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CALCULATIONAL MODELS FOR ESTIMATING RADIATION DOSES TO MAN FROM AIRBORNE RADIOACTIVE MATERIALS RESULTING FROM URANIUM MILLING OPERATIONS

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ERRATA

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Table 3, "Inhalation Dose Conversion Factors," on page 31 of this guide has the following typographical errors:

1. Under "Uranium Ore Dust," the ^{238}U bone dose value in the second row of the first column should read 7.29E+01 instead of 7.92E+01.

2. Under "Coarse Tailings Particulates," the first value for ^{226}Ra for the whole body dose should read 3.90E+01 instead of 4.90E+01.

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*See Table 12.

A. INTRODUCTION

The NRC staff is required to make analyses of radiation doses to the public, or individual members thereof, resulting from the radioactive effluents from uranium mills for the following purposes:

1. Evaluating compliance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations,"

2. Evaluating compliance with the "as low as is reasonably achievable" (ALARA) criterion embodied in 10 CFR Part 20, "Standards for Protection Against Radiation," and

3. Evaluating overall radiological impact as part of the complete environmental impact assessment required by the National Environmental Policy Act (NEPA) of 1969 (Public Law 91-190, 83 Stat. 852).

This regulatory guide describes basic features of calculational models used by the NRC staff for such evaluations and suggests values for various parameters used in the estimation of radiation doses to man from uranium milling operations. Specifically, this guide addresses the calculation of radiation doses to man from previously estimated environmental radioactivity concentrations in air. The environmental radioactivity concentrations in air required for this calculation result from extensive and detailed analyses of effluent release rates and atmospheric dispersion phenomena.

Information on the approach used for estimating source terms is included in the Final Generic Environmental Impact Statement on Uranium Milling, NUREG-0706 (Ref. 1). The methodology used by the staff for calculating atmospheric dispersion is documented in the MILDOS code user's manual, NUREG/CR-2011 (Ref. 2).

B. DISCUSSION

This guide describes models used by the NRC staff to estimate the radiological impacts resulting from uranium mills for the purpose of evaluating compliance with 40 CFR Part 190 and 10 CFR Part 20 and of assessing overall environmental radiological impacts in accordance with NEPA.

1. URANIUM MILL SOURCE TERMS

A uranium mill, unlike other types of fuel cycle facilities, goes through phases in its life cycle in which both the composition and the magnitude of its radioactive emissions (and associated impacts) vary greatly. For this reason, the NRC staff will perform impact evaluations for each individual mill at different phases of its existence. The three principal uranium mill life-cycle phases discussed in this guide are (1) operational (milling), (2) tailings pile drying and stabilization, and (3) reclamation.

Typically, a uranium mill will operate for a period of years during which there will be radon and particulate releases from the ore storage pile, the mill itself, and the tailings disposal area. During this operational period, both particulate and radon releases from the tailings pile may be somewhat curtailed by maintaining the pile at least partially under water. Mechanical sprinkler systems or chemical stabilizing agents may also be used to inhibit the suspension in air of radioactive tailings dust by the wind.

When actual milling ceases, the tailings pile is normally allowed to dry by natural evaporation until it is ready for stabilization. When the tailings are wet, there are essentially no particulate releases from the tailings pile. However, as the tailings pile dries, releases of radon and particulates from this source may increase, reaching their maximum prior to implementation of measures required to achieve long-term stabilization. After stabilization and reclamation of the tailings area, there should be no further radioactive particulate releases. However, small quantities of radon may continue to diffuse upward from the tailings and may be released to the atmosphere. These continuing radon releases, though small, are likely to persist for tens of thousands of years.

Depending on the specific details of the site, facility, effluent controls, and stabilization program, maximum individual particulate exposure could occur either during the last year of actual milling or the last year prior to stabilization of the tailings. Maximum individual doses due to radon releases are likely to occur during the last year prior to stabilization.

The radioactive isotopes comprising uranium mill radioactivity releases are mostly those belonging to the ^{238}U and ^{235}U decay series. The ^{235}U series radionuclides amount to less than 5 percent of total releases and are routinely

disregarded because of their insignificant contribution to overall radiological impact.

2. CRITICAL EXPOSURE PATHWAYS

Three exposure pathways of concern for airborne releases from uranium mills are (1) inhalation of airborne radioactive material, (2) ingestion of vegetable and animal products contaminated via deposition, and (3) direct external exposure to radiation emitted by airborne activity and activity deposited on ground surfaces. Liquid exposure pathways are not usually of concern because there are usually no discharges to surface water of liquid effluents. Liquid pathways may exist, however, and methodology similar to that used in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," should be used for evaluating intakes via the liquid pathway. However, ingestion dose factors from Table 6 should be used in converting intakes to doses.

All individual exposure pathways of significance will be evaluated at locations where the exposure pathway and a dose receptor actually exist at the time the analysis is made. Also, the applicant may take into account any real phenomena or actual exposure conditions that may be present. Such conditions could include actual values for agricultural productivity, dietary habits and food sources, occupancy times, measured environmental transport factors, or similar values determined for a specific site. However, if the analysis is based on existing conditions and if potential changes in land use and food pathways could result in significantly higher exposures, the applicant should provide reasonable assurance that a monitoring and surveillance program will be performed on a regular and continuing basis to determine if such changes have occurred.

3. REQUIRED DOSE ESTIMATES

3.1 Individual Doses

Evaluations of the dose received by an exposed individual are made to satisfy the requirements of both 40 CFR Part 190 and 10 CFR Part 20. The Environmental Protection Agency (EPA) regulation, 40 CFR Part 190, speaks to

individual radiation doses from all pathways and all nuclear power and fuel cycle facilities combined, except that exposure from radon and its daughters need not be included. The NRC regulation, 10 CFR Part 20, includes a requirement to keep all radiation exposures "as low as is reasonably achievable" (ALARA). ALARA is a general concept that has not to date been interpreted in the form of numerical design objectives for uranium mills as it has been for light-water-cooled nuclear reactors (see Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"). However, a case-by-case evaluation will be made to ensure that doses are kept as low as is reasonably achievable. ALARA evaluations will address all releases, including radon and its daughters, and will consider population doses as well as individual doses.

For the purpose of evaluating compliance with 40 CFR Part 190, the whole body and organ doses to any individual for all pathways combined and from all activity releases except radon and its daughters are evaluated for (1) the last year of actual mill operation and (2) the last year prior to tailings pile reclamation. These evaluations are adequate for assessing ALARA compliance except that exposure to radon and its daughters should be included and radon and daughter exposure for the first year after tailings pile reclamation should also be evaluated. Postreclamation exposure to radon and its daughters should be evaluated at the location of greatest radon concentration where unrestricted land use after mill decommissioning may be permitted.

Exposed individuals are characterized by food consumption, occupancy, and other uses of the region in the vicinity of the mill site. All physiological and metabolic parameters for the exposed individuals are assumed to have those characteristics that represent the averages for the various age groups in the general population. Although specific individuals will almost certainly display dietary, recreational, and other living habits considerably different from those suggested here and actual physiological and metabolic parameters may vary considerably, the NRC staff considers the use of these reference values to be acceptable because the actual physiological and metabolic characteristics of specific individuals cannot usually be determined. Applicants are encouraged to use information and data applicable to a specific region or site when possible.

When site-specific information and data are used, their origin or derivation should be documented for the NRC staff's review.

In this guide, the term "dose" is used instead of the more precise term "dose equivalent." When applied to the evaluation of internal deposition of radioactivity, the term "dose," as used here, includes the prospective dose component arising from retention in the body beyond the period of environmental exposure, i.e., the committed dose equivalent. The committed dose equivalent is evaluated over a period of 50 years.

The committed dose equivalent per unit intake, either by inhalation or ingestion, usually varies by age as well as by organ. For the purpose of calculating collective (population) doses, the population has been assumed to be composed of four age groups: infants (0 to 1 year), children (1 to 11 years), teenagers (11 to 17 years), and adults (17 years and older). Four sets of ingestion-dose conversion factors are presented in this guide, one for each of these four age groups. Available data are not sufficient to permit the calculation of age-specific dose conversion factors for inhalation exposure, and adult dose conversion factors are assumed to apply for all age groups for this exposure pathway.

3.2 Population Doses

Evaluations of population doses resulting from uranium milling operations are required to satisfy NEPA requirements for assessing the total environmental impact associated with the operation of each facility. Calculated estimates of resulting population doses therefore need to reflect, insofar as practicable, the overall radiological impact of each uranium mill over the duration of its existence.

For a typical uranium mill, the total radiological impact is composed of the impacts of the three major phases of its existence: the operational phase, the prereclamation phase, and the postreclamation phase. The first two phases may involve substantial releases of radon gas and particulates but are of relatively short duration. The postreclamation phase involves only small releases of radon, but these releases may persist for periods of tens of thousands of years. For each phase, the average annual radiological impact will be estimated by the NRC staff using the following basic procedure:

1. Annual average releases over the duration of the particular mill phase will be estimated for each radionuclide.

2. The radiological impact resulting from 1 year of average releases will be evaluated in terms of population dose using the EPA concept of "environmental dose commitment" (Ref. 3). The environmental dose commitment will be evaluated for a period of 100 years following release as per the procedure used by EPA in setting the standards in 40 CFR Part 190.

The total dose commitments for the operational and prereclamation phases will be calculated by multiplying the annual population dose commitments by the number of years the mill is expected to be in each phase. The sum of these two products represents an approximation of the combined radiological impact of the facility prior to tailings pile reclamation. The annual population dose commitments from postreclamation radon releases are also calculated and represent the continuously recurring impact of this residual activity source.

Consideration of particulate releases will generally be limited geographically to the area within 80 km (50 mi) of the mill site. Within this area, exposure pathways requiring assessment include all those considered in the evaluation of maximum individual exposure. Outside the 80-km (50-mi) radius, only radon and daughters require consideration and these are treated separately from particulate releases (see Regulatory Position 3.2).

4. USE OF THIS GUIDE

Present NRC staff practice with regard to the calculation of radioactive emission rates from uranium milling facilities involves the characterization of such releases by radionuclide, particle size, and density (Ref. 1). The data required as input for use of the calculational models described in this guide consist of annual average air concentrations resulting directly from such releases at specific locations (not including resuspended air concentrations of radioactive materials previously deposited on ground surfaces). The required input air concentrations for a particular location are denoted in this guide by the symbol C_{adip} (in pCi/m^3), where the subscripts indicate air concentration (a), direct (d), radionuclide (i), and particle size (p). Direct air concentrations required are those for values of the subscripts i and p as identified and defined in Table 1.

The primary calculational tool employed by the staff in performing radiological impact evaluations of uranium milling operations is the MILDOS code (Ref. 2), a modified version of the Argonne National Laboratory Uranium Dispersion and Dosimetry (UDAD) Code (Ref. 4). As used by the NRC staff, the MILDOS code has only five primary radionuclides in the ^{238}U decay chain that are treated explicitly as source terms. These radionuclides are ^{238}U, ^{230}Th, ^{226}Ra, ^{210}Pb, and ^{222}Rn. Release rates for these radionuclides are required for each potential onsite source (for particle sizes 1 through 4 in Table 1). For ^{222}Rn daughters, which grow in during transport of ^{222}Rn from the site, the resulting ingrowth concentrations (particle size 5 in Table 1) are also required. These ^{222}Rn daughters include ^{218}Po, ^{214}Pb, ^{214}Bi, ^{210}Pb, and ^{210}Po. The dosimetry model accounts for releases and ingrowth of other radionuclides, using assumptions of secular equilibrium.

Appendix A identifies and describes the various other site-specific information and data routinely used by the NRC staff in performing radiological impact assessments for uranium milling facilities. Appendix B provides a more detailed discussion of the method used in this guide for calculating environmental dose commitments. Appendix C provides a detailed explanation of the derivation of the radon dose conversion factor used in this guide.

C. REGULATORY POSITION

Equations and other data by which the NRC staff will estimate radiation exposure for individuals and the population in general from uranium mills are presented below. These equations are appropriate for the exposure pathways that the staff routinely considers in its evaluations. In addition, other pathways that may be present because of unique conditions at a specific site should be considered if they are likely to provide a significant contribution to total dose. A pathway is considered significant if a conservative evaluation yields an additional dose increment of more than 10 percent of the total from all other pathways considered in this guide.

1. CONCENTRATIONS IN ENVIRONMENTAL MEDIA

As discussed in Section B.4, annual average direct air concentrations are required as input data for use in the equations that follow. These equations

yield resulting concentrations in environmental media of interest, including total ground surface concentrations, air concentrations, and concentrations in edible vegetation, meat, and milk. These concentration calculations are explicitly performed only for certain radionuclides of the ^{238}U decay chain. Concentrations in environmental media of other radionuclides of the chain are inferred from those for which concentrations are explicitly calculated.

The basic calculational procedure first involves treatment of the direct air concentrations to obtain ground surface concentrations and resuspended air concentrations. Resuspension of radioactive materials deposited on ground surfaces is not treated as a loss mechanism for ground concentrations. For this reason, deposition of resuspended air concentrations onto ground surfaces is not considered. Resuspended particulate concentrations in air are added to the airborne concentrations arising directly from the source to obtain total air concentrations. The calculated total air concentrations are then used to obtain total deposition rates onto vegetation (resuspension losses of activity deposited on vegetation are assumed to be accounted for by the application of a weathering half-life). Total deposition rates and ground concentrations are used to compute concentrations in various vegetation types, including hay and forage. Radionuclide concentrations in hay and animal forage are initial inputs for the calculation of radionuclide concentrations in meat and milk ingested by man. This basic calculational process, the resulting environmental media concentrations, and the exposure pathways for which they are used are indicated schematically in Figure 1.

1.1 Radionuclide Accumulation on the Ground

Radionuclide ground concentrations are computed from the calculated airborne particulate concentrations arising directly from onsite sources (not including air concentrations resulting from resuspension). Resuspended particulate concentrations are not considered for evaluating ground concentrations. The direct deposition rate of radionuclide i is calculated, using the following relationship:

$$D_{di} = \sum_{p} C_{adip} V_p \qquad (1)$$

where

C_{adip} is the calculated direct air concentration of radionuclide i in particle size p in pCi/m^3;

D_{di} is the resulting direct deposition rate of radionuclide i in pCi/m^2 per sec; and

V_p is the deposition velocity of particle size p in m/sec (see Table 1).

The concentration of radionuclide i on a ground surface due to constant deposition at the rate D_{di} over time interval t is obtained from

$$C_{gi}(t) = D_{di}\left[\frac{1 - \exp[-(\lambda_i + \lambda_e)t]}{\lambda_i + \lambda_e}\right] \qquad (2)$$

where

$C_{gi}(t)$ is the calculated ground surface concentration of radionuclide i at time t in pCi/m^2;

t is the time interval over which deposition has occurred in sec;

λ_e is the assumed rate constant for environmental loss in sec^{-1}; and

λ_i is the radioactive decay constant* for radionuclide i in sec^{-1}.

The environmental loss constant λ_e corresponds to an assumed half-time for loss of environmental availability of 50 years (Ref. 1). This parameter accounts for downward migration in soil and loss of availability due to chemical binding. It is assumed to apply to all radionuclides deposited on the ground.

*Radiological decay constants employed by the NRC staff are obtained from data given in Reference 5.

Ground concentrations are explicitly computed only for ^{238}U, ^{230}Th, ^{226}Ra, and ^{210}Pb. For all other radionuclides, the ground concentration is assumed equal to that of the first parent radionuclide for which the ground concentration is explicitly calculated. For ^{210}Pb, ingrowth from deposited ^{226}Ra can be significant. The concentration of ^{210}Pb on the ground due to ^{226}Ra deposition is calculated by the staff, using the standard Bateman equation and ignoring the very-short-lived daughter radionuclides. This is equivalent to assuming that ^{226}Ra decays directly to ^{210}Pb. Using $i = 6$ for ^{226}Ra and $i = 12$ for ^{210}Pb (see Table 1), the following equation is obtained:

$$C_{g12}(Pb \leftarrow Ra) = \frac{\lambda_{12} D_{d6}}{\lambda_6^*} \left[\frac{1 - e^{-\lambda_{12}^* t}}{\lambda_{12}^*} + \frac{e^{-\lambda_6^* t} - e^{-\lambda_{12}^* t}}{\lambda_6^* - \lambda_{12}^*} \right] \qquad (3)$$

where

$C_{g12}(Pb \leftarrow Ra)$ is the incremental ^{210}Pb ground concentration resulting from ^{226}Ra deposition in pCi/m^2; and

λ_n^* is the effective rate constant for loss by radioactive decay and migration of a ground-deposited radionuclide and is equal to $\lambda_n + \lambda_e$ in sec^{-1}.

1.2 Total Air Concentrations

For use of the models described in this guide, air concentrations arising directly from onsite sources are required for each receptor location as a function of particle size (for particulates). Direct air concentrations are assumed to include the effects of depletion by deposition (particulates) or ingrowth and decay in transit (for radon and its daughters). In order to compute inhalation doses, the total air concentration of each radionuclide at each location (as a function of particle size) is computed as the sum of the direct air concentration and the resuspended air concentration:

$$C_{aip}(t) = C_{adip} + C_{arip}(t) \qquad (4)$$

where

C_{adip} is the calculated direct air concentration of radionuclide i in particle size p in pCi/m^3;

$C_{aip}(t)$ is the calculated total air concentration of radionuclide i in particle size p at time t in pCi/m^3; and

$C_{arip}(t)$ is the calculated resuspended air concentration of radionuclide i in particle size p at time t in pCi/m^3.

The resuspended air concentration is computed using a time-dependent and particle-size-dependent resuspension factor, which, for deposits of age t years, is defined by

$$R_p(t) = (0.01/V_p)10^{-5}\, e^{-\lambda_R t} \quad (\text{for } t \leq 1.82 \text{ yr}) \qquad (5a)$$

$$R_p(t) = (0.01/V_p)10^{-9} \quad (\text{for } t > 1.82 \text{ yr}) \qquad (5b)$$

where

$R_p(t)$ is the ratio of the resuspended air concentration to the ground concentration for a ground deposit of age t yr for particle size p in m^{-1};

λ_R is the assumed decay constant of the resuspension factor (equivalent to a 50-day half-life), 5.06 yr^{-1};

0.01 is the deposition velocity for the particle size for which the initial resuspension factor value is $10^{-5}/m$ in m/sec;

10^{-5} is the initial value of the resuspension factor for particles with a deposition velocity of 0.01 m/sec in m^{-1};

10^{-9} is the terminal value of the resuspension factor for particles with a deposition velocity of 0.01 m/sec in m^{-1}; and

1.82 is the time required to reach the terminal resuspension factor in yr.

The basic formulation of the above expression for the resuspension factor, the initial and final values, and the assigned decay constant derive from experimental observations (Ref. 1). The decrease with age primarily accounts for agglomeration with other larger particles. The inverse relationship to deposition velocity physically accounts for decreased resuspendibility of larger particles; mathematically, it eliminates mass balance problems for the 35-μm particle size. Based on this formulation, the resuspended air concentration is given by

$$C_{arip}(t) = 0.01 C_{adip} 10^{-5} \left[\frac{1 - \exp[-(\lambda_i^* + \lambda_R)(t - a)]}{(\lambda_i^* + \lambda_R)} + 10^{-4}\ \delta(t)\ \frac{\exp[-\lambda_i^*(t - a)] - \exp(-\lambda_i^* t)}{\lambda_i^*} \right] (3.156 \times 10^7) \quad (6)$$

where

- a is equal to $(t - 1.82)$ if $t > 1.82$ and is otherwise equal to zero in yr;
- $\delta(t)$ is zero if $t \leq 1.82$ and is unity otherwise, dimensionless;
- λ_i^* is the effective removal constant for radionuclide i on soil in yr^{-1}; and
- 3.156×10^7 is the number of seconds per year.

Equation 6 yields the resuspended air concentration of radionuclide i in particle size p because of deposition over time span t in years. Total air concentrations are computed using Equations 6 and 4 (in that order) for all particulates in particle sizes 1 through 4 as given in Table 1. Particulate daughters of ^{222}Rn (particle size 5 in Table 1) are not assumed to be depleted because of deposition and are also not assumed to resuspend.

1.3 Vegetation Concentrations

As illustrated in Figure 1, vegetation concentrations are derived from ground concentrations and total deposition rates. Total deposition rates are given by the following summation:

$$D_i = \sum_p C_{aip} V_p \quad (7)$$

where

- D_i is the total deposition rate, including deposition of resuspended activity, of radionuclide i in pCi/m^2 per sec.

Concentrations of released particulate materials can be environmentally transferred to the edible portions of vegetables or to hay or pasture grass consumed by animals by two mechanisms--direct foliar retention and root uptake. Five categories of vegetation are treated by the staff. They are edible above-ground vegetables, potatoes, other edible below-ground vegetables, pasture grass, and hay. Vegetation concentrations are computed using the following equation:

$$C_{vi} = D_i F_r E_v \left[\frac{1 - \exp(-\lambda_w t_v)}{Y_v \lambda_w} \right] + C_{gi} \frac{B_{vi}}{p} \qquad (8)$$

where

B_{vi} is the soil-to-plant transfer coefficient for radionuclide i and vegetation type v (pCi/kg(wet) plant per pCi/kg(dry) soil);

C_{vi} is the resulting concentration of radionuclide i in vegetation v in pCi/kg(wet weight);

E_v is the fraction of the foliar deposition reaching edible portions of vegetation v, dimensionless;

F_r is the fraction of the total deposition retained on plant surfaces, 0.2, dimensionless;

p is the assumed soil areal density for surface mixing, 240 kg(dry weight)/m^2;

t_v is the assumed duration of exposure while vegetation v is growing in sec;

Y_v is the assumed yield density of vegetation v in kg(wet weight)/m^2; and

λ_w is the decay constant accounting for weathering losses (equivalent to a 14-day half-life), 5.73 x 10^{-7} sec^{-1}.

The value of E_v is assumed to be 1.0 for all above-ground vegetation and 0.1 for all below-ground vegetables (Ref. 6). The value of t_v is taken to be 60 days, except for pasture grass for which a value of 30 days is assumed. The yield density Y_v is taken to be 2.0 kg/m^2, except for pasture grass for which a value of 0.75 kg/m^2 is applied. Values of the soil-to-plant transfer coefficients B_{vi} are provided in Table 2.

1.4 Meat and Milk Concentrations

Radioactive materials can be deposited on grasses, hay, or silage that are eaten by meat animals that are in turn eaten by man. The equation used to estimate radionuclide concentrations in meat is

$$C_{bi} = QF_{bi}(F_{pg}C_{pgi} + F_hC_{hi}) \quad (9)$$

where

C_{bi} is the resulting average concentration of radionuclide i in meat in pCi/kg;

C_{hi} is the concentration of radionuclide i in hay (or other stored feed) in pCi/kg(wet weight);

C_{pgi} is the concentration of radionuclide i in pasture grass in pCi/kg(wet weight);

F_{bi} is the feed-to-meat transfer coefficient for radionuclide i in pCi/kg per pCi/day ingested (see Table 2);

F_{pg},F_h are the fractions of the total annual feed requirement assumed to be satisfied by pasture grass or locally grown stored feed (hay), respectively, dimensionless; and

Q is the assumed feed ingestion rate, 50 kg(wet weight)/day (Ref. 6).

