk of the body, at or above the waist for whole body exposure estimates.

7.1.4 A minimum of two TLD/OSL control badges is stored along with the individual TLD/OSL badges at the guard house.

7.2 READING OF TLD/OSL BADGES

7.2.1 TLD/OSL badges will be read by a NVLAP certified vender.

7.2.2 The TLD/OSL vender issues a quarterly Occupational Exposure Report

for a Monitoring Period within thirty days from the end of the calendar quarter, and a summary Occupational Radiation Exposure Report within sixty days from the end of the calendar year, which contains the following:

- Personnel Identification
- Name
- Badge Number
- Social Security Number
- Birth Date
- Date of Badge Issue
- Date of Badge Return
- Dose measured for the reporting period, calendar quarter, calendar year, and lifetime occupational dose.
- Deep Dose Equivalent (DDE)
- Shallow Dose Equivalent (SDE)
- Permissible accumulated dose.

7.2.3 In the event a measurement exceeds the action level, the TLD/OSL vender will notify the RSO, or the Assistant RSO by telephone who will then inform the RSO, and an investigation will be undertaken which will either substantiate the occurrence of measurement or describe the conditions that caused the measurement to be non-representative of the exposure that the badge holder actually sustained.
A Dosimeter High Call List, when measurement exceeds the action level, is also received, via fax, reviewed by the RSO, and retained in the files by the Assistant RSO.

7.2.4 In the event the measurement is valid and exceeds the action level, the RSO will initiate such corrective action as is necessary in his/her opinion to reduce the exposure condition for the individual involved.

7.2.5 An incident of exposure of an individual to radiation in excess of the maximum permissible dose stipulated by Radiation Regulations will be reported in writing to the Colorado Department of Public Health and Environment (CDPHE) within 30 days from the date of the discovery of such exposure. This report shall include the following:

- The name, date of birth and social security number of the affected person(s)
- The amount of exposure
- The cause of the exposure
- Corrective actions taken or planned to prevent reoccurrence.

7.2.6 Concurrently with the transmittal of the overexposure report to the CDPHE, a written report will be transmitted to the affected individual(s) in accordance with the requirements of the CDPHE, Radiation Control Regulations.

Personal Radiation Dosimeters Procedure RH-210

Appendix A

Lost or Broken TLD/OSL Badge Report Form RH-210A

Lost or Broken TLD/OSL Badge Report Form

Employee Name _____ Company _____

Employee Number _____ Date _____

Hire Date _____ Termination Date _____

WEARER _____ was not left in controlled area for the following reason:

LOST BADGE_____ BROKEN BADGE_____ LEFT ELSEWHERE_____

Describe the specific area, if known, where the badge/ring was left or lost: _____

_____

REMARKS: _____

SAFETY DEPARTMENT ONLY

If lost badge was found, complete the following:

Date Found _____ Area Found _____

If blank badge was issued, complete the following:

Badge Number_____ Date Issued_____

Reason for issue of blank badge: _____

_____

_____
Assistant RSO or RST

_____
Date

_____
RSO

_____
Date

1.0 PURPOSE

This procedure describes the radiation dose calculation for personnel at the Piñon Ridge Uranium Mill. The Radiation Dose Calculations Procedure is divided into 3 sub-procedures (RH-301 through -302).

2.0 APPLICABILITY

This procedure applies to Mill personnel that are exposed to radiation. Doses will be calculated and assigned for any personnel reaching 10 % of any exposure limit including Annual Limits on Intake (ALI), Derived Air Concentrations (DAC), and working levels (WL) related to radionuclide concentrations in air, results of bioassay samples and measurements as well as external exposure monitoring results.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 USNRC Regulatory Guide 8.30, "Health Physics Surveys in Uranium Recovery Facilities," 2002

4.0 RESPONSIBILITY

4.1 The Radiation Safety Officer (RSO) is responsible for:

4.1.1 Calculating the radiation doses of all personnel at the Mill that receive radiation doses in excess of 10% of the applicable limits, which is the level at which personal radiation monitoring is required.

4.1.2 Providing guidance to the Radiation Security Technician (RST) on how to monitor the radiation exposures of personnel at the Mill.

4.1.3 Review all radiation doses and trends in radiation doses to maintain radiation doses to levels that are As Low As Reasonably Achievable (ALARA).

4.2 The Radiation Security Technician (RST) is responsible for:

4.2.1 Monitoring radiation exposures to personnel the Mill.

5.0 PREREQUISITE INFORMATION

5.1 *FREQUENCY*

5.1.1 Dose assessments and assignments will be performed monthly, or more frequently as determined by the RSO.

6.0 PROCEDURES

This procedure is divided into three separate sub-procedures as follows:

- RH-301 Worker Exposure to Long-Lived Radionuclides in Airborne Particulate Matter
- RH-302 Radionuclide Concentrations in Air Samples Based on Gross Alpha Measurements and Laboratory Analyses
- RH-303 Dose Calculation Spreadsheets

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

1.0 INTRODUCTION

Individual worker exposures to long-lived radionuclides in airborne particulate matter are calculated based on measured radionuclide concentrations in air and worker time-logs. Airborne radionuclide concentrations are measured for a worker, or a group of workers performing the same task in the same location, using breathing zone samplers sometimes called personal lapel samplers. Concentrations at specific locations in the Mill are measured using general air samplers.

Workers record the hours spent in particular locations in weekly time logs. The time logs are also used to track whether the individual wore a breathing zone sampler and/or used respiratory protection for part or all of the day. Worker exposures are calculated by multiplying the number of hours worked by the radionuclide concentrations determined from breathing zone samples. Where no concentration data are available from breathing zone samplers, worker exposures are calculated by multiplying the number of hours spent in particular locations in the Mill by the concentrations measured using general air samples.

2.0 OTHER DOCUMENTS

2.1 *APPENDICES*

Appendix A – Determination that a Sampler is an Outlier Form RH-301B

Appendix B – Breathing Zone Sample Return Report Form RH-301A

3.0 BREATHING ZONE SAMPLES

Individual workers are issued breathing zone samplers when they are expected to be in a work environment in which they could receive an exposure to airborne radionuclides equal to or greater than 12 Derived Air Concentration (DAC)-hours in any week. The worker wears the sampler for a specified number of hours, generally for the entire work day. In some cases the sampler may be worn for only part of the day or for more than one day depending on the area, the expected radionuclide concentrations, and the intended use of the data. The Radiation/Security Technician (RST) issues the breathing zone samplers as described in Procedure RH-150. The worker returns the breathing zone sampler at the end of his/her shift and completes the attached Form RH-301A. The Safety Department then submits the sample to the lab for analysis as soon as possible after the end of the shift.

The RST maintains a log of all samples which includes date, sample time, individual wearing the sampler, respirator protection used, work being performed, and, where applicable, the location of the work. This information is provided to the individual performing the dose calculations.

4.0 GENERAL AIR SAMPLES

General air samples are taken in several specific locations in Mill buildings as determined by the Radiation Safety Officer (RSO). The general air sampling method and specific locations are designated in Procedure RH-130. These samples are used to determine

whether respiratory protection is necessary (i.e., concentrations greater than 25 percent of the area-specific DAC) and as a concentration measurement for workers at that location. The data are used in estimating the individual worker's radionuclide intakes for the purpose of dose calculation. General air samples are obtained from all work areas on a daily, weekly, or monthly schedule, depending on the location, the potential for exposure of workers, and the average fraction of the DAC based on prior sampling, as per Regulatory Guide 8.25, Air Sampling in the Workplace and Regulatory Guide 8.30, Health Physics Surveys in Uranium Recovery Facilities.

The data from general air samples are used in the intake calculations for workers when breathing zone data are not available. The worker's work-hour logs are used to estimate the number of hours the worker was exposed to the air concentrations represented by the general air sample.

The employee work hours are assigned to general air sample results based on work hour logs that record time spent in each particular area of the mill. The inhalation intake calculation spreadsheet notes whether the concentration value was based on a breathing zone sample or on the average value for general air samples.

5.0 TREATMENT OF ANOMALOUS SAMPLE RESULTS

Occasionally the results of a breathing zone or general air sample show gross alpha or specific nuclide concentrations that are inconsistent with previous data, either significantly higher or lower. In such instances, the RSO or Assistant RSO investigates the potential cause for the anomalous sample. In general the sample is considered an outlier if it is more than 10 times higher or 10 times lower than the previous average values for the particular job category or if the sample is greater than 50 percent of the DAC.

For breathing zone outliers, the RSO or Assistant RSO takes the following steps:

5.1 Verify that the sampler was handled appropriately by checking with the RST who issued it and the wearer.

5.2 Submit a Data Verification Request to the Laboratory

5.3 Check the sample submittal form to verify that the data on the form are correct (e.g., correct sample volume)

5.4 Review work records and/or interview the wearer of the breathing zone sampler to determine whether anything out of the ordinary occurred and to verify that the wearer was actually in the area to which he was initially assigned.

