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HEALTH PHYSICS SURVEYS IN URANIUM RECOVERY FACILITIES

A. INTRODUCTION

This guide is being revised to describe health physics surveys that are acceptable to the NRC staff for protecting workers at uranium recovery (UR) facilities (e.g., uranium mills, in-situ leach (ISL) facilities, ion exchange recovery facilities, heap leach facilities) from radiation and the chemical toxicity of uranium while on the job. The guidance can also be applied, in part, to other types of UR facilities and portions of conversion facilities since some of the processes used in these facilities are similar to those in UR facilities.

Section 40.32, "General Requirements for Issuance of Specific Licenses," of 10 CFR Part 40, "Domestic Licensing of Source Material," indicates that the NRC will approve an application to operate a UR facility (e.g., uranium milling, uranium hexafluoride facility) if the applicant is qualified by reason of training and experience to be able to protect health and minimize danger to life and property and if the applicant's proposed equipment, facilities, and procedures are also adequate.

The following sections of the NRC's regulations in 10 CFR Part 20, "Standards for Protection Against Radiation," deal with the protection of workers: § 20.1501 requires adequate surveys, § 20.1201provides occupational dose limits for adults, § 20.1208 provides dose limits for declared pregnant women, § 20.1502 requires personnel radiation dosimeters in certain instances, § 20.1902 requires posting of warning signs, § 20.1602 requires controlling access to areas with high radiation levels,

§ 20.2106 requires records of radiation surveys and personnel monitoring reports, and § 20.2203 requires reports of over exposures.

This guide does not cover surveys to prevent the release of radioactive material to unrestricted areas or surveys to measure the exposure of the public to radioactive materials in effluents, except for surveys of the skin and clothing of workers leaving the UR facility and surveys of equipment and packages leaving the UR facility.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Parts 20, which were approved by the Office of Management and Budget, approval number 3150-0014. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

B. DISCUSSION

Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills" (Ref. 1), outlines the type of information that applicants for a UR facility license should include in their applications and suggests a uniform format for presenting that information. This regulatory guide describes occupational health physics (radiation protection) surveys acceptable to the NRC licensing staff that an applicant may use for describing surveys in Section 5.5, "Radiation Safety," of Regulatory Guide 3.5. Also see Regulatory Guide 3.46, “Standard Format and Content of License Applications, Including Environmental Reports, for In Situ Uranium Solution Mining” (Ref. 2).

The contents of this guide conform with NRC's current licensing practice. The contents of this guide are also based to a large extent on the International Atomic Energy Agency (IAEA) "Manual of Radiological Safety in Uranium and Thorium Mines and Mills” (Ref. 3).

Respiratory protection, uranium bioassay, and programs for maintaining occupational exposures to radiation as low as reasonably achievable are not included in this guide. Those subjects are covered in Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection” (Ref. 4); Regulatory Guide 8.22, "Bioassay at Uranium Mills" (Ref. 5); American National Standard HPS N13.22-1995, ”Bioassay Programs for Uranium” (Ref. 6); American National Standard HPS N13.30-1996, “Performance Criteria for Radiobioassay” (Ref. 7); and Regulatory Guide 8.31, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Is Reasonably Achievable” (Ref. 8).

C. REGULATORY POSITION

1. OCCUPATIONAL DOSE LIMITS

In 10 CFR Part 20, "Standards for Protection Against Radiation," 10 CFR 20.1201 establishes radiation dose limits for occupationally exposed adults. These dose limits apply to the sum of the dose received from external exposure and the dose from internally deposited radioactive material. These dose limits are summarized in Table 1. The occupational dose limits for minors according to 10 CFR 20.1207 are 10% of the dose limit for adults, and 10 CFR 20.1208 establishes a dose limit for the embryo/fetus of 0.5 rem (0.005 Sv) during the entire declared pregnancy.

The "total effective dose equivalent" is defined as the sum of the "deep-dose equivalent" (for external exposures) and the "committed effective dose equivalent" (for internal exposures). The limit of 50 rems (0.5 Sv) specified in 10 CFR 20.1201(a)(1)(ii) applies to the sum of the "deep-dose equivalent" and the "committed dose equivalent" to any individual organ or tissue other than the lens of the eyes. The requirements in 10 CFR 20.1202 are for summing external and internal doses if the licensee is required to monitor under both 10 CFR 20.1502(a) and (b) to demonstrate compliance with the dose limits of 10 CFR 20.1201.

In addition to these limits, 10 CFR 20.1201(e) establishes a limit for the intake of soluble uranium of 10 milligrams per week, based on chemical toxicity to the kidney.

The Part 20 requirements for recording individual monitoring results are in 10 CFR 20.2106. When monitoring is required under 10 CFR 20.1502, the monitoring results must be recorded on NRC Form 5 or its equivalent according to 10 CFR 20.2106(c).

TABLE 1
Dose Limits and Associated Terminology

Type of Exposure	10 CFR Part 20 Designation	Dose Limit
Total Whole Body Dose (Sum of External and Internal)	Total Effective Dose Equivalent (TEDE) TEDE = DDE + CEDE	5 rem/year
External Dose	Deep Dose Equivalent (DDE)	(a)
Internal Whole Body Dose	Committed Effective Dose Equivalent (CEDE)	(a)
Total Organ Dose (Sum of External and Internal)	Total Organ Dose Equivalent (TODE) TODE = DDE + CDE	50 rem/year
Internal Organ Dose	Committed Dose Equivalent (CDE)	(a)
Skin Dose	Shallow Dose Equivalent (SDE), Skin of Whole Body	50 rem/year
Extremity Dose	Shallow Dose Equivalent (SDE), Maximum Extremity	50 rem/year
Eye Dose	Eye Dose Equivalent to Lens of the Eye (LDE)	15 rem/year

(a) Included in limits for whole body and individual organs. In the absence of any internal exposure, external dose is limited to 5 rem per year. In the absence of any external exposure, internal exposure is limited to 2000 DAC-hours per year or 1 annual limit on intake (ALI) (50 rem/yr non-stochastic, 5 rem/yr stochastic).

2. SURVEYS

2.1 Surveys for Airborne Uranium Ore Dust

Surveys for airborne uranium ore dust are necessary to (1) demonstrate compliance with the occupational dose limits for workers specified in 10 CFR 20.1201, (2) meet the posting requirements for airborne radioactivity areas in 10 CFR 20.1902(d), (3) determine whether precautionary procedures such as process or other engineering controls, increased surveillance, limitation on exposure times, use of respiratory protection equipment, or other precautions should be considered to meet 10 CFR 20.1701 and 20.1702, and (4) determine whether exposures to radioactive materials are being maintained as low as is reasonably achievable as stated in 10 CFR 20.1101 and 20.1702.

The Derived Air Concentration (DAC) applicable to limiting exposure to airborne uranium ore dust in restricted areas is given in paragraph 3 of the Note to Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," of 10 CFR Part 20. If gross alpha counting of the air sample is performed, the DAC value is 6×10^{-11} microcuries (μCi) of alpha activity per milliliter (ml) of air. This concentration applies to the alpha emissions of uranium-238, uranium-234, thorium-230, and radium-226. If chemical separation of uranium followed by alpha counting, alpha spectrometry, or fluorometric procedures are used to determine the uranium concentration alone, the DAC value is 3×10^{-11} μCi of natural uranium per ml of air. In mass units, the concentration is 45 micrograms (μg) of natural uranium per cubic meter of air.[1] The uranium ore dust concentration is applicable to areas where ore is handled prior to chemical separation of the uranium from the ore. Where the ore crushing and grinding circuits, chemical leaching areas, and yellowcake areas are physically isolated from each other, the ore dust concentration obviously applies to the ore handling areas.

Where ore handling and yellowcake processing are not physically isolated from each other, the concentration value of 6×10^{-11} μCi/ml may be used provided that gross alpha counting is performed. For other methods of analysis that include only measurements of uranium, it is necessary to determine the fraction of the alpha activity that is due to ore dust. For example, in a UR facility that produces little ore dust because it has a wet ore grinding process but has significant emissions from yellowcake processing equipment, the natural uranium concentration of 3×10^{-11} μCi of natural uranium per ml of air may be applicable throughout the plant. If uranium ore dust concentrations are below 10% of the applicable concentration value in Appendix B to Part 20 (i.e., below 3×10^{-12} μCi/ml), uranium ore dust may be considered to be not present, and the appropriate value for natural uranium (3×10^{-11} μCi/ml) may be used instead. If ore dust concentrations exceed 10% of the Appendix B value, the airborne mixture may either be considered entirely ore dust (for which the concentration value of 6×10^{-11} μCi/ml applies) or a new concentration value for the mixture, DAC_m, may be calculated using Equation 1.