The equation used to estimate milk concentrations from cows ingesting contaminated feed is

$$C_{mi} = QF_{mi}(F_{pg}C_{pgi} + F_hC_{hi}) \quad (10)$$

where

C_{mi} is the resulting average concentration of radionuclide i in milk in pCi/L; and

F_{mi} is the feed-to-milk transfer coefficient for radionuclide i in pCi/L per pCi/day ingested (see Table 2).

1.5 Concentrations at Different Times

Maximum doses to individuals are calculated for the last year of mill operation and for the last year prior to tailings pile reclamation. This section explains the procedures used by the NRC staff to obtain annual average environmental media concentrations for these years.

In order to estimate average environmental media concentrations during the final year of actual mill operation, for an operational lifetime of T_o years, the value of the time variable t appearing in Equations 2, 3, 4, and 6 is set equal to T_o (in appropriate units). The resulting concentration values are those predicted for the end of the final year of operation and are assumed to represent average values existing over that year.

Environmental concentrations existing during the final prereclamation year result from postoperational releases and residual contamination due to releases during the period of mill operation. Because direct air concentrations from operational releases vanish, environmental concentrations due to operational releases at the time of reclamation arise only from residual ground and resuspended air concentrations. Ground concentrations at the end of the milling period are calculated using Equations 2 and 3, with the value of t set to T_o, the operational lifetime. Residual ground concentrations at the end of the final prereclamation year are then determined by

$$C_{gi}(T_d) = C_{gi}(T_o)\exp[-\lambda_i^*(T_d)] \qquad (11)$$

where

$C_{gi}(T_d)$ is the residual ground concentration of radionuclide i resulting from operational releases at the end of the T_d-year drying period in pCi/m^2;

$C_{gi}(T_o)$ is the ground concentration of radionuclide i at the time of mill shutdown in pCi/m^2; and

T_d is the duration of time required to dry the tailings pile prior to reclamation per yr.

Residual resuspended air concentrations resulting from operational releases are determined at the end of the final prereclamation year by

$$C_{arip}(T_d) = 0.01 C_{adip} 10^{-9} \exp[-\lambda_i^*(T_d)] \times \left[\frac{1 - \exp(-\lambda_i^* T_o)}{\lambda_i^*}\right] (3.156 \times 10^7) \qquad (12)$$

where

C_{adip} is the direct air concentration of radionuclide i in particle size p resulting from operational releases in pCi/m^3; and

$C_{arip}(T_d)$ is the residual resuspended air concentration of radionuclide i in particle size p resulting from operational releases at the end of the T_d-year drying period in pCi/m^3.

Ground and resuspended air concentrations resulting from postoperational releases at the end of the final prereclamation year are calculated using Equations 2, 3, 4, and 6 with the value of t equal to T_d. These concentrations are then incremented by the residual concentrations due to operational releases. These residual concentrations are calculated using Equations 11 and 12 to obtain the required totals. Total air concentrations and concentrations in vegetation, meat, and milk are then calculated from the total ground and resuspended air concentrations.

2. DOSE CALCULATIONS FOR INDIVIDUALS

Doses to individuals are calculated for inhalation, external exposure to air and ground concentrations, and ingestion of vegetables, milk, and meat. Internal doses are calculated using dose conversion factors that yield the 50-year committed dose equivalent, i.e., the entire dose received over a period of 50 years following either inhalation or ingestion. The annual doses are actually the 50-year committed dose equivalents resulting from a 1-year exposure period. The 1-year exposure period is taken to be the year when environmental concentrations resulting from plant operations are expected to be at their highest level.

2.1 Inhalation Doses

Inhalation doses are calculated from the total radionuclide concentration in air, including resuspended material. The inhalation dose conversion factors for radioactive particulate materials used in this analysis are presented in Table 3. With the exception of the dose conversion factors presented for "mass average lung," these dose conversion factors have been computed by Argonne National Laboratory's UDAD computer code (Ref. 4) in accordance with the Task Group Lung Model (TGLM) of the International Commission on Radiological Protection (Ref. 7). Dose conversion factors for the mass average lung have been computed by mass-averaging the UDAD-calculated dose conversion factors for the four regions of the TGLM: nasopharyngeal, tracheobronchial, pulmonary, and lymph. Ordinarily, the dose computed specifically for the pulmonary region is reported or presented as the "lung" dose. For the principal lung dose contributors (uranium and thorium), doses computed for the mass average lung are slightly higher than those calculated for the pulmonary region. The net overall effect, considering all radionuclides, is thus a slight increase in the reported lung dose.

In addition to the physical characteristics of the particulate matter involved, use of the TGLM demands the assignment of a solubility class, denoted by Y (years, slowly soluble or insoluble), W (weeks, moderately soluble), or D (days, quite soluble). Solubility classifications have been assigned on the basis of experimental data reported and summarized by Kalkwarf in NUREG/CR-0530 (Ref. 8). These data indicate that thorium, lead, and polonium are 100% class Y in ore, yellowcake, or tailings dusts. Radium was determined to be best characterized by the split-solubility classification 10% class D, 90% class Y. Uranium in ore dust was determined to be 100% class W; uranium solubility for tailings dusts was not analyzed and is assumed to be class Y. Data for uranium in yellowcake were mixed and showed a pronounced dependence on the specific source of the yellowcake sample. Results reported by Kalkwarf indicate a split-solubility classification is appropriate, and on review of those results (particularly those given on page 55 of Reference 8), the staff has assumed uranium in yellowcake to be 50% class D and 50% class Y. The computed inhalation dose conversion factors are given in Table 3.

Doses to the bronchial epithelium from ^{222}Rn and short-lived daughters are computed based on the assumption of indoor exposure with 100% occupancy.

The dose conversion factor for bronchial epithelium exposure from ^{222}Rn is derived as follows (see Appendix C for detailed basis):

1. 1 pCi/m^3 ^{222}Rn in outdoor air will yield an average indoor concentration of about 5 x 10^{-6} Working Level (WL).*
2. Continuous exposure to 1 WL = 25 cumulative working-level months (WLM) per year.
3. 1 WLM = 5000 mrem (Ref. 9).

Therefore,

$$1 \text{ pCi/m}^3 \ ^{222}Rn \times (5 \times 10^{-6} \frac{\text{WL}}{\text{pCi/m}^3}) \times (25 \frac{\text{WLM}}{\text{WL}})$$

$$\times (5000 \frac{\text{mrem}}{\text{WLM}}) = 0.625 \text{ mrem}$$

and the ^{222}Rn bronchial epithelium dose conversion factor is taken to be 0.625 mrem/yr per pCi/m^3.

Inhalation doses are computed by the staff by use of the following equation:

$$d_j(\text{inh}) = \sum_{ip} C_{aip} DCF_{ijp}(\text{inh}) \qquad (13)$$

where

$d_j(\text{inh})$ is the inhalation dose to organ j in mrem/yr; and

$DCF_{ijp}(\text{inh})$ is the inhalation dose conversion factor for radionuclide i, organ j, and particle size p in mrem/yr per pCi/m^3.

2.2 External Doses

External doses resulting from exposure to air and ground activity concentrations are computed, using the dose conversion factors presented in Table 4 and assuming 100 percent occupancy at a given location. Indoor exposure is assumed to occur 14 hours per day at a dose rate of 70 percent of the outdoor

*One WL concentration is defined as any combination of short-lived radioactive decay products of ^{222}Rn per liter of air that will release 1.3 x 10^5 MeV of alpha-particle energy during their radioactive decay to ^{210}Pb.

dose rate, which is equivalent to a dose reduction factor for structural shielding of 0.825. The following equation is used by the staff to calculate external doses:

$$d_j(\text{ext}) = 0.825 \sum_i C_{ai} DCF_{ij}(\text{cld}) + C_{gi} DCF_{ij}(\text{gnd}) \qquad (14)$$

where

C_{ai} is the total air concentration of radionuclide i in pCi/m^3;

$d_j(\text{ext})$ is the external dose to organ j in mrem/yr;

$DCF_{ij}(\text{cld})$ is the dose conversion factor for cloud exposure from radionuclide i to organ j in mrem/yr per pCi/m^3;

$DCF_{ij}(\text{gnd})$ is the dose conversion factor for ground exposure from radionuclide i to organ j in mrem/yr per pCi/m^2; and

0.825 is the effective reduction factor because of structural shielding for indoor exposure periods.

2.3 Ingestion Doses

Ingestion doses are routinely calculated for ingestion of vegetables and meat (beef, unprocessed pork, and lamb). Milk ingestion doses are also computed if that pathway exists at the time of licensing. Ingestion doses are based on environmental concentrations established using Equations 8, 9, and 10, ingestion rates given in Table 5, and dose conversion factors given in Table 6. Ingestion doses from vegetable consumption are computed under the assumption that an average of 50 percent of the initial activity will be lost in food preparation (Ref. 6), usually involving washing, peeling, boiling, etc. The following equation is used to compute the annual radionuclide intake via ingestion:

$$I_{ik} = U_{mk} C_{mi} + U_{bk} C_{bi} + 0.5 \sum_v U_{vk} C_{vi} \qquad (15)$$

where

I_{ik} is the activity ingestion rate of radionuclide i by an individual in age group k in pCi/yr;

U_{mk}, U_{bk} are milk (in L/yr) and meat (in kg/yr) ingestion rates for an individual in age group k;

U_{vk} is the ingestion rate of vegetable category v for age group k in kg(wet weight)/yr; and

0.5 is the fraction of vegetable activity remaining after food preparation, dimensionless.

Ingestion doses are then computed by

$$d_{jk}(ing) = \sum_i I_{ik} DCF_{ijk}(ing) \qquad (16)$$

where

$d_{jk}(ing)$ is the ingestion dose for organ j of an individual in age group k in mrem/yr; and

$DCF_{ijk}(ing)$ is the ingestion dose conversion factor for radionuclide i in organ j of an individual in age group k in units of mrem/pCi ingested.

2.4 Individual Dose Totals

Individual doses are calculated by the NRC staff for purposes of evaluating compliance with 10 CFR Part 20 and 40 CFR Part 190. For evaluating compliance with 40 CFR Part 190, dose contributions from ^{222}Rn and daughters are excluded. Total doses to individuals are calculated for both purposes using the following equation, which sums the dose contributions from inhalation, external dose, and ingestion:

$$d_{jk}(tot) = d_j(inh) + d_j(ext) + d_{jk}(ing) \qquad (17)$$

where

$d_{jk}(tot)$ is the total dose to organ j of an individual in age group k from all exposure pathways in mrem/yr.

To evaluate compliance with 40 CFR Part 190, the staff will compute total doses to appropriate individual receptors, using the above equation and all other models, data, and assumptions described in this guide, except that--

1. all dose contributions from radiation emitted by ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, and ^{214}Po will be excluded, and

2. all dose contributions from radiation emitted by ^{210}Pb, ^{210}Bi, and ^{210}Po formed by decay of released ^{222}Rn will be excluded.

With reference to Table 1 of this guide, the dose contributions eliminated for the purpose of evaluating compliance with 40 CFR Part 190 include those due to any radiation emitted by (a) radionuclides for which i = 7, 8, 9, 10, or 11 and (b) radionuclides present in particle size category p = 5 (radon daughters). The staff will add to dose totals computed for evaluating compliance with 40 CFR Part 190 any known significant doses resulting from any other light-water-cooled nuclear power generating or fuel cycle facilities, as appropriate (excluding doses from ^{222}Rn and its daughters as stipulated above and excluding doses from any radioactive materials released by nuclear or other facilities or operations not included under 40 CFR Part 190).

3. POPULATION DOSE CALCULATIONS

Population doses are calculated, using the environmental dose commitment concept with an integrating period of 100 years (Ref. 3). Under this approach, radiological impacts for a given release of activity are integrated over a time interval of 100 years following the release. The 100-year environmental dose commitment resulting from average release rates over a 1-year period is computed for (1) the period of actual uranium milling and (2) the period of time after the cessation of milling during which tailings are allowed to dry prior to final stabilization and reclamation. The NRC staff's rationale for the selection and use of a 100-year integrating period and the staff's technique for computing environmental dose commitments are addressed in Appendix B to this guide.

Population doses resulting from particulate and radon releases are evaluated over the general region of the facility site for the first two phases of the mill life cycle: operational (milling) and prereclamation. For these two time intervals and for the postreclamation era, annual population dose commitments resulting from transcontinental dispersion of ^{222}Rn are also evaluated.

3.1 Regional Population Doses

Population doses resulting from environmental radioactivity concentrations in the region of the site are evaluated for all exposure pathways considered in the evaluation of maximum individual doses; other pathways should also be considered if they are likely to result in an increase of more than 10 percent to the total result. Regional population dose commitments are generally computed on the basis of the population and agricultural productivity within a distance of 80 km (50 mi). Individual localized population centers lying beyond this distance should also be considered if their inclusion would increase the population dose estimates by more than 10 percent.

3.1.1 Inhalation and External Doses

Inhalation and external doses are computed by the NRC staff, using the identical models, equations, data, and assumptions as previously described for individual dose calculations in Regulatory Positions 1 and 2 of this guide. The procedure for calculating regional population doses from those pathways is to (1) divide the geographical site region into segments by radius and direction, (2) establish average individual doses within each segment, (3) multiply these individual doses by the estimated population lying within each segment, and (4) sum over all segments.

The population distribution required is that projected for the final year of mill operation. The appropriate population projection should be presented for each segment formed by radii extending outward from the site and bisecting the 16 compass directions (forming 22.5° sectors) and concentric circles drawn at distances of 1, 2, 3, 4, 5, 10, 20, 30, 40, 50, 60, 70, and 80 km. The 13 circles and 16 radii then form a grid composed of 192 individual segments. Average doses over the population within each segment are computed by the NRC staff along the segment directional centerline at a distance midway between the inner and outer boundaries of each annulus.

The population dose in the site region from inhalation and external exposure pathways is computed by the staff using the following equation:

$$M_j(\text{inh} + \text{ext}) = 10^{-3} \sum_s . P_s[d_{js}(\text{inh}) + d_{js}(\text{ext})] \qquad (18)$$

where

$d_{js}(ext)$ is the average external dose to organ j in segment s in mrem/yr;

$d_{js}(inh)$ is the average inhalation dose to organ j in segment s in mrem/yr;

$M_j(inh+ext)$ is the resulting population dose from inhalation and external exposure pathways in rem/yr;

P_s is the population residing in segment s; and

10^{-3} is the conversion factor from millirem to rem.

3.1.2 Food Ingestion Doses

Collective population doses from food ingestion are calculated on the basis of the region's agricultural productivity rather than its population. This is because the total population dose from food pathways is proportional to the total quantity of radionuclides in all food produced in the region rather than the number of people exposed. The model employed by the NRC staff considers population doses resulting from radioactive contamination of vegetable, meat, and milk products produced in the region. For population dose calculations, the vegetable category includes fruit and grain crops as well. The procedure followed by the staff to compute food ingestion doses is similar to that used for inhalation and external doses and is composed of the following procedural steps:

1. The site region is divided into segments and each segment is assigned a productivity rate for each food category (vegetables, meat, and milk in kg/yr per km^2);

2. The average activity concentrations for each food type are computed and multiplied by the segment productivity factor and by the segment area;

3. Total activity content of the regional food production is then determined by summing over the segments; and

4. Population doses are determined assuming that all food produced in the region is consumed by a population with the same age distribution as the U.S. population.

Agricultural productivity data required for use in this analysis are generally available on a county-by-county basis for a relatively recent year.

The available raw data should be projected forward in time to provide a reasonable estimate of productivity during the final year of mill operation. If other means are not available, the NRC staff considers it acceptable to assume that regional agricultural productivity will remain in constant proportion to the U.S. population. Should other site-specific data not be available, the staff will rely on the statewide average productivity data presented in Table 7. The following equation is used to obtain segment average radionuclide concentrations in vegetables:

$$C_{vis}(avg) = \sum_{v} W_{vs} C_{vis} \qquad (19)$$

where

C_{vis} is the average concentration of radionuclide i in vegetable type v produced in segment s in pCi/kg(wet weight);

$C_{vis}(avg)$ is the average concentration of radionuclide i averaged over all types of vegetables in segment s in pCi/kg; and

W_{vs} is the weighting factor for vegetable type v in segment s (fraction of total production), dimensionless.

When relying on the state-average production data given in Table 7, the NRC staff will use values of W_v that have been selected to roughly correspond to the fractions of the three vegetable types in the average diet. From Reference 1, these W_v values are 0.78 for above-ground vegetables, 0.20 for potatoes, and 0.02 for other below-ground vegetables.

The gross activity content of the regional food production for each food type (vegetables, meat, or milk) is obtained by

$$Q_{fi} = \sum_{s} G_{fs} A_s C_{fis} \qquad (20)$$

where

A_s is the area of segment s in km^2;

C_{fis} is the concentration of radionuclide i in food category f in segment s in pCi/kg(wet weight);

G_{fs} is the productivity factor for food f in segment s in kg/yr per km^2; and

Q_{fi} is the gross activity content of radionuclide i in food f in pCi/yr.

Since the food produced may be eaten at different rates by different age groups and since ingestion dose conversion factors are also age dependent, it is necessary to establish the fractions of the Q_{fi} values determined by Equation 20 that are ingested by the various age groups. The following relationship applies:

$$F_{fk} = \frac{F_{pk}U_{fk}}{\sum_{k} F_{pk}U_{fk}} \qquad (21)$$

where

F_{fk} is the fraction of the production of food type f ingested by individuals in age group k, dimensionless;

F_{pk} is the fraction of the regional population belonging to age group k, dimensionless; and

U_{fk} is the average consumption rate in kg/yr or L/yr (for milk or other liquids) of food type f for an individual in age group k (see Table 8 for values). In the absence of suitable site-specific information, the NRC staff will assume average consumption rates for the population at large as given in Table 8 and population age fractions and fractional consumption rates as given in Table 9.

Using values obtained from Equations 20 and 21, total population ingestion doses from all food categories are calculated by

$$M_j(ing) = 10^{-3} \sum_{fik} E_f Q_{fi} F_{fk} DCF_{ijk}(ing) \qquad (22)$$

where

E_f is a factor to account for activity remaining after food preparation, dimensionless; and

$M_j(ing)$ is the resulting regional population dose from food ingestion for organ j in rem/yr.

The value of E_f is assumed to be 0.5 for vegetables and 1.0 for meat and milk. Fractions of the population belonging to the various age groups used in Equation 20 are determined from U.S. census data in the absence of site-specific information (see Table 9 for values).

3.2 Continental Population Doses

Substantial contributions to the total population dose may arise from the transport of released ^{222}Rn across the North American continent. Formation of long-lived ^{210}Pb from ^{222}Rn may result in both inhalation and ingestion doses not only to people in the United States, but to people in Canada and Mexico as well (Ref. 10). In order to estimate population doses occurring beyond the immediate region of the site, the staff makes use of the data presented in Table 10. These data consist of estimates of population doses resulting from 1,000-Ci releases of ^{222}Rn from four specific locations in the western United States. The location closest to the mill site should be used. The population doses provided are those that would have resulted from releases during calendar year 1978, including doses to Canadian and Mexican populations, and are based on the use of the environmental dose commitment concept with an integrating period of 100 years.

For projected releases of ^{222}Rn in future years, resulting population doses are computed by assuming those doses to be proportional to the U.S. population (use the population data provided in Table 11). The anticipated annual ^{222}Rn release in kCi is multiplied by the appropriate population doses from Table 10, and these results are then multiplied by the ratio of the projected U.S. population for the year of release to the 1978 U.S. population.

3.3 Total Population Dose Commitments

Population doses over the site region and the North American continent are computed on an annual basis for the operational (milling), prereclamation (pile drying), and postreclamation phases. The total radiological impact due to emissions during the first two phases is estimated by multiplying the annual

impacts by the durations and summing. Total annual impacts for each of the three phases are obtained by

$$M_j = M_j(\text{inh} + \text{ext}) + M_j(\text{ing}) + M_j(\text{Rn}) \qquad (23)$$

where

M_j is the annual committed population dose to organ j in rem/yr; and

$M_j(\text{Rn})$ is the annual continental population dose from ^{222}Rn and its daughters to organ j in rem/yr.

Total impacts over the first two phases are obtained by

$$M_j(\text{m\&d}) = T_o M_j(\text{m}) + T_d M_j(\text{d}) \qquad (24)$$

where

$M_j(\text{d})$ is the annual committed population dose to organ j during the drying phase in rem/yr;

$M_j(\text{m})$ is the annual committed population dose to organ j during the milling phase in rem/yr;

$M_j(\text{m\&d})$ is the aggregate committed population dose to organ j over the milling and drying phases in rem; and

T_o, T_d are the durations of the operational and pile-drying phases, respectively, in yr.

The calculation, compilation, and presentation of these population doses is considered by the NRC staff to represent a reasonably complete description of the radiological impact incurred by the operation of a typical uranium mill.

D. IMPLEMENTATION

The models specified in this guide are being used by the NRC staff in evaluating radiological impact in connection with applications for uranium mill licenses and renewals.

Table 1

ISOTOPES AND PARTICLE SIZES FOR WHICH DIRECT AIR CONCENTRATIONS (C_{adip} VALUES) ARE REQUIRED AS INPUT DATA

Particle Size Group*	Diameter Range, μm	Mean Diameter, μm	Density, g/cm^3	Unit Density Activity--Median Aerodynamic Equivalent Diameter (AMAD), μm	Deposition Velocity, m/sec
p = 1	-	1.0	8.9	3.0	1.0×10^{-2}
p = 2	-	1.0	2.4	1.5	1.0×10^{-2}
p = 3	1 to 10	5.0	2.4	7.75	1.0×10^{-2}
p = 4	10 to 80	35.0	2.4	54.0	8.82×10^{-2}
p = 5	-	0.3	1.0	0.3	0.3×10^{-2}

		Particle Size Group Index**				
i	Radionuclide	p = 1	p = 2	p = 3	p = 4	p = 5
1	uranium-238	C & R	C & R	C & R	C & R	-
2	thorium-234	se	se	se	se	-
3	protactinium-234	se	se	se	se	-
4	uranium-234	se	se	se	se	-
5	thorium-230	C & R	C & R	C & R	C & R	-
6	radium-226	C & R	C & R	C & R	C & R	-
7	radon-222***	se	se	se	se	-
8	polonium-218	se	se	se	se	C & R
9	lead-214	se	se	se	se	C & R
10	bismuth-214	se	se	se	se	C & R
11	polonium-214	se	se	se	se	se
12	lead-210	C & R	C & R	C & R	C & R	C & R
13	bismuth-210	se	se	se	se	C & R
14	polonium-210	se	se	se	se	C & R

*Particle size groups are assigned to effluents as follows: p = 1 for yellowcake dust; p = 2, 3, or 4 for fugitive ore and tailings dusts; p = 5 for ^{222}Rn air ingrowth concentrations of particulate daughters.