5.5 Compare the anomalous concentration with previous breathing zone or general air sample results.

5.6 Use the data in the intake calculation for the worker unless it is clear that the sample is invalid, e.g. the sampler was dropped in a drum of yellowcake or the pump was not pulling the expected amount of air through the filter.

For anomalous general air samples, the area will be re-sampled as soon as practicable under the conditions similar to those existing when the initial sample was taken.

If the RSO or Assistant RSO determines on the basis of his investigation that the sample is not valid, that decision will be documented on Form RH-301A.

6.0 APPLICATION OF RESPIRATORY PROTECTION FACTORS

Workers are required to wear respiratory protection in areas where the gross alpha or uranium concentration might exceed 25 percent of the area-specific DAC based on analysis of general air samples. The area-specific DACs are as given in Procedure 302, Table 3.

The protection factor for the type of respirator used is applied to the measured concentration for the number of hours the respirator was worn during the period of time covered by the breathing zone or general air sample. Where quantitative fit tests have been performed the protection factors in Part 4 of the Colorado Radiation Control Regulations (6 CCR 1007-1) will be used. Where only qualitative fit tests have been performed the default protection factor provided in the regulations will be used.

7.0 SAMPLE COMPOSITES

Based on the judgement of the RSO or Assistant RSO, breathing zone samples for a particular individual may be composited to increase the volume of air represented by the sample and to improve the Lower Limit of Detection for the air concentration. Such composites are noted on the dose calculation spread sheets.

8.0 DOSE CALCULATION

The dose calculations are described in detail in Procedure RH-303. Intakes in DAC-hours are calculated at least on a quarterly basis for the purpose of determining the need for an in-vivo bioassay and work restrictions. The Committed Effective Dose Equivalent (CEDE) for intake of radionuclides is calculated quarterly based on the specific radionuclide concentrations and the number of hours the worker is exposed. The specific radionuclide concentrations are derived from the general air or breathing zone sample laboratory analyses for gross alpha and specific radionuclides as described in the attached radionuclide apportionment method (Procedure RH-302).

Worker Exposed to Long-Lived Radionuclides in Airborne Particulate Matter RH-301

Appendix A

Determination that Sampler is an Outlier Form RH-301A

Worker Exposed to Long-Lived Radionuclides in Airborne Particulate Matter RH-301

Appendix B

Breathing Zone Sample Return Report, Form 301B

Breathing Zone Sample Return Report

Name_____

Sample date_____

Time on_____ Time off_____

Respiratory protection worn yes no

Other workers performing the same task:

Any unusual circumstances:_____

Signature of wearer_____

1.0 INTRODUCTION

Airborne concentrations of radioactive materials are determined from breathing zone and general air samples that are obtained by passing a measured volume of air through a filter. The filter is then analyzed in the laboratory to determine its radionuclide content. The concentration of radioactive materials in air is determined by dividing the radioactivity on the filter by the volume of air that flowed through it.

The gross alpha activity on an air filter is apportioned among U-238, U-234, Th-232 and their long lived decay products according to the relative concentrations in the bulk material being processed at the Mill and based on the results of specific radionuclide analysis of the filter. That is, the measured specific radionuclide concentrations, if any, will be subtracted from the final gross alpha concentration and the remainder apportioned among the other nuclides. For the purpose of determining airborne radionuclide concentrations, U-nat activity is equal to the activity of U-238 plus U-234 and the U-235 decay series is ignored.

Tables 1 and 2 list the uranium and thorium series decay products along with their dose conversion factors. Assuming all of the radionuclides in each of the decay series are in secular equilibrium with the parent, the short-lived alpha emitting decay products and the beta/gamma emitters contribute no more than 2 percent of the dose from uranium and thorium and do not need to be accounted for in the analysis.

The following sections describe the method by which radionuclide concentrations are determined from the laboratory analysis data. Section 2 provides a description of the Decision Flow Charts (Figures 1 and 2). Section 3 includes the methods by which the ratios of radionuclides in airborne dust are determined. Section 4 describes the methods by which the gross alpha concentrations are apportioned among the radionuclides potentially present on the filter.

2.0 DECISION FLOW CHART

The attached Figures 1 and 2 are Decision Flow Charts for the determination of radionuclide concentrations in general air or breathing zone samples. Figure 1 applies to all samples. Figure 2 applies to the product areas. Routine and special breathing zone and general air samples are taken from all areas of the plant periodically according to the requirements of the radioactive materials license and the type of work being performed. The methods by which the radionuclide concentrations are attributed to the various radionuclides potentially present in the air are described in Section 4.

2.1 Samples taken from locations on the site other than the product area are routinely counted for gross alpha activity. (In rare cases air filters may have excessive dust or debris deposits so gross alpha analysis is not feasible. These samples are analyzed radiochemically for specific radionuclides.) In areas where the gross alpha concentration may exceed the area-specific Derived Air Concentration (DAC) an initial gross alpha count will be requested. The area-specific DACs, given in Table 3 for caldasite processing, will depend on the relative radionuclide concentrations in the material to be processed. The activity on the filter will

include Rn-222 and Rn-220 decay products if the sample is counted within a few hours of collection. Rn-220 decay products will not decay out for about 50 hours after the sample is collected.

2.1.1 If the initial gross alpha result indicates that the radionuclide concentration may exceed the DAC listed in Table 3 for the particular area of the plant from which it was taken, the sample is analyzed for U-nat, Th-230, Th-232, and Ra-226. The isotopic concentrations are apportioned as per Method 6 (Section 4.7). The gross alpha measurement is not used in determining the airborne concentration. [Specific isotopic analysis precludes a final gross alpha analysis, i.e., after Rn-222 progeny have decayed out.]

2.1.2 If the initial gross alpha result indicates that the area-specific DAC listed in Table 3 will not be exceeded, a final gross alpha measurement is made after the sample has been stored for at least two days to allow the Rn-222 and Rn-220 progeny to decay out.

2.2 If the final gross alpha measurement indicates concentrations greater than 10 percent of the U-nat DAC given in Table 3, the sample is analyzed to determine the mass of U-nat.

2.2.1 If the U-nat concentration accounts for more than 90 percent of the gross alpha, the remainder of the gross alpha is apportioned among Th-230, Th-232+D, and Ra-226 as per Method 4 (Section 4.4). The decay products of Th-232, Ra-228 and Th-228, are assumed to be in equilibrium with the parent. Filters are not specifically analyzed for these isotopes.

2.2.2 If the U-nat concentration accounts for less than 90 percent of the gross alpha, the sample is also analyzed for Th-230 and Th-232. The remainder of the gross alpha is attributed to Ra-226 (Method 3, Section 4.3).

2.2.3 If the final gross alpha measurement indicates concentrations less than 10 percent of the U-nat (Y) DAC given in Table 3 but greater than 10 percent of the Th-230 (Y) DAC given in Table 3, the filter is analyzed for Th-230 and Th-232. The remainder of the gross alpha is apportioned between U-nat and Ra-226 according to Method 2 (Section 4.2).

2.2.4 If the final gross alpha measurement indicates concentrations less than 10 percent of the Th-230 DAC, no specific radionuclide analyses are required. The gross alpha activity is apportioned according to Method 1 (Section 4.1).

2.2.5 If the sample has been taken from the product area where uranium constitutes nearly all of the airborne radioactivity and yellowcake appears to be on the filter, it is analyzed for uranium and the uranium concentration calculated directly from the laboratory data (Method 5a, Section 4.5). If no visible yellowcake residue is present, the filter is analyzed for gross alpha activity and the total gross alpha activity is attributed to U-nat (Method 5b, Section 4.6). The results of analyses of air

samples from the yellowcake stacks demonstrate that the concentrations of the other radionuclides in the U-238 decay series as well as Th-232 constitute less than one percent of the activity on average and can be omitted in the analysis.

3.0 CALCULATION OF RADIONUCLIDE RATIOS FOR THE PURPOSE OF APPORTIONING THE GROSS ALPHA MEASUREMENT

Energy Fuels processes a variety of feed materials with different relative concentrations of natural thorium and natural uranium. For most Colorado Plateau uranium ores, the natural thorium concentrations are negligible (i.e. 0.02 percent thorium).