[1] Micrograms of uranium can be converted to microcuries by using the specific activity of natural uranium: 6.77×10^{-7} mCi/mg.

$$DAC_m = \left[\frac{f_{nu}}{DAC_{nu}} + \frac{f_{od}}{DAC_{od}} \right]^{-1} \qquad \text{Equation 1}$$

where

DAC_m = regulatory concentration value for natural uranium

DAC_{od} = regulatory concentration value (in radio-metric units) for natural uranium in ore dust

f_{nu} = fraction of alpha activity from natural uranium as yellowcake, i.e., $C_{nu}/(C_{nu} + C_{od})$. C_{nu}: alpha concentration from natural uranium; C_{od}: alpha concentration from ore dust.

f_{od} = fraction of alpha activity from natural uranium in ore dust, i.e., $C_{od}/(C_{nu} + Co_{od})$

Since this equation would only be used with the 6 x 10^{-11} µCi/ml value of C_{od}, f_{od} is calculated as the fraction of the uranium alpha activity only. This equation was derived from, and is thus equivalent to, the inequality shown in paragraph 1 of the Note to Appendix B to 10 CFR Part 20 (see Appendix A of this guide).

In areas that are not "airborne radioactivity areas," an acceptable sampling program for airborne uranium ore dust includes monthly grab samples of 30 minutes duration in worker-occupied areas while ore is being actively handled. As an alternative, weekly grab samples of 5 minutes duration, each using a high-volume sampler (roughly 30 cfm), are acceptable as long as the licensee can demonstrate that the volume sampled is accurately known. The quantity of air sampled and the method of analysis should allow a lower limit of detection (LLD) of 3 x 10^{-12} µCi of natural uranium per ml of air (or 4.5 µg of uranium per m^3 of air). Appendix B to this guide shows how to calculate the LLD when a fluorometric analysis for uranium is used. If any area is an "airborne radioactivity area," as defined in 10 CFR 20.1003, 30-minute samples should be taken weekly if workers occupy the area. Air samples from outdoor areas such as the ore pad should be collected quarterly.

Only ore dust samples representative of the air inhaled by the workers present are acceptable. Samples taken at a height of about 3 to 6 feet and positioned between the source and the worker are normally considered representative. Samples should be taken while normal ore handling is taking place. The state of operation of major equipment during sampling should be recorded. In large rooms, several locations should be sampled. Special breathing zone sampling (lapel sampling or other sampling of the immediate breathing zone of a particular worker) is not necessary for ore dust; however, it may be warranted in special situations.

During the first year of operation, new UR facilities will need a more extensive air sampling program to determine the locations that provide measurements of the concentration representative of the concentration to which workers are exposed.

Sample analysis should usually be completed within two working days after sample collection. Unusual results should be reported promptly to the Radiation Safety Officer (RSO).[2]

Intake and exposure calculations for ore dust are discussed in Regulatory Position 3 of this guide.

2.2 Surveys for Airborne Yellowcake

It is generally accepted that uranium dissolved in the lung or absorbed by the gastrointestinal tract enters the bloodstream and is distributed to various body organs. The rate of dissolution for yellowcake appears to depend on its temperature history during processing. Yellowcake dried at low temperature, which is predominantly composed of ammonium diuranate, or in the new processes uranyl peroxide, both are more soluble in body fluids than yellowcake dried at higher temperature; and a relatively large fraction is rapidly transferred to kidney tissues (Refs. 9 to 11). If the intake of such yellowcake is controlled to protect the kidney from the chemical toxicity of uranium, radiological protection criteria for natural uranium will also be satisfied. For purposes of compliance with 10 CFR Part 20, yellowcake undried or dried at low temperature should be classified as soluble.

Yellowcake dried at high temperature is a mixture of compounds that contains a major portion of more insoluble uranium oxides. Radiation dose to the lung and other organs is the limiting consideration rather than chemical toxicity; this is primarily due to the large insoluble component. For compliance purposes, yellowcake dried at 400°C (752°F) and above should be classified as insoluble (Refs. 12 and 13).

Thus, surveys for airborne yellowcake are necessary to demonstrate compliance with the occupational dose limits in 10 CFR 20.1201. Surveys are also necessary to establish the boundaries of airborne radioactivity areas and to determine whether surveillance, limitation on working times, provisions of respiratory equipment, or other precautions should be considered in compliance with 10 CFR 20.1701 and 20.1702.

The recommended survey program for yellowcake uses a combination of general air sampling and breathing zone sampling during routine and nonroutine operations that may involve considerable intake, such as those that require a radiation work permit (RWP).

Grab samples for yellowcake with a duration of 30 minutes should be performed weekly in airborne radioactivity areas and monthly in areas not designated as airborne radioactivity areas. As an alternative, weekly grab samples of 5 minutes duration using a high-volume sampler (roughly 30 cfm) are acceptable in areas that are not airborne radioactivity areas instead of monthly 30-minute samples as long as the licensee can demonstrate that the volume of air sampled is accurately known.

Breathing zone sampling for specific jobs should be used to monitor intakes of individual workers doing special high-exposure jobs if the special jobs are likely to involve more than 12

[2] The title "Radiation Safety Officer" is used by many licensees and, in this guide, means the person responsible for conducting health physics survey programs; other titles are equally acceptable.

DAC-hours in any one week. An example of a job during which such breathing zone sampling may be used is maintenance of yellowcake drying and packaging equipment.

Samples should be representative of the air inhaled by the workers. The state of operation of major equipment during sampling should be recorded.

The quantity of air sampled and the method of analysis should allow a lower limit of detection of at least 3×10^{-12} μCi/ml (10% of the Appendix B to 10 CFR Part 20 concentration for natural uranium). Appendix B to this guide shows a calculation of the LLD.

Sample analysis should usually be completed within 2 working days after sample collection to permit prompt corrective action if needed. Unusual results should be reported promptly to the RSO.

2.3 Surveys for Radon-222 and Its Daughters

In UR facilities, significant air concentrations of radon-222 and its daughters may occur near ore storage bins and crushing and grinding circuits or anywhere large quantities of ore are found, particularly dry ore. In addition, any poorly ventilated room can have high radon[3] daughter concentrations even if large quantities of ore are not present.

NRC regulations permit measurements of concentrations of either radon itself or the radon daughters. Thus either type of measurement is acceptable. However, at UR facilities, measurements of daughters are considered by the staff to be more appropriate. Measurements of radon daughter concentrations are more appropriate because radon daughter concentrations are easy to measure and because radon daughter concentrations are the best indicator of worker dose. The dose from radon will be negligible in comparison with the dose from radon daughters (Ref. 14, p. 78, and Ref. 15).

Monthly measurements of radon daughter concentrations should be made where radon daughters routinely exceed 10% of the limit or 0.03 working level above background. If radon daughter concentrations are normally greater than 0.08 working level (25% of limit) or radon concentrations are above 3×10^{-8} μCi/ml, the sampling frequency should be increased to weekly. Sampling should continue to be performed weekly until four consecutive weekly samples indicate concentrations of radon daughters below 0.08 working level or radon below 3×10^{-8} μCi/ml. After that, radon daughter surveys may be resumed on a monthly basis.

Quarterly sampling for radon daughters should be made where previous measurements have shown the daughters are not generally present in concentrations exceeding 0.03 working level (10% of the limit) but where proximity to sources of radon daughters might allow them to be present. For example, quarterly measurements might be appropriate for a shop area attached to the crushing and grinding circuit building.

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[3] The term “radon” used in this guide means “radon-222.”

Radon daughter samples should be representative of worker exposures. Samples should be taken from locations where workers are most often present. The state of operation of major equipment during sampling and the time of day the sample was taken should be recorded.

The LLD for radon daughter measurements should be 0.03 working level. Appendix B to this guide shows how to calculate the LLD for a radon daughter measurement. Measured values less than the LLD, including negative values, should still be recorded on data sheets. The LLD is set high enough to provide a high degree of confidence that 95% of the measured values above the LLD truly represent radon daughters and are not "false positive" values. However, the most accurate average for a sampling location is obtained by averaging all representative values, including values obtained that are below the LLD.

The modified Kusnetz method for measuring radon daughter working levels is a suitable method for UR facilities. The procedure consists of sampling radon daughters on a high-efficiency filter paper for 5 minutes and, after a delay of 40 to 90 minutes, measuring the alpha counts on the filter during a 1-minute interval. The original Kusnetz method measured the alpha count rate. In the modified Kusnetz method, the rate meter is replaced by a scaler. This improves the sensitivity to a practical lower limit of 0.03 working level for a 1-minute count on a 10-liter (0.01 cubic meter) sample. This is about a factor of 10 lower than that originally obtained using the original Kusnetz method. A 4-minute count gives a lower limit of about 0.003 working level (Ref. 3). High-efficiency membrane or glass fiber filters should be used to minimize loss of alpha counts by absorption in the filter. However, a correction factor to account for alpha absorption in the filter paper should still be used. Care should be taken to avoid contamination of the alpha counter.