**The entry "C & R" indicates that the particular C_{adip} value is explicitly calculated by the staff and required as input for use of the models, equations, and data described in this guide. The entry "se" indicates that radionuclide is assumed to be in secular equilibrium with the next-higher-up parent for which the direct air concentration is explicitly calculated.

***The air concentration of ^{222}Rn is also calculated by the staff and is required as input for use of this guide; ^{222}Rn gas is not assigned a particle size.

Table 2

ENVIRONMENTAL TRANSFER COEFFICIENTS*

	Transfer Coefficent			
	U	Th	Ra	Pb
Plant/Soil (B_{vi})				
(pCi/kg plant - wet weight)/(pCi/kg soil - dry weight)				
Edible Above Ground	2.5×10^{-3}	4.2×10^{-3}	1.4×10^{-2}	4.0×10^{-3}
Potatoes	2.5×10^{-3}	4.2×10^{-3}	3.0×10^{-3}	4.0×10^{-3}
Other Below Ground	2.5×10^{-3}	4.2×10^{-3}	1.4×10^{-2}	4.0×10^{-3}
Pasture Grass	2.5×10^{-3}	4.2×10^{-3}	1.8×10^{-2}	2.8×10^{-2}
Stored Feed (Hay)	2.5×10^{-3}	4.2×10^{-3}	8.2×10^{-2}	3.6×10^{-2}
Beef/Feed (F_{bi})				
(pCi/kg per pCi/day)	3.4×10^{-4}	2.0×10^{-4}	5.1×10^{-4}	7.1×10^{-4}
Milk/Feed (F_{mi})				
(pCi/L per pCi/day)	6.1×10^{-4}	5.0×10^{-6}	5.9×10^{-4}	1.2×10^{-4}

*Sources for these data include References 11-14.

Table 3

INHALATION DOSE CONVERSION FACTORS

	Conversion Factor, mrem/yr per pCi/m³					
Radon Decay Products						
Particle Size = 0.3 micron Density = 1.0 g/cm³ AMAD = 1.0 microns	^{210}Pb	^{210}Po				
Whole Body	7.46E+00	1.29E+00				
Bone	2.32E+02*	5.24E+00				
Kidney	1.93E+02	3.87E+01				
Liver	5.91E+01	1.15E+01				
Mass Average Lung	6.27E+01	2.66E+02				
Yellowcake Dust						
Particle Size = 1.0 micron Density = 8.9 g/cm³ AMAD = 3 microns	^{238}U	^{234}U	^{230}Th	^{226}Ra	^{210}Pb	^{210}Po
Whole Body	9.82E+00	1.12E+01	1.37E+02	3.58E+01	4.66E+00	5.95E-01
Bone	1.66E+02	1.81E+02	4.90E+03	3.58E+02	1.45E+02	2.43E+00
Kidney	3.78E+01	4.30E+01	1.37E+03	1.26E+00	1.21E+02	1.79E+01
Liver	0.0	0.0	2.82E+02	4.47E-02	3.69E+01	5.34E+00
Mass Average Lung	1.07E+3	1.21E+3	2.37E+03	4.88E+03	5.69E+02	3.13E+02
Uranium Ore Dust						
Particle Size = 1.0 micron Density = 2.4 g/cm³ AMAD = 1.5 microns	^{238}U	^{234}U	^{230}Th	^{226}Ra	^{210}Pb	^{210}Po
Whole Body	4.32E+00	4.92E+00	1.66E+02	3.09E+01	4.36E+00	4.71E-01
Bone	7.92E+01	7.95E+01	5.95E+03	3.09E+02	1.35E+02	1.92E+00
Kidney	1.66E+01	1.89E+01	1.67E+03	1.09E+00	1.13E+02	1.42E+01
Liver	0.0	0.0	3.43E+02	3.87E-02	3.45E+01	4.22E+00
Mass Average Lung	1.58E+02	1.80E+02	3.22E+03	6.61E+03	7.72E+02	4.20E+02
Fine Tailings Particulates						
Particle Size = 5.0 microns Density = 2.4 g/cm³ AMAD = 7.75 microns	^{238}U	^{234}U	^{230}Th	^{226}Ra	^{210}Pb	^{210}Po
Whole Body	1.16E+00	1.32E+00	1.01E+02	4.00E+01	4.84E+00	7.10E-01
Bone	1.96E+01	2.14E+01	3.60E+03	4.00E+02	1.50E+02	2.89E+00
Kidney	4.47E+00	5.10E+00	1.00E+03	1.41E+00	1.25E+02	2.13E+01
Liver	0.0	0.0	2.07E+02	4.97E-02	3.83E+01	6.36E+00
Mass Average Lung	1.24E+03	1.42E+03	1.38E+03	2.84E+03	3.30E+02	1.88E+02
Coarse Tailings Particulates						
Particle Size = 35.0 microns Density = 2.4 g/cm³ AMAD = 54 microns	^{238}U	^{234}U	^{230}Th	^{226}Ra	^{210}Pb	^{210}Po
Whole Body	7.92E-01	9.02E-01	5.77E+01	4.90E+01	4.43E+00	7.28E-01
Bone	1.34E+01	1.46E+01	2.07E+03	3.90E+02	1.38E+02	2.96E+00
Kidney	3.05E+00	3.47E+00	5.73E+02	1.38E+00	1.15E+02	2.19E+01
Liver	0.0	0.0	1.19E+02	4.85E-02	3.51E+01	6.52E+00
Mass Average Lung	3.33E+02	3.80E+02	3.71E+02	7.64E+02	8.70E+01	5.75E+01

*Read 2.32E+02 as $2.32 \times 10^2 = 232$.

Table 4

DOSE CONVERSION FACTORS FOR EXTERNAL EXPOSURE

Radionuclide	Dose Factor for External Dose from Air Concentrations mrem/yr per pCi/m^3	
	Skin	Whole Body*
^{238}U	1.05E-05**	1.57E-06
^{234}Th	6.63E-05	5.24E-05
^{234m}Pa	8.57E-05	6.64E-05
^{234}U	1.36E-05	2.49E-06
^{230}Th	1.29E-09	3.59E-06
^{226}Ra	6.00E-05	4.90E-05
^{222}Rn	3.46E-10	2.83E-06
^{218}Po	8.18E-07	6.34E-07
^{214}Pb	2.06E-03	1.67E-03
^{214}Bi	1.36E-02	1.16E-02
^{214}Po	9.89E-07	7.66E-07
^{210}Pb	4.17E-05	1.43E-05

Radionuclide	Dose Factor for External Dose from Ground Concentrations mrem/yr per pCi/m^2	
	Skin	Whole Body*
^{238}U	2.13E-06	3.17E-07
^{234}Th	2.10E-06	1.66E-06
^{234m}Pa	1.60E-06	1.24E-06
^{234}U	2.60E-06	4.78E-07
^{230}Th	2.20E-06	6.12E-07
^{226}Ra	1.16E-06	9.47E-07
^{222}Rn	6.15E-08	5.03E-08
^{218}Po	1.42E-08	1.10E-08
^{214}Pb	3.89E-05	3.16E-05
^{214}Bi	2.18E-04	1.85E-04
^{214}Po	1.72E-08	1.33E-08
^{210}Pb	6.65E-06	2.27E-06

*Doses to internal body organs are assumed to be the same as computed for the whole body.
**Read as 1.05×10^{-5} or 0.0000105.

Table 5

FOOD CONSUMPTION RATES USED FOR CALCULATING DOSES TO INDIVIDUALS

	Ingestion Rate by Age Group,* kg/yr			
	Infant	Child	Teen	Adult
Vegetables (Total)	-	47.8	76.1	105.
Edible Above Ground	-	17.3	28.9	39.9
Potatoes	-	27.2	42.2	60.4
Other Below Ground	-	3.3	5.0	5.0
Meat (Beef, Fresh Pork, and Lamb)	-	27.6	44.8	78.3
Milk (L/yr)	208.0	208.0	246.0	130.0

*All data are taken from Reference 6. Ingestion rates are averages for typical farm households. No allowance is routinely credited for portions of year when locally grown or home-grown food may not be available.

Table 6

INGESTION DOSE CONVERSION FACTORS

Age Group	Organ	Internal Dose Conversion Factor by Organ and Age, mrem per pCi ingested							
		^{238}U	^{234}U	^{234}Th	^{230}Th	^{226}Ra*	^{210}Pb	^{210}Bi	^{210}Po
Infant	Wh. Bod	3.33E-04	3.80E-04	2.00E-08	1.06E-04	1.07E-02	2.38E-03	3.58E-07	7.41E-04
	Bone	4.47E-03	4.88E-03	6.92E-07	3.80E-03	9.44E-02	5.28E-02	4.16E-06	3.10E-03
	Liver	0	0	3.77E-08	1.90E-04	4.76E-05	1.42E-02	2.68E-05	5.93E-03
	Kidney	9.28E-04	1.06E-03	1.39E-07	9.12E-04	8.71E-04	4.33E-02	2.08E-04	1.26E-02
Child	Wh. Bod	1.94E-04	2.21E-04	9.88E-09	9.91E-05	9.87E-03	2.09E-03	1.69E-07	3.67E-04
	Bone	3.27E-03	3.57E-03	3.42E-07	3.55E-03	8.76E-02	4.75E-02	1.97E-06	1.52E-03
	Liver	0	0	1.51E-08	1.78E-04	1.84E-05	1.22E-02	1.02E-05	2.43E-03
	Kidney	5.24E-04	5.98E-04	8.02E-08	8.67E-04	4.88E-04	3.67E-02	1.15E-04	7.56E-03
Teenager	Wh. Bod	6.49E-05	7.39E-05	3.31E-09	6.00E-05	5.00E-03	7.01E-04	5.66E-08	1.23E-04
	Bone	1.09E-03	1.19E-03	1.14E-07	2.16E-03	4.09E-02	1.81E-02	6.59E-07	5.09E-04
	Liver	0	0	6.68E-09	1.23E-04	8.13E-06	5.44E-03	4.51E-06	1.07E-03
	Kidney	2.50E-04	2.85E-04	3.81E-08	5.99E-04	2.32E-04	1.72E-02	5.48E-05	3.60E-03
Adult	Wh. Bod	4.54E-05	5.17E-05	2.13E-09	5.70E-05	4.60E-03	5.44E-04	3.96E-08	8.59E-05
	Bone	7.67E-04	8.36E-04	8.01E-08	2.06E-03	4.60E-02	1.53E-02	4.61E-07	3.56E-04
	Liver	0	0	4.71E-09	1.17E-04	5.74E-06	4.37E-03	3.18E-06	7.56E-04
	Kidney	1.75E-04	1.99E-04	2.67E-08	5.65E-04	1.63E-04	1.23E-02	3.83E-05	2.52E-03

*Adult whole body and bone dose conversion factors for ^{226}Ra have been obtained from Reference 6 and are based on applicable models and data from Reference 15. ^{226}Ra whole body and bone dose conversion factors for other age groups have been computed by assuming the same proportion to adult whole body and bone dose factors as given in Reference 16. All other dose conversion factors are directly from Reference 16.

Table 7

AVERAGE AGRICULTURAL PRODUCTIVITY FACTORS FOR VARIOUS STATES

State	State-Average Productivity,* kg/yr per km^2		
	Vegetables	Meat	Milk
Arizona	580	1,040	1,130
Colorado	2,800	3,200	1,400
Idaho	14,200	2,000	3,400
Montana	1,800	2,000	370
Nevada	18	510	230
New Mexico	280	1,150	460
South Dakota	2,400	6,400	3,600
Texas	1,200	5,300	2,100
Utah	370	790	1,800
Washington	10,700	1,600	6,000
Wyoming	320	1,400	230

*Data presented are based on a staff survey and analysis of available data on agricultural productivity for 1973.

Table 8

FOOD CONSUMPTION RATES USED FOR CALCULATING DOSES TO POPULATIONS

Food Category	Average Consumption Rates,* kg/yr) Infants	Children	Teens	Adults
Vegetable Pathway				
Berries and Tree Fruit	0	54.1	63.9	49.2
Fresh Vegetables**				
1. Potatoes	0	27.2	42.3	60.4
2. Other root veg.	0	3.4	5.0	5.0
3. Leafy vegetables	0	5.8	9.4	13.9
4. Other above-ground vegetables	0	11.4	19.5	26.0
Processed Vegetables				
1. Potatoes	0	2.3	3.6	5.2
2. Other root veg.	0	0.9	1.4	1.4
3. Leafy vegetables	0	0.4	0.6	0.8
4. Other above-ground vegetables	0	14.4	24.6	32.8
Grain, Rice, and Wheat	0	118.2	136.2	90.8
Total Vegetables	0	238.1	306.5	285.5
Meat Pathway				
Beef and Lamb**	0	21.8	35.9	64.0
Unprocessed Pork**	0	5.9	8.9	14.3
Poultry and Processed Pork	0	21.0	33.2	49.6
Total Meat	0	48.7	78.0	127.9
Milk Pathway (L/yr)				
Fresh Milk**	207.6	207.6	246.0	129.6
Milk Products	0	27.2	45.4	46.7
Total Milk	207.6	234.8	291.4	176.3

*All data are taken from Reference 6 and are representative of average consumption rates by individuals at farm residences.

**These food categories are evaluated for individual doses from ingestion pathways.

Table 9

AGE DISTRIBUTION OF POPULATION, AVERAGE AND PER CAPITA CONSUMPTION RATES, AND FRACTIONS USED IN THE ABSENCE OF SITE-SPECIFIC DATA

Age Group	Fraction of Population*	Average Total Consumption Rates,** kg/yr		
		Vegetables	Meat	Milk
Infants	0.0179	0	0	207.6
Children	0.1647	238.1	48.7	234.8
Teenagers	0.1957	306.5	78.0	291.4
Adults	0.6217	285.5	127.9	176.3

	Fraction of Regional Production Ingested by Each Age Group		
Age Group	Vegetables	Meat	Milk
Infants	0	0	0.0178
Children	0.1418	0.0780	0.1850
Teenagers	0.2167	0.1485	0.2728
Adults	0.6415	0.7735	0.5244

*Age fractions given reflect average values for the entire U.S. population indicated by 1970 census data, as reported in Reference 17.

**Consumption rates given are from Table 8 and are not those used for, or appropriate to, the calculation of maximum individual doses.

Table 10

CONTINENTAL POPULATION DOSES PER kCi OF ^{222}Rn RELEASED IN 1978

Release Site	Population Dose Resulting from a 1-kCi Release of ^{222}Rn During 1978, organ-rem*			
	Bronchial Epithelium	Whole Body	Pulmonary Lung	Bone
Casper, Wyoming	56.	8.8	2.0	120.
Falls City, Texas	72.	5.8	1.6	77.
Grants, New Mexico	52.	8.2	1.8	110.
Wellpinit, Washington	43.	9.0	1.7	120.
Average	56.	8.0	1.8	110.

* Values given are based on data reported in Reference 10 and amended for inclusion in Reference 1. Exposure pathways considered include inhalation and ingestion. Isotopes considered include ^{222}Rn and its short-lived daughters, ^{210}Pb, ^{210}Bi, and ^{210}Po. A 100-year integrating period was used in the application of the environmental dose commitment concept.

Table 11

PROJECTED POPULATION OF THE UNITED STATES, 1978-2100

Year	Projected U.S. Population, millions*	Year	Projected U.S. Population, millions*
1978	218.4	1992	247.4
1979	220.2	1993	249.3
1980	222.2	1994	251.1
1981	224.2	1995	252.8
1982	226.3	1996	254.4
1983	228.5	1997	255.9
1984	230.7	1998	257.5
1985	232.9	1999	258.9
1986	235.1	2000	260.4
1987	237.2	2025	287.5
1988	239.4	2050	291.1
1989	241.5	2075	291.9
1990	243.5	2100	293.0
1991	245.5		

*Population projections through the year 2000 are from Reference 18. Later projections were obtained from Reference 10 and are based on a predicted growth rate obtained from Reference 19.

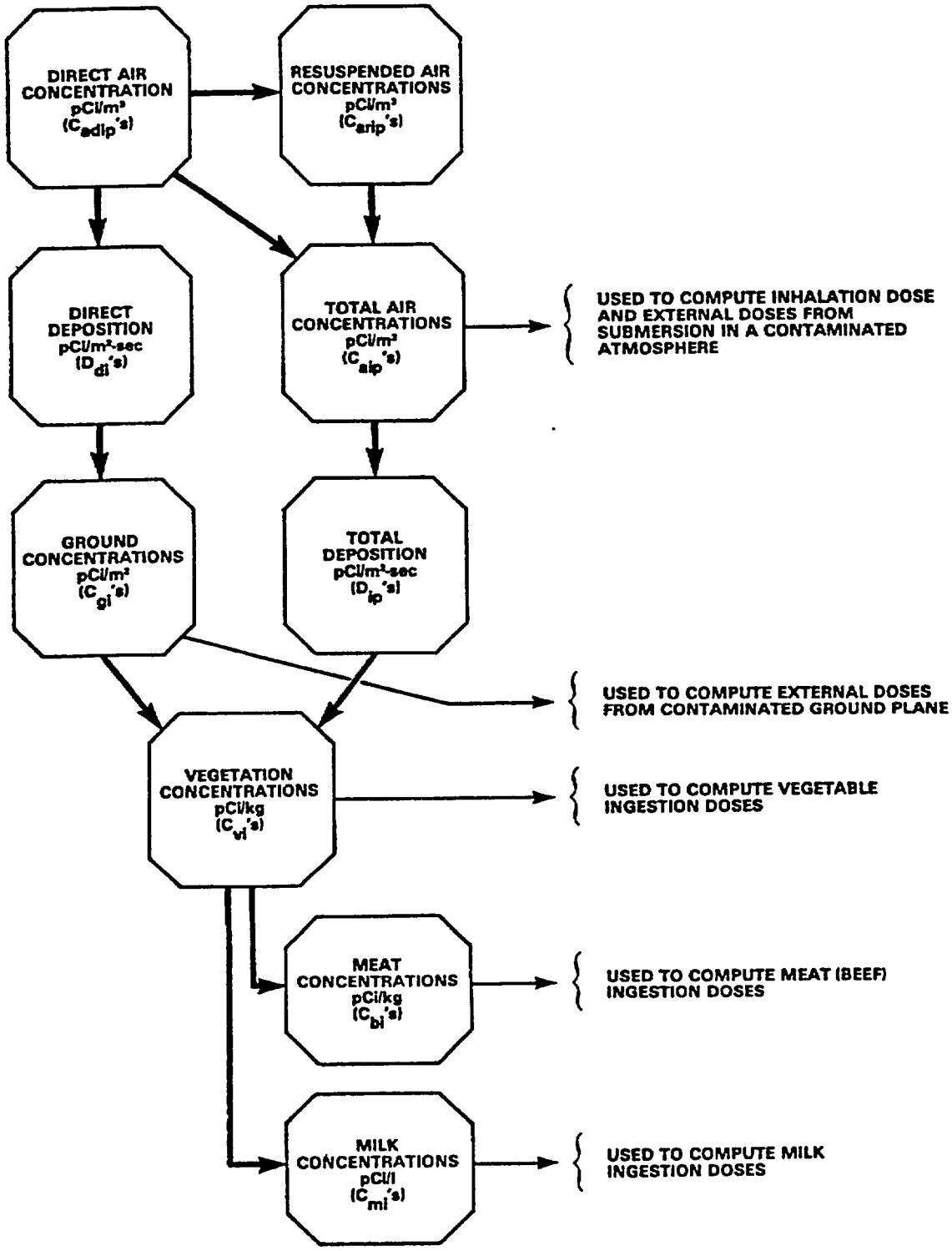

Figure 1. Schematic Diagram of Information Flow and Use For Dose Calculations

Table 12

CONVERSION FACTORS INTO SI UNITS

	Old Units*	New SI Units	Conversion Factor from Old to New Unit
Activity Concentrations (Environmental)			
Airborne Particulates and Gas	$pCi \cdot m^{-3}$	$Bq \cdot m^{-3}$	3.70E-02
Liquids (Water, Milk, etc.)	$pCi \cdot L^{-1}$	$Bq \cdot L^{-1}$	3.70E-02
Solids (Soil, Sediment, Vegetation, Food Stuff, etc.)	$pCi \cdot kg^{-1}$	$Bq \cdot kg^{-1}$	3.70E-02
Activity Concentrations (Effluent) Gas (Air)	$(\mu Ci \cdot mL^{-1})$**	$Bq \cdot m^{-3}$	3.70E+10
Liquid	$(\mu Ci \cdot mL^{-1})$**	$Bq \cdot L^{-1}$	3.70E+07
Exposure Rate (Environmental)	$\mu R \cdot h^{-1}$	$C \cdot kg^{-1} \cdot h^{-1}$	2.58E-10
Absorbed Dose	mrad	Gy	1.00E-05
Dose Equivalent	mrem	Sv	1.00E-05
Dose Equivalent Rate (Commitment)	$mrem \cdot yr^{-1}$	$Sv \cdot yr^{-1}$	1.00E-05

*Sanctioned for temporary use.
**Adopted because of established convention and use in maximum permissible concentration (MPC) tabulations.