The theoretical fraction of the total gross alpha activity for each of the nuclides in a material that contains natural thorium as well as uranium will be calculated as follows:

3.1 Assume a total of 4 long-lived alpha emitters for the U-238 decay series (U-238, U-234, Th-230, and Ra-226).

3.2 In the bulk ore, the ratio of radionuclides can be calculated as follows:

Total Gross Alpha activity Concentration (TGAC) = [U-238 pCi/g)(4)]

Fraction from each nuclide:

$F_{U\text{-}nat} = 0.50$

$F_{Th\text{-}230} = 0.25$

$F_{Ra\text{-}226} = 0.25$

4.0 GROSS ALPHA APPORTIONMENT

As noted in the previous sections, all samples except those from the yellowcake area, are routinely analyzed for gross alpha. In some cases, specific radionuclide analyses are also performed. The spreadsheets for worker dose calculations list the results for uranium, thorium, and radium analyses whenever these laboratory analyses were performed. At least 100 percent of the gross alpha activity from air samples is accounted for in the determination of the air concentrations of radionuclides except where specific radionuclide analyses are performed for U-nat, Th-230, Th-232, and Ra-226. The measured concentrations will be used in the dose calculations in such cases regardless of the gross alpha concentration. Gross alpha concentration measurements are generally less accurate than the specific radionuclide analyses. The only long-lived alpha emitter that is not accounted for by the analyses for U-nat, Th-230, Th-232, and Ra-226 is Po-210. The Po-210 inhalation dose coefficient is a factor of approximately 10 lower than the U-238 dose coefficient and a factor of 30 lower than the Th-230 dose coefficient. Therefore, Po-210 would not contribute significantly to the dose.

In cases where uranium and/or thorium concentrations are specifically measured and the concentrations exceed the gross alpha concentration, i.e., the gross alpha concentration to be apportioned to the other nuclide(s) is a negative value, the remaining nuclide(s) will be assigned a concentration of zero.

Nuclide Apportionment

4.1 ***METHOD 1*: Used for samples analyzed only for gross alpha activity**

Activity of U-nat (U-238 + U-234) = [gross alpha activity][2 x F_{U-238}]

Activity of Th-230 = [gross alpha activity][F_{U-238}]

Activity of Ra-226 = [gross alpha activity][F_{U-238}]

Activity of Th-232+D = [gross alpha activity][$F_{Th-232+D}$]

4.2 ***METHOD 2*: Used for samples analyzed for gross alpha, Th-230, and Th-232 activities**

Activity of Th-230 = measured value

Activity of Th-232 = measured value (decay products assumed to be in equilibrium)

Activity of U-nat = [gross alpha activity – (Th-230 + Th-232 activity)][0.67]

Activity of Ra-226 = [gross alpha activity – (Th-230 + Th-232 activity)][0.33]

4.3 ***METHOD 3*: Used for samples analyzed for gross alpha, Th-230, Th-232, and U-nat**

Activity of Th-230 = measured value

Activity of Th-232 = measured value (decay products assumed to be in equilibrium)

Activity of U-nat = measured mass (g) x 0.68 μCi/g

Activity of Ra-226 = gross alpha – (Th-230 + Th-232 + U-nat activity)

4.4 ***METHOD 4*: Used for samples analyzed for gross alpha and U-nat**

Activity of U-nat = measured U-nat mass (g) x 0.68 μCi/g

Activity of Th-230 = [gross alpha – U-nat activity][$0.5(1 – 3xF_{Th-232})$]

Activity of Ra-226 = [gross alpha – U-nat activity][$0.5(1 – 3xF_{Th-232})$]

Activity of Th-232+D = [gross alpha – U-nat activity][3 x F_{Th-232}]

4.5 ***METHOD 5a*: Used for samples from the product area analyzed for U-nat**

Activity of U-nat = measured U-nat mass (g) x 0.68 μCi/g

4.6 ***METHOD 5b*: Used for samples from the product area analyzed only for gross alpha**

Activity of U-nat = measured gross alpha activity

4.7 ***METHOD 6*: Used for samples analyzed for Th-230, Th-232, U-nat, and Ra-226**

Activity of U-nat = measured mass (g) x 0.68 μCi/g

Activity of Th-230 = measured value

Activity of Ra-226 = measured value

Activity of Th-232 = measured value (decay products assumed to be in equilibrium)

5.0 SOLUBILITY CLASS DETERMINATION

In all cases, Ra-226 is assumed to be International Commission of Radiological Protection (ICRP) 68 lung clearance Class M (equivalent to Federal Guidance Report [FGR] No. 11 solubility class W) and Th-230, lung clearance class S (equivalent to FGR No. 11 solubility class Y). The lung clearance class for U-nat depends on the process and the location in the Mill. Based on information in U. S. Nuclear Regulatory Commission Regulatory Guides 8.22 and 8.30, NUREG 0874, and NUREG 3598, the current assumptions regarding solubility or lung clearance class for uranium are as follows: Class Y (equivalent to ICRP 68 lung clearance class S) for locations where ore is handled prior to chemical separation and for yellowcake dried at >400° C and Class D (equivalent to ICRP clearance class F) for those locations with undried yellowcake or yellowcake dried at <400° C. Where the solubility for a specific bulk material (e.g. yellowcake) has been analyzed, the lung clearance class will be determined based on the laboratory data.

6.0 ALTERNATE APPORTIONMENT IF LABORATORY DATA REQUIRED FOR THE PARTICULAR METHOD ARE NOT AVAILABLE

In the event laboratory data on specific radionuclide concentrations are not available for a sample either due to loss of the sample or inability to perform the required analyses according to Figure 1, the method of apportioning the gross alpha activity will default to Method 1 (Section 4.1). If gross alpha and Th-230/Th-232 concentrations are available but not U-nat, the gross alpha will be apportioned according to Method 2 (Section 4.2). If U-nat concentrations but not Th-230/Th-232 concentrations are available, Method 4 (Section 4.4) will be used to apportion the gross alpha count.

Table 1: U-238 Decay Series Dose Coefficients (ICRP 68)

Nuclide	Half-life	Type of emissions	ICRP 68 Dose Coefficients (Sv/Bq)	Considered in dose calculation?
U-238	4.51E9 y	Alpha	7.3E-6 (S)	Yes
Th-234	24 d	Beta	7.3E-9 (S)	No
Pa-234m	1.17 m	Beta	None given	No
U-234	2.35E5 y	Alpha	8.5E-6 (S)	Yes
Th-230	8.0E4 y	Alpha	1.3E-5 (S)	Yes
Ra-226	1.6E3 y	Alpha	1.6E-5 (M)	Yes
Rn-222	3.8 d	Alpha	NA	Yes – but taken into account separately as WL
Po-218	3.0 m	Alpha	NA	Yes (see Rn-222)
Pb-214	26.8 m	Beta/gamma	NA	Yes (see Rn-222)
Bi-214	19.7 m	Beta/gamma	NA	Yes (see Rn-222)
Po-214	164 μs	Alpha	NA	Yes (see Rn-222)
Pb-210	22 y	Beta	8.7E-7 (F)	No
Bi-210	5 d	Beta	8.4E-8 (M)	No
Po-210	138 d	Alpha	3.0E-6 (M)	Yes – not specifically, but in the gross alpha except where specific radionuclide analyses are performed.
Total Dose Coefficient			4.9 E-5	
Total Dose Coefficient for nuclides not taken into account			9.6E-7	
Fraction Dose Coefficient not taken into account			0.02	

Table 2: Th-232 Decay Series Dose Coefficients

Nuclide	Half-life	Type of emissions	ICRP 68 Dose Coefficients (Sv/Bq)	Considered in dose calculation?
Th-232	1.41E10 y	Alpha	2.3E-5 (S)	Yes
Ra-228	5.75 y	Beta	2.6E-6 (M)	Yes
Ac-228	6.13 h	Beta	1.6E-8 (M)	No
Th-228	1.91 y	Alpha	3.9E-5 (S)	Yes
Ra-224	3.66 d	Alpha	2.9E-6 (M)	Yes (assumed in equilibrium with Ra-228)
Rn-220	56 s	Alpha	NA	Yes – but taken into account separately as WL
Po-216	0.15 s	Alpha	NA	Yes (see Rn-220)
Pb-212	10.64 h	Beta	NA	Yes (see Rn-220)
Bi-212	60.6 m	Alpha/beta	NA	Yes (see Rn-220)
Tl-208 (0.36)	3.1 m	Beta/gamma	NA	Yes (see Rn-220)
Po-212 (0.64)	304 ns	Alpha	NA	Yes (see Rn-220)
Total Dose coefficient			6.8 E-5	
Total Dose Coefficient for nuclides not taken into account			1.6 E-8	
Fraction Dose Coefficient not taken into account			<0.001	

Table 3: Area-specific Derived Air Concentrations for Caldasite Ore (From the *Colorado Rules and Regulations Pertaining to Radiation Control*)

Area	DAC	Justification
Ore handling	2E-11 µCi/ml (Class Y U-nat DAC)	More conservative than the 6E-11 gross alpha DAC for uranium ore; takes into account the potential for Th-232 in the caldasite ore
Crusher/FOB	2E-11µCi/ml	Same as above
Grind	2E-11µCi/ml	Same as above
Leach	1E-11 µCi/ml (50% of the U-nat DAC)	Takes into account the presence of some recycle solutions that could have some tailings
Kiln	7E-12 µCi/ml	Combined DAC for caldasite ore (See Note a, below)
Solvent Extraction	1E-11 µCi/ml (50% of the Class Y U-nat DAC)	Same as for Leach area
Sulfation	7E-12 µCi/ml	Same as for Kiln
Zirconium Product	7E-12 µCi/ml	Same as for Kiln
Uranium Product	2 E-11µCi/ml (Class Y U-nat DAC)	

Note: The DAC for the kiln, sulfation, and product areas where Th-232 may be present was calculated as follows:
DAC for Th-232 (Y) = 1 E-12 µCi/ml; DAC for U-238 (Y) = 2 E-11 µCi/ml
Ratio of Th-232 to U-238 = 77 pCi/g/701 pCi/g = 0.11
Fraction of Th-232 + U-238 activity due to Th-232 = 0.1: fraction due to U-238 = 0.9
Assume a gross alpha concentration = 1 E-11 µCi/ml
Fraction of DAC = [9 E-12 µCi/ml/2E-11µCi/ml] + [1E-12 µCi/ml/1 E-12 µCi/ml] = 1.45
DAC = 1E-11 µCi/ml/1.45 = 7 E-12 µCi/ml
(The combined DAC for the Kiln, Sulfation, and Product areas is conservative in that it accounts only for U-238. In fact a much greater proportion of the gross alpha than 0.9 is due to U-nat since it includes two alpha emitting isotopes, U-238 and U-234.)