The modified Kusnetz method is discussed in more detail in References 3 and 16. Other acceptable methods discussed in Reference 2 are the original Kusnetz method with greater than 10 liters of air sampled, the modified Tsivoglou method, and the Rolle method. The modified Tsivoglou method is slightly more accurate but is also more complicated than the modified Kusnetz method. The Rolle method is quicker than the Kusnetz method, but is less sensitive. Alpha spectroscopy yields acceptable results, but the instruments are expensive and fragile and lack portability. The "instant working level" meters are also acceptable if an LLD of 0.03 working level can be achieved.

2.4 Surveys for External Radiation

Most, but not all, UR facility workers receive external gamma radiation doses of less than 1 rem per year (Ref. 3). Gamma radiation exposure rates are generally below 1 milliroentgen per hour (mR/hr) in contact with incoming ore and are about 1.2 mR/hr in contact with fresh yellowcake (Ref. 3). During the buildup of the uranium daughters thorium-234 and protactinium-234 in fresh yellowcake, the radiation levels increase somewhat for several months following yellowcake production.

Gamma radiation surveys should be performed semi-annually throughout a UR facility at locations representative of workers’ exposure to determine where to post "radiation area" boundaries in accordance with 10 CFR 20.1902(a) and to determine external radiation dosimetry requirements, in accordance with 10 CFR 20.1502. At new UR facilities, a gamma radiation survey should be performed shortly after plant operation starts.

If the semiannual survey reveals any areas accessible to personnel where the gamma exposure rates are high enough that a major portion of the body of an individual could receive a dose in excess of 0.005 rem (0.05mSv) in an hour at 30 centimeters (12 inches) from the radiation source or from any surface that the radiation penetrates, the area must be designated a "radiation area," as defined in 10 CFR 20.1003. Few UR facilities will have radiation dose rates this high, but such dose rates have been found where radium-226 builds up in part of the circuit.

The survey frequency in radiation areas should be quarterly. Survey measurements should be representative of where workers might stand so that their whole-body radiation exposures can be estimated. Thus, measurements should generally be made at about 30 centimeters (12 inches) from the surfaces.[4] Surface "contact" exposure rate measurements are not required for establishing radiation area boundaries or estimating personnel whole-body exposures because these exposures would not be representative of the exposures workers would receive.

A list of the radiation levels in each area of the plant should be prepared after each survey. The number of areas on the list should be held to a manageable number. In general, a minimum of 20 survey locations is necessary to characterize the radiation levels in a UR facility.

Personnel monitoring and recording of monitoring results are generally required for any individual likely to exceed 10 percent of the limits stated in Regulatory Position 1 of this guide. For all workers who are required to be monitored, the licensee is required to advise each worker annually of the worker's dose as shown in records maintained by the licensee pursuant to the provisions of 10 CFR 20.2106 (required by 10 CFR 19.13).

In addition to gamma surveys, beta surveys of specific operations that involve direct handling of large quantities of aged yellowcake are advised to ensure that extremity and skin exposures for workers who will perform those operations are not unduly high. Beta surveys should be used to determine the need for protective clothing for these operations (e.g., thick rubber gloves). Beta surveys should also be used to determine whether procedures could be changed to reduce beta dose while still allowing the worker to do the operation efficiently. Because of these needs, beta dose rates, unlike gamma dose rates, are usually measured on the surface and at short distances rather than at 30 centimeters (12 inches). Beta surveys need be done only once for an operation but should be repeated for an operation any time the equipment or operating procedure is modified in a way that may have changed the beta dose that would be received by the worker.

The beta dose rate on the surface of yellowcake just after separation from ore is negligible, as shown in Figure 1; but this dose rate rises steadily thereafter. The beta dose rate from yellowcake aged for a few months after chemical separation from the ore so that equilibrium with protactinium-234 and thorium-234 has been reached is about 150 mrem/hr (Ref. 11). Figure 2 shows the beta dose rate from aged yellowcake as a function of distance from the surface (Ref. 18). The diameter of the yellowcake source used to measure the dose rates shown in Figure 2 was 9.5 cm. Rubber work gloves (thickness: 0.04 cm or 50 mg/cm^2) will reduce the beta dose to the hands from aged yellowcake by about 15%.

[4] See 10 CFR 20.1903 and item 6(a) of Regulatory Guide 10.6, "Guide for the Preparation of Applications for use of Sealed Sources and Devices for Performing Industrial Radiography" (Ref. 17).

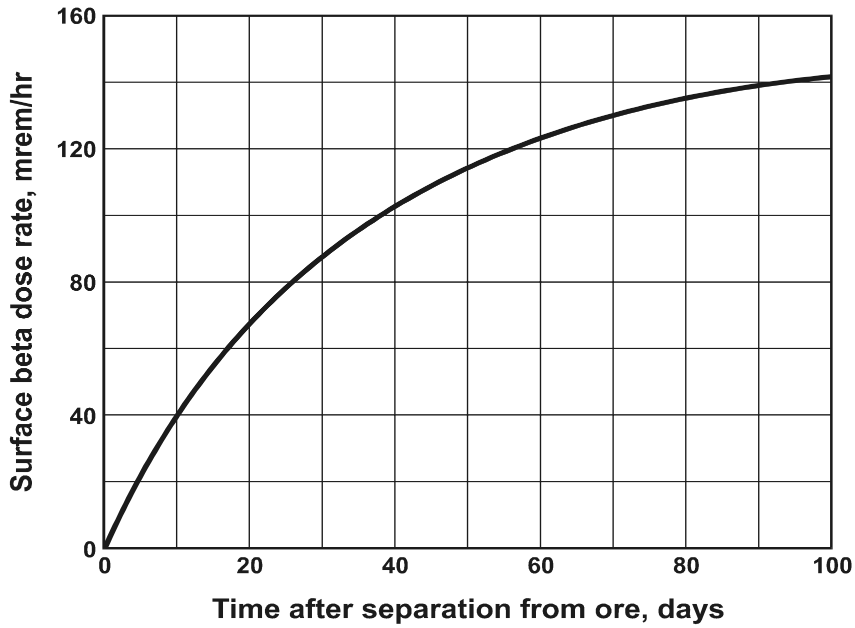

Figure 1. Beta Dose Rate on the Surface of Yellowcake

This curve was prepared by S. McGuire, NRC staff, by calculating the buildup of thorium-234 and protactinium-234 from the parent uranium-238, and the buildup of thorium-231 from the parent uranium-235. The surface beta dose rate was normalized to 150 mrem/hr (Figure 2 shows the measured value on the surface). Since measurements show that less than 1% of the thorium, radium, and lead initially present in the ore remains after the chemical separation process, betas from thorium-234, lead -210, and lead-214 in the ore before separation are negligible in the yellowcake after separation (Ref 19).

Conditions requiring individual monitoring of external and internal occupational dose are specified in 10 CFR 20.1502.

It is usually acceptable to substitute evaluations of beta doses based on Figures 1 and 2 in place of beta surveys using radiation survey instruments.

It should be noted that commercially available film badge and TLD services often have not been able to measure beta radiation in the mixed beta-gamma field of a UR facility (see, for example, Tables A-11 and A-12 of Reference 20 and Tables 6 and 9 of Reference 21). Workers' beta doses should be estimated from the beta surveys described above rather than from personnel monitoring reports.

2.5 Surveys for Surface Contamination in Restricted Area

NRC regulations provide no specific limit on surface contamination levels in restricted areas. However, yellowcake or ore dust lying on surfaces can become resuspended and contribute to the intake of radionuclides, which is limited by 10 CFR 20.1204.