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SYMBOLS

Symbol	Description
a	Equal to $(t - 1.82)$ if $t > 1.82$ and otherwise equal to zero in yr
A_s	Area of segment s in km^2
B_{vi}	Soil-to-plant transfer coefficient for radionuclide i and vegetation type v, (pCi/kg(wet) plant per pCi/kg(dry) soil)
C_{adip}	Calculated direct air concentration of radionuclide i in particle size p resulting from operational releases in pCi/m^3
C_{ai}	Total air concentration of radionuclide i in pCi/m^3
$C_{aip}(t)$	Calculated total air concentration of radionuclide i in particle size p at time t in pCi/m^3
$C_{arip}(t)$	Calculated resuspended air concentration of radionuclide i in particle size p at time t in pCi/m^3
$C_{arip}(T_d)$	Residual resuspended air concentration of radionuclide i in particle size p resulting from operational releases at the end of the T_d-year drying period in pCi/m^3
C_{bi}	Resulting average concentration of radionuclide i in meat in pCi/kg
C_{fis}	Concentration of radionuclide i in food category f in segment s in pCi/kg(wet weight)
$C_{gi}(t)$	Calculated ground surface concentration of radionuclide i at time t in pCi/m^2
$C_{gi}(T_d)$	Residual ground concentration of radionuclide i resulting from operational releases at the end of the T_d-year drying period in pCi/m^2
$C_{gi}(T_o)$	Ground concentration of radionuclide i at the time of mill shutdown in pCi/m^2
$C_{g12}(Pb \leftarrow Ra)$	Incremental ^{210}Pb ground concentration resulting from ^{226}Ra deposition in pCi/m^2
C_{hi}	Concentration of radionuclide i in hay (or other stored feed) in pCi/kg(wet weight)
C_{mi}	Resulting average concentration of radionuclide i in milk in pCi/L

SYMBOLS (Continued)

C_{pgi}	Concentration of radionuclide i in pasture grass in pCi/kg (wet weight)
C_{vi}	Resulting concentration of radionuclide i in vegetation v in pCi/kg(wet weight)
C_{vis}	Average concentration of radionuclide i in vegetable type v produced in segment s in pCi/kg(wet weight)
$C_{vis}(avg)$	Average concentration of radionuclide i averaged over all types of vegetables in segment s in pCi/kg(wet weight)
$DCF_{ij}(cld)$	Dose conversion factor for cloud exposure from radionuclide i to organ j in mrem/yr per pCi/m^3
$DCF_{ij}(gnd)$	Dose conversion factor for ground exposure from radionuclide i to organ j in mrem/yr per pCi/m^2
$DCF_{ijk}(ing)$	Ingestion dose conversion factor for radionuclide i in organ j of an individual in age group k in mrem/pCi ingested
$DCF_{ijp}(inh)$	Inhalation dose conversion factor for radionuclide i, organ j, and particle size p in mrem/yr per pCi/m^3
D_{di}	Resulting direct deposition rate of radionuclide i in pCi/m^2 per sec
D_i	Total deposition rate, including deposition of resuspended activity, of radionuclide i in pCi/m^2 per sec
$d_j(ext)$	External dose to organ j in mrem/yr
$d_j(inh)$	Inhalation dose to organ j in mrem/yr
$d_{jk}(ing)$	Ingestion dose for organ j of an individual in age group k in mrem/yr
$d_{jk}(tot)$	Total dose to organ j of an individual in age group k from all exposure pathways in mrem/yr
$d_{js}(ext)$	Average external dose to organ j in segment s in mrem/yr
$d_{js}(inh)$	Average inhalation dose to organ j in segment s in mrem/yr
E_f	Factor to account for activity remaining after food preparation, dimensionless
E_v	Fraction of the foliar deposition reaching edible portions of vegetation v, dimensionless

SYMBOLS (Continued)

F_{bi}	Feed-to-meat transfer coefficient for radionuclide i, in pCi/kg per pCi/day ingested (see Table 2)
F_{fk}	Fraction of the production of food type f ingested by individuals in age group k, dimensionless
F_{mi}	Feed-to-milk transfer coefficient for radionuclide i in pCi/L per pCi/day ingested (see Table 2)
F_{pg},F_h	Fractions of the total annual feed requirement assumed to be satisfied by pasture grass or locally grown stored feed (hay), respectively, dimensionless
F_{pk}	Fraction of the regional population belonging to age group k, dimensionless
F_r	Fraction of the total deposition retained on plant surfaces, 0.2, dimensionless
G_{fs}	Productivity factor for food f in segment s in kg/yr per km^2
I_{ik}	Activity ingestion rate of radionuclide i by an individua in age group k in pCi/yr
M_j	Annual committed population dose to organ j in rem/yr
$M_j(d)$	Annual committed population dose to organ j during the drying phase in rem/yr
$M_j(ing)$	Resulting regional population dose from food ingestion for organ j in rem/yr
$M_j(inh + ext)$	Resulting population dose from inhalation and external exposure pathways in rem/yr
$M_j(m)$	Annual committed population dose to organ j during the milling phase in rem/yr
$M_j(m\&d)$	Aggregate committed population dose to organ j over the milling and drying phases in rem
$M_j(Rn)$	Annual continental population dose from ^{222}Rn and its daughters to organ j in rem/yr
p	Assumed soil areal density for surface mixing, 240 kg(dry weight)/m^2
P_s	Population residing in segment s
Q	Assumed feed ingestion rate at 50 kg(wet weight)/day

SYMBOLS (Continued)

Q_{fi}	Gross activity content of radionuclide i in food f in pCi/yr
$R_p(t)$	Ratio of the resuspended air concentration to the ground concentration for a ground deposit of age t yr for particle size p in m^{-1}
t	Time interval over which deposition has occurred in sec
T_d	Duration of time required to dry the tailings pile prior to reclamation in yr
T_o	Duration of the operational phase in yr
t_v	Assumed duration of exposure while vegetation v is growing in sec
U_{fk}	Average consumption rate of food type f for an individual in age group k (see Table 8 for values) in L/yr or kg/yr
U_{mk}, U_{bk}	Milk (in L/yr) and meat (in kg/yr) ingestion rates for an individual in age group k
U_{vk}	Ingestion rate of vegetable category v for age group k, in kg(wet weight)/yr
V_p	Deposition velocity of particle size p in m/sec (see Table 1)
W_{vs}	Weighting factor for vegetable type v in segment s (fraction of total production), dimensionless
Y_v	Assumed yield density of vegetation v, in kg/m^2 (wet weight)
$\delta(t)$	Zero if $t \leq 1.82$ and unity otherwise, dimensionless
λ_e	Assumed rate constant for environmental loss in sec^{-1}
λ_i	Radioactive decay constant for radionuclide i in sec^{-1}
λ_i^*	Effective removal constant for radionuclide i on soil in yr^{-1}
λ_n^*	Effective rate constant for loss by radioactive decay and migration of a ground-deposited radionuclide and equal to $\lambda_n + \lambda_e$ in sec^{-1}
λ_R	Assumed decay constant of the resuspension factor (equivalent to a 50-day half-life), 5.06 yr^{-1}
λ_w	Decay constant accounting for weathering losses (equivalent to a 14-day half-life), 5.73×10^{-7} sec^{-1}

VALUES OF CONSTANTS

Terminal value of the resuspension factor for particles with a deposition velocity of 0.01 m/sec	$10^{-9}m^{-1}$
Initial value of the resuspension factor for particles with a deposition velocity of 0.01 m/sec	$10^{-5}m^{-1}$
Deposition velocity for the particle size for which the initial resuspension factor value is $10^{-5}/m$	0.01m/sec
Fraction of vegetable activity remaining after food preparation, dimensionless	0.5
Effective reduction factor because of structural shielding for indoor exposure periods	0.825
Time required to reach the terminal resuspension factor	1.82 yr

APPENDIX A

SITE-SPECIFIC INFORMATION AND DATA USED BY THE NRC STAFF IN PERFORMING RADIOLOGICAL IMPACT EVALUATIONS FOR URANIUM MILLING OPERATIONS

Table A-1 lists and partially describes most of the information and data commonly used by the NRC staff in performing its uranium mill radiological impact evaluations. All the data detailed in Table A-1 are not always available on a site-specific basis, in which case the staff will employ conservative estimates or assumptions. In some situations, the data identified in Table A-1 may not be adequate, so the staff will attempt to secure additional information. This situation may arise, for instance, when operations at more than one site are involved and the staff is required to evaluate combined impacts. In most cases, however, provision of the data identified in Table A-1 allows the staff to completely fulfill its responsibilities with regard to the preparation of a thorough, knowledgeable, and technically sound radiological impact evaluation.

Table A-1

PLANT, PLANT OPERATIONS, METEOROLOGICAL, AND ENVIRONMENTAL DATA ROUTINELY USED BY THE NRC STAFF IN PERFORMING RADIOLOGICAL IMPACT EVALUATIONS

I. PHYSICAL PLANT DATA

A. Detailed site plot plan (overlaid on topographic map with scale and true north arrow) clearly identifying all locations of--

1. Site property boundaries
2. Raw ore storage pads
3. Primary crushers
4. Secondary crushers
5. Crushed ore storage areas
6. Ore grinders
7. Yellowcake dryer and yellowcake dryer stack*
8. Yellowcake packaging area and exhaust stack
9. Tailings impoundments and their boundaries
10. Any heap leach piles and their boundaries
11. Restricted area boundaries if different from site property boundaries
12. Fences

B. Plant operations data

1. General data

a. Ore processing rates for all crushers and grinders, MT/d; hr/d and d/yr operational
b. Raw ore grade, % U_3O_8 by weight, average and range
c. Fractions of uranium, thorium, radium, and lead in raw ore expected to flow through to tailings
d. Expected yellowcake purity, % U_3O_8 by weight, average and range, MT/yr produced
e. Expected calendar years of initial ore milling, final ore milling, and completion of tailings area reclamation

* Part of the input to the NRC staff's impact assessment computer code consists of X, Y, and Z coordinates for various release and receptor locations. The staff routinely determines these coordinates with respect to the topographic elevation at the location of the yellowcake dryer stack. A list of all such locations should be given in the radiological assessment.

Table A-1 (Continued)

2. Ore storage data

   a. Areas of each pile or bin complex, m^2
   b. Ore storage masses
   c. Ore grades, % U_3O_8 by weight
   d. Antidusting measures routinely implemented
   e. Anticipated dusting rates, MT/yr
   f. Anticipated ^{222}Rn releases, Ci/yr
   g. Fractions of input ore sent to storage

3. Crushing, grinding data

   a. Description of ventilation air filtration equipment
   b. Design efficiency of exhaust filters
   c. Minimum efficiencies of exhaust filters
   d. Filter testing procedure and schedule if applicable
   e. Fraction of time filters not operational or used
   f. Any measured effluent concentrations
   g. Stack heights and airflows
   h. Anticipated release rates, kg/hr or kg/MT yellowcake processed
   i. Anticipated ^{222}Rn release rate, Ci/yr
   j. Fractions of ore throughput reaching filters as dust

4. Yellowcake drying and packaging data

   a. Processing rates, MT/hr, for drying and packaging if different
   b. Hr/d and d/yr drying and packaging operations are carried out
   c. Description of all ventilation air filtration equipment with design, expected, and minimum efficiencies
   d. Filtration equipment testing procedures and frequencies
   e. Any measured effluent concentrations
   f. Stack heights and airflows
   g. Anticipated release rates, kg/hr, for the dryer stack, the packaging area ventilation exhaust, and any yellowcake storage area ventilation exhausts
   h. Annual yellowcake yield, MT/yr

5. Tailings impoundment system (including evaporation or settling ponds) data

   a. Complete physical, chemical, hydrological, and radiological description
   b. Total area, surface areas expected to be under water, saturated, moist, and dry (indicate surface moisture contents used as basis of estimates)

Table A-1 (Continued)

c. Description of antidusting measures routinely implemented and their expected effectiveness
d. Anticipated dusting rates for saturated, moist, and dry surface areas, g/m^2 per sec
e. Anticipated ^{222}Rn release rates for underwater, saturated, moist, and dry surface areas, Ci/yr per m^2
f. Estimated drying time required prior to initiation of reclamation procedures and basis
g. Estimated time required to stabilize and reclaim after drying and basis
h. Postreclamation estimated ^{222}Rn release rate, Ci/yr per m^2, and basis

II. METEOROLOGICAL DATA

A. Joint frequency data

1. National Weather Service (NWS) station data

a. Locations of all NWS stations within 80 km (50 mi)
b. Available joint frequency distribution data by wind direction, wind speed, and stability class (3-dimensional numerical array)
c. Period of record by month and year
d. Height of data measurement

2. Onsite meteorological data

a. Location and heights of instrumentation
b. Description of instrumentation
c. Minimum of 1 full year of onsite joint frequency distribution data broken down by wind direction, wind speed, and stability class (3-dimensional array) with a joint data recovery of 90 percent or more

B. Miscellaneous data

1. Annual average mixing depth heights
2. Description (general) of regional climatology, particularly including frequencies and durations of extreme wind speeds

III. ENVIRONMENTAL DATA

A. A detailed topographic map of the area within 8 km (5 mi) of the site showing the locations of all--

1. Site boundaries
2. Lands owned, leased, or otherwise controlled (including mill site claims) by the applicant

Table A-1 (Continued)

3. Lands privately owned
4. Lands under the jurisdiction of the U.S. Bureau of Land Management
5. Lands otherwise publicly held
6. Lands useable and available for grazing
7. Private residences or other structures used by the general public
8. Vegetable or other crops, identified by type
9. Private, public, and industrial water wells and natural springs
10. Milk animals (cows or goats)

B. Regional data (within 80 km)

1. Population distributions by direction (16) and radius (for 1, 2, 3, 4, 5, 10, 20, 30, 40, 50, 60, 70, and 80 km) for a recent year (no earlier than 1970), for the last year of expected milling (approximate), and for the last year prior to completion of tailings area reclamation (approximate) with expected age group fractions (if available)

2. Available county food production data, kg/yr, for vegetables (by type and totals), meat (all types), and milk; any available future predictions by local governmental, industrial, or institutional organizations

APPENDIX B

STAFF METHODOLOGY FOR THE COMPUTATION OF 100-YEAR ENVIRONMENTAL DOSE COMMITMENTS

A primary objective of the NRC staff's radiological impact analysis is to estimate the aggregate radiological impact of the evaluated facilities. In attempting to achieve this goal, the staff employs the concept of environmental dose commitment (EDC) and uses an integrating period of 100 years. In adopting this general calculational approach, the staff has also endeavored to select and employ a specific calculational scheme suitable for routine use, both by the NRC staff and by uranium milling license applicants. The specific technique used by the staff is, for this reason, greatly simplified but somewhat less comprehensive in comparison with other published approaches for EDC computation. This appendix describes the staff's technique for EDC evaluation and addresses the rationale for selecting a 100-year integrating period.

Ordinarily, to compute maximum individual doses, the staff uses environmental concentrations calculated for the final year of the particular phase of milling operations. The duration of the operational (milling) phase is most often estimated to be 15 to 20 years, while drying of tailings piles in the prestabilization phase may require from 2 to 5 years or slightly longer. The lengths of these time intervals define the value of the time variable "t" that appears in Equations 2, 3, 4, and 6 of Regulatory Position 1, Concentrations in Environmental Media, of this guide.

The staff technique for evaluating regional population EDCs for an integrating period of 100 years following activity release involves artificially setting the value of t to 101 years. The specific procedural steps taken by the staff in the calculation of 100-year EDCs are then as otherwise described in Regulatory Positions 1 and 3 and as follows:

1. Obtain all necessary input direct air concentrations, as identified in Table 1 of the guide, for average release rates (by radionuclide) over the time interval of the phase being evaluated.

2. Evaluate all required environmental media concentrations by means of the equations provided for this purpose in Regulatory Position 1, using a value of 101 years for the variable t appearing in Equations 2, 3, 4, and 6.

3. Based on the environmental media concentrations computed for t = 101 years, using appropriate population, agricultural, and other data as described in Regulatory Position 3, calculate the regional population doses for all exposure pathways for an exposure period of 1 year.

4. Sum the computed doses, as appropriate, over all exposure pathways.

These calculational procedures actually result in the computation of the population dose commitments resulting from a 1-year exposure period to environmental concentrations existing during the 101st year of releases at the constant rates employed. The similarity of this result to the desired EDC (the population dose commitments resulting from a 100-year period of exposure to environmental concentrations resulting from constant releases over a 1-year time period) is illustrated in Table B-1, which provides a comparison of staff and conventional methodologies for EDC computation. This table has been organized to display the component parts of each calculational method. Line-by-line equivalence of these component parts can be readily demonstrated under conditions of constant population, population distribution, and agricultural productivity in the site region.

The staff has elected to use the approach described, rather than the more conventional approach, and a 100-year integrating period, primarily for the following reasons:

1. The major exposure pathways are dominated by doses resulting from airborne activity, which decreases rapidly in the absence of a continuing source (the resuspension factor has a half-life of about 50 days);

2. The major dose impact of ground concentrations arises from the food ingestion pathways, which depend on estimates of agricultural productivity (forecast data for food productivity in specific areas are rare and are considered to be potentially unreliable);

3. Inordinate computational difficulties are involved in routinely taking into account growth trends not amenable to description by very simple mathematical functions; and

4. The vast majority of resulting population exposure results from environmental concentrations at distances between 20 and 80 km (32 and 50 mi) from the site at which routine atmospheric dispersion calculations cannot generally yield results with sufficient accuracy to justify accounting for minor perturbations.

Table B-1

COMPARISON OF STAFF AND CONVENTIONAL TECHNIQUES FOR ENVIRONMENTAL DOSE COMMITMENT CALCULATION

	NRC Staff EDC Calculational Technique* (Defined as population dose commitments resulting from a 1-year period of exposure to environmental concentrations present during the 101st year of constant releases)			Conventional EDC Calculational Technique (Defined as population dose commitments resulting from a 100-year period of exposure to environmental concentrations resulting from constant releases over a 1-year period)		
Line	Exposure Interval, yr	Release Interval, yr	Average Time Difference, yr	Exposure Interval, yr	Release Interval, yr	Average Time Difference, yr
1	100 - 101	100 - 101	0	0 - 1	0 - 1	0
2	"	99 - 100	1	1 - 2	"	1
3	"	98 - 99	2	2 - 3	"	2
4	"	97 - 98	3	3 - 4	"	3
5	"	96 - 97	4	4 - 5	"	4
6	"	95 - 96	5	5 - 6	"	5
.	.	.	.	.	.	.
.	.	.	.	.	.	.
.	.	.	.	.	.	.
.	.	.	.	.	.	.
.	.	.	.	.	.	.
94	"	7 - 8	93	93 - 94	"	93
95	"	6 - 7	94	94 - 95	"	94
96	"	5 - 6	95	95 - 96	"	95
97	"	4 - 5	96	96 - 97	"	96
98	"	3 - 4	97	97 - 98	"	97
99	"	2 - 3	98	98 - 99	"	98
100	"	1 - 2	99	99 - 100	"	99
101	"	0 - 1	100	100 - 101	"	100

*This table has been purposely organized to portray a line-by-line similarity between staff and conventional EDC computation methods. Computation by both methods is broken down into component parts that, under conditions described in the text, can be shown to be mathematically identical.

APPENDIX C

RADON DOSE CONVERSION FACTOR

The basis on which the NRC staff has relied for its radon daughter inhalation dose conversion factor consists of the following major component parts:

1. The indoor working level (WL) concentration resulting from an outdoor ^{222}Rn concentration of 1 pCi/m^3 is approximately 5.0×10^{-6} WL.

2. The number of cumulative working level months (WLM) exposure per year for an average individual at a constant concentration of one WL is 25 WLM/yr.

3. The committed dose equivalent to the bronchial epithelium (basal cell nuclei of segmented bronchi) per unit WLM exposure is 5000 mrem (5 rem).

These component parts enter into the following equation, which yields the ^{222}Rn inhalation dose conversion factor used by the staff:

$$\frac{5.0 \times 10^{-6}\ \text{WL}}{\text{pCi/m}^3} \times \frac{25\ \text{WLM/yr}}{\text{WL}} \times \frac{5000\ \text{mrem}}{\text{WLM}} = \frac{0.625\ \text{mrem/yr}}{\text{pCi/m}^3}$$

Each of the three components identified above are derived from the following sources and data:

1. 5×10^{-6} WL per pCi/m^3 of ^{222}Rn is established by the assumed indoor air concentration ratios for ^{222}Rn, ^{218}Po, ^{214}Pb, and ^{214}Bi of 1.0/0.90/0.51 and 0.35. These concentration ratios and the derived conversion factor are representative of conditions in a reasonably well-ventilated structure (Refs. 1 and 2 for Appendix C).

2. 25 WLM/yr per WL concentration is derived from the assumption that an average individual's average breathing rate will be about 50 percent of that of a working miner. A WLM is defined, in terms of exposure to a working miner, as one month's occupational exposure to a 1-WL concentration. This assumed breathing rate would result in an average individual receiving about

0.5 WLM as a result of the same length of exposure to air at a 1-WL concentration. The following relationship applies:

$$(8760 \text{ hr/yr}) \times \frac{12 \text{ WLM/yr-WL}}{40 \text{ hr/wk} \times 52 \text{ wk/yr}} \times 0.5 = 25 \text{ WLM/yr-WL}$$

3. Five rem/WLM is the value derived from applying a quality factor (QF) of 10 for alpha radiation to convert from rad to rem (Refs. 1, 2, and 3 of Appendix C) to the figure of 0.5 rad/WLM as reported in the BEIR Report (page 148 of Ref. 3 of Appendix C).

The NRC staff considers the above basis for its ^{222}Rn inhalation dose conversion factor to be both sound and reasonable. The staff acknowledges that radon dosimetry is extremely complex and strongly influenced by assumed environmental and biological conditions. In view of the large variations induced by rather small changes in the assumed free-ion fraction, relative equilibrium, thickness of the intervening tissue and mucous layers, etc., the staff has endeavored to use physical, environmental, and other data reasonably representative of average conditions.

REFERENCES FOR APPENDIX C

1. Environmental Protection Agency, "Potential Radiological Impact of Airborne Releases and Direct Gamma Radiation to Individuals Living Near Inactive Uranium Mill Tailings Piles," EPA Report EPA-520/1-76-001, January 1976.

2. Environmental Protection Agency, "Environmental Analysis of the Uranium Fuel Cycle, Part I--Fuel Supply," EPA Report EPA-520/9-73-003-B, October 1973.

3. National Academy of Sciences--National Research Council, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Report of the Advisory Committee on the Biological Effects of Ionizing Radiations (BEIR), November 1972.

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

The proposed action consists of the development and publication of a routine methodology for assessing the radiological impacts of routine radioactive releases from uranium mills. These radiological impacts include doses to exposed individuals, doses to the population within an 80-km (50-mi) radius, doses to the population of the entire United States, and doses to the population of the North American Continent. Evaluations made using the published methodology would serve several regulatory and licensing purposes for which the methodology must be suitable. These purposes include evaluating compliance with 40 CFR Part 190 and NRC regulations, evaluating impacts of releases as part of the overall ALARA evaluation, and evaluation of environmental impacts to meet NEPA requirements.

1.2 Need

Radiological impact evaluations for routine releases from uranium mills have been carried out in the past, and numerous new and repeat evaluations will probably be required in the future. Past evaluations have been prepared by NRC personnel or by personnel from national laboratories under contract. These assessments have lacked a uniformity of approach and purpose for numerous reasons, the most important being the absence of a standardized routine procedure. Other reasons include, but are not limited to, the evolution of new models, techniques, and data; the development of new concerns requiring new methods of analysis; and the problems associated with having evaluations prepared by different groups of people. This situation needed to be corrected. The proposed action includes the publication of state-of-the-art analytical models, including environmental transport models and data, models and data for human dosimetry, and appropriate data for receptor characteristics. An example

of the problems to be addressed by this effort is the evaluation of the long-term time-integrated impact of mill tailings piles, heretofore assessed by NRC only in terms of the impact during a single year.