Figure 1
Decision Flow Chart
Determination of Radionuclide Concentrations
In Routine Air Samples

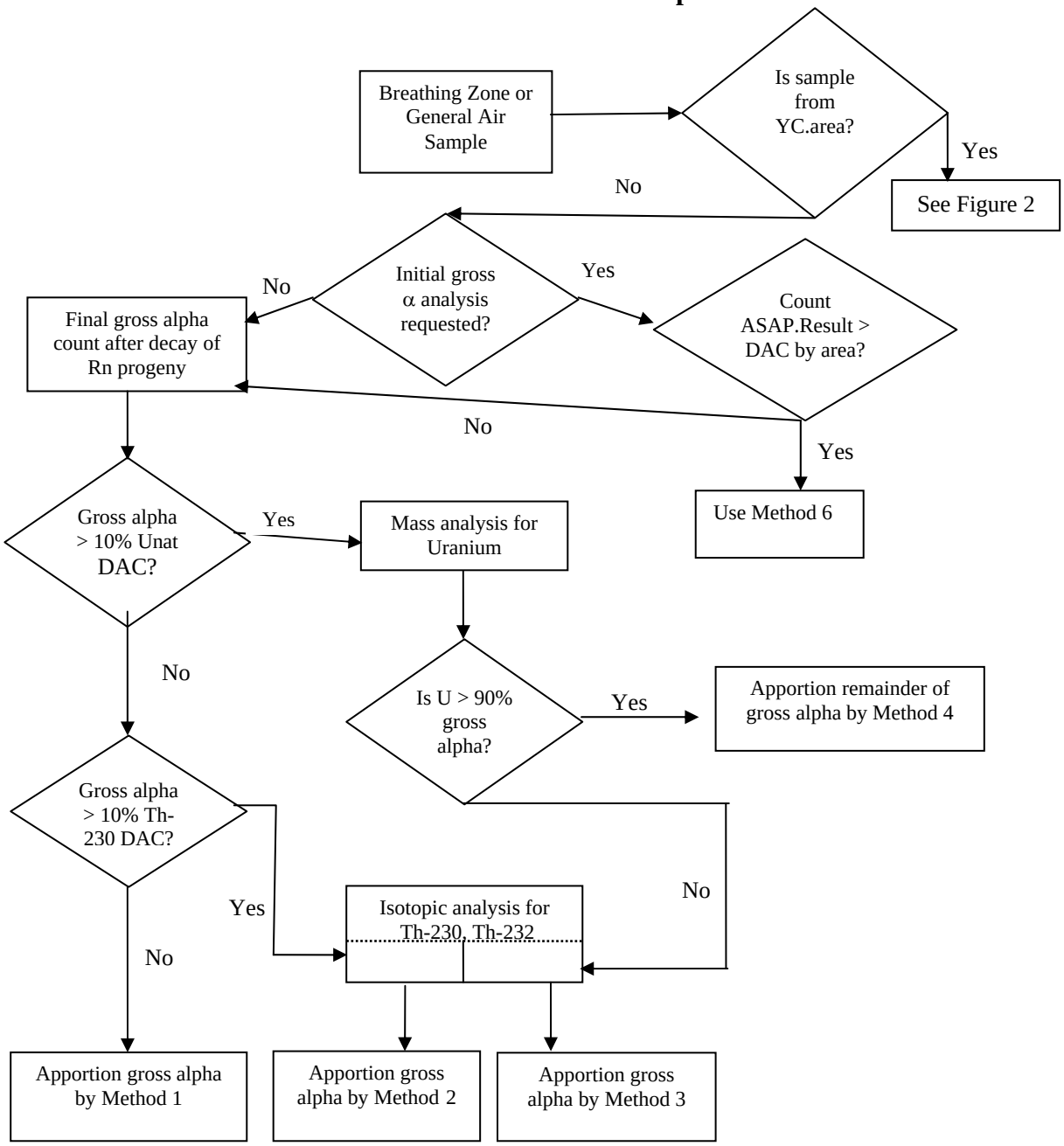
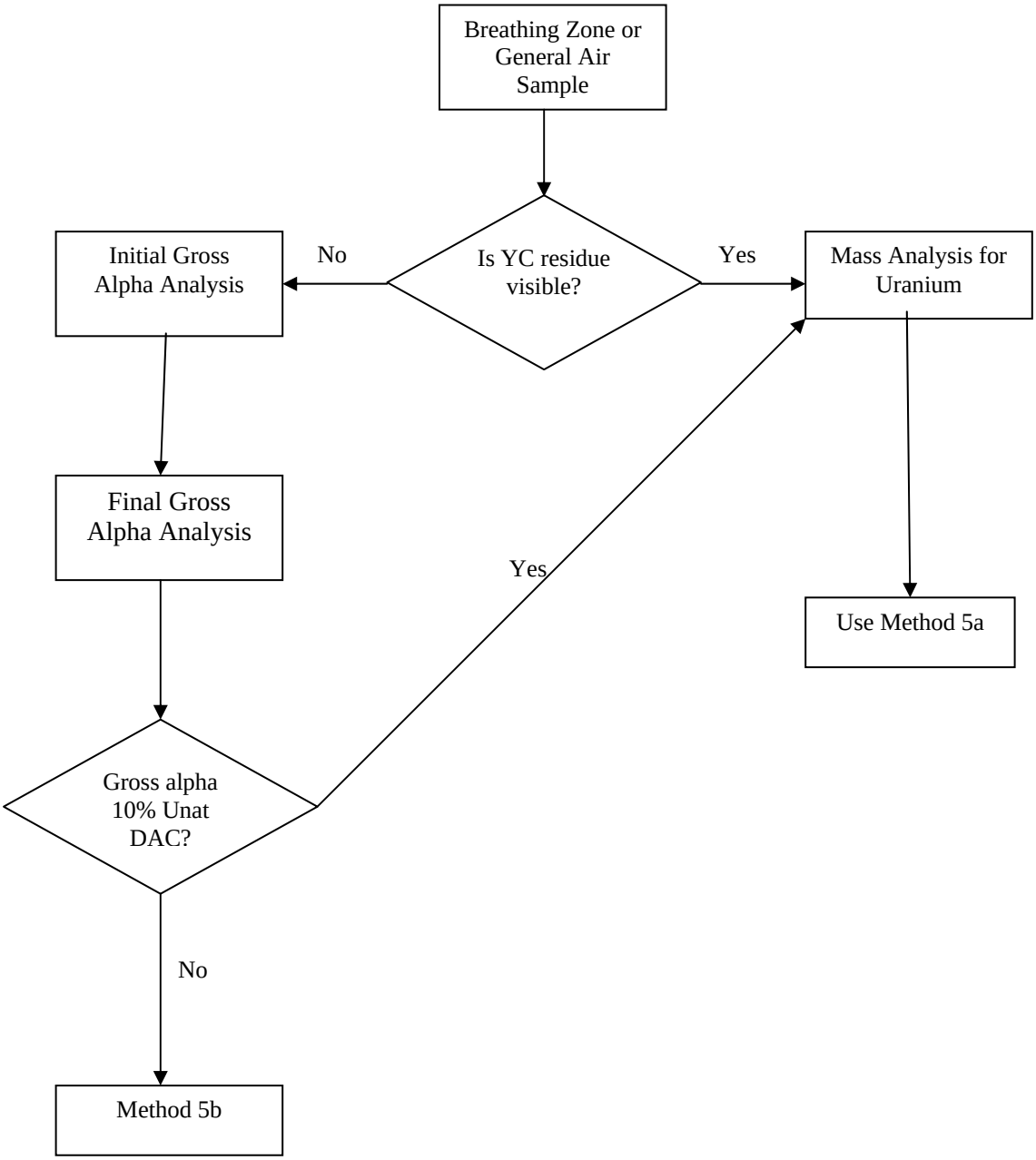


Figure 2
Decision Flow Chart
Determination of Uranium Concentrations
In Product Area Air Samples
At the Milling Facility



1.0 INTRODUCTION

Worker radiation doses are calculated based on inhalation intake of radionuclides in airborne particulate matter, inhalation intake of radon decay products and direct radiation dose from external sources. Spreadsheets are used to estimate the inhalation doses. External radiation doses are taken directly from the vendor reports and entered into a dose calculation spreadsheet.