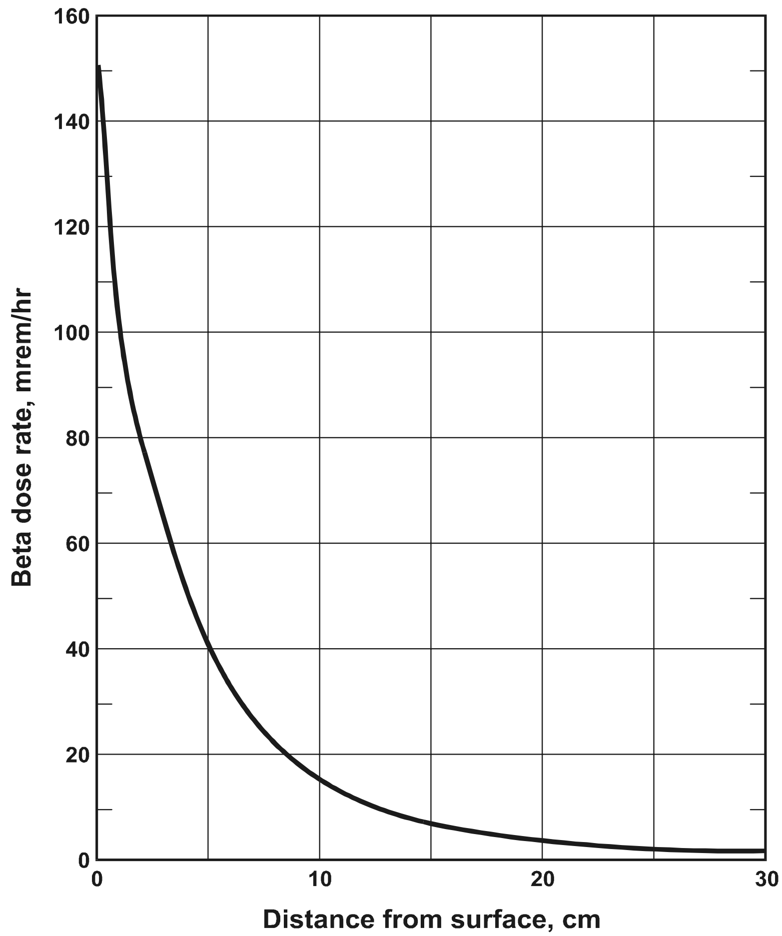

Figure 2. Beta Dose Rate from Yellowcake Separate from Ore for More than 100 Days (from Reference 10)

In ore handling areas, surface contamination is not a problem because of the very low specific activity of the ore. In fact, cleanup attempts by methods such as sweeping are likely to produce a more serious hazard through resuspension in the air than if the ore dust were allowed to remain where it lies. When necessary, cleanup may be performed by hosing down the ore dust into floor sumps or by using vacuum suction systems with filtered exhausts.

In leaching and chemical separation areas there is usually little dust and little difficulty with surface contamination.

In the precipitation circuit and the yellowcake drying and barreling areas, surface contamination can be a problem because of the concentrated nature of the yellowcake. The International Atomic Energy Agency (IAEA) recommends (Ref. 2) a limit for alpha contamination on such areas as walls, floors, benches, and clothing of 10^{-3} μCi/cm^2 (220,000 dpm/100 cm^2), which is equivalent to about 2 mg/cm^2 of natural uranium. Based on experience, the IAEA concluded that if surface contamination levels are kept below this value, the contribution to airborne radioactivity from surface contamination will be well below applicable limits. The British

National Radiological Protection Board also recommends a limit of 10^{-3} µCi/cm^2 for uranium alpha contamination in active areas of plants (Ref. 22), based on calculations using resuspension factors rather than experience.

The NRC staff considers surface contamination levels of 10^{-3} µCi/cm^2 acceptable to meet the ALARA concept in UR facilities. The levels are low enough to ensure little contribution to airborne radioactivity, yet are practical to meet. Such an amount of yellowcake surface contamination is readily visible because of the low specific activity of uranium and does not require a survey instrument for detection. It is recommended that surfaces where yellowcake may accumulate be painted in contrasting colors because surveys for surface contamination in work areas are visual rather than by instrument.

In yellowcake areas, daily visual inspections should be made for locating yellowcake contamination on surfaces. Visible yellowcake should be cleaned up promptly, especially where contamination will be disturbed and resuspended on walkways, railings, tools, vibrating machinery, and similar surfaces. Spills should be cleaned up before the yellowcake dries so that resuspension during cleanup will be lessened.

In rooms where work with uranium is not performed, such as eating rooms, change rooms, control rooms, and offices, a lower level of surface contamination is likely to be present. These areas should be spot-checked weekly for removable surface contamination using smear tests. The areas should be promptly cleaned if surface contamination levels exceed the values shown in Table 2.8.

TABLE 2

Surface Contamination Levels for Uranium and Daughters on Equipment To Be Released for Unrestricted Use, on Clothing, and on Nonoperating Areas of UR Facilities*

Average**	5,000 dpm alpha per 100 cm^2	Average over no more than 1m^2
Maximum**	15,000 dpm alpha per 100 cm^2	Applies to an area of not more than 100 cm^2
Removable	1,000 dpm alpha per 100 cm^2	Determined by smearing with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the smear

* These values are taken from Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Ref. 23), and from "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct Source. or Special Nuclear Material," Division of Fuel Cycle and Material Safety, USNRC, Washington, DC 20555, August 1987 (Ref. 24). Available in NRC Public Document Room for inspection and copying for a fee.

** The value includes both fixed and removable contamination.

(The contamination levels in Table 2 are given in units of dpm/100 cm^2 because this is the minimum area typically surveyed. When performing a smear or wipe test, the area should roughly approximate 100 cm^2. However, there is no need to be precise about the area to be smeared.)

2.6 Surveys for Contamination of Skin and Personal Clothing

Contamination of skin and personal clothing should be controlled to prevent the spread of contamination to unrestricted areas (e.g., the workers' cars and homes). Alpha radiation from uranium on the skin or clothing is not a direct radiation hazard because the alpha particles do not penetrate the dead layer of the skin. Rather, uranium is primarily a hazard if it is inhaled or swallowed.

Visual examination for yellowcake is not sufficient evidence that the worker's skin or clothing is sufficiently free of contamination to permit the workers to leave the work environment. Normally such contamination can be adequately controlled if yellowcake workers wash their hands before eating, shower before going home, and do not wear street clothes while working with yellowcake in a UR facility. Before leaving the restricted area, everyone who has worked with yellowcake during the day should either shower or monitor their skin after changing clothes. If the worker does not change clothes, the clothes should also be monitored. The soles of the shoes of anyone entering the yellowcake area of a UR facility should be monitored before leaving the Restricted Area of a UR facility. An alpha survey instrument should be available at the exit of the employee change room and at the exit of a UR facility. In addition, the licensee should at least quarterly use a calibrated alpha survey instrument to perform an unannounced spot survey for alpha contamination on selected yellowcake workers leaving theUR facility.

Limits on acceptable levels of alpha contamination of skin and clothing are found in Table 2. They are to be used in the following manner: All alpha contamination on skin and clothing should be considered to be removable so that the limit of 1,000 dpm alpha per 100 cm^2 applies.[5] The worker must shower or wash if the limit is exceeded. The value of 5,000 dpm alpha contamination per 100 cm^2 should be used for the soles of shoes using a portable alpha survey instrument to measure total alpha activity. If alpha levels exceed the value in Table 2, the clothing should be laundered before leaving the site. If the soles of shoes exceed the value in Table 2, the shoes should be brushed or scrubbed until they are below the limit.

2.7 Surveys of Equipment Prior to Release to Unrestricted Areas

Surface contamination surveys should be conducted before potentially contaminated equipment is released to unrestricted areas. The surface contamination limits listed in Table 2 are recommended.[6] If contamination above these limits is detected, the equipment should be decontaminated until additional efforts do not significantly reduce contamination levels.

The licensee should develop methods to prevent potentially contaminated equipment from leaving the restricted area without being monitored. In some cases this is facilitated if parking areas for workers and visitors are located outside the restricted area.

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[5] This value is comparable to the limit of 10^{-5} $\mu Ci/cm^2$ or 2.200 dpm per 100 cm^2, which is recommended by the International Atomic Energy Agency on page 15 of Reference 3 and the United Kingdom Atomic Energy Authority in Reference 25.

[6] See Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Ref. 23), and "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" (Ref. 24).

2.8 Surveys of Packages Prepared for Shipment

After being filled, yellowcake packages should be washed down to remove surface contamination. Surveys of external surfaces of yellowcake packages prepared for shipment should be carried out before shipment. The surveys conducted should be adequate to ensure that the wash-downs are reducing surface contamination levels to less than Department of Transportation (DOT) limits, but do not necessarily include a survey of each package. The bottoms of all barrels should be surveyed to determine the effectiveness of the wash-downs.

Contamination on packages should not exceed DOT limits in 49 CFR 173.443. The average measured removable alpha contamination determined by wiping the external surface of the package with an absorbent material should be below 2200 dpm/100 cm^2 if a non-exclusive-use vehicle is to be used (49 CFR 173.443(a) and (a)(1)) or 22,000 dpm/100 cm^2 if an exclusive-use vehicle is to be used (49 CFR 173.443(b) and (a)(1)). Packages having higher contamination levels should be cleaned and resurveyed prior to shipment. Visible yellowcake should be cleaned off.

2.9 Ventilation Surveys

A properly operating ventilation system is the most effective means of worker protection from inhalation hazards at a UR facility. The operation of the ventilation system should be checked each day by the radiation safety staff during the daily walk-through of the UR facility.

Whenever equipment or procedures in theUR facility are changed in a manner that affects ventilation, a survey should be made of the ventilation rates in the area to ensure that the ventilation system is operating effectively.