1.3 Value/Impact

1.3.1 NRC

The document conveying the results of the proposed action will be a useful tool and should result in substantial benefits to NRC. These include upgrading the quality of future evaluations, particularly with regard to uniformity, completeness, and the application of more up-to-date methods and data. Other benefits will include greater flexibility in personnel assignments and reduced allocations of personnel time to completing evaluations.

1.3.2 Other Government Agencies

Other agencies will have available a reliable reference document explaining NRC's evaluation techniques. If evaluations can be conducted more uniformly, other agencies concerned with radiological and health impacts would benefit from these evaluations as they become more familiar with a routine approach and require less time to review NRC evaluations.

1.3.3 Industrial and Public Interest Groups

Clearly predictable impacts on these groups include the costs involved in familiarizing themselves with the proposed regulatory guide. Benefits will be derived from more easily predicting and understanding the results of NRC evaluations. Some differences from past evaluation techniques have been incorporated in this guide, but based on public comment, the degree and effects of such alterations appear to be minimal.

1.3.4 Public

The public will derive a benefit from the availability of a reference document explaining NRC evaluation techniques, and a further benefit will be derived from the increase in quality of NRC evaluations and subsequent licensing decisions and regulatory requirements.

2. TECHNICAL APPROACH

The technical approach to be used is based in part on contract work prepared by staffs of the Argonne National Laboratory, the Pacific Northwest Laboratory, and the Oak Ridge National Laboratory. This approach reflects techniques currently being adopted for use in review of uranium milling license applications and license renewal applications by the Office of Nuclear Material Safety and Safeguards. Comments on the technical approach were solicited by the issuance of Draft Regulatory Guide RH 802-4 for public comment. The comments received were evaluated and modifications were made to the guide where appropriate.

3. PROCEDURAL APPROACH

In its preliminary value/impact assessment, the staff considered several procedural approaches for carrying out the proposed action and selected the publication of a regulatory guide.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

The product document establishes routine procedures by which NRC will evaluate radiological impacts of routine airborne releases from uranium mills. These evaluations will be and are being used in "as low as is reasonably achievable" determinations to evaluate compliance with NRC regulations, to evaluate compliance with EPA's 40 CFR Part 190 regulation, and to evaluate environmental impacts as part of NRC's overall NEPA determination.

4.2 Need for NEPA Assessment

The proposed action on calculational models did not require an environmental impact statement as it was not "a major Commission action significantly affecting the quality of the environment" as detailed in paragraph 51.5(a)(10) of 10 CFR Part 51.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No potential conflicts with other agencies have been identified. However, the proposed regulatory guide will be a principal tool in the implementation of EPA regulation 40 CFR Part 190. Implementation of 40 CFR Part 190 is an NRC responsibility.

There is some possibility that backfitting requirements may result from implementation of 40 CFR Part 190. Such possible requirements will not result from the proposed action, but rather from the EPA regulation.

6. SUMMARY AND CONCLUSIONS

Guidance on routine procedures for evaluating the radiological impact of routine airborne releases of radioactive material from uranium mills should be developed and published in a regulatory guide.

U.S. NUCLEAR REGULATORY COMMISSION

May 1986

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 3.56
(Task CE 309-4)

GENERAL GUIDANCE FOR DESIGNING, TESTING, OPERATING, AND MAINTAINING EMISSION CONTROL DEVICES AT URANIUM MILLS

A. INTRODUCTION

Regulations applicable to uranium milling are contained in 10 CFR Part 20, "Standards for Protection Against Radiation," and in 10 CFR Part 40, "Domestic Licensing of Source Material."

Paragraph 20.1(c) of 10 CFR Part 20 states that licensees should make every reasonable effort to keep radiation exposures, as well as releases of radioactive material to unrestricted areas, as low as is reasonably achievable. Paragraph 20.105(c) of 10 CFR Part 20 requires that licensees engaged in uranium fuel cycle operations subject to the provisions of 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," comply with that part. Part 190 of Title 40 requires that the maximum annual radiation dose to individual members of the public resulting from fuel cycle operations be limited to 25 millirems to the whole body and to all organs except the thyroid, which must be limited to 75 millirems. Criterion 8 of Appendix A to 10 CFR Part 40 requires that milling operations be conducted so that all airborne effluent releases are reduced to levels as low as is reasonably achievable.

Air in the immediate vicinity of such uranium milling operations as ore crushing, ore grinding, and yellowcake drying and packaging frequently contains radioactive materials in excess of that permissible for release to unrestricted areas. Emission control devices are installed in ventilation systems of uranium mills to limit releases of these radioactive materials to the environment.

General guidance for filing an application for an NRC source material license authorizing uranium milling operations is provided in § 40.31 of 10 CFR Part 40. An applicant for a new license or renewal of an existing license for a uranium mill is required by § 40.31 to provide detailed information on the proposed equipment, facilities, and procedures at the installation. This information is used by the NRC to determine whether the applicant's proposed equipment, facilities, and procedures are adequate to protect the health and safety of the public and and to determine if they will significantly affect the quality of the environment. Calculations by the NRC of the environmental impact from the proposed uranium milling operations are based on the estimated rate of production of radioactive airborne particulates adjusted to reflect the removal efficiency of the emission control devices installed in the plant ventilation systems. This requires reliable information on the efficiency of these devices. It also requires reliable information on the production of airborne radioactive particulates during the proposed operations.

Section 40.65 of 10 CFR Part 40 requires mill operators to submit semiannual reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in gaseous effluents. This information may be used by the NRC to estimate maximum potential annual radiation doses to the public resulting from effluent releases and thereby determine compliance with paragraphs 20.1(c) and 20.105(c) of 10 CFR Part 20 and with Criterion 8 of Appendix A to 10 CFR Part 40. The quantity of radionuclides released is based on scheduled sampling of effluents discharged into exhaust stacks. The reliability of these data for estimating radiation exposures depends on maintaining uniform operation of the emission control devices during the reporting time interval because these effluents are not continuously sampled.

All emission control devices used in uranium mill ventilation systems need to perform reliably under expected operating conditions to meet the objectives discussed above. This guide describes procedures acceptable to the NRC staff for designing, testing, operating, and maintaining these emission control devices to ensure the reliability of their performance.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Parts 20 or 40, which provide the regulatory basis for this guide. The information collection requirements in 10 CFR Parts 20 and 40 have been cleared under OMB Clearance Nos. 3150-0014 and 3150-0020, respectively.

B. DISCUSSION

The milling of uranium ores results in the production of airborne particulates containing uranium and its daughters in several areas of a typical uranium mill. These areas encompass (1) ore storage, handling, and crushing, (2) ore grinding, leaching, and concentrating processes; (3) yellowcake precipitation, drying, and packaging, and (4) miscellaneous mill locations such as maintenance shops, laboratories, and general laundries. Milling operations must be conducted so that all airborne effluent releases are reduced to levels as low as is reasonably achievable (ALARA). The primary means of accomplishing this is the control of emissions at the source.

The most significant sources of radioactive airborne particulates occur in ore handling and crushing areas and in yellowcake drying and packaging areas. These sources are generally controlled by separate ventilation systems in each area that remove these airborne particulates through local hoods, hooded conveyor belts, etc., into emission control devices where they are removed from the air streams. The cleaned air is then discharged by fans into the atmosphere through local exhaust stacks.

Emission control devices are available in a wide range of designs to meet variations in air cleaning requirements. Degree of removal required, quantity and characteristics of the contaminant to be removed, and conditions of the air stream all have a bearing on the device selected for any given application. Emission control devices used at ore crushing and grinding operations include bag or fiber filters (baghouses), orifice or baffle scrubbers, and wet impingement scrubbers. Water spray systems are also used at these operations to minimize the generation of dust. Wet impingement scrubbers or venturi scrubbers are generally employed at yellowcake drying and packaging areas.

All emission control devices used in a uranium mill ventilation system need to be designed for reliable performance under the expected operating conditions. Initial testing and proper maintenance are primary factors in ensuring the reliability of these components. Periodic testing during operation to verify the efficiency of these components is another important means of ensuring reliability. Built-in features that will facilitate convenient in-place testing of these devices are important in ventilation system design.

Emission control devices used in a uranium mill ventilation system need to be sufficiently instrumented to measure and monitor their operating characteristics. Frequent checks of all significant operating parameters are necessary to determine whether or not conditions are within a range prescribed to ensure that this equipment is operating consistently near peak efficiency. When checks indicate that the equipment is not operating within this range, it is necessary to take action to restore parameters to the prescribed range. To ensure that timely actions are taken, instrumentation is often supplemented by audible alarms that are preset to signal when prescribed operating range limits are exceeded. When the required actions cannot be taken without shutdown and repair of this equipment, it will be necessary to suspend milling operations that are the source of the emissions until corrective actions have been taken. Criterion 8 of Appendix A to 10 CFR Part 40 requires suspension of yellowcake drying and packaging operations as soon as practicable when shutdown and repair of the emission control system is necessary. The installation of automatic shutdown instrumentation on processes and systems at which operating parameters on emission control devices may exceed acceptable limits could prevent excessive releases that may result from continuous operations under these circumstances, e.g., those associated with the production of yellowcake. The installation of backup or redundant emission control systems would permit continuous operation during repair and maintenance of the primary system.

A preventive maintenance program is important for emission control devices used in uranium mill ventilation systems. A program designed to identify deficiencies in operation of these devices so that corrective action can be taken to reduce the frequency of off-normal operation can provide a measure of confidence in the operating characteristics of these devices. This program may require periodic updating to reflect actual in-plant experience, equipment manufacturers' guidelines, and NRC guidance. For example, a preventive maintenance program can consist of the equipment supplier's recommendations supplemented by provisions derived from the licensee's own routine inspection and maintenance records.

The key to proper maintenance of emission control devices is frequent inspection. It is important that a regular program of inspection be established and followed and records be kept of all inspections and the resulting maintenance. Inspection intervals will depend on the type of emission control device, the manufacturer's recommendation, and the process area where the unit is installed. These inspections need to be performed as frequently as experience shows to be necessary but not less than annually.

Considerable maintenance time can be expended on trouble shooting and correction of malfunctions of emission control devices. The ability to locate and correct malfunctioning components of these devices requires a thorough understanding of the system.

Throughout the manufacturing industry, there are many models of each type emission control device used at uranium mills. These models range in size in order to meet the different air capacity needs at the mills. In addition, some design features of each manufacturer are unique. Accordingly, the specific design and the testing, operating, and maintenance procedures for each model are beyond the scope of this guide. General guidance is presented, however, for each type of emission control device based on typical models in present-day use. Background information for this guidance can be found in the Bibliography. The licensee may substitute procedures based on specific operating parameters of the model in use at the facility for those described in this guide.

1. DESIGN AND OPERATION

1.1 Bag or Fabric Filters (Baghouses)

Bag or fabric filters, usually in the form of baghouses, remove particulates from a gas stream by filtering the airborne particulates (by impaction or diffusion) through a porous flexible fabric made of a woven or felted material. These collected particles form a structure of their own, supported by the filter, and have the ability to intercept and retain other particles. The increase in retention efficiency is accompanied by an increase in pressure drop through the filter. The baghouses are equipped with one of several automatic cleaning mechanisms for periodically dislodging collected material from filter components to prevent excessive resistance to the gas flow (i.e., excessive pressure drop) that would otherwise develop. The dislodged material settles in storage hoppers before the filter components are placed back on stream. The automatic cleaning cycle can be initiated by either a differential pressure switch or a timer, which may be interlocked with the main fan motor for the baghouse.

The cleaning mechanisms employed in baghouses are based on either mechanical shaking of the filter components or pneumatic vibration of these components by high-pressure air applied in reverse flow, reverse jet, or reverse pulse modes. The effectiveness of these compressed air systems depends on maintaining a sufficient reservoir of compressed air at the pressure specified by the baghouse manufacturer. Higher pressures than specified could cause failure of the filter fabric, while lower pressures can result in poor filter cleaning. These problems are minimized by pressure-regulating devices used in the compressed air systems.

The most critical parameter to be observed during baghouse operation is the pressure drop. Proper operation of the baghouse requires, at a minimum, maintaining the differential pressure of this device in the correct range specified by the manufacturer. A manometer or a differential-pressure gauge and transmitter are usually provided for this purpose. This instrumentation is often supplemented by an audible alarm system designed to signal and alert mill operators when prescribed differential-pressure ranges are exceeded. Lower differential pressures indicate potential deficiencies such as damaged filters or other air bypass channels that should be corrected. Higher differential pressures indicate that cleaning operations are inadequate. This can be corrected by increasing the frequency of the automatic cleaning cycle through adjustment of the differential-pressure switch or timer of the baghouse installation.

1.2 Wet Scrubbers

Wet scrubbers remove particulates from a gas stream by effecting intimate contact between the gas stream and a scrubbing liquor, usually water. The basic operations that take place within a wet scrubber are (1) saturation of the incoming gas, (2) contacting and capture of the particulates in the scrubbing liquor, and (3) separating the entrained particulate-laden liquid from the gas stream. The basic types of wet scrubbers are distinguished by the mechanisms used for transfer of particulates from the gas stream to the liquid stream. Most scrubber systems require some type of treatment and disposal of the particulate-laden scrubbing liquor.

Several water spray systems may be used in wet scrubber operations. Water from the main water spray system is directed either into a screen or throat to contact the particulate-laden gas stream. In applications where inlet gas temperatures are inordinately high, preconditioning of the incoming gas to the scrubber may be necessary to provide adequate humidity and thereby maintain particulate collection efficiency. This may be accomplished by use of an auxiliary water spray system upstream of the scrubber particulate scavenging area. Where particulate buildup is likely to occur in the entrainment separator, a wash system may be necessary to avoid this condition. The wash system is usually composed of low-pressure spray nozzles using recycled scrubbing liquor or fresh water for cleansing.

Orifice, wet impingement, or venturi wet scrubbers are generally used in uranium mill ventilation systems. In orifice-type wet scrubbers, the gas stream is made to impinge upon a surface of scrubbing water and is then passed through various constrictions where its velocity may be increased and where greater liquid-particulate interaction may occur. The gas stream finally discharges through a chamber section where entrained droplets are disengaged. In wet impingement scrubbers, the gas stream is wetted with water from low-pressure spray nozzles in the scrubber inlet and then passed through perforated plates at high velocity to impinge on baffle plates or vanes where liquid droplets containing particulate matter coalesce and drain to a sump. Solid particles are washed to the sump by either intermittent or continuous sprays. Prior to exiting from the scrubber, the gas stream passes through an entrainment separator to remove entrained liquid droplets. In a venturi scrubber, the gas stream flows through a throatlike passage where the gas is accelerated in velocity. The scrubbing liquor is

added at or ahead of the venturi throat and is sheared into fine droplets by the high-velocity gas stream, resulting in liquid-particulate interaction. The gas and liquor droplets then pass through a cyclone separator where entrained droplets containing particulate matter are removed from the gas stream.

Although each type of scrubber discussed above has unique design features, their collection efficiencies are influenced in similar ways by incremental changes in certain common operating parameters, principally gas and liquid flow as well as pressure drop. A decrease in either the gas or liquid flow rate could result in insufficient gas cleaning. Collection efficiency can also diminish if the liquid-to-gas flow rate ratio falls below design values. An increase in pressure drop across the scrubber will enhance the collection efficiency for the same size distribution and concentration of particulates in the gas stream. Proper operation of these wet scrubbers requires monitoring of these parameters to determine that they are within ranges prescribed to ensure equipment performance consistently near optimum collection efficiency. Instrumentation used to monitor these parameters is often supplemented by audible alarm systems designed to signal and alert mill operators of the need for corrective action when prescribed operating ranges are exceeded. In some cases automatic control systems with interlocks may be necessary. For example, the scrubber fan could be interlocked to shut down in the event of an indication of water flow failure. These circumstances would require suspending particulate-producing processes in the ventilation zone serviced by the scrubber until corrective action could be taken or switching to a redundant scrubber unit.

Daily operational data summaries on baghouse and wet scrubber performance are useful in providing a continuous record of performance of these devices. Other formats that contain equivalent information such as recorder charts can also be used for this purpose. Criterion 8 of Appendix A to 10 CFR Part 40 requires that checks of all parameters that determine the efficiency of yellowcake stack emission control equipment operation be made and logged hourly. In addition, data from checks made of all operating parameters necessary to enable timely identification of malfunctions can be of value in ensuring proper operation of baghouses and wet scrubbers and in updating preventive maintenance programs for these devices to reflect actual operating experience.

2. MAINTENANCE

2.1 Bag or Fabric Filters (Baghouses)

The frequency of needed maintenance for baghouses can be determined from manufacturers' recommendations and operating experience. In order of decreasing frequency, the principal baghouse components requiring maintenance are (1) filter bags, (2) flow controls, (3) hoppers, and (4) cleaning mechanisms. Symptoms of potential operating problems requiring corrective maintenance are almost always one of the following: (1) excessive emissions, (2) short filter bag life, and (3) high pressure drop. These symptoms may indicate malfunctioning in more than one component. For example, high pressure drop may be attributable to difficulties with the filter bag cleaning mechanism, low compressed air pressure, high humidity, weak shaking action, loose filter bag tension or excessive reentrainment of dust. Many other factors can cause excessive pressure drop, and several options are usually available for appropriate corrective action.

2.2 Wet Scrubbers

The major maintenance problems with wet scrubbers are (1) excessive buildup of solids in the wet/dry zones and entrainment separator, (2) plugged water spray nozzles, (3) abrasion in areas of high velocity such as throats and orifices, and (4) corrosion on scrubber vessel internal surfaces. A buildup of solids often occurs around the wet/dry interfaces of ducts where the gas stream contacts the wetted scrubber housing. Instrumentation such as liquid and gas pressure indicators can exhibit rapid solids buildup and therefore require regular cleaning to ensure proper system operation and performance. Increased pressure drop, reduced gas flow, and subsequent system malfunction are all possible consequences of a buildup of solids in the entrainment separator. Water spray nozzles frequently wear or clog, which produces an uneven liquid pattern and requires their replacement. Venturi and impingement scrubbers tend to show signs of abrasion in areas downstream of gas and liquid acceleration. Corrosion can occur from the high moisture and airborne liquid incident on components, in particular where protective liners may have deteriorated.

A regular schedule of routine inspection of key components and operating parameters is an essential ingredient of a maintenance program for ensuring the reliability of performance of typical baghouses and wet scrubbers. Examples of some typical maintenance activities for baghouses and wet scrubbers used at uranium mills are presented in Appendices A and B, respectively. These activities are in addition to those procedures recommended by manufacturers for routine lubrication, inspection, and replacement of component parts.

3. TESTING

To ensure proper selection of emission control devices, it is necessary for potential users to supply manufacturers with a list of specifications for the given application, including gas flow rates, liquid flow rates (where scrubbers are under consideration), temperature, pressure, pressure drop, concentration of particulates, particle size distribution, emission levels, and collection efficiency. The manufacturers, in turn, should design and supply these devices based on test data already available for prototype equipment used under similar circumstances.

If relevant test data are not available, it is generally advisable for the manufacturer and potential user to run mutually agreed-upon pilot plant or prototype tests with a gas stream typical of the gas stream to be cleansed to ensure that proper equipment is supplied to meet the desired collection efficiency. After installation of the device, it may be tested in place to confirm its particulate removal efficiency. Periodic in-place testing will ensure continued effectiveness of the device. In this way, reliable data will be available to the licensee for estimating the environmental impact of uranium milling operations before and after the commencement of operations.

Collection efficiency for baghouses and wet scrubbers used in uranium mills is usually based on inlet and outlet particulate concentrations in a dry gas corrected to standard temperature and pressure. Inlet and outlet particulate concentrations are preferably sampled simultaneously if practicable. The procedure of choice for determination of particulate concentrations is described in Method 5, "Determination of Particulate Emissions From Stationary Sources," of Appendix A to 40 CFR Part 60, "Standards of Performance for New Stationary Sources." In this procedure, particulate matter is withdrawn isokinetically from the gas stream and collected on a glass fiber filter maintained in a prescribed elevated temperature range. The particulate mass, which includes any material that condenses at or above the filtration temperature, is determined gravimetrically after removal of uncombined water. If a preoperational in-place determination of collection efficiency is desired, a procedure mutually acceptable to the user and manufacturer may be used.

4. QUALITY ASSURANCE

Components of uranium mills do not require a formal quality assurance program; however, particular quality assurance requirements may be imposed by the NRC as license conditions if deemed necessary to protect health. A quality assurance program for emission control devices need only be an extension of the overall quality assurance program usually submitted by an applicant for a license to ensure that the emission control devices are designed and the testing, operating, and maintenance procedures are implemented to maintain uniform operation of these devices within prescribed ranges under expected operating conditions.

C. REGULATORY POSITION

Emissions from milling operations must be controlled so that all airborne effluent releases are reduced to levels as low as is reasonably achievable. An important means of accomplishing this is by means of emission control devices in mill ventilation systems. The design and the testing, operating, and maintenance procedures for these emission control devices should ensure that these devices are operating consistently near peak operational efficiency.

1. DESIGN AND OPERATION

In addition to the requirement in Criterion 8 of Appendix A to 10 CFR Part 40 that requires checks to be made and logged hourly of all parameters that determine the efficiency of yellowcake stack emission control equipment operation, other emission control devices should be sufficiently instrumented to monitor all operating parameters necessary to enable timely identification of malfunctions. Consideration should be given to centralizing equipment instrumentation and controls, where feasible, to facilitate ease of changing and evaluating operating parameters.

Instrumentation may be supplemented by audible alarms that are preset to signal when prescribed operating range limits are exceeded.

Consideration should be given to installation of automatic shutdown instrumentation on processes and systems so that, when operating parameters on emission control devices exceed preset limits, operations would cease.

Equipment used in the emission control system should be clearly marked to allow easy identification. Up-to-date system drawings should be available to identify the location of valves and instruments. A record of system modification or changes should also be available.

Consideration should be given to keeping records of operating data in order to evaluate system performance and to provide a basis for establishing or modifying a preventive maintenance program.

Written procedures should be available for equipment operation and for operator actions if malfunctions occur. Checkoff lists should be considered for complex or infrequent modes of operation. Some operational procedures that may be considered for typical baghouses and wet scrubbers used at uranium mills are presented in Appendix C.

Equipment operators should be instructed in the function of each device and its operating characteristics. They should also be made aware of consequences of malfunctions and misoperation as well as of corrective measures that may be taken by the operator.

Equipment operators should be made aware of modifications to the equipment, changes in procedures, and problems encountered during system operation.

2. MAINTENANCE

A preventive maintenance program should be developed and implemented to sustain proper equipment performance and to reduce unscheduled repairs. Inspections should be performed at least annually, more frequently if necessary, on all components.

In the development of the maintenance program, consideration should be given to the type of emission control device, the manufacturer's recommendations, and the process at which the unit is installed. This program may require periodic updating to reflect onsite maintenance experience.

Schedules and written procedures should be available for maintenance work. Maintenance personnel should be trained in the implementation of maintenance procedures. They should be trained to recognize the symptoms that indicate potential problems, to determine the cause of the difficulty, and to remedy it with the help, if necessary, of the manufacturer or other outside resource.