2.0 OTHER DOCUMENTS

2.1 *APPENDICES*

Appendix A – Example Total Summary Table

Appendix B – Example Organ Dose Calculation Spreadsheet

3.0 INHALATION OF LONG-LIVED RADIONUCLIDES

Radiation doses to workers are calculated for inhalation of long-lived radionuclides in airborne particulate matter, inhalation of radon decay products, and direct radiation exposure. The individual worker's intake of long-lived radionuclides is calculated based on breathing zone and/or general air samples as described in Procedure RH-301. Annual average concentrations of long-lived radionuclides in air for particular areas in the Mill are calculated based on the general air samples. The measured concentrations are averaged quarterly and used to calculate quarterly radionuclide intakes. Concentrations measured on breathing zone samples are used for assigning a specific radionuclide concentration to a particular employee, or group of employees performing a specific task in the same location.

The inhalation intake calculation spreadsheet tracks the radionuclide intake by worker using the air concentration data, occupancy data from the worker time logs, and use of respiratory protection. A sample inhalation intake form is attached. Table 1 provides a key to the inhalation intake calculation spreadsheet.

3.1 *RADON DECAY PRODUCT DOSE CALCULATION*

Doses due to Rn-222 decay product exposure are calculated as follows:

Exp. (WLM) = [conc. (WL)][(hours of exp.)][(12 months/y)]/[2000 h/year]

Dose = [exp. (WLM)][5 rem]/[4 WLM]

Doses due to Rn-220 decay product exposure are calculated as follows:

Exp. (WLM) = [conc. (WL)][hours of exp.][12 months/y]/[2000 h/year]

Dose = [exp. (WLM)][5rem]/[12 WLM/y]

where:

WLM = exposure in Working Level Months

WL = Working Level measured concentration of Rn-222 or Rn-220 decay products

Annual Limit of Intake (ALI) for Rn-222 decay products = 4 WLM

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

ALI for Rn-220 decay products = 12 WLM

Rn-220 decay product concentrations will be required to be measured and included in the dose estimate only when the natural thorium concentration of the material being processed or stored on site exceeds 0.02 percent by weight.

3.2 *DOSE CALCULATION SPREADSHEET*

As noted in Procedure RH-300, committed effective dose equivalents (CEDEs) for workers are calculated using International Commission on Radiological Protection (ICRP) 68 effective dose coefficients to demonstrate compliance with the stochastic occupational dose limit of 5 rem per year. The dose limit for individual organs (non-stochastic dose limit) is 50 rem per year (except skin and lens of the eye that have separate non-stochastic dose limits based on external doses). Regulatory guidance notes that organ doses only need to be calculated when the CEDE exceeds 1 rem (Reg Guide 8.34, *Monitoring Criteria and Methods to Calculate Occupational Radiation Doses*, July 1992).

Sample spreadsheets showing the CEDE and Organ Dose Equivalent (ODE) calculations for 2001 are attached. Tables 2 and 3 describe the columns in the spreadsheets.

This procedure and the spreadsheets use the terminology specified in the regulations (i.e. terms associated with ICRP 30 dose coefficients) even though the more recent ICRP dose calculation system uses slightly different terminology. Table 4 gives a glossary of equivalent terms.

Table 1: Inhalation Intake Estimates Spreadsheet Explanation

Col. A	Employee ID #
Col. B	Employee Name/Job Assignment
Col. C	Work Area #
Col. D	Sample Collection Date
Col. E	Sample ID – multiple numbers indicate a composite sample
Col. F	Natural Uranium concentration (S, formerly Y) – Air Sample Results (μCi/ml)
Col. G	Radium-226 concentration (M, formerly W) – Air Sample Results (μCi/ml)
Col. H	Thorium-230 concentration (S, formerly Y) – Air Sample Results (μCi/ml)
Col. I	Thorium-232 Concentration (S, formerly Y) – Air Sample Results (μCi/ml)
Col. J	Uranium (F, formerly D) concentration – Air Sample Results (μCi/ml)
Col. K	Gross Alpha Concentration – Air Sample Results (μCi/ml)
Col. L	Method Code
Col. M	BZ Sample VOL – sample volume (liters)
Col. N	BZ Sample MIN – sample minutes = Col. M/2 lpm
Col. O	EXPOSURE HRS – sample hours = Col. N/60 min/hr or Assigned exposure hours
Col. P	Liters Breathed = Col. O x 1200 l/hr (breathing rate)
Col. Q	PF – protection factor assigned to respiratory protection
Col. R	Note – explanation of source of respiratory protection information (i.e. work log or radiation safety technician)
Col. S	Hours worked – Hours worked during quarter based on work records. This is used as a cross check to verify totals assigned to exposure
Col. T	Estimated inhalation intake - μCi U-nat class S (formerly Y) = Col. P x Col. F x 1000 ml/l Total = sum of Col. T
Col. U	Estimated inhalation intake - μCi Ra-226 Class M (formerly W) = Col. P x Col. G x 1000 ml/l Total = sum of Col. U
Col. V	Estimated inhalation intake - μCi Th-230 Class S (formerly Y) = Col. P x Col. H x 1000 ml/l Total = sum of Col. V
Col. W	Estimated inhalation intake - μCi Th-232 Class S (formerly Y) = Col. P x Col. I x 1000 ml/l Total = sum of Col. W
Col. X	Estimated inhalation intake - μCi U-nat Class F (formerly D) = Col. P x Col. J x 1000 ml/l Total = sum of Col. X
Col. Y	Estimated inhalation intake of soluble uranium – mg U-nat Class F (formerly D) Intake = Col. X /[(6.77E-7 μCi/μg)(1000 μg/mg)]
Col. Z	Estimated total intake of equilibrium equivalent Rn-222 – entered from the WL spreadsheet
Col. AA	Estimated intake of equilibrium equivalent Rn-222 by quarter – entered from WL spreadsheet

Table 2: Total Summary Table Explanation

Column	Content	Explanation (Reference to Intake Spreadsheet Column in Italics)
Col. A	Emp. #	
Col. B	Emp. Name	
Col. C	Job Description	
Col. D	Hours worked	
Col. E	U-238 (Y) intake	Total *Col. T*/2 Performed to apportion equal amounts of U-nat (S, formerly Y)to U-238 and U-234
Col. F	U-234 (Y) intake	Total *Col. T*/2 Performed to apportion equal amounts of U-nat (S, formerly Y) to U-238 and U-234
Col. G	U-238 (D) intake	Total *Col. X*/2 Performed to apportion equal amounts of U-nat (F, formerly D) to U-238 and U-234
Col. H	U-234 (D) intake	Total *Col. X*/2 Performed to apportion equal amounts of U-nat (F, formerly D) to U-238 and U-234
Col. I	Ra-226 (W) intake	Total *Col. U*
Col. J	Th-230 (Y) intake	Total *Col. V*
Col. K	Th-232 (Y) intake	Total *Col. W*
Col. L	Rn-222 Exposure (WLM)	*Total Col Z*
Col. M	Rn-220 Exposure (WLM)	[Rn-220 WL (from Rn-220 data sheet)][2000 hr/y/170 h/M]
Col. N	Rn-222 CEDE	[Col. L (Rn-222 intake)][5 rem/4 WLM]
Col. O	Rn-220 effective dose	[Col. M(Rn-220 intake)][5 rem/12 WLM]
Col. P	Total CEDE	Sum[Total intake for nuclide I x Effective Dose Coefficient from ICRP 68 spreadsheet for nuclide I] + Col. N + Col. O
Col. Q	DDE	From Vendor TLD report
Col. R	TEDE	Col. P + Col. Q
Col. S	U intake (mg)	[Col. E + Col. F + Col. G + Col. H]/[6.67E-4 μCi/mg]
Col. T	Class D U intake (mg)	[Col. G + Col. H]/[6.67E-4 μCi/mg]
Col. U	U intake (mg/wk)	From Intake Calculation Spreadsheet *Col. Y*