2.10 Surveys for Contamination on Respirators

Before being reused, respirator face pieces and hoods should be surveyed for alpha contamination by a standard wipe or smear technique. Removable alpha contamination levels should be less than 100 dpm/100 cm^2 (Ref. 26, Section 9.6).

2.11 Summary of Survey Frequencies

Table 3 summarizes the survey frequencies given in this guide.

3. INTAKE AND EXPOSURE CALCULATIONS

The internal dose component needed for evaluating the total effective dose equivalent is the committed effective dose equivalent. The committed effective dose equivalent is the 50-year effective dose equivalent that results when radioactive material is taken into the body, whether through inhalation, ingestion, absorption through the skin, accidental injection, or introduction through a wound. The contributions from all occupational intakes for these modes of intake are added over the yearly time period for which the total committed effective dose equivalent is being evaluated. The regulatory requirements for determining the internal dose are contained in 10 CFR 20.1204.

This guide presents two alternative methods for calculating committed effective dose equivalent from inhalation. The first method uses stochastic inhalation ALIs from 10 CFR Part 20. The second method uses DACs from 10 CFR Part 20. The methods are equivalent and either may be used.

Method 1: Use of Stochastic Inhalation ALIs from 10 CFR Part 20

ALI values have been established for individual radionuclides and are presented in Table 1 in Appendix B to 10 CFR Part 20. The ALI values for inhalation, presented in Column 2 in Table 1, correspond to a committed effective dose equivalent of 5 rems (0.05 Sv) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue, whichever is more limiting. If the ALI value presented in Table 1 is limited by the 50-rem committed dose equivalent, the controlling organ is listed directly below the ALI value, and the stochastic ALI value based on the 5-rem committed effective dose equivalent is listed in parentheses directly below the organ name. If a stochastic ALI is listed in parentheses, that value should be used to calculate the committed effective dose equivalent. The committed effective dose equivalent for each radionuclide may be calculated, using the estimated radionuclide intake, by Equation 2.

TABLE 3
SUMMARY OF SURVEY FREQUENCIES

Type of Survey	Type of Area	Survey Frequency	Lower Limit of Detection
1. Uranium ore dust	Airborne radioactivity areas Other indoor process areas Outdoor areas	Weekly grab samples Monthly grab samples Quarterly grab samples	5 x 10^{-12} μCi/ml (uranium)
2. Yellowcake	Airborne radioactivity areas Other indoor process areas Special maintenance involving high airborne concentrations of yellowcake	Weekly grab samples Monthly grab samples Extra breathing zone grab samples	1x10^{-11} μCi/ml
3. Radon daughters	Areas that exceed 0.08 working level Areas that exceed 0.03 working level Areas below 0.03 working level	Weekly radon daughter grab samples Monthly radon daughter grab samples Quarterly radon daughter grab samples	0.03 WL
4. External radiation: Gamma Beta	Throughout UR facility Radiation areas Where workers are in close contact with yellowcake	Semiannually Quarterly Survey by operation done once plus whenever procedures change	0.1 mrad/hr 1 mrem/hr
5. Surface contamination	Yellowcake areas Eating rooms, change rooms, control rooms, offices	Daily Weekly	Visual 500 dpm alpha per 100 cm^2
6. Skin and personal clothing	Yellowcake workers who shower Yellowcake workers who do not shower	Quarterly Each day before leaving	500 dpm alpha per 100 cm^2
7. Equipment to be released	Equipment to be released that may be contaminated	Once before release	500 dpm alpha per 100 cm^2
8. Package containing yellowcake	Packages	Spot check before release	500 dpm alpha per 100 cm^2
9. Ventilation	All areas with airborne radioactivity	Daily	Not applicable
10. Respirators	Respirator face pieces and hoods	Before reuse	100 dpm alpha per 100 cm^2

$$H_{i,E} = \frac{5\, I_i}{ALI_{i,E}}$$ Equation 2

where

$H_{i,E}$ = Committed effective dose equivalent from radionuclide i (rems)

I_i = Intake of radionuclide i by inhalation during the calendar year (µCi) (If multiple intakes occurred during the year, is the sum of all intakes.)

$ALI_{i,E}$ = Value of the stochastic inhalation ALI (based on the committed effective dose equivalent) from Column 2 of Table 1 in Appendix B to Part 20 (µCi)

5 = Committed effective dose equivalent from intake of 1 ALI (rems)

If intakes of more than one radionuclide occurred, the total committed effective dose equivalent will be the sum of the committed effective dose equivalents for all radionuclides.

The ALIs in Part 20 are based on a particle distribution with a 1-micro-meter activity median aerodynamic diameter. Those ALIs may be used regardless of the actual median diameter. However, the NRC allows adjustment of ALIs to account for particle size, but only with prior approval (10 CFR 20.1204(c)).

Some noble gases in Appendix B to 10 CFR Part 20 do not have inhalation ALI values listed and are listed as "submersion" class. For these radionuclides, the internal dose is negligible compared to the external dose. These radionuclides may be excluded from the determination of the internal dose.

Method 2: Use of DACs from 10 CFR Part 20

Committed effective dose equivalent may also be calculated from exposures expressed in terms of DAC-hours. If the DAC in Appendix B to 10 CFR Part 20 for a radionuclide represents a stochastic value (i.e., the corresponding ALI does not have the name of an organ below it), the DAC may be used directly. If Appendix B to 10 CFR Part 20 does not list a stochastic DAC (which will be the case any time there is a stochastic ALI value in parentheses), it is preferred (but not required) that the licensee calculate and use a stochastic DAC. The stochastic DAC can be calculated from the stochastic ALI (the ALI in parentheses) by using Equations 3 and 4.

$$DAC_{stoc,i} = \frac{ALI_{stoc,i}}{2.4\text{x}10^{9}}$$ Equation 3

where

$DAC_{stoc,i}$ = The stochastic DAC for radionuclide i (microcuries/ml)

$ALI_{stoc,i}$ = The stochastic ALI for radionuclide i (microcuries)

2.4×10^9 = The volume of air inhaled by a worker in a work year (ml)

$$H_{i,E} = \frac{5\, C_i\, t}{2000 DAC_{stoc,i}} \qquad \text{Equation 4}$$

where

$H_{i,E}$ = Committed effective dose equivalent from radionuclide i (rems)

C_i = The airborne concentration of radionuclide i to which the worker is exposed (microcuries/ml)

t = The duration of the exposure (hours)

2000 = The number of hours in a work year

5 = Committed effective dose equivalent from annual intake of 1 ALI or 2000 DAC-hours (rems)

If there is a mixture of several radionuclides, it is permissible to disregard certain radionuclides in the mixture that are present in relatively small quantities (10 CFR 20.1204(g)). These radionuclides may be disregarded if the following conditions are met: (1) the concentration of any radionuclide disregarded is less than 10% of its DAC; (2) the sum of these percentages for all the radionuclides disregarded in the mixture does not exceed 30%; and (3) the licensee uses the total activity of the mixture in demonstrating compliance with the dose limits and monitoring requirements.

4. ADMINISTRATIVE ACTION LEVELS

The licensee should establish administrative action levels to protect workers. Action levels should be established as shown below. A record of each investigation made and the actions taken, if any, should be kept until license termination.

4.1 Uranium Ore[7] Dust

The RSO should establish an action level for each ore dust sampling location. The action level for the location should be set somewhat above the normal fluctuations that occur when theUR facility

[7] As defined in NRC guidance, ore is a natural or native matter that may be mined and treated for the extraction of any of its constituents or any other matter from which source material is extracted in a licensed uranium or thorium mill.

is operating properly. If any sample is above the action level for that location, the RSO should find out why and should take corrective action if appropriate.

4.2 Yellowcake

Similarly, for yellowcake the RSO should establish an action level for each sampling location. In addition, action levels should be established for maintenance activities where breathing zone sampling is used. The action level for maintenance activities can be expressed either in airborne concentration or in DAC-hours. If any action level is exceeded, the RSO should find out why and should take corrective action, if appropriate.

4.3 Radon Daughters

The RSO should establish an action level for radon daughters for each sampling location. If the action level for any location is exceeded, the RSO should find out why and should take corrective action, if appropriate.

4.4 Time-Weighted Exposure to Airborne Radioactivity

If any worker's time-weighted exposure, calculated by either of the two options in Method 2 of Regulatory Position 3 of this guide, exceeds 25% of the exposure limits, as listed in Table 1 of this guide, the RSO should determine the causes of the exposure, should investigate why the exposure was higher than previous exposures in performing the work, and should take corrective action if appropriate. This action level will be on a weekly basis for soluble uranium (yellowcake dried at less than 400ºC), a quarterly basis for uranium ore dust and yellowcake combined, and an annual basis for radon daughters of 4 Working Level Months or 2000 DAC-hours.