3. TESTING

Emission control devices should be tested in place at least annually to verify collection efficiency. Collection efficiency for baghouses and wet scrubbers used in uranium mills should be based on inlet and outlet radioactive particulate concentrations in a dry gas corrected to standard temperature and pressure. Inlet and outlet (radioactive or uranium) particulate concentrations should be sampled simultaneously, if practicable.

The test should be performed in accordance with Method 5 of Appendix A to 40 CFR Part 60 or an acceptable equivalent.

If a preoperational in-place determination of collection efficiency is desired, a procedure mutually acceptable to the user and manufacturer may be used.

4. QUALITY ASSURANCE

The overall quality assurance program submitted by an applicant for a license should include provisions for (1) documentation, review, and evaluation of design, testing, operating, and maintenance data for emission control devices and (2) timely initiation of corrective actions necessary to maintain uniform operation of these devices within prescribed ranges under expected operating conditions.

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 in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide will be used by the NRC staff in evaluating procedures for designing, testing, operating, and maintaining emission control devices used at uranium mills.

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*Available from the Industrial Gas Cleaning Institute, Inc., 700 N. Fairfax Street, Alexandria, VA 22314.

APPENDIX A

TYPICAL MAINTENANCE ACTIVITIES FOR BAGHOUSES

COMPONENT	ACTIVITIES
Baghouse Housing	• Inspect exhaust from filters for visible dust. • Inspect gasketing on filter housing to ensure against leakage.
Compressed Air System	• Inspect for air leakage (low pressure) and check valves.* • Check alignment of air pulse holes with center of bag filters.*
Dust Collection Hopper	• Inspect for dust and debris buildup in ducts to hopper. • Rod out dust buildup on all accessible hopper surfaces. • Check operation of the discharge mechanism.
Manometer	• Inspect for blockage.
Filter Bags	• Inspect individual filter bags and attachment hardware.

*Activities applicable to pulse or jet baghouses. The remainder are applicable to all baghouses.

APPENDIX B

TYPICAL MAINTENANCE ACTIVITIES FOR WET SCRUBBERS

COMPONENT	ACTIVITIES
Scrubber Body	• Inspect for wear, particularly in areas downstream of gas and liquid acceleration. • Inspect for corrosion on all scrubber internal surfaces. • Inspect for excessive buildup, in particular in the wet/dry zone.
Nozzles	• Inspect for buildup and damage.
Entrainment Separator	• Check operation. • Inspect structural supports for integrity.
Pumps	• Inspect pumps for wear, seal water, packing, and smooth operation.
Instruments	• Inspect the condition of all instruments with regard to solids buildup.

APPENDIX C

TYPICAL OPERATIONAL SURVEILLANCE PROGRAM FOR EMISSION CONTROL DEVICES

EMISSION CONTROL DEVICE	SURVEILLANCE ACTIVITY
Baghouses	• Monitoring differential pressure. Adjusting timer or differential-pressure switch to adjust frequency of automatic cleaning cycle as needed. • Monitoring differential-pressure alarm lights in control area. • Monitoring compressed air pressure gauge on high-pressure air system. • Monitoring air flow instrumentation in control area.
Wet Scrubbers	• Monitoring differential pressure. • Monitoring differential-pressure alarm lights in control area. • Monitoring air flow instrumentation and alarm lights in control area. • Monitoring water flowmeters. • Monitoring water pressure alarm lights in control area. • Monitoring control area process control indicator lights for possible process shutdown in the event of water flow failures at preconditioning sprays or at the scrubber.

VALUE/IMPACT STATEMENT

The NRC staff performed a value/impact assessment to determine the proper procedural approach for providing guidance on designing, testing, operating, and maintaining emission control devices at uranium mills. The assessment resulted in a decision to develop a regulatory guide describing procedures for designing, testing, operating, and maintaining emission control devices at uranium mills. The results of this assessment were included in a draft regulatory guide on this subject, CE. 309-4, that was issued for public comment in May 1985. Comments received from the public and additional NRC staff review have shown no need to change the value/impact statement published with the proposed regulatory guide. Therefore, the value/impact statement published with the proposed guide is still applicable. A copy of the draft regulatory guide (identified by its task number, CE 309-4) and its associated value/impact statement is available for inspection and copying for a fee at the NRC Public Document Room at 1717 H Street NW., Washington, DC.

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REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 4.14

RADIOLOGICAL EFFLUENT AND ENVIRONMENTAL MONITORING AT URANIUM MILLS

A. INTRODUCTION

Uranium mill operators are required by Nuclear Regulatory Commission (NRC) regulations and license conditions to conduct radiological effluent and environmental monitoring programs. Regulations applicable to uranium milling are contained in 10 CFR Part 20, "Standards for Protection Against Radiation," and Part 40, "Domestic Licensing of Source Material." For example, § 40.65, "Effluent Monitoring Reporting Requirements," of 10 CFR Part 40 requires the submission to the Commission of semiannual reports containing information required to estimate doses to the public from effluent releases.

Information on radiation doses and the radionuclides in a mill's effluents and environment both prior to and during operations is needed by the NRC staff:

1. To estimate maximum potential annual radiation doses to the public resulting from effluent releases.

2. To ascertain whether the regulatory requirements of the NRC (including 10 CFR Part 20 dose limits, release limits, and the "as low as is reasonably achievable" requirement), mill license conditions, and the requirements of 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," have been met.

3. To evaluate the performance of effluent controls, including stabilization of active and inactive tailings piles.

4. To evaluate the environmental impact of milling operations, both during operations and after decommissioning.

5. To establish baseline data to aid in evaluation of decommissioning operations or decontamination following any unusual releases such as a tailings dam failure.

This guide describes programs acceptable to the NRC staff for measuring and reporting releases of radioactive materials to the environment from typical uranium mills.

The programs described in this guide are not requirements. Licensing requirements are determined by the NRC staff on a case-by-case basis during individual licensing reviews. Individual applicants or licensees may propose alternatives for new or existing monitoring programs that need not necessarily be consistent with this guide. The justification for such alternatives will be reviewed by the NRC staff, and the acceptability of proposed alternatives will be determined on a case-by-case basis during individual licensing reviews. For example, it is anticipated that operational monitoring programs that do not include at least three continuous air samples at the site boundary will include more extensive stack sampling and more sampling locations than are described in this guide as well as meteorological data and additional environmental monitoring requirements.

B. DISCUSSION

The radiation dose an individual receives can be determined only if the radionuclides to which an individual is exposed are known. Therefore, monitoring programs should provide accurate information on the specific radionuclides in effluents from a mill, its ore piles, and its tailings retention system and in the surrounding environment.

Methods of sampling and analysis for the radionuclides associated with uranium milling are discussed in sources listed in the bibliography. The listing of these documents is not meant to be all inclusive, nor does it constitute an endorsement by the NRC staff of all of the methods in all of the listings. Rather, these listings are provided as sources of information to aid the licensee in developing a monitoring program.

The sampling program described below is divided into two parts: preoperational monitoring and operational

* The substantial number of changes in this revision has made it impractical to indicate the changes with lines in the margin.

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. This guide was revised as a result of substantive comments received from the public and additional staff review.

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monitoring. Preoperational data is submitted to the NRC as part of the application process. Operational data is reported as required by § 40.65 of 10 CFR Part 40 and specific license conditions and at times of license renewal.

C. REGULATORY POSITION

1. PREOPERATIONAL MONITORING

An acceptable preoperational monitoring program is described below and summarized in Table 1. At least twelve consecutive months of data, including complete soil sampling, direct radiation, and radon flux data, should be submitted to the NRC staff prior to any major site construction. A complete preoperational report with twelve consecutive months of data should be submitted prior to beginning milling operations. Prior to the start of local mining operations, if possible, monitoring data, including airborne radon measurements, should be submitted to the NRC staff.

Applicants may propose alternatives to this preoperational program. However, equivalent alternatives should be proposed for the operational program so that the programs remain compatible.

1.1 Preoperational Sampling Program

1.1.1 Air Samples

Air particulate samples should be collected continuously at a minimum of three locations at or near the site boundary. If there are residences or occupiable structures within 10 kilometers of the site, a continuous outdoor air sample should be collected at or near the structure with the highest predicted airborne radionuclide concentration due to milling operations and at or near at least one structure in any area where predicted doses exceed 5 percent of the standards in 40 CFR Part 190. A continuous air sample should also be collected at a remote location that represents background conditions at the mill site; in general, a suitable location would be in the least prevalent wind direction from the site and unaffected by mining or other milling operations.

Normally, filters for continuous ambient air samples are changed weekly or more often as required by dust loading.

The sampling locations should be determined according to the projected site and milling operation. Preoperational sampling locations should be the same as operational locations. The following factors should be considered in determining the sampling locations: (1) average meteorological conditions (windspeed, wind direction, atmospheric stability), (2) prevailing wind direction, (3) site boundaries nearest to mill, ore piles, and tailings piles, (4) direction of nearest occupiable structure (see footnotes of Tables 1 and 2), and (5) location of estimated maximum concentrations of radioactive materials.

Samples should be collected continuously, or for at least one week per month, for analysis of radon-222. The sampling locations should be the same as those for the continuous air particulate samples.

1.1.2 Water Samples

Samples of ground water should be collected quarterly from at least three sampling wells located hydrologically down gradient from the proposed tailings area, at least three locations near other sides of the tailings area, and one well located hydrologically up gradient from the tailings area (to serve as a background sample). The location of the ground-water sampling wells should be determined by hydrological analysis of the potential movement of seepage from the tailings area, and the basis for choosing these locations should be presented when data is reported. Wells drilled close to the tailings for the specific purpose of obtaining representative samples of ground water that may be affected by the mill tailings are preferable to existing wells.

Ground-water samples should also be collected quarterly from each well within two kilometers of the proposed tailings area that is or could be used for drinking water, watering of livestock, or crop irrigation.

Samples of surface water should be collected quarterly from each onsite water impoundment (such as a pond or lake) and any offsite water impoundment that may be subject to seepage from tailings, drainage from potentially contaminated areas, or drainage from a tailings impoundment failure.

Samples should be collected at least monthly from streams, rivers, any other surface waters or drainage systems crossing the site boundary, and any offsite surface waters that may be subject to drainage from potentially contaminated areas or from a tailings impoundment failure. Any stream beds that are dry part of the year should be sampled when water is flowing. Samples should be collected at the site boundary or at a location immediately downstream of the area of potential influence.

1.1.3 Vegetation, Food, and Fish Samples

Forage vegetation should be sampled at least three times during the grazing season in grazing areas in three different sectors having the highest predicted airborne radionuclide concentration due to milling operations.

At least three samples should be collected at time of harvest or slaughter or removal of animals from grazing for each type of crop (including vegetable gardens) or livestock raised within three kilometers of the mill site.

Fish (if any) samples should be collected semiannually from any bodies of water that may be subject to seepage or surface drainage from potentially contaminated areas or that could be affected by a tailings impoundment failure.

1.1.4 Soil and Sediment Samples

Prior to initiation of mill construction (and if possible prior to mining), one set of soil samples should be collected as follows:

a. Surface-soil samples (to a depth of five centimeters) should be collected using a consistent technique at 300-

meter intervals in each of the eight compass directions out to a distance of 1500 meters from the center of the milling area. The center is defined as the point midway between the proposed mill and the tailings area.

b. Surface-soil samples should also be collected at each of the locations chosen for air particulate samples.

c. Subsurface samples (to a depth of 1 meter) should be collected at the center of the milling area and at a distance of 750 meters in each of the four compass directions.

Soil sampling should be repeated for each location disturbed by site excavation, leveling, or contouring.

One set of sediment samples should be collected from the same surface-water locations as described in Section 1.1.2. For surface water passing through the site, sediment should be sampled upstream and downstream of the site. Samples should be collected following spring runoff and in late summer, preferably following an extended period of low flow. In each location, several sediment samples should be collected in a traverse across the body of water and composited for analysis.

1.1.5 Direct Radiation

Prior to initiation of mill construction (and if possible prior to mining), gamma exposure rate measurements should be made at 150-meter intervals in each of the eight compass directions out to a distance of 1500 meters from the center of the milling area. Measurements should also be made at the sites chosen for air particulate samples.

Measurements should be repeated for each location disturbed by site excavation, leveling, or contouring.

Gamma exposure measurements should be made with passive integrating devices (such as thermoluminescent dosimeters), pressurized ionization chambers, or properly calibrated portable survey instruments.

Direct radiation measurements should be made in dry weather, not during periods following rainfall or when soil is abnormally wet.

1.1.6 Radon Flux Measurements

Radon-222 flux measurements should be made in three separate months during normal weather conditions in the spring through the fall when the ground is thawed. The measurements should be made at the center of the milling area and at locations 750 and 1500 meters from the center in each of the four compass directions. Measurements should not be taken when the ground is frozen or covered with ice or snow or following periods of rain.

1.2 Analysis of Preoperational Samples

Air particulate samples should be analyzed for natural uranium, thorium-230, radium-226, and lead-210.

Air samples collected for radon should be analyzed for radon-222.

The results of analyses of air samples should be used to determine the radionuclide concentrations for the sampling locations.

All ground-water samples collected near the tailings area should be analyzed for dissolved natural uranium, thorium-230, radium-226, polonium-210, and lead-210. Ground-water samples from sources that could be used as drinking water for humans or livestock or crop irrigation should also be analyzed for suspended natural uranium, thorium-230, radium-226, polonium-210, and lead-210.

Surface-water samples from water impoundments should be analyzed quarterly for natural uranium, thorium-230, and radium-226 and semiannually for lead-210 and polonium-210. The samples should be analyzed separately for dissolved and suspended radionuclides.

Surface-water samples from flowing surface water should be analyzed monthly for natural uranium, thorium-230 and radium-226 and semiannually for lead-210 and polonium-210. The samples should be analyzed separately for dissolved and suspended radionuclides.

The results of analyses of water samples should be used to determine the radionuclide concentrations for the sampling locations.

Vegetation, food, and fish (edible portion) samples should be analyzed for natural uranium, thorium-230, radium-226, lead-210, and polonium-210.

All soil samples should be analyzed for radium-226. Soil samples collected at air particulate sampling locations and ten percent of all other soil samples (including at least one subsurface set) should be analyzed for natural uranium, thorium-230, and lead-210. Analysis of extra soil samples may be necessary for repeat samples collected at locations disturbed by site excavation, leveling, or contouring.

Sediment samples should be analyzed for natural uranium, thorium-230, radium-226, and lead-210.

2. OPERATIONAL MONITORING

An acceptable monitoring program to be conducted during construction and after the beginning of milling operations is described below and summarized in Table 2. The results of this program should be summarized quarterly and submitted to NRC semiannually pursuant to § 40.65 of 10 CFR Part 40. An acceptable reporting format is shown in Table 3.

2.1 Operational Sampling Program

2.1.1 Stack Sampling

Effluents from the yellowcake dryer and packaging stack should be sampled at least quarterly during normal operations. The sampling should be isokinetic, representative,

and adequate for determination of the release rates and concentrations of uranium. The sampling should also be adequate for the determination of release rates and concentrations of thorium-230, radium-226, and lead-210 if this data cannot be obtained from other sources.

Other stacks should be sampled at least semiannually. The samples should be representative (not necessarily isokinetic) and adequate for the determination of the release rates and concentrations of uranium, thorium-230, radium-226, and lead-210.

All stack flow rates should be measured at the time of sampling.

2.1.2 Air Samples

Air particulate samples should be collected continuously at (1) a minimum of three locations at or near the site boundary, (2) the residence or occupiable structure within 10 kilometers of the site with the highest predicted airborne radionuclide concentration, (3) at least one residence or occupiable structure where predicted doses exceed 5 percent of the standards in 40 CFR Part 190, and (4) a remote location representing background conditions. The sampling locations should be the same as those for the preoperational air samples (see Section 1.1.1). The sampling should be adequate for the determination of natural uranium, thorium-230, radium-226, and lead-210.

Normally, filters for continuous ambient air samples are changed weekly or more often as required by dust loading.

Samples should be collected continuously at the same locations, or for at least one week per month, for analysis of radon-222.

2.1.3 Water Samples

Samples of ground water should be collected from at least three sampling wells located hydrologically down gradient from the tailings area and from one background well located hydrologically up gradient. The samples should be collected monthly through the first year of operation and quarterly thereafter from the same downslope and background wells that were used for preoperational samples (see Section 1.1.2).

Samples should be collected at least quarterly from each well within two kilometers of the tailings area that is or could be used for drinking water, watering of livestock, or crop irrigation.

Samples should be collected at least quarterly from each onsite water impoundment (such as a pond or lake) and any offsite water impoundment that may be subject to seepage from tailings, drainage from potentially contaminated areas, or drainage from a tailings impoundment failure.

Samples should be collected at least monthly from any surface water crossing the site boundary and offsite streams or rivers that may be subject to drainage from potentially contaminated areas or from a tailings impoundment failure. Stream beds that are dry part of the year should be sampled when water is flowing. Operational samples should be collected upstream and downstream of the area of potential influence.

Any unusual releases (such as surface seepage) that are not part of normal operations should be sampled.

.2.1.4 Vegetation, Food, and Fish Samples

Where a significant pathway to man is identified in individual licensing cases, vegetation, food, and fish samples should be collected as described below.

Forage vegetation should be sampled at least three times during the grazing season in grazing areas in three different sectors having the highest predicted airborne radionuclide concentration due to milling operations.

At least three samples should be collected at the time of harvest or slaughter or removal of animals from grazing for each type of crop (including vegetable gardens) or livestock raised within three kilometers of the mill site.

Fish (if any) samples should be collected semiannually from any bodies of water that may be subject to seepage or surface drainage from potentially contaminated areas or that could be affected by a tailings impoundment failure.

2.1.5 Soil and Sediment Samples

Surface-soil samples should be collected annually using a consistent technique at each of the locations chosen for air particulate samples as described in Section 2.1.2.

Sediment samples should be collected annually from the surface-water locations described in Section 2.1.3.

2.1.6 Direct Radiation

Gamma exposure rates should be measured quarterly at the sites chosen for air particulate samples as described in Section 2.1.2. Passive integrating devices (such as thermoluminescent dosimeters), pressurized ionization chambers, or properly calibrated portable survey instruments should be used (see Regulatory Guide 4.13).

2.2 Analysis of Operational Samples

Samples from the yellowcake dryer and packaging stack should be analyzed for natural uranium. Samples should also be analyzed for thorium-230, radium-226, and lead-210 if this data cannot be obtained from other sources such as isotopic analysis of yellowcake product. Samples from other stacks should be analyzed for natural uranium, thorium-230, radium-226, and lead-210.

Air particulate samples should be analyzed for natural uranium, thorium-230, radium-226, and lead-210.

Air samples collected for radon should be analyzed for radon-222.

The results of analyses of air samples should be used to determine the radionuclide release rates for the stacks and the radionuclide concentrations for the stacks and other sampling locations.

Water samples should be analyzed for natural uranium, thorium-230, radium-226, polonium-210, and lead-210.

Ground-water samples from sources not expected to be used as drinking water should be analyzed for dissolved radionuclides. Ground-water samples from sources that could be used as drinking water for humans or livestock and all surface-water samples should be analyzed separately for dissolved and suspended radionuclides. These results should be used to determine radionuclide concentrations for ground water and natural bodies of water.

All vegetation, food, and fish (edible portion) samples should be analyzed for radium-226 and lead-210.

All soil samples should be analyzed for natural uranium, radium-226, and lead-210.

All sediment samples should be analyzed for natural uranium, thorium-230, radium-226, and lead-210.

3. QUALITY OF SAMPLES

Provisions should be made to ensure that representative samples are obtained by use of proper sampling equipment, proper locations of sampling points, and proper sampling procedures (see bibliography).

Air samples may be composited for analysis if (1) they are collected at the same location and (2) they represent a sampling period of one calendar quarter or less. Air samples should not be composited if (1) they represent a sampling period of more than one calendar quarter, (2) they are from different sampling locations, or (3) the samples are to be analyzed for radon-222.

Samples collected for analysis of radon-222 should be analyzed quickly enough to minimize decay losses.

Samples other than air samples should not be composited.

4. SOLUBILITY OF AIRBORNE RADIOACTIVE MATERIAL

Table II of Appendix B, "Concentrations in Air and Water Above Natural Background," to 10 CFR Part 20 lists separate values for soluble and insoluble radioactive materials in effluents. In making comparisons between airborne effluent concentrations and the values given in Table II of Appendix B to 10 CFR Part 20, the maximum permissible concentrations for insoluble materials should be used.

5. LOWER LIMIT OF DETECTION

The lower limits of detection for stack effluent samples should be 10% of the appropriate concentration limits listed in Table II of Appendix B to 10 CFR Part 20.

The lower limits of detection for analysis of other samples should be as follows:

U-natural, Th-230, Ra-226 in air	- 1×10^{-16} μCi/ml
Pb-210 in air	- 2×10^{-15} μCi/ml
Rn-222	- 2×10^{-10} μCi/ml
U-natural, Th-230, Ra-226 in water	- 2×10^{-10} μCi/ml
Po-210 in water	- 1×10^{-9} μCi/ml
Pb-210 in water	- 1×10^{-9} μCi/ml
U-natural, Th-230, Ra-226, Pb-210 in soil and sediment (dry)	- 2×10^{-7} μCi/g
U-natural, Th-230 in vegetation, food, and fish (wet)	- 2×10^{-7} μCi/kg
Ra-226 in vegetation, food, and fish (wet)	- 5×10^{-8} μCi/kg
Po-210, Pb-210 in vegetation, food, and fish (wet)	- 1×10^{-6} μCi/kg

Obviously, if the actual concentrations of radionuclides being sampled are higher than the lower limits of detection indicated above, the sampling and analysis procedures need only be adequate to measure the actual concentrations. In such cases, the standard deviation estimated for random error of the analysis should be no greater than 10% of the measured value.

An acceptable method for calculating lower limits of detection is described in the appendix to this guide.

6. PRECISION AND ACCURACY OF RESULTS

6.1 Error Estimates

The random error associated with the analysis of samples should always be calculated. The calculation should take into account all significant random uncertainties, not merely counting error.

If the analyst estimates that systematic errors associated with the analysis are significant relative to the random error, the magnitude of the systematic error should be estimated.

6.2 Calibration

Individual written procedures should be prepared and used for specific methods of calibrating all sampling and measuring equipment, including ancillary equipment. The procedures should ensure that the equipment will operate with adequate accuracy and stability over the range of its intended use. Calibration procedures may be compilations

of published standard practices, manufacturers' instructions that accompany purchased equipment, or procedures written in-house. Calibration procedures should identify the specific equipment or group of instruments to which the procedures apply.

To the extent possible, calibration of measuring equipment should be performed using radionuclide standards certified by the National Bureau of Standards or standards obtained from suppliers who participate in measurement assurance activities with the National Bureau of Standards (see Regulatory Guide 4.15).