Table 3: Organ Dose Calculation Spreadsheet Explanation

Column	Content	Explanation (Reference to Intake Spreadsheet Column in Italics)
Col. A	Emp. #	
Col. B	Emp. Name	
Col. C	Job Descrip.	
Col. D	Hours worked	
Col. E	U-238 (Y) intake	Total *Col. T*/2, Performed to apportion equal amounts of U-nat (S, formerly Y)to U-238 and U-234
Col. F	U-234 (Y) intake	Total *Col. T*/2, Performed to apportion equal amounts of U-nat (S, formerly Y) to U-238 and U-234
Col. G	U-238 (D) intake	Total *Col. X*/2, Performed to apportion equal amounts of U-nat (F, formerly D) to U-238 and U-234
Col. H	U-234 (D) intake	Total *Col. X*/2, Performed to apportion equal amounts of U-nat (F, formerly D) to U-238 and U-234
Col. I	Ra-226 (W) intake	Total *Col. U*
Col. J	Th-230 (Y) intake	Total *Col. V*
Col. K	Th-232 (Y) intake	Total *Col. W*
Col. L	Rn-222 intake	Total *Col Z*
Col. M	Rn-220 intake	[Rn-220 WL (from thoron data sheet)][2000 hr/y/170 h/M]
Col. N	Rn-222 CEDE	[Col. L][5 rem/4 WLM]
Col. O	Rn-220 CEDE	[Col. M][5 rem/12 WLM]
Col. P	Rn-222 Lung Dose	[Col. N]/0.12 Note: the organ (tissue) weighting factor for the lung is 0.12
Col. Q	Rn-220 Lung Dose	[Col. O]/0.12
Col. R	Total Bone Dose	Sum[Total intake for nuclide I x Bone Dose Coefficient from ICRP 68 spreadsheet for nuclide I]
Col. S	Total Lung Dose	Sum[Total intake for nuclide I x Lung Dose Coefficient from ICRP 68 spreadsheet for nuclide I] + Col. P + Col. Q

Table 4: ICRP Terminology

Conventional Terminology	Equivalent ICRP 68 Terminology
Committed Effective Dose Equivalent (CEDE)	Effective Dose (ED)
Organ Dose Equivalent (ODE)	Committed Equivalent Organ Dose
Solubility Class Y (years)	Lung Clearance Class S (slow)
Solubility Class W (weeks)	Lung Clearance Class M (moderate)
Solubility Class D (days)	Lung Clearance Class F (fast)
Dose Conversion Factor	Dose Coefficient

Determination That Sample Is an Outlier

Sample ID_____

Type of sample_____

Location_____

Date of Sample Collection_____

Gross alpha concentration _____
or
U-nat concentration_____

Reason for determination that sample is an outlier:

_____

_____

_____

_____

_____

Dose Calculation Spreadsheets Procedure RH-303

Appendix A

Example Total Summary Table

Example Total Summary Table

Job Description	Hours Worked	U-238 uCi (Y)	U-234 uCi (Y)	U-238 uCi (D)	U-234 uCi (D)	Ra-226 uCi (W)	Th-230 uCi (Y)	Th-232 uCi (Y)	Rn-222 WLM	Rn-220 WLM
Analyst	2941.75	0.00037	0.00037	5E-07	5E-07	0.00068	0.00072	0.00024	0.1589	0.00588
Analyst	1344.5	0.00019	0.00019	0	0	0.00019	0.0001	8.8E-06	0.0565	0.00588
Lab Manager	2762.5	0.00021	0.00021	5E-06	5E-06	0.00021	0.00014	1E-05	0.1309	0.00588
Analyst	3049	0.00037	0.00037	1E-06	1E-06	0.00065	0.00068	0.00022	0.1229	0.00588
Operator	1375	0.00029	0.00029	2E-05	2E-05	0.00077	0.00032	7.9E-05	0.0597	0.00588
Utility-LC	2117	0.00038	0.00038	6E-06	6E-06	0.00045	0.0002	1.4E-05	0.0757	0.00588
General Mill Foreman	2717	0.0003	0.0003	0.0002	0.0002	0.00033	0.00016	3.2E-05	0.1523	0.00588
Utillity-LL	2173	0.00049	0.00049	4E-05	4E-05	0.00044	0.00019	8.2E-06	0.0855	0.00588
QA/QC Coord.	2229.75	0.00012	0.00012	3E-08	3E-08	0.00011	0.00011	1E-05	0.088	0.00588
Maint. Foreman	2083.25	0.00024	0.00024	0.0001	0.0001	0.00028	0.00017	3.5E-05	0.099	0.00588
Sr RST	2179.5	0.00032	0.00032	3E-06	3E-06	0.00032	9.5E-05	1.8E-05	0.086	0.00588
Foreman	2229.25	0.00031	0.00031	0	0	0.00031	0.00023	1.3E-05	0.1059	0.00588
Accounting Clerk	2253	0.00011	0.00011	0	0	0.00011	0.00011	9.4E-06	0.1035	0.00588
RST	2029	0.00019	0.00019	6E-08	6E-08	0.00021	0.00024	2.6E-05	0.0765	0.00588
SST	1962	0.00017	0.00017	0	0	0.00017	7.1E-05	6E-06	0.0718	0.00588
SST	1786	0.00023	0.00023	0	0	0.00023	8.5E-05	5.6E-06	0.0662	0.00588
EC/RSO	2559	0.00015	0.00015	3E-06	3E-06	0.00016	0.00013	1.5E-05	0.1121	0.00588
Sr RST	2485	0.00051	0.00051	0.0002	0.0002	0.00041	0.00018	2.8E-05	0.1235	0.00588
HSS	2378	0.00015	0.00015	7E-06	7E-06	0.00016	0.00014	1.2E-05	0.1161	0.00588
Maintenance	1450.5	0.00027	0.00027	8E-05	8E-05	0.00037	0.00015	1E-05	0.0637	0.00588
SST	1929	0.00017	0.00017	0	0	0.00017	6.7E-05	5.7E-06	0.0627	0.00588
Shift Foreman	1951.25	0.0002	0.0002	0.0003	0.0003	0.00028	0.00012	2.5E-05	0.1062	0.00588
Analyst	2534	0.00033	0.00033	0	0	0.0006	0.00062	0.0002	0.1064	0.00588
SF/met tech	2146.5	0.00056	0.00056	8E-05	8E-05	0.00045	0.00022	1E-05	0.1056	0.00588
Op-ADU/SF	2215	0.00112	0.00112	0.0006	0.0006	0.00042	0.00023	1.6E-05	0.0837	0.00588
SST	1928.5	0.00018	0.00018	0	0	0.00018	6.8E-05	5.8E-06	0.0741	0.00588
Maintenance	2008.5	0.00028	0.00028	5E-05	5E-05	0.00029	0.00016	3.5E-05	0.0793	0.00588
Maint. Foreman	2205.25	0.00033	0.00033	3E-05	3E-05	0.00039	0.00019	4.1E-05	0.0847	0.00588
Warehouse Superv.	2445	0.00013	0.00013	0	0	0.00013	0.00013	1.3E-05	0.2472	0.00588
Analyst	2151.3	0.0003	0.0003	3E-05	3E-05	0.00051	0.0005	0.00016	0.0919	0.00588
Instrument	1710	0.00019	0.00019	0	0	0.00019	0.00019	1.6E-05	0.0683	0.00588
Utility/Crusher	1037	0.00013	0.00013	2E-06	2E-06	0.00013	5.5E-05	4.5E-06	0.0358	0.00588

Example Total Summary Table (continued)

Rn-222 CEDE rem	Rn-220 CEDE rem	CEDE rem	DDE rem	TEDE rem	U mg	Uranium mg (D)	Uranium mg/wk		
0.199	0.0025	0.2858	0.082	0.368	0.001	0.001455	2.91E-05		
0.071	0.0025	0.0925	0.082	0.174	0	0	0	**AVG TEDE**	
0.164	0.0025	0.1883	0.087	0.275	0.014	0.013666	0.000273	0.260201949	Full-time
0.154	0.0025	0.237	0.09	0.327	0.003	0.003323	6.65E-05	**MAX TEDE**	
0.075	0.0025	0.1253	0.075	0.2	0.069	0.069231	0.001385	0.476007354	All Staff
0.095	0.0025	0.1353	0.197	0.332	0.019	0.019192	0.000384	**Median TEDE**	
0.19	0.0025	0.2255	0.1	0.325	0.554	0.55386	0.011077	0.262996899	Full-time
0.107	0.0025	0.1532	0.205	0.358	0.107	0.107383	0.002148		
0.11	0.0025	0.1267	0.086	0.213	9E-05	8.52E-05	1.7E-06		
0.124	0.0025	0.1548	0.086	0.241	0.325	0.32549	0.00651		
0.108	0.0025	0.1385	0.101	0.239	0.01	0.00986	0.000197		
0.132	0.0025	0.1687	0.148	0.317	0	0	0		
0.129	0.0025	0.1457	0.115	0.261	0	0	0		
0.096	0.0025	0.1253	0.121	0.246	2E-04	0.000177	3.54E-06		
0.09	0.0025	0.1081	0.113	0.221	0	0	0		
0.083	0.0025	0.1058	0.094	0.2	0	0	0		
0.14	0.0025	0.1604	0.077	0.237	0.01	0.010238	0.000205		
0.154	0.0025	0.2038	0	0.204	0.67	0.670352	0.013407		
0.145	0.0025	0.1661	0.097	0.263	0.022	0.021956	0.000439		
0.08	0.0025	0.1107	0.126	0.237	0.228	0.228175	0.004564		
0.078	0.0025	0.0966	0.094	0.191	0	0	0		
0.133	0.0025	0.1594	0.126	0.285	0.744	0.744151	0.014883		
0.133	0.0025	0.2092	0.099	0.308	0	0	0		
0.132	0.0025	0.1847	0.115	0.3	0	0.246234	0.004925		
0.105	0.0025	0.1925	0.161	0.353	0	1.837194	0.036744		
0.093	0.0025	0.1112	0.114	0.225	0	0	0		
0.099	0.0025	0.1318	0.108	0.24	0	0.15167	0.003033		
0.106	0.0025	0.1452	0.104	0.249	0	0.097832	0.001957		
0.309	0.0025	0.3282	0.09	0.418	0	0	0		
0.115	0.0025	0.1784	0.096	0.274	9E-04	0.078779	0.001576		
0.085	0.0025	0.1116	0.025	0.137	0.001	0	0		
0.045	0.0025	0.0591	0.073	0.132	2E-04	0.006075	0.000121		