4.5 Gamma Dose Rates

The RSO should establish an action level for each location where the gamma dose rate is periodically measured. If the action level for any location is exceeded, the RSO should determine the cause of the elevation and should take corrective action, if appropriate.

4.6 Dosimeter Results

The RSO should establish action levels for the monthly or quarterly dosimeter results, whichever is established in approved procedures. If the action level for any person is exceeded, the RSO should determine the cause and take corrective action, if appropriate.

4.7 Contamination on Skin and Clothing

If alpha contamination of the skin or clothing of workers leaving a UR facility is found to exceed 1000 dpm/100 cm^2, an investigation of the cause of the contamination should be made and corrective action taken, if appropriate.

4.8 Low Airborne Radioactivity Concentrations

Abnormally low concentrations of airborne radioactivity (uranium ore dust, yellowcake, and radon daughters) should also be investigated since very low concentrations may indicate an equipment malfunction or procedural error. The RSO should establish action levels for low readings of airborne radioactivity. If concentrations are below these action levels, the RSO should determine the reason and should take corrective action, if appropriate.

5. REPORTING REQUIREMENTS

Each licensee is required to notify the NRC as soon as possible of exposures, radiation levels, and concentrations of radioactive materials exceeding the constraints or limits as required in Subpart M of 10 CFR Part 20 and in 10 CFR 40.60.

6. ESTABLISHING "AIRBORNE RADIOACTIVITY AREAS"

In general, yellowcake drying and packaging rooms and enclosures should always be considered to be airborne radioactivity areas because of the high concentrations that can result if any equipment malfunctions. On the other hand, ore crushing and grinding areas and areas outside yellowcake drying and packaging areas will not normally need to be classified as airborne radioactivity areas when normal engineering controls are used.

Any area, room, or enclosure is an "airborne radioactivity area" as defined in 10 CFR 20.1003 if (1) at any time the airborne uranium concentration exceeds 5×10^{-11} μCi/ml in the case of ore dust or 1×10^{-10} μCi/ml in the case of yellowcake (i.e., the values in Appendix B to 10 CFR Part 20) or (2) the concentration exceeds 25% of the values in Appendix B to 10 CFR Part 20 averaged over the number of hours in any one week in which individuals are present in such area, room, or enclosure. For example, an area that is occupied 20 hours per week (out of the 40 hours used as a basis for the limits) is an airborne radioactivity area if the concentration of uranium in yellowcake exceeds 0.5×10^{-10} μCi/ml of air. The licensee should maintain records to show that occupancy is in fact thus limited.

If combinations of radon daughters, ore dust, and yellowcake are present (see Regulatory Position 2.3 of this guide), their concentrations, divided by the appropriate value from Table 1 of Appendix B to 10 CFR Part 20, should be added. If the sum of these fractions exceeds unity or if the sum exceeds 0.25 after adjustment for the occupancy factor, the area is an airborne radioactivity area.

7. POSTING OF CAUTION SIGNS, LABELS, AND NOTICES TO EMPLOYEES

The radiation protection staff should periodically survey to ensure that signs, labels, required notices to employees, copies of licenses, and other items are properly posted as required by 10 CFR 19.11 and 10 CFR Part 20.

The UR facility and tailings area should be fenced to restrict access, and the fence should be posted with "Caution, Radioactive Material Area" signs as required in 10 CFR 20.1902. If the fence and all entrances are posted and also state the words "Any area within this UR facility may contain radioactive material," the entire area is posted adequately to meet the requirement in 10 CFR 20.1902. Additional posting of each room with "Radioactive Material" signs is not necessary.

"Radiation Areas" and "Airborne Radioactivity Areas" must be posted in accordance with 10 CFR 20.1902. The licensee should avoid posting radiation area signs and airborne radioactivity area signs in areas that do not require them. The purpose of the signs is to warn workers where additional precautions to avoid radiation exposure are appropriate. Posting all areas in a UR facility with such signs defeats this purpose.

8. CALIBRATION OF SURVEY INSTRUMENTS

Portable survey instruments should be placed on a routine maintenance and calibration program to ensure that properly calibrated and operable survey instruments are available at all times for use by the health physics staff.

Survey instruments should be checked for constancy of operation with a radiation check source prior to each usage. If the instrument response to the radiation check source differs from the reference reading by more than 20%, the instrument should be repaired if necessary and recalibrated (Ref. 27, paragraph 4.6).

This constancy check should be supplemented by calibrations at 12-month intervals or at the manufacturer's suggested interval, whichever is shorter (Ref. 27, paragraph 4.7.1). An adequate calibration of survey instruments cannot be performed solely with built-in check sources. Electronic calibrations that do not involve a source of radiation will not determine the proper functioning and response of all components of an instrument. However, an initial calibration with a gamma source and periodic tests using electronic input signals may be considered adequate for the high dose ranges on survey instruments if those ranges are not used routinely. Each instrument should be calibrated at two points at about one-third and two-thirds of each linear scale routinely used or with a calibration at one point near the midpoint of each decade on logarithmic scales that are routinely used. Digital readout instruments with either manual or automatic scale switching should be calibrated in the same manner as are meter-dial instruments. Digital readout instruments without scale switching should be calibrated in the same manner as are logarithmic readout instruments. Survey instruments should be calibrated following repair. A survey instrument may be considered properly calibrated when the instrument readings are within ±20% of the calculated or known values for each point checked (see Appendix A to Regulatory Guide 10.6, "Guide for the Preparation of Applications for Use of Sealed Sources and Devices for Performing Industrial Radiography" (Ref. 17)).

Calibration for beta dose rate measurements may be performed in the following manner. A usual technique for making a beta survey is to note the difference between the open-window and closed-window reading on a GM or ionization chamber survey meter. The difference is considered to be an indication of the beta dose rate. This approach is incorrect if the survey meter has been calibrated with a gamma source alone. A correction factor must be applied to determine the beta dose rate.

To determine the beta correction factor, use Figure 2 in this guide. Place the detector of the survey meter at the surface of an extended yellowcake source that has been separated from ore for at least 100 days. Use a piece of paper or thin plastic between the detector and yellowcake to avoid contaminating the detector. Note the difference between the open-window and closed-window readings. Compute a correction factor that applies to the surface dose rate that will make the difference between the open-window and closed-window readings equal to the surface beta dose rate of 150 mrad/hr, as shown in Figure 2. To determine the correction factor that applies at a distance from the surface, place the axis of the detector at 2 cm from the surface. Note the difference between the open-window and closed-window readings. Compute a correction factor that will make the difference between the open-window and closed-window, readings equal to 75 mrad/hr, as shown in Figure 2. A sample calculation is shown in Appendix C to this guide.

Errors in estimates of the volume of air that has passed through filters should be avoided by accurate calibration of the flow rate and by preventing or correcting for the loss of flow caused by accumulation of material on the filter. As material accumulates on filter paper the air flow rate will drop. Thus less air volume will be sampled. Air flow rates through filters should be determined by calibrating pumps with the filter paper in place once every 6 months to ±20% accuracy. These calibrations should be done in accordance with the manufacturer's recommendations. Further information on these calibrations is contained in Regulatory Guide 8.25, "Calibration and Error Limits of Air Sampling Instruments for Total Volume of Air Sampled" (Ref. 28).

The fluorometric analysis should be calibrated by processing a known standard uranium solution and a blank sample with each batch. Every quarter, the fluorometer response should be checked by a complete serial dilution.

Alpha counting systems used for radon daughter measurements should be calibrated at least monthly by using a known standard alpha source.

Alpha survey meters used to monitor and detect contamination on skin and equipment should receive a response check before each use, a constancy check each week to determine whether the instrument is within the acceptable error band, and a calibration in accordance with the manufacturer's recommendations or annually, whichever is shorter (Ref. 27, paragraph 4.7.1).

9. PROTECTIVE CLOTHING

Workers working with yellowcake should be provided with protective clothing such as coveralls and shoes or shoe covers. Rubber work gloves should be used when aged yellowcake will be handled in order to reduce the beta dose and to avoid contamination of the skin with uranium.

Protective clothing should be changed and discarded or laundered weekly or whenever yellowcake is visible on the clothing. Potentially contaminated clothing should not be sent to a laundry that is not specifically authorized by the NRC or an Agreement State to process clothing contaminated with uranium unless the clothing has been surveyed and found to have less uranium contamination than the values in Table 2 of this guide.

10. QUALITY ASSURANCE PROGRAM

The licensee should ensure the accuracy of survey measurements by having a quality assurance program. Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations)--Effluent Streams and the Environment" (Ref. 29), should be consulted for guidance on quality assurance.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except when an applicant proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide reflecting public comments will be used in the evaluation of applications for new UR facilities and renewal applications.