Calibrations should be performed at regular intervals, at least semiannually, or at the manufacturer's suggested interval, whichever is more frequent. Frequency of calibration should be based on the stability of the system. If appropriate, equipment may be calibrated before and after use instead of at arbitrarily scheduled intervals. Equipment should be recalibrated or replaced after any repairs or whenever it is suspected of being out of adjustment, excessively worn, or otherwise damaged and not operating properly. Functional tests, i.e., routine checks performed to demonstrate that a given instrument is in working condition, may be performed using sources that are not certified by the National Bureau of Standards.

6.3 Quality of Results

A continuous program should be prepared and implemented for ensuring the quality of results and for keeping random and systematic uncertainties to a minimum. The procedures should ensure that samples and measurements are obtained in a uniform manner and that samples are not changed prior to analysis because of handling or because of their storage environment. Tests should be applied to analytical processes, including duplicate analysis of selected effluent samples and periodic cross-check analyses with independent laboratories (see Regulatory Guide 4.15).

7. RECORDING AND REPORTING RESULTS

This section provides guidelines for recording all results. Reports submitted to NRC should be prepared using these guidelines and the format shown in Table 3 of this guide.

7.1 Sampling and Analysis Results

7.1.1 Air and Stack Samples

For each air or stack sample, the following should be recorded:

1. Location of sample.
2. Dates during which sample was collected.
3. The concentrations of natural uranium, thorium-230, radium-226, lead-210, and radon-222 for all samples except stack samples.
4. The cconcentration of natural uranium, thorium-230, radium-226, and lead-210 for stack effluent samples.
5. The percentage of the appropriate concentration limit as shown in Table II of Appendix B to 10 CFR Part 20.
6. The estimated release rate of natural uranium, thorium-230, radium-226, and lead-210 for stack effluent samples.
7. The flow rate of each stack.

7.1.2 Liquid Samples

For each liquid sample, the following should be recorded:

1. Location of sample.
2. Type of sample (ground or surface water).
3. Date of sample collection.
4. The concentrations of natural uranium, thorium-230, radium-226, polonium-210, and lead-210. (If separate analyses were conducted for dissolved and suspended radionuclides, report each result separately.)

7.1.3 Other Samples

For other samples, the following should be recorded:

1. Location of sample.
2. Date of sample collection.
3. Type of sample (vegetation, soil, radon-222 flux, gamma exposure rate, etc.).
4. Analytical result (radionuclide concentration, gamma exposure rate, radon flux rate, etc.).

7.1.4 Error Estimates

Reported results should always include estimates of uncertainty. The magnitude of the random error of the analysis to the 95% uncertainty level should be reported for each result. If significant, an estimate of the magnitude of the systematic error should also be reported.

7.2 Supplemental Information

The following information should be included in each monitoring report submitted to NRC:

1. Name of facility, location, docket number, and license number.
2. Description of sampling equipment and discussion of how sampling locations were chosen.

3. Description of sampling procedures, including sampling times, rates, and volumes.

4. Description of analytical procedures.

5. Description of calculational methods.

6. Discussion of random and systematic error estimates, including methods of calculation and sources of systematic error.

7. The values of the lower limits of detection, along with a description of the calculation of the lower limit of detection.

8. The values of maximum permissible concentration from Table II of Appendix B to 10 CFR Part 20 used in any calculations.

9. Discussion of the program for ensuring the quality of results.

10. Description of calibration procedures.

11. Discussion of any unusual releases, including the circumstances of the release and any data available on the quantities of radionuclides released.

7.3 Units

Radionuclide quantities should be reported in curies. Radionuclide concentrations should be reported in microcuries per milliliter for air and water, microcuries per gram for soil and sediment, and microcuries per kilogram for vegetation, food, or fish. Direct radiation exposure rates should be reported in milliroentgens per calendar quarter. Radon flux rates should be reported in picocuries per square meter per second. Stack flow rates should be reported in cubic meters per second. (In the International System of Units, a curie equals 3.7×10^{10} becquerels, a microcurie equals 3.7×10^{4} becquerels, and a milliliter equals 10^{-6} cubic meters.)

Estimates of random error should be reported in the same units as the result itself. Estimates of systematic error should be reported as a percentage of the result.

Note: The Commission has discontinued the use in 10 CFR Part 20 of the special curie definitions for natural uranium and natural thorium (39 FR 23990, June 28, 1974). Reports to the Commission should use units consistent with this change.

7.4 Significant Figures

Results should not be reported with excessive significant figures, so that they appear more certain than they actually are. The reported estimate of error should contain no more than two significant figures. The reported result itself should have the same number of decimal places as the reported error.

7.5 Format

Reports should be submitted according to the format shown in Table 3.

The term "not detected," "less than the lower limit of detection (LLD)," or similar terms should never be used. Each reported result should be a value and its associated error estimate, including values less than the lower limit of detection or less than zero.

TABLE 1

PREOPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
AIR						
Particulates	Three	At or near the site boundaries	Continuous[a]	Weekly filter change or more frequently as required by dust loading	Quarterly composites of weekly samples	Natural uranium, Ra-226, Th-230, and Pb-210
	One	At or close to the nearest[b] residence(s) or occupiable offsite structure(s) (if within 10 km of site)	Continuous	Weekly filter change or more frequently as required by dust loading	Quarterly composites of weekly samples	Natural uranium, Ra-226, Th-230, and Pb-210
	One	At a control or background location remote from site[c]	Continuous	Weekly filter change or more frequently as required by dust loading	Quarterly composites of weekly samples	Natural uranium, Ra-226, Th-230, and Pb-210
Radon Gas[d]	Five or more	Same locations as for air particulates	Continuous or at least one week per month representing about the same period each month	Continuous	Each sample or continuous	Rn-222
WATER						
Ground Water[e]	Six or more	Wells located around future tailings disposal area. At least three wells hydrologically down gradient from disposal area. At least three located on other sides of tailings disposal area.[f]	Grab	Quarterly	Quarterly	Dissolved natural uranium, Ra-226, Th-230, Pb-210, and Po-210
	One from each well	Wells within 2 km of tailings disposal area that are or could be used for potable water supplies, watering of livestock, or crop irrigation.	Grab	Quarterly	Quarterly	Dissolved and suspended natural uranium, Ra-226, Th-230, Pb-210, and Po-210
	One	Well located hydrologically up gradient from tailings disposal area to serve as control or background location.	Grab	Quarterly	Quarterly	Dissolved natural uranium, Ra-226, Th-230, Pb-210, and Po-210

TABLE 1 (Continued)

PREOPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
Surface Water[g]	One from each body of water	Large permanent onsite water impoundments or offsite impoundments that may be subject to direct surface drainage from potentially contaminated areas or that could be affected by a tailings impoundment failure.	Grab	Quarterly	Quarterly	Suspended and dissolved natural uranium, Ra-226 and Th-230
					Semiannually	Suspended and dissolved Pb-210 and Po-210
Surface Water	One from each body of water	Surface waters passing through the site(n) or offsite surface waters that may be subject to drainage from potentially contaminated areas or that could be affected by a tailings impoundment failure.	Grab	Monthly	Monthly	Suspended and dissolved natural uranium, Ra-226, Th-230
					Semiannually	Suspended and dissolved Pb-210 and Po-210
VEGETATION, FOOD, AND FISH						
Vegetation	Three	Grazing areas near the site in different sectors that will have the highest predicted air particulate concentrations during milling operations.	Grab	Three times during grazing season	Three times	Natural uranium, Ra-226, Th-230, Pb-210, and Po-210
Food	Three of each type	Crops, livestock, etc. raised within 3 km of mill site	Grab	Time of harvest or slaughter	Once	Natural uranium, Ra-226, Th-230, Pb-210, and Po-210
Fish	Each body of water	Collection of fish (if any) from lakes, rivers, and streams in the site environs that may be subject to seepage or direct surface runoff from potentially contaminated areas or that could be affected by a tailings impoundment failure	Grab	Semiannually	Twice	Natural uranium, Ra-226, Th-230, Pb-210, and Po-210

TABLE 1 (Continued)

PREOPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
SOIL AND SEDIMENT						
Surface Soil[k]	Up to forty	300-meter intervals to a distance of 1500 meters in each of 8 directions from center of milling area	Grab	Once prior to site construction. Repeat for location disturbed by excavation, leveling, or contouring	Once	All samples for Ra-226, 10% of samples natural uranium, Th-230, and Pb-210
Surface Soil	Five or more	At same locations used for collection of air particulate samples.	Grab	Once prior to site construction	Once	Natural uranium, Ra-226, Th-230, and Pb-210
Subsurface Soil Profile[l]	Five	At center reference location and at distances of 750 meters in each of 4 directions.	Grab	Once prior to site construction. Repeat for locations disturbed by construction.	Once	Ra-226 (all samples) Natural uranium, Th-230, and Pb-210 (one set of samples)
Sediment[m]	Two from each stream	Up and downstream of surface waters passing through site or from offsite surface waters that may be subject to direct runoff from potentially contaminated areas or that could be affected by a tailings impoundment failure	Grab	Once following spring runoff and late summer following period of extended low flow	Twice	Natural uranium, Ra-226, Th-230, and Pb-210
	One from each water impoundment	Onsite water impoundments (lakes, ponds, etc), or offsite impoundments that may be subject to direct surface runoff from potentially contaminated areas or that could be affected by tailings impoundment failure	Grab	Once prior to site construction	Once	Natural uranium, Ra-226, Th-230, and Pb-210
DIRECT RADIATION	Up to eighty	150-meter intervals to a distance of 1500 meters in each of 8 directions from center of milling area or at a point equidistant from milling area[i] and tailings disposal area.		Once prior to site construction. Repeat for areas disturbed by site preparation or construction.	Once	Gamma exposure rate, using passive integrating device such as TLD, pressurized ionization chamber, or properly calibrated portable survey instrument.

TABLE 1 (Continued)

PREOPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
	Five or more	At same locations used for collection of particulate samples		Once prior to site construction	Once	Gamma exposure rate, using passive integrating device, pressurized ionization chamber, or properly calibrated portable survey instrument.
RADON FLUX[n]	Up to ten	At center reference location and at distances of 750 and 1500 meters in each of 4 directions.		One sample during each of three months.	Each sample	Radon-222 flux

TABLE 2

OPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
STACKS						
Particulates	One for each stack	Yellowcake dryer and packaging stack(s)	Isokinetic	Quarterly	Each sample	Natural uranium, Th-230, Ra-226, and Pb-210 if not available from other sources. Measure stack flow rate semiannually.
Particulates	One for each stack	Other stacks	Representative grab	Semiannually	Each sample	Natural uranium Th-230, Ra-226, and Pb-210. Measure stack flow.
AIR						
Particulates	Three	Locations at or near the site boundaries and in different sectors that have the highest predicted concentrations of airborne particulates[b]	Continuous[a]	Weekly filter change, or more frequently as required by dust loading	Quarterly composite, by location, of weekly samples	Natural uranium, Ra-226, Th-230, and Pb-210
	One or more	At the nearest residence(s) or occupiable structure(s)	Continuous	Weekly filter change, or more frequently as required by dust loading	Quarterly composite, by location, of weekly samples	Natural uranium, Ra-226, Th-230, and Pb-210
	One	Control Location(s)[c]	Continuous	Weekly filter change, or more frequently as required by dust loading	Quarterly composite, by location, of weekly samples	Natural uranium, Ra-226, Th-230, and Pb-210
Radon Gas	Five or more	Same locations as for air particulates	Continuous or at least one week per month[d]	At least one week per calendar month representing approximately the same period each month	Monthly	Rn-222
WATER						
Ground Water	Three or more	Hydrologically down gradient and relatively close to the tailings impoundment[f]	Grab	Monthly (first year) Quarterly (after first year)	Monthly (first year) Quarterly (after first year)	Dissolved natural uranium, Ra-226, Th-230, Pb-210, and Po-210[e]
	At least one control sample	Hydrologically up gradient (i.e., not influenced by seepage from tailings)	Grab	Quarterly	Quarterly	Dissolved natural uranium, Ra-226, Th-230, Pb-210 and Po-210

TABLE 2 (Continued)

OPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
	One from each well	Each well used for drinking water or watering of livestock or crops within 2 km of the tailings impoundment	Grab	Quarterly	Quarterly	Dissolved and suspended natural uranium, Ra-226, Th-230, Pb-210, and Po-210
Surface Water	Two from each water body	Surface waters passing through the mill site or offsite surface waters that are sufficiently close to the site to be subject to surface drainage from potentially contaminated areas or that could be influenced by seepage from the tailings disposal area.(h) One sample collected upstream of mill site and one sample collected at the downstream site boundary or at a location immediately downstream of location of potential influence	Grab	Quarterly	Quarterly	Dissolved and suspended natural uranium, Ra-226, Th-230, Pb-210, and Po-210(g)
	One from each water body	Large water impoundments (i.e., lakes, reservoirs) near the mill site that are sufficiently close to the site to be subject to drainage from potentially contaminated areas or that could be influenced by seepage from the tailings disposal area.	Grab	Quarterly	Quarterly	Dissolved and suspended natural uranium, Ra-226, Th-230, Pb-210, and Po-210
VEGETATION, FOOD, AND FISH						
Vegetation or Forage(o)	Three or more	From animal grazing areas near the mill site in the direction of the highest predicted airborne radionuclide concentrations	Grab	Three times during grazing season	Each sample	Ra-226 and Pb-210

TABLE 2 (Continued)

OPERATIONAL RADIOLOGICAL MONITORING PROGRAM FOR URANIUM MILLS

Type of Sample	Sample Collection				Sample Analysis	
	Number	Location	Method	Frequency	Frequency	Type of Analysis
Food	Three of each type	Crops, livestock, etc. raised within 3 km of mill site	Grab	Time of harvest or slaughter	Once	Ra-226 and Pb-210
Fish	Each body of water	Collection of fish (if any) from lakes, rivers, and streams in the site environs that may be subject to seepage or direct surface runoff from potentially contaminated areas or that could be affected by a tailings impoundment failure	Grab	Semiannually	Twice	Ra-226 and Pb-210
SOIL AND SEDIMENT						
Soil	Five or more	Same as for air particulate samples(k)	Grab	Annually	Annually	Natural uranium, Ra-226, and Pb-210
Sediment	One or two from each water body	Same as surface water samples(m)	Grab	Annually	Annually	Natural uranium, Th-230, Ra-226, and Pb-210
DIRECT RADIATION	Five or more	Same as for air particulate samples	Continuous passive integrating device	Quarterly change of passive dosimeters	Quarterly	Gamma exposure rate

Footnotes for Tables 1 and 2:

(a) Continuous collection means continuous sampler operation with filter change weekly or as required by dust loading, whichever is more frequent.

(b) The term "nearest" as used here means the location with the highest predicted airborne radionuclide concentrations during milling operations.

(c) Care should be taken in selection of the control sampling location so that it is representative of the site conditions. In general, a location in the least prevalent wind direction from the site should provide a suitable location for a control sampling site.

(d) Various methods are acceptable; for example: (1) Continuous collection of a gaseous air sample with samples being changed about every 48 hours for a 1-week period or (2) continuous sampling.

(e) If the sample contains appreciable suspended material, it should be filtered as soon as possible following collection through a membrane filter and the filtrate acidified to 1% hydrochloric acid.

(f) The location of the ground-water sampling wells should be determined by a hydrological analysis of the potential movement of seepage from the tailings disposal area. In general, the objective is to place monitor wells in all directions around the tailings area with the emphasis on the down gradient locations.

(g) Surface-water samples to be analyzed for dissolved and suspended fractions should be filtered as soon as possible following collection through a membrane filter and the filtrate acidified to 1% hydrochloric acid.

(h) Natural drainage systems (dry washes) that carry surface runoff from the site following a precipitation event should be sampled following the event but at a frequency not greater than monthly.

(i) The milling area refers to the area that includes ore storage pads, mill buildings, and other processing areas.

(j) Thermoluminescent dosimeters should contain two or more chips or otherwise provide for two readings per exposure period (see Regulatory Guide 4.13).

(k) Surface soil samples should be collected using a consistent technique to a depth of 5 cm.

(l) Subsurface soil profile samples should be collected to a depth of one meter. Samples should be divided into three equal sections for analysis.

(m) Several samples should be collected at each location and composited for a representative sample.

(n) Radon exhalation measurements should not be taken during periods when the ground is frozen or covered with ice or snow or following periods of rain. It is recommended that these measurements be taken in the spring through the fall during normal weather conditions.

(o) Vegetation or forage sampling need be carried out only if dose calculations indicate that the ingestion pathway from grazing animals is a potentially significant exposure pathway (an exposure pathway should be considered important if the predicted dose to an individual would exceed 5% of the applicable radiation protection standard).

TABLE 3[a]

SAMPLE FORMAT FOR REPORTING MONITORING DATA

1. STACK SAMPLES

For each sample analyzed, report the following information:

a. Date sample was collected
b. Location of sample collection
c. Stack flow rate (m^3/sec)

Radionuclide	Concentration (µCi/ml)	Error Estimate[b] (µCi/ml)	Release Rate (Ci/qr)	Error Estimate (Ci/qr)	LLD[c] (µCi/ml)	% MPC[c]
U-nat						
Th-230						
Ra-226						
Pb-210						

2. AIR SAMPLES

For each sample analyzed, report the following information:

a. Date sample was collected
b. Location of sample collection

Radionuclide	Concentration (µCi/ml)	Error Estimate (µCi/ml)	LLD (µCi/ml)	% MPC
U-nat				
Th-230				
Ra-226				
Pb-210				
Rn-222				

[a] This table illustrates format only. It is not a complete list of data to be reported. (See text of guide and Tables 1 and 2.)

[b] Error estimate should be calculated at 95% uncertainty level, based on all sources of random error, not merely counting error. Significant systematic error should be reported separately. See Sections 6.1, 7.1.4, and 7.3.

[c] All calculations of lower limits of detection (LLD) and percentages of maximum permissible concentration (MPC) should be included as supplemental information.

TABLE 3 (Continued)

SAMPLE FORMAT FOR REPORTING MONITORING DATA

3. LIQUID SAMPLES

For each sample analyzed, report the following information:

a. Date sample was collected
b. Location of sample collection
c. Type of sample (for example: surface, ground, drinking, stock, or irrigation)

Radionuclide	Concentration (μCi/ml)	Error Estimate (μCi/ml)	LLD (μCi/ml)
U-nat (dissolved)			
U-nat (suspended)[d]			
Th-230 (dissolved)			
Th-230 (suspended)[d]			
Ra-226 (dissolved)			
Ra-226 (suspended)[d]			
Pb-210 (dissolved)			
Pb-210 (suspended)[d]			
Po-210 (dissolved)			
Po-210 (suspended)[d]			

4. VEGETATION, FOOD, AND FISH SAMPLES

For each sample analyzed, report the following information:

a. Date sample was collected
b. Location of sample collection
c. Type of sample and portion analyzed

Radionuclide	Concentration (μCi/kg wet)	Error Estimate (μCi/kg)	LLD (μCi/kg)
U-nat			
Th-230			
Ra-226			
Pb-210			
Po-210			

[d] Not all samples must be analyzed for suspended radionuclides. See Sections 1.2 and 2.2 of this guide.

TABLE 3 (Continued)

SAMPLE FORMAT FOR REPORTING MONITORING DATA

5. SOIL AND SEDIMENT SAMPLES

For each sample analyzed, report the following information:

a. Date sample was collected
b. Location of sample collection
c. Type of sample and portion analyzed

Radionuclide	Concentration (μCi/g)	Error Estimate (μCi/g)	LLD (μCi/g)
U-nat			
Th-230			
Ra-226			
Pb-210			
Po-210			

6. DIRECT RADIATION MEASUREMENTS

For each measurement, report the dates covered by the measurement and the following information:

Location	Exposure Rate (mR/qr)	Error Estimate (mR/qr)

7. RADON FLUX MEASUREMENTS

For each measurement, report the dates covered by the measurement and the following information:

Location	Flux (pCi/m^2-sec)	Error Estimate (pCi/m^2-sec)

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APPENDIX

LOWER LIMIT OF DETECTION

For the purposes of this guide, the Lower Limit of Detection (LLD) is defined as the smallest concentration of radioactive material sampled that has a 95% probability of being detected, with only a 5% probability that a blank sample will yield a response interpreted to mean that radioactive material is present. (Radioactive material is "detected" if it yields an instrument response that leads the analyst to conclude that activity above the system background is present.)

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ S_b}{3.7 \times 10^4\ EVY \exp(-\lambda \Delta t)}$$

where

LLD is the lower limit of detection (microcuries per milliliter);

S_b is the standard deviation of the instrument background counting rate (counts per second);

3.7×10^4 is the number of disintegrations per second per microcurie;

E is the counting efficiency (counts per disintegration);

V is the sample volume (milliliters);

Y is the fractional radiochemical yield (when applicable);

λ is the radioactive decay constant for the particular radionuclide; and

Δt is the elapsed time between sample collection and counting.

The value of S_b used in the calculation of the LLD for a particular measurement system should be based on the actual observed variance of the instrument background counting rate rather than an unverified theoretically predicted variance.

Since the LLD is a function of sample volume, counting efficiency, radiochemical yield, etc., it may vary for different sampling and analysis procedures. Whenever there is a significant change in the parameters of the measurement system, the LLD should be recalculated.*

*For a more complete discussion of the LLD, see "HASL Procedures Manual," John H. Harley, editor, USERDA, HASL-300 (revised annually) and Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination–Application to Radiochemistry," *Anal. Chem.* 40, 1968, pp. 586-93, and Donn, J. J. and R. L. Wolke, "The Statistical Interpretation of Counting Data from Measurements of Low-Level Radioactivity," *Health Physics*, Vol. 32, 1977, pp. 1-14.

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BIOASSAY AT URANIUM MILLS

A. INTRODUCTION

Section 20.108, "Orders Requiring Furnishing of Bioassay Services," of 10 CFR Part 20, "Standards for Protection Against Radiation," states that, where necessary or desirable in order to aid in determining the extent of an individual's exposure to concentrations of radioactive material, the NRC may incorporate appropriate provisions in any license directing the licensee to make available to the individual appropriate bioassay services. Paragraphs 20.103(a)(1) and 20.103(a)(2) require licensees to limit intakes of materials such as uranium by individuals in restricted areas to the limits specified in Appendix B to 10 CFR Part 20. As specified in paragraph 20.103(a)(3), compliance with these limits must be determined through air sampling and, as appropriate, through bioassays.

Paragraph 20.103(b)(2) permits licensees to make allowance for the use of respiratory protection equipment in determining the magnitude of intake provided such equipment is used as stipulated in paragraphs 20.103(c) through (g). These paragraphs require the licensee to perform bioassays, as appropriate, to evaluate individual exposure and to assess the protection actually provided. Respiratory protection devices do not always offer efficient protection. If a device is defective, is inappropriate for the particular contaminant involved, does not fit the wearer properly, or is carelessly put in place, the wearer may unknowingly receive a significant inhalation exposure. Therefore, if the potential intake was sufficiently large, bioassay procedures should be performed to determine whether such devices were in fact effective.