Dose Calculation Spreadsheets Procedure RH-303

Appendix B

Example Organ Dose Calculation Spreadsheet

Example Organ Dose Calculation Spreadsheet

Empl #	Name	Job Description	Hours Worked	U-238 uCi (Y)	U-234 uCi (Y)	U-238 uCi (D)	U-234 uCi (D)	Ra-226 uCi (W)	Th-230 uCi (Y)	Th-232 uCi (Y)	Rn-222 WLM	Rn-220 WLM
1		Analyst	2941.75	0.00037	0.00037	4.9E-07	4.9E-07	0.00068	0.00072	0.0002	0.1589	0.00588
2		Analyst	1344.5	0.00019	0.00019	0	0	0.00019	0.0001	9E-06	0.0565	0.00588
3		Lab Manager	2762.5	0.00021	0.00021	4.6E-06	4.6E-06	0.00021	0.00014	1E-05	0.1309	0.00588
4		Analyst	3049	0.00037	0.00037	1.1E-06	1.1E-06	0.00065	0.00068	0.0002	0.1229	0.00588
5		Operator	1375	0.00029	0.00029	2.3E-05	2.3E-05	0.00077	0.00032	8E-05	0.0597	0.00588
6		Utility-LC	2117	0.00038	0.00038	6.5E-06	6.5E-06	0.00045	0.0002	1E-05	0.0757	0.00588
7		General Mill Foreman	2717	0.0003	0.0003	0.00019	0.00019	0.00033	0.00016	3E-05	0.1523	0.00588
8		Utillity-LL	2173	0.00049	0.00049	3.6E-05	3.6E-05	0.00044	0.00019	8E-06	0.0855	0.00588
9		QA/QC Coord.	2229.75	0.00012	0.00012	2.9E-08	2.9E-08	0.00011	0.00011	1E-05	0.088	0.00588
10		Maint. Foreman	2083.25	0.00024	0.00024	0.00011	0.00011	0.00028	0.00017	3E-05	0.099	0.00588
11		Sr RST	2179.5	0.00032	0.00032	3.3E-06	3.3E-06	0.00032	9.5E-05	2E-05	0.086	0.00588
12		Foreman	2229.25	0.00031	0.00031	0	0	0.00031	0.00023	1E-05	0.1059	0.00588
13		Accounting Clerk	2253	0.00011	0.00011	0	0	0.00011	0.00011	9E-06	0.1035	0.00588
14		RST	2029	0.00019	0.00019	6E-08	6E-08	0.00021	0.00024	3E-05	0.0765	0.00588
15		SST	1962	0.00017	0.00017	0	0	0.00017	7.1E-05	6E-06	0.0718	0.00588
16		SST	1786	0.00023	0.00023	0	0	0.00023	8.5E-05	6E-06	0.0662	0.00588
17		EC/RSO	2559	0.00015	0.00015	3.5E-06	3.5E-06	0.00016	0.00013	2E-05	0.1121	0.00588
18		Sr RST	2485	0.00051	0.00051	0.00023	0.00023	0.00041	0.00018	3E-05	0.1235	0.00588
19		HSS	2378	0.00015	0.00015	7.4E-06	7.4E-06	0.00016	0.00014	1E-05	0.1161	0.00588
20		Maintenance	1450.5	0.00027	0.00027	7.7E-05	7.7E-05	0.00037	0.00015	1E-05	0.0637	0.00588
21		SST	1929	0.00017	0.00017	0	0	0.00017	6.7E-05	6E-06	0.0627	0.00588
22		Shift Foreman	1951.25	0.0002	0.0002	0.00025	0.00025	0.00028	0.00012	3E-05	0.1062	0.00588
23		Analyst	2534	0.00033	0.00033	0	0	0.0006	0.00062	0.0002	0.1064	0.00588
24		SF/met tech	2146.5	0.00056	0.00056	8.3E-05	8.3E-05	0.00045	0.00022	1E-05	0.1056	0.00588
25		Op-ADU/SF	2215	0.00112	0.00112	0.00062	0.00062	0.00042	0.00023	2E-05	0.0837	0.00588
26		SST	1928.5	0.00018	0.00018	0	0	0.00018	6.8E-05	6E-06	0.0741	0.00588
27		Maintenance	2008.5	0.00028	0.00028	5.1E-05	5.1E-05	0.00029	0.00016	3E-05	0.0793	0.00588
28		Maint. Foreman	2205.25	0.00033	0.00033	3.3E-05	3.3E-05	0.00039	0.00019	4E-05	0.0847	0.00588
29		Warehouse Superv.	2445	0.00013	0.00013	0	0	0.00013	0.00013	1E-05	0.2472	0.00588
30		Analyst	2151.3	0.0003	0.0003	2.7E-05	2.7E-05	0.00051	0.0005	0.0002	0.0919	0.00588
31		Instrument	1710	0.00019	0.00019	0	0	0.00019	0.00019	2E-05	0.0683	0.00588
32		Utility/Crusher	1037	0.00013	0.00013	2.1E-06	2.1E-06	0.00013	5.5E-05	5E-06	0.0358	0.00588

Example Organ Dose Calculation Spreadsheet (continued)

Rn-222 CEDE	Rn-220 CEDE	Rn-222 Lung DE	Rn-220 Lung DE	Total Bone DE	Total Lung DE
rem	rem	rem	rem	rem	rem
0.19857	0.00245	1.65473	0.020425	0.9463232	2.2377933
0.07059	0.00245	0.58822	0.020425	0.1152485	0.7526974
0.16368	0.00245	1.36398	0.020425	0.1541625	1.5458005
0.15367	0.00245	1.28058	0.020425	0.8902153	1.8396989
0.07466	0.00245	0.62219	0.020425	0.4123341	0.9799667
0.09463	0.00245	0.78858	0.020425	0.2242548	1.0925034
0.19033	0.00245	1.58605	0.020425	0.2135364	1.8421811
0.10691	0.00245	0.89093	0.020425	0.2047313	1.2444673
0.11004	0.00245	0.91696	0.020425	0.1157994	1.0385645
0.12375	0.00245	1.03121	0.020425	0.2109103	1.2555285
0.10756	0.00245	0.89632	0.020425	0.1206037	1.1349008
0.13242	0.00245	1.10348	0.020425	0.2436997	1.3689431
0.12933	0.00245	1.07775	0.020425	0.1192049	1.1957153
0.09557	0.00245	0.79638	0.020425	0.2636143	1.0041016
0.08976	0.00245	0.74801	0.020425	0.0802286	0.8885285
0.08278	0.00245	0.68984	0.020425	0.0955675	0.8669682
0.14011	0.00245	1.16754	0.020425	0.1412058	1.3148519
0.15435	0.00245	1.28624	0.020425	0.234432	1.6576812
0.14518	0.00245	1.20981	0.020425	0.149032	1.3611882
0.07963	0.00245	0.6636	0.020425	0.1728501	0.8946159
0.07841	0.00245	0.65345	0.020425	0.0761192	0.793069
0.13281	0.00245	1.10672	0.020425	0.169032	1.2976349
0.13304	0.00245	1.10863	0.020425	0.8142698	1.6196763
0.13201	0.00245	1.10008	0.020425	0.2481707	1.4993469
0.10466	0.00245	0.87216	0.020425	0.2935905	1.5453229
0.09257	0.00245	0.7714	0.020425	0.0775499	0.9147726
0.09912	0.00245	0.826	0.020425	0.1983085	1.0669183
0.10585	0.00245	0.8821	0.020425	0.2380971	1.1722099
0.30899	0.00245	2.57489	0.020425	0.1447316	2.7132428
0.11488	0.00245	0.95733	0.020425	0.6487653	1.3871909
0.08541	0.00245	0.71179	0.020425	0.2042732	0.8993606
0.0448	0.00245	0.37332	0.020425	0.0618068	0.4828031

1.0 PURPOSE

This procedure provides for prenatal radiation exposure training of women of reproductive age, for radiation monitoring of declared pregnant women, and for the calculation of dose to the embryo-fetus.