REFERENCES

1. USNRC, "Standard Format and Content of License Applications for Uranium Mills," Regulatory Guide 3.5, Revision 1, November 1977.[1]

2. USNRC, "Standard Format and Content of License Applications, Including Environmental Reports, for In Situ Uranium Solution Mining," June 1982.[1]

3. International Atomic Energy Agency, *Manual on Radiological Safety in Uranium and Thorium Mines and Mills*, IAEA Safety Series No. 43, Vienna, 1976.[2]

4. USNRC, "Acceptable Programs for Respiratory Protection," Regulatory Guide 8.15, Revision 1, October 1999.[1]

5. USNRC, "Bioassay at Uranium Mills," Regulatory Guide 8.22, Revision 1, August 1988.[1]

6. Health Physics Society, "Bioassay Programs for Uranium," American National Standard HPS N13.22-1995.

7. Health Physics Society, "Performance Criteria for Radiobioassay," American National Standard HPS N13.30-1996.

8. USNRC, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Is Reasonably Achievable," Regulatory Guide 8.31, Revision 1, May 2002.[1]

9. D. R. Kalkwarf, "Solubility Classification of Airborne Products from Uranium Ores and Tailings Piles," NUREG/CR-0530, USNRC, January 1979.[3]

10. F. Eidson and J.A. Mewhinney, "In Vitro Dissolution of Uranium Product Samples from Four Uranium Mills," NUREG/CR-0414, USNRC, October 1978.[3]

11. N. A. Dennis, H. M. Blauer, and J. E. Kent, "Dissolution Fractions and Half-times of Single Source Yellow-Cake in Simulated Lung Fluids," *Health Physics*, vol. 42, p. 469, April 1982.

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[1] Requests for single copies of draft or active regulatory guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section; or by fax to (301)415-2289; email <DISTRIBUTION@NRC.GOV>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or 1-(800)397-4209; fax (301)415-3548; e-mail <PDR@NRC.GOV>.

[2] Available from UNIPUB, P.O.Box 433, Murray Hill Station, New York, NY 10016.

[3] Copies are available at current rates from the U.S. Government Printing Office, or from the National Technical Information Service, or the NRC Public Document Room. See the information in Footnote 1 above.

12. R. C. Merritt, *The Extractive Metallurgy of Uranium*, Colorado School of Mines Research Institute, pp. 252-254, 1971.

13. L. M. Steckel and C. M. West, "Characterization of Y-12 Uranium Process Materials Correlated with In Vitro Experience," Report Y-1544-A, U.S. Atomic Energy Commission, 1966.[4]

14. National Council on Radiation Protection and Measurements, "Natural Background Radiation in the United States," NCRP Report No. 45, Washington, DC, 1975.

15. International Commission on Radiological Protection, "Occupational Limits for Inhalation of Radon-222, Radon-220 and their Short-Lived Daughters," ICRP Publication 32, Pergamon Press, Oxford, England, 1981.

16. American National Standards Institute, "Radiation Protection in Uranium Mines," ANSI N 13.8-1973 (Reaffirmed in 1989).

17. USNRC, "Guide for the Preparation of Applications for Use of Sealed Sources and Devices for Performing Industrial Radiography," Regulatory Guide 10.6, Revision 1, December 1981.[1]

18. D. Haggard, Battelle-Pacific Northwest Laboratory, letter to Dr. Stephen A. McGuire, USNRC, June 29, 1982.[4]

19. M.H. Momeni et al., "Radioisotopic Composition of Yellowcake," NUREG/CR-1216, USNRC, 1979.[3]

20. L. L. Nichols, "A Test of the Performance of Personnel Dosimeters," Battelle-Pacific Northwest Laboratories Report BNWL-2159, April 1977.

21. P. Plato and G. Hudson, "Performance Testing of Personnel Dosimetry Services," NUREG/CR-1064, USNRC, 1980.[3]

22. A.D. Wrixon, G.S. Linsley, K.C. Binns, and D.F. White, "Derived Limits for Surface Contamination, "British National Radiological Protection Board Report NRPB-DL2, November 1979.

23. USNRC, "Termination of Operating Licenses for Nuclear Reactors," Regulatory Guide 1.86, June 1974.[1]

24. USNRC, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," Division of Fuel Cycle and Material Safety, August 1987.[4]

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[4] Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike (first floor), Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or 1-(800)397-4209; fax (301)415-3548; e-mail <PDR@NRC.GOV>.

25. United Kingdom Atomic Energy Authority, Health and Safety Code, “Maximum Permissible Doses from Inhaled and Ingested Radioactive Materials,”Authority Code No. E.1.2, Issue No. 1, London, June 1961.

26. J.L. Caplin et al., “Manual of Respiratory Protection Against Airborne Radioactive Material,” NRC Report NUREG-0041, October 1976.[3]

27. American National Standards Institute, “Radiation Protection Instrumentation Test and Calibration,” ANSI N323-1978.

28. USNRC, “Airsampling in the Workplace,” Regulatory Guide 8.25, Revision 2, June 1992.[1]

29. USNRC, “Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment,” Regulatory Guide 4.15, Revision 1, February 1979.[1]

APPENDIX A

DERIVATION OF EQUATION FOR DAC_m

The equation for DAC_m is derived here. The equation for mixtures in paragraph 1 of the Note to Appendix B of Part 20 is:

$$\frac{C_a}{DAC_a} + \frac{C_b}{DAC_b} + \frac{C_c}{DAC_c} \leq 1 \qquad \text{Equation A-1}$$

Consider a mixture of natural uranium as yellowcake with a concentration of C_{nu} and ore dust with a concentration C_{od}. If the sum of the concentrations equals the DAC for the mixture

$$\frac{C_{nu} + C_{od}}{DAC_m} = 1 \qquad \text{Equation A-2}$$

the equality in the first equation will apply. Therefore:

$$\frac{C_{nu}}{DAC_{nu}} + \frac{C_{od}}{DAC_{od}} = \frac{C_{nu} + C_{od}}{DAC_m} \qquad \text{Equation A-3}$$

Solve for DAC_m

$$DAC_m = \frac{C_{nu} + C_{od}}{\dfrac{C_{nu}}{DAC_{nu}} + \dfrac{C_{od}}{DAC_{od}}} \qquad \text{Equation A-4}$$

Divide the numerator and denominator of the right-hand side by $C_{nu} + C_{od}$

$$DAC_m = \frac{1}{\dfrac{C_{nu}}{(C_{nu} + C_{od})(DAC_{nu})} + \dfrac{C_{od}}{(C_{nu} + C_{od})(DAC_{od})}} \qquad \text{Equation A-5}$$

The term

$$\frac{C_{nu}}{C_{nu} + C_{od}}$$

can be recognized as f_{nu}, the fraction of activity from natural uranium as yellowcake. Therefore:

$$DAC_m = \left[\frac{f_{nu}}{DAC_{nu}} + \frac{f_{od}}{DAC_{od}} \right]^{-1} \qquad \text{Equation A-6}$$

APPENDIX B

LOWER LIMIT OF DETECTION

For the purpose of this guide, the lower limit of detection (LLD) is defined as the smallest concentration of radioactive material that has a 95% probability of being detected.[1] Radioactive material is "detected" if the value measured on an instrument is high enough to conclude that activity above the system background is probably present.

For a particular measurement where radioactive disintegrations are detected (which may include a radiochemical separation):

$$LLD = \frac{3 + 4.65S_b}{3.7x10^4 EVYe^{-\lambda t}} \qquad \text{Equation B-1}$$

where:

LLD = the lower limit of detection (μCi/ml)

S_b = the standard deviation of background count rate (counts per second)

3.7×10^4 = the number of disintegrations/sec/μCi (this term is omitted if Sb is given in terms of microcuries)

E = the counting efficiency (counts per disintegration)

V = the sample volume (ml)

Y = the fractional radiochemical yield (if applicable)

λ = the decay constant for the particular radionuclide

t = the elapsed time between sample collection and counting

Example: LLD for Uranium when Fluorometric Analysis Is Used

Work this example in terms of microcuries of natural uranium. The LLD could just as well be calculated in terms of micrograms of uranium. A conversion factor of 6.77×10^{-7} μCi/μg for natural uranium can be used if the uranium quantity is known in micrograms.

First, determine the standard deviation of the background count rate S_b. To do this, perform a fluorometric analysis for several clean filter papers that have not been used to collect air samples. At least 5 filter papers would have to be analyzed over many months. The value of S_b will be in terms of microamperes because fluorometers usually give readings in microamperes. The value

[1] The definition of LLD was chosen to be consistent with the NRC position stated in Tables 1 and 3 of Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants" (Ref. B-1). The basis for the definition is given in References B-2 and B-3 of this guide. The definition is also used in other regulatory guides, among them Regulatory Guides 4.14, "Radiological Effluent and Environmental Monitoring at Uranium Mills" (Ref. B-4), and 8.14, "Personnel Neutron Dosimeters" (Ref. B-5), as well as in NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (pages 6-32 through 6-37) (Ref. B-6), and Appendix A to ANSI N13.30 (1996a) (Ref. B-7).

of S_b can then be converted either to microcuries or to counts per second by using a calibration factor.

A sample calculation is shown here. The fluorometric reading for 10 clean filter papers are as follows:

SAMPLE NUMBER	FLUOROMETRIC READING (X_i) (microamperes)
1	0.082
2	0.072
3	0.05
4	0.05
5	0.05
6	0.04
7	0.086
8	0.088
9	0.08
10	0.018

Calculate the standard deviation S_b by Equation B-2 (or by pocket calculator):

$$S_b^2 = \frac{1}{n-1} \sum_{i=1}^{n} (X_i - \overline{X})^2 \qquad \text{Equation B-2}$$

where:

n = the number of samples

X_i = the reading for sample i

$\overline{X}$ = the average of the readings

For the data above, the standard deviation is:

$$S_b = 0.023\ \mu A$$

Convert S_b to micrograms of uranium. On this fluorometer 0.1 µg of U_3O_8 gives a reading of 0.67 µA. The fluorometer will read 6.7 µA/µg of U_3O_8. This compound is 85% uranium by weight (238 x 3 = 714, 16 x 8 = 128, 714/842 = 0.85). Therefore, the fluorometer will read 7.9 µA/µg of uranium (6.7/0.85 = 7.9).

Now calculate the standard deviation in micrograms of uranium:

$$S_b = \frac{0.023 \mu A}{7.9 \mu A / \mu g}$$

$$= 0.0029\ \mu g \text{ of uranium}$$

To convert to microcuries, use a conversion factor of 6.77×10^{-7} μCi/µg of uranium. Therefore:

$S_b = 0.0029\ \mu g \times 6.77 \times 10^{-7}\ \mu Ci/\mu g$

$= 1.97 \times 10^{-9}\ \mu Ci$

In the equation for LLD, the counting efficiency E will be 1. (The term E is not applicable to a fluorometric analysis.)

The sample volume V will be equal to the collection rate of the air sampler times the sample collection time. Assume a low-volume air sampler with an air flow rate of 10 liters per minute and a 30-minute sample collection time.

V= 10 liters/min x 30 minutes

= 300 liters

= 300,000 ml

For a fluorometric analysis, the radiochemical yield is not applicable, and Y may be set equal to 1.

The exponential term for radioactive decay $e^{-\lambda t}$ will also be equal to 1 because the half-life of uranium is so long that the amount of decay between collection and analysis will be negligible.

Therefore

$$LLD = \frac{3 + 4.65 x 1.97 x 10^{-9} \mu Ci}{300{,}000 ml}$$

$= 5 \times 10^{-12}\ \mu Ci$ of uranium/ml of air

This LLD is about 100 times more sensitive than recommended in the guide as an acceptable lower limit of detection.

Example: LLD for radon daughter when the modified Kusnetz method is used.

The background standard deviation is established by using blank filters. Assume the alpha counts on 10 blank filters counted for 1 minute each are as shown below:

Sample Number	Alpha Counts
1	2
2	3
3	1
4	3
5	2
6	2
7	2
8	3
9	2
10	4

For these filters Sb can be calculated to be 0.84 counts for a 1-minute count.

Assume the counting efficiency E is 0.27. Consider a low-volume sampler with a flow rate of 5 liters per minute and a 5-minute collection time. Therefore, the sample volume will be 25,000 ml. The radiochemical yield Y is not applicable, and is set equal to 1.

To calculate radioactive decay the value of λ can be taken to be roughly 0.026 per minute (for lead-214, the radon daughter with the longest half-life). The value of t is taken to be 60 minutes. It will be accurate enough to use 60 minutes for this value even though it could be as short as 40 minutes or as long as 90 minutes. Therefore $e^{-\lambda t}$ equals 0.21. The lower limit of detection can now be calculated:

$$\text{LLD} = \frac{3 + 4.65 \text{ x } 0.84 \text{ counts / min}}{0.27 \text{ counts / dis x } 25 \text{ liters x } 1 \text{ x } 0.21}$$

$$= 4.5 \text{ dpm/liter}$$

To convert this LLD to working levels (WL), divide by the factor from Figure 1 in ANSI N13.8-1989 (Ref. 9.) The factor is 110 dpm/liter/WL for a sample counted 60 minutes after collection. Therefore:

LLD = 0.025 WL

This is below the LLD for radon daughters recommended in thie guide.

APPENDIX B REFERENCES

B-1. USNRC, "Environmental Technical Specifications for Nuclear Power Plants," Regulatory Guide 4.8, December 1975.[1]

B-2. J.H. Harley, Editor, "EML Procedures Manual,"DOE Report HASL-300, p. D-08-01, revised annually.

B-3. L.A. Currie, "Limits for Qualitative Detection and Quantitative Determination -- Application to Radioactivity," *Analytical Chemistry*, Vol. 40, pp. 586-593, 1968.

B-4. USNRC, "Radiological Effluent and Environmental Monitoring at Uranium Mills," Regulatory Guide 4.14, Revision 1, April 1980.[1]

B-5 USNRC, "Personnel Neutron Dosimiters," Regulatory Guide 8.14, Revision 1, August 1977.[1]

B-6 USNRC, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," NUREG-1575, Revision 1, August 2000.[2]

B-7. ANSI, Appendix A, "Performance Criteria for Radiobioassay," ANSI N13.30, 1996a.

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APPENDIX C

BETA CORRECTION FACTOR FOR SURVEY INSTRUMENT

Here is an example for calculating the beta correction factor for the survey instrument.

At the surface, the closed-window reading is 3 mR/hr. The open-window reading is 28 mR/hr. The difference is 25 mR/hr. Since the beta dose rate at the surface is 150 mrem/hr, the correction factor Cf_{sur} can be calculated from the equation below:

Observed dose rate x CF = actual dose rate

25 mR/hr x Cf_{sur} = 150 mrem/hr

Cf_{sur} = (150 mrem/hr)/ 25 mR/hr

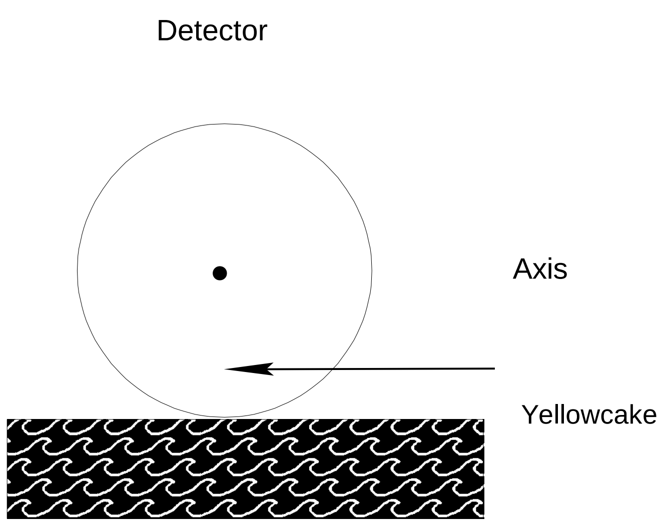
Cf_{sur} = 6 mrem/mR (at the surface)

At 2 cm: Place the axis of the detector at 2 cm from the surface of the yellowcake. The closed-window reading is 3 mR/hr. The open-window reading is 23 mR/hr. The difference is 20 mR/hr. Since the beta dose rate at 2 cm is 75 mrem/hr, the correction factor CF_{2cm} can be calculated:

CF_{2cm} = (75 mrem/hr) / (20 mR/hr)

CF_{2cm} = 3.75 mrem/mR (at 2 cm)

The value obtained at 2 cm will generally be accurate enough to use at all distances greater than 2 cm.



REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this Revision 1 to Regulatory Guide 8.30. A value/impact statement accompanied Regulatory Guide 8.30 when it was issued in June 1983.

Revision 1 to Regulatory Guide 8.30 is needed to conform with the revised 10 CFR Part 20, "Standards for Protection Against Radiation." The regulatory analysis prepared for 10 CFR Part 20 provides the regulatory basis for this Revision 1 of Regulatory Guide 8.30, and it examines the cost and benefits of the rule as implemented by the guide. A copy of the "Regulatory Analysis for the Revision of 10 CFR Part 20" (PNL-6712, November 1988), is available for inspection and copying for a fee in the NRC's Public Document Room at 11555 Rockville Pike (first floor), Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or 1-(800)397-4209; fax (301)415-3548; e-mail <PDR@NRC.GOV>.