2.0 APPLICABILITY

This procedure applies to women of reproductive age who are Energy Fuels employees or contractors working at the Piñon Ridge Mill.

3.0 OTHER DOCUMENTS

3.1 *REFERENCES*

3.1.1 Colorado Radiation Control Regulations (6 CCR 1007-1) Part 4 Standards for Protection Against Radiation.

3.1.2 CDPHE Part 10 Notices Instructions and Reports to Workers.

3.1.3 U.S. Nuclear Regulatory Commission Regulatory Guide 8.13, Instruction Concerning Prenatal Radiation Exposure, Revision 3, June 1999.

3.2 *APPENDICES*

Appendix A - Energy Fuels Resources Policy on Declared Pregnant Women

Appendix B - Declaration of Pregnancy Form RH-310A

4.0 EQUIPMENT AND MATERIALS

4.1 *MATERIALS*

4.1.1 Regulatory Guide 8.13 or the equivalent.

5.0 RESPONSIBILITY

5.1 The Radiation Safety Officer (RSO) or Assistant RSO is responsible for:

5.1.1 Providing Prenatal Radiation Exposure Training

5.1.2 Calculating the dose to the embryo-fetus

5.1.3 Recording the radiation dose to declared pregnant women and the embryo-fetus

5.2 The Radiation/Security Technician (RST) is responsible for:

5.2.1 Radiation monitoring of declared pregnant women

5.2.2 Filing and retain Form RH-310A

5.2.3 Filing and retain the radiation doses for the embryo-fetus

6.0 PREREQUISITE INFORMATION

6.1 *DEFINITIONS*

6.1.1 CEDE - Committed Effective Dose Equivalent.

6.1.2 TEDE - Total Effective Dose Equivalent.

6.1.3 DDE - Deep Dose Equivalent.

APPROVALS	*Signature*	*Date*
RSO		
Plant Manager		

6.1.4 Declared Pregnant Woman - Woman who has voluntarily informed Energy Fuels in writing of her pregnancy and the estimated date of conception.

6.2 ***FREQUENCY***

6.2.1 One-time training of females of reproductive age.

6.2.2 Upon receipt of declaration of pregnancy provide additional radiation monitoring.

6.2.3 During and at the end of a pregnancy of a radiation worker the RSO or Assistant RSO calculates the radiation dose to the embryo-fetus.

7.0 PROCEDURES

7.1 The RSO or Assistant RSO shall provide prenatal radiation exposure instruction to all females of reproductive age who are likely in a year to receive an occupational dose in excess of 0.1 rem (6 CCR 1007-1 10.3). The instruction may include providing each trainee a copy of the U.S. Nuclear Regulatory Commission Regulatory Guide 8.13, *Instruction Concerning Prenatal Radiation Exposure,* and/or a copy of this procedure. Classroom or individual instruction, and/or videotaped presentations may be used in place of the written materials. The RSO or Assistant RSO may be contacted by the declared pregnant woman for additional information. Training will be documented on Form RH-010A which will be filed and retained.

7.2 After a woman who receives occupational radiation exposures at the mill knows she is pregnant, she is requested, but not required, to declare her pregnancy on the attached form RH-310A or the equivalent. That declaration must be in writing, contain the estimated date of conception, and be given to the RSO or Assistant RSO. If the woman chooses not to declare her pregnancy, she and her embryo-fetus will continue to be subject to the same radiation dose limits that apply to other occupational workers, i.e. 5 rem/yr.

7.3 The RSO or Assistant RSO shall monitor the radiation exposures of declared pregnant women if they are likely to receive a deep dose equivalent in excess of 0.1 rem.

7.4 The dose to an embryo-fetus during the entire pregnancy due to occupational exposure of a declared pregnant woman shall not exceed 0.5 rem. If the dose to the embryo-fetus is equal to or greater than 0.45 rem before her declaration of pregnancy, then an additional dose may not exceed 0.05 rem for the remainder of the pregnancy (6 CCR 1007-1 4.13).

7.5 The RSO or Assistant RSO are to ensure that the dose to the declared pregnant woman is substantially uniform so that all the 0.5 rem allowed dose does not occur in a short period during the pregnancy. Monthly doses greater than 0.1 rems are to be justified.

7.6 Calculate the dose to the embryo-fetus as follows:

7.6.1 The 9-month gestation internal dose equivalent is calculated by:

$CEDE_g = (I)(TF)(DF)$

Where:

$CEDE_g$= 9-month gestation committed effective dose equivalent
I = Radionuclide intake by the pregnant woman during her pregnancy in µCi.
TF = Transfer fraction
DF = Dose factor in rem/µCi

The transfer fraction and dose factor applicable to an inhalation dose at the Mill were derived using USNRC Regulatory Guide 8.36. They are presented in the following table where one can specify the intakes for each radionuclide, calculate the dose for each radionuclide, and the total 9-month gestation dose equivalent.

9-Month Gestation Dose Equivalent

Radionuclide	Intake in µCi	Transfer fraction	Dose Factor[(a)]	rem/ nuclide
U-234 (S)		7.90E-02	5.84E-01	
U-238 (S)		7.90E-02	5.10E-01	
U-234 (F)		4.88E-01	5.84E-01	
U-238 (F)		4.88E-01	5.10E-01	
Ra-226 (M)		2.22E-01	1.69E+00	
Th-230 (S)		5.01E-02	1.26E+01	
Th-232 (S)		5.01E-02	2.26E+01	
			TOTAL DOSE	

Where:
(S) = Slow absorption
(M) = Moderate absorption
(F) = Fast absorption
(a) = ICRP 68 dose factors.

7.6.2 The <u>pre-existing body-burden dose equivalent</u> from embryo-fetal burdens existing at the time of conception is calculated by:

$CEDE_c = (SA)(DF)$

Where:

$CEDE_c$ = The committed effective dose equivalent from embryo-fetal burdens at the time of conception

SA = The nuclide burden existing at time of conception in μCi for each radionuclide

DF = The dose factors listed in the above table.

7.6.3 The Deep Dose Equivalent (DDE) for the embryo-fetus is the whole-body dose to the pregnant woman and the embryo fetus during her pregnancy, usually measured using Thermoluminescent Dosimeters (TLD) or Optically Stimulated Luminescence Dosimeters (OSL).

7.6.4 The Total Effective Dose Equivalent (TEDE) to the embryo-fetus from mill operations is the 9-month gestation dose minus the preexisting body-burden dose, plus the deep dose equivalent as shown below:

$$TEDE = CEDE_g - CEDE_c + DDE$$

7.7 The records of the dose to the embryo-fetus are to be kept by the Assistant RSO and retained.

Pregnant Women Procedure RH-310

Appendix A

Energy Fuels Resources Policy on Declared Pregnant Women Form RH-310A

Energy Fuels Resources Policy
On
Declared Pregnant Women

June 2009

Energy Fuels Resources will limit the occupational radiation exposures to the embryo-fetus of declared pregnant women to 0.5 rem (5 mSv) for their entire pregnancy. A declared pregnant woman is a woman who has voluntarily informed Energy Fuels in writing of her pregnancy and the estimated date of conception. Energy Fuels will avoid substantial variation above a uniform monthly dose rate during the pregnancy so that all or most of the 0.5 rem allowed dose would not occur in a short period during the pregnancy. The declared pregnant woman may be reassigned to a job involving a lower radiation exposure during her pregnancy than was received on her job before her declaration of pregnancy. The written declaration of pregnancy is to be given to Energy Fuels Radiation Safety Officer or Assistant Radiation Safety Officer.

_____
Vice President of Regulatory Affairs

Pregnant Women Procedure RH-310

Appendix B

Declaration of Pregnancy Form RH-310B

Declaration Of Pregnancy Form

To: Energy Fuels Radiation Safety Officer or Assistant Radiation Safety Officer

In accordance with Colorado CDPHE Radiation Control Regulations RH 4.13, "Dose to an Embryo/Fetus," I declare that I am pregnant. I believe I became pregnant in

_____ _____.
month year

I understand the radiation dose to my embryo-fetus during my entire pregnancy will not be allowed to exceed 0.5 rem (5 millisievert) unless that dose has already been exceeded between the time of conception and submitting this letter. I also understand that meeting the lower dose limit may require a change in job or job responsibilities during my pregnancy.

_____
(Your Signature)

_____
(Your Name Printed)

_____
(Date)