This guide describes a bioassay program acceptable to the NRC staff for uranium mills (and applicable portions of uranium conversion facilities where the possibility of exposure to yellowcake dust exists), including exposure conditions with and without the use of respiratory protection devices.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 20, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 20 have been cleared under OMB Clearance No. 3150-0014.

B. DISCUSSION

This guide is based on information from the references, public comments received on the versions published in July 1978 and January 1987, data submitted by the milling industry, and an analysis by the staff of the Office of Nuclear Regulatory Research (NUREG-0874, "Internal Dosimetry Model for Applications to Bioassay at Uranium Mills," Ref. 1). Information acquired in the future may result in revisions to this guide; in particular, if bioassay results accumulated over a sufficiently long period of time indicate that workers at uranium mills are being adequately protected from airborne uranium by means of ventilation equipment and effective air sampling programs, the guide may be revised accordingly.

C. REGULATORY POSITION

1. DEFINITIONS

Recent solubility studies have revealed notable differences in the dissolution rates of yellowcake produced under different thermal conditions. For the purpose of this guide, the following distinction is made:

a. Low-fired yellowcake is defined as yellowcake dried at temperatures less than 400° C.

b. High-fired (calcined) yellowcake is defined as yellowcake dried at temperatures of 400° C or more.

Two important areas in a uranium mill where workers are exposed to uranium are defined as follows:[1]

a. Ore-dust areas, under normal conditions, are defined as those areas beginning with the transfer of ore from the ore pad to the crusher through the final thickening stage of the leaching operation.

b. Yellowcake areas are defined as those areas that contain uranium extracted from the ore in a solution form from the ion exchange or solvent extraction stage through final packaging.

2. WORKING CONDITIONS UNDER WHICH BIOASSAYS SHOULD BE PERFORMED

Routine bioassays are considered by the NRC staff to be necessary for 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. Baseline urinalysis bioassays should be performed for each worker prior to initial assignments for such work. Bioassays should be performed if there is any reason to suspect an inhalation exposure exceeding that resulting from exposure to an average yellowcake concentration[2] of 10^{-10} μCi/mL (3.7 x 10^{-6} Bq/mL) for a 40-hour workweek or to an average ore-dust concentration of 10^{-10} μCi/mL (3.7 x 10^{-6} Bq/mL) (based on the concentration of gross alpha activity in air) for a period of 1 calendar quarter; if respiratory protection is used to maintain inhalation exposures below these quantities, bioassay should be performed to verify the effectiveness of the respirators.

3. TYPES OF BIOASSAY

Urinalysis should be performed to monitor exposures to uranium in ore dust as well as in yellowcake as they clear from the kidney before elimination renders them undetectable. In vivo thorax measurements should be made to detect the presence of (1) the more insoluble yellowcake component and (2) uranium in ore dust in the lung when air-sampling results indicate an exposure exceeding that resulting from exposure to such materials at an average concentration of 10^{-10} μCi/mL (3.7 x 10^{-6} Bq/mL) (based on the concentration of gross alpha activity in air) in a period of 1 calendar quarter.

4. FREQUENCY

4.1 General Considerations

The prescribed frequency of urinalysis and in vivo lung measurements is a function of the dissolution rates of the inhaled ore dust or yellowcake in the lungs. Workers in the yellowcake concentrate areas may be exposed to transient levels of airborne uranium that may cause chemical damage to the kidney. Therefore, urinalysis should be performed with sufficient frequency to detect such exposures before elimination from the body renders them undetectable. Guidance on selecting appropriate frequencies is available in NUREG-0874 (Ref. 1). The applicant may use the simplified system of frequencies and action levels presented in this guide.

4.2 Urinalysis for Workers from Yellowcake Areas

Specimens from workers, regardless of whether or not respiratory protection devices were used, should be collected and evaluated at least once per month, and additional special specimens should be collected and evaluated if for any reason an inhalation exposure exceeding that resulting from an exposure to an average yellowcake concentration of 10^{-10} μCi/mL (3.7 x 10^{-6} Bq/mL) for a 40-hour workweek is suspected or air sampling data are not available.

4.3 Urinalysis for Workers from Ore-Dust Areas Exclusively

Specimens from workers, regardless of whether or not respiratory protection devices were used, should be collected and evaluated at least once per month, and additional special specimens should be collected and evaluated if for any reason an inhalation exposure exceeding that resulting from an exposure to an average ore-dust concentration of 10^{-10} μCi/mL (3.7 x 10^{-6} Bq/mL) (based on the concentration of gross alpha activity in air) for a period of 1 calendar quarter is suspected.

4.4 In Vivo Lung (Thorax) Measurements

The lung counting procedure should be capable of detecting (at the lower limit of detection (LLD)) 9 nCi (330 Bq) or less of uranium in the lungs.

When urinalysis results call for in vivo measurements (see Section 5), they should be performed as quickly as possible to determine if corrective measures are required.

When air monitoring or exposure calculations call for in vivo measurements (see Section 3), they should be performed as quickly as practicable but no later than 3 months after such indication.

4.5 Measurement Detection Limits

The measurement sensitivity for urine analyses should be such that the LLD (for a probability of 0.05 for a Type I or a Type II statistical error) is 5 μg of uranium per liter of urine or

[1] If these definitions do not apply to a specific milling operation, the applicant may submit different definitions for consideration.

[2] The 1 x 10^{-10} μCi/mL (3.7 x 10^{-6} Bq/mL) value is not exactly consistent with the 0.2 mg/m^3 concentration limit for soluble uranium in Footnote 4 of Appendix B to 10 CFR Part 20 because of the rounding off of values in Appendix B. Since the 1 x 10^{-10} μCi/mL limit is more restrictive, this value has been used in the calculation of all the action levels (weekly and quarterly) in this guide. For compliance purposes, Footnote 4 to Appendix B sets the weekly limit for soluble uranium compounds, which can be converted to radiological units using the specific activity of natural uranium (6.77 x 10^{-7} Ci/g or 2.5 x 10^4 Bq/g). As now defined in 10 CFR Part 20, the curie of natural uranium differs from the original definition in ICRP-2 (Ref. 2). The present definition of the curie of natural uranium in 10 CFR Part 20 refers to the total activity of all uranium isotopes in the natural uranium mixture. When natural uranium is defined to be 0.711% by weight ^{235}U and the ^{234}U is assumed to be in secular equilibrium with ^{238}U, 1 Ci of natural uranium is composed of 0.489 Ci ^{234}U, 0.0225 Ci ^{235}U, and 0.489 Ci ^{238}U. Actual percentages of ^{235}U may be 0.711 ±0.1%.

less (see Appendix A for an example of the determination of LLD). The LLD for uranium counting in vivo should be 9 nCi (330 Bq) or less of uranium in the lungs.

5. ACTION BASED ON BIOASSAY RESULTS

Bioassay results should be promptly and carefully reviewed by qualified personnel, and appropriate action should be taken if the results exceed preselected levels. The corrective actions to be taken depend on the amount of uranium detected. Action levels and actions in Tables 1 and 2 are acceptable as a basis for a uranium mill bioassay program. Proposals for other action levels and actions from an applicant will be considered on a specific-case basis if accompanied by a description of how the information in NUREG-0874 (Ref. 1) was used to derive those different criteria.

It should be assumed that any confirmed positive urinalysis results are an indication of soluble uranium to which the kidney has been exposed.

5.1 Urinalysis for Workers from High-Fired-Yellowcake Areas

The corrective actions to be taken depend on the amount of uranium detected and are given in Table 1. Figure 1 and other information in NUREG-0874 (Ref. 1) may be used to determine acceptable action levels for a single intake as a function of time for workers from high-fired-yellowcake areas.

5.2 Urinalysis for Workers from Low-Fired-Yellowcake Areas

The corrective actions to be taken depend on the amount of uranium detected and are given in Table 1. Figure 2 and other information in NUREG-0874 (Ref. 1) may be used to obtain acceptable action levels for a single intake as a function of time for workers from low-fired-yellowcake areas.

5.3 Urinalysis for Workers from Ore-Dust Areas Exclusively

The corrective actions to be taken depend on the amount of uranium detected and are given in Table 1. Figure 3 and information in NUREG-0874 (Ref. 1) may be used to obtain acceptable action levels for a single intake as a function of time for workers from ore-dust areas.

5.4 In Vivo

It should be assumed that positive in vivo results indicate the quantity of uranium in relatively insoluble form that has accumulated in the lung. Corrective action should be taken in accordance with Table 2 of this guide.

6. TIME OF SPECIMEN COLLECTION AND AVAILABILITY OF RESULTS

Routine and special urine specimens for analysis of uranium compounds pertinent to mill operations should usually be collected at least 36 hours after the most recent occupancy in the mill. The 36-hour delay is necessary to avoid uranium that is eliminated without uptake in kidney tissues. (However, if compounds are encountered that mainly produce a very short-lived component, Morrow (Ref. 3, p. 6) recommends the use of two action levels: a 1 μg/L Monday morning urinary excretion rate and an exposure-associated urinary output of 100 μg/L during the first 24 hours after the exposure. Tables 1 and 2 would not necessarily be applicable to these results.) Sufficient volume should be collected for four analyses, each of which should be capable of achieving an LLD of 5 μg/L (see Appendix A).

Urinalysis results should be available to the person responsible for conducting the bioassay program within 20 days after specimen collection. If the urinalyses are performed by an outside laboratory, results exceeding 35 μg/L should be reported by telephone.

In vivo results should be available to the person conducting the bioassay program within 20 days after measurement. Results exceeding 16 nCi (590 Bq) should be reported by telephone.

7. PREVENTION OF SPECIMEN CONTAMINATION[3]

7.1 Collection

The specimens should be collected before the worker enters the work area and in an area free of uranium contamination. The collection may occur at an area outside the mill specifically designated to be maintained contamination free. The hands should be carefully washed prior to voiding. Disposable collection containers should be used.

Under unusual circumstances where specimens cannot be collected in this manner, the worker should shower immediately prior to voiding. When a shower is not possible, disposable plastic or rubber gloves should be worn during voiding.

7.2 Laboratory Analysis

All laboratory analyses should be performed in a laboratory essentially free of uranium contamination using containers and equipment essentially free of such contamination. Both on-site and off-site laboratories should maintain the quality control procedures specified in Section 8 of this guide. Use of the laboratory, containers, and equipment for process or environmental samples should be restricted to low-level samples. (Note: The laboratory may be located within the restricted area provided these conditions are met.)

7.3 In Vivo Counting Precautions

For in vivo measurements, employee and clothing contamination are major sources of measurement bias. Care must be taken to minimize these factors. Only new clothing or clothing washed in a facility separate from those used for

[3]The appropriate actions specified in Table 1 should be taken for any result that is confirmed by a second analysis even though specimen contamination is believed to be the cause of the elevated result.

potentially contaminated clothing should be worn during the in vivo measurement. If the in vivo measurement results indicate contamination, the subject should reshower, use clean clothing, and be recounted.

8. QUALITY CONTROL

A quality control program for bioassay measurements should be incorporated in each uranium mill bioassay program. A quality control program consistent with that recommended in the draft standard ANSI/HPS-N13.30 (Ref. 4) will be acceptable. Alternatively, the following specific quality control program for bioassay at uranium mills will be acceptable.

8.1 Urinalysis

Each batch of specimens sent to the laboratory for analysis should be accompanied by at least two control urine specimens. When possible, these control specimens should be taken from individuals who are not and have not been occupationally exposed to uranium; otherwise simulated controls known to contain a uranium concentration less than 1 μg/L may be used. Aliquots of each of these control urine specimens should be taken; one should be a "blank," one should be spiked with uranium to obtain a concentration of 10 to 20 μg/L, and one should be spiked to 40 to 60 μg/L, the actual spiked concentrations being recorded confidentially and not available to the analytical laboratory. When results are received, the licensee should ensure that each reading is corrected for the reading of the corresponding blank, that the net reading of each spiked sample is recorded, and that an 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 for the spiked samples accompanying each batch of urine specimens should be within 30% of the spiked values. Otherwise, the most recent batch of affected samples should be rerun, and steps should be taken to correct the procedures for spiking or the procedures for laboratory analyses, or both.

In order to provide adequate quality control within the analytical laboratory as well as to provide a check on the quality control program of the mill, the analytical laboratory should duplicate the analysis of 10% to 20% of the samples received, including the blanks and spikes received from the mill. In addition, the laboratory should measure its own reagent and urine blanks and spiked standards as appropriate to check its own procedures, provide its own calibration factors, check its LLDs, and evaluate its results for each batch. The laboratory should report the results of its own blank and standard samples along with the other results reported to the mill.

8.2 In Vivo

For in vivo measurements, a quality control program using persons known to have no lung or systemic uranium burdens and phantoms spiked with known amounts of uranium should be used to test the counting system before measurements on each group of employees.

9. USE OF RESPIRATORY PROTECTION DEVICES

Licensees using respiratory protection devices in accordance with paragraph 20.103(c) of 10 CFR Part 20 are to conduct bioassay programs in accordance with paragraph 20.103(c)(2) and NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials" (Ref. 5).

Under certain conditions, bioassay measurements should be performed to ensure the proper evaluation of personnel exposure and to evaluate the actual effectiveness provided by respiratory protection devices. If a worker wearing such a device is subjected for a period of 1 week to an average concentration greater than 10^{-10} μCi/mL (3.7×10^{-6} Bq/mL), as given in Table 1, Column 1, of Appendix B to 10 CFR Part 20 for soluble natural uranium, urinalysis should be performed to test the actual effectiveness of the device. This special bioassay measurement should also be performed if for any reason the magnitude of the exposure that would have occurred if no respiratory protection device had been worn is unknown. The time that the sample for this special measurement was collected should be recorded; it should be consistent with the need to relate bioassay results to kidney exposure (see Section 6).

The appropriate urinalysis or in vivo measurement given in Section 3 of this guide should not be reduced because of the use of respiratory protection devices.

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 in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described in this guide will be used in the evaluation of existing bioassay programs of uranium mill licensees or proposed programs of applicants for such licenses.

Table 1

CORRECTIVE ACTIONS BASED ON MONTHLY URINARY URANIUM RESULTS[a]

Urinary Uranium Concentration	Interpretation	Actions
Less than 15 µg/L	Uranium confinement and air 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 measures 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 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 than 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 of 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 tests 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 (F_1, 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 available literature. The use of alternative values of F_1, F_2, and F_3 specific for a particular operation are acceptable provided (1) details regarding their determination are described and mentioned in employee exposure records (see paragraph 20.401(c)(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 NUREG-0874 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.

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 affected workers or increased uranium confinement control measures. (Normally workers with a lung burden greater than 16 nCi (590 Bq) are not allowed by their employer to resume work in airborne activity areas until the burden is reduced to less than 9 nCi or 330 Bq.) 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 (590 Bq).

[a]The model used in NUREG-0874 (Ref. 1) employs fractional composition values (F_1, 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 available literature. The use of alternative values of F_1, F_2, and F_3 specific for a particular operation are acceptable provided (1) details regarding their determination are described and mentioned in employee exposure records (see paragraph 20.401(c)(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.

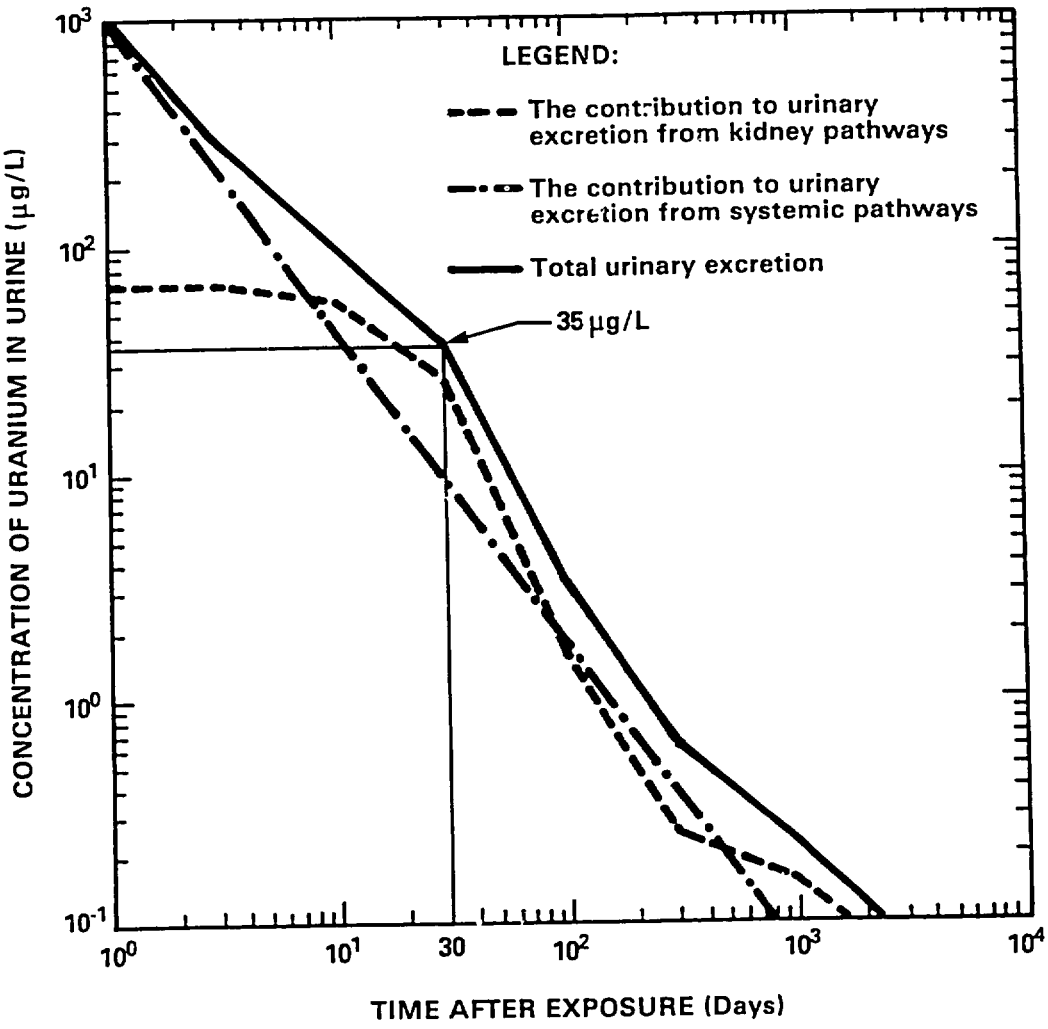

Figure 1 Uranium Concentration in Urine Following Single Exposure to High-Fired Yellowcake (Intake = 160,000 µg U = 1 ALI) (from NUREG-0874, Ref. 1)

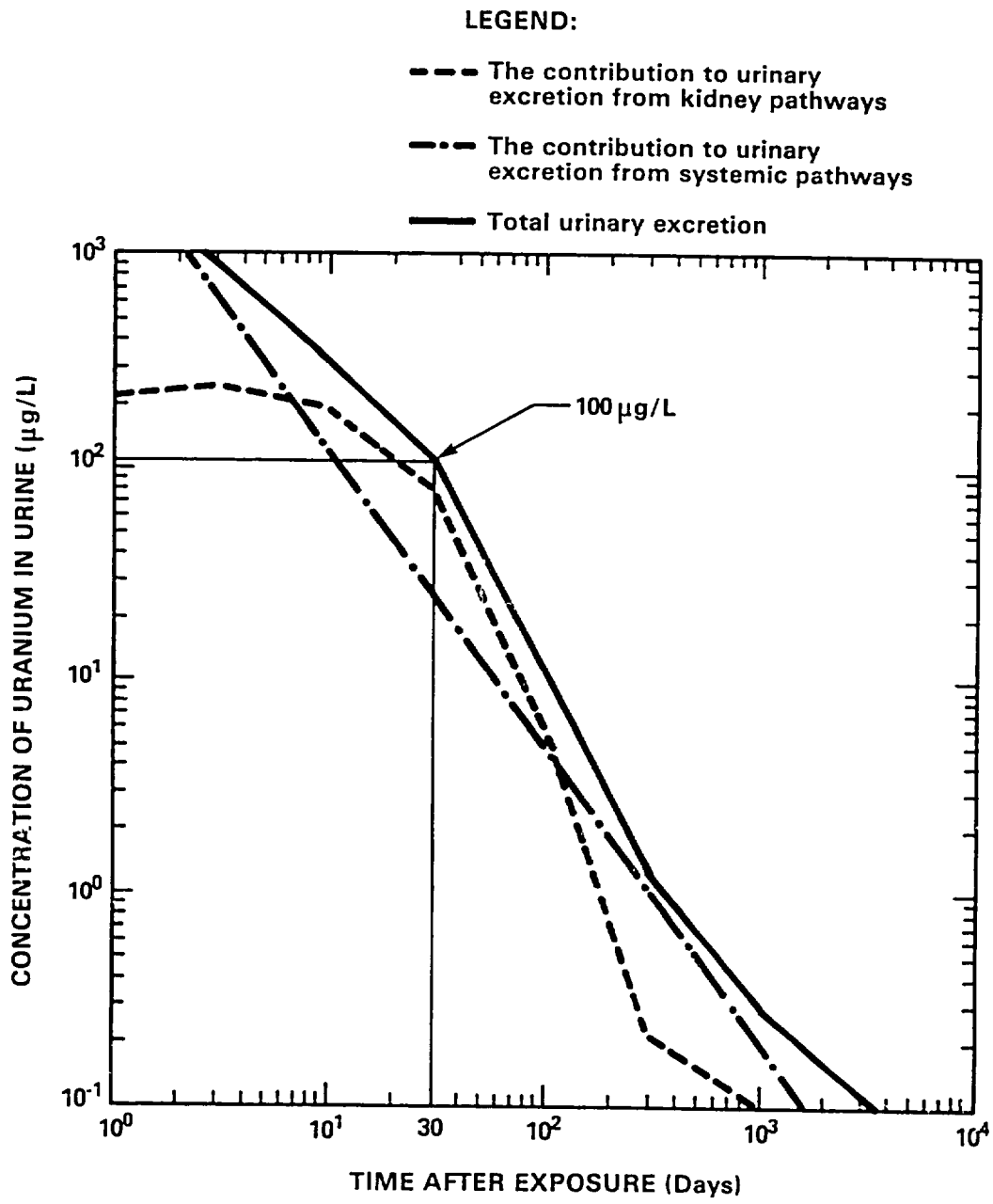

Figure 2 Uranium Concentration in Urine Following Single Exposure to Low-Fired Yellowcake (Intake = 260,000 µg U = 1 ALI) (from NUREG-0874, Ref. 1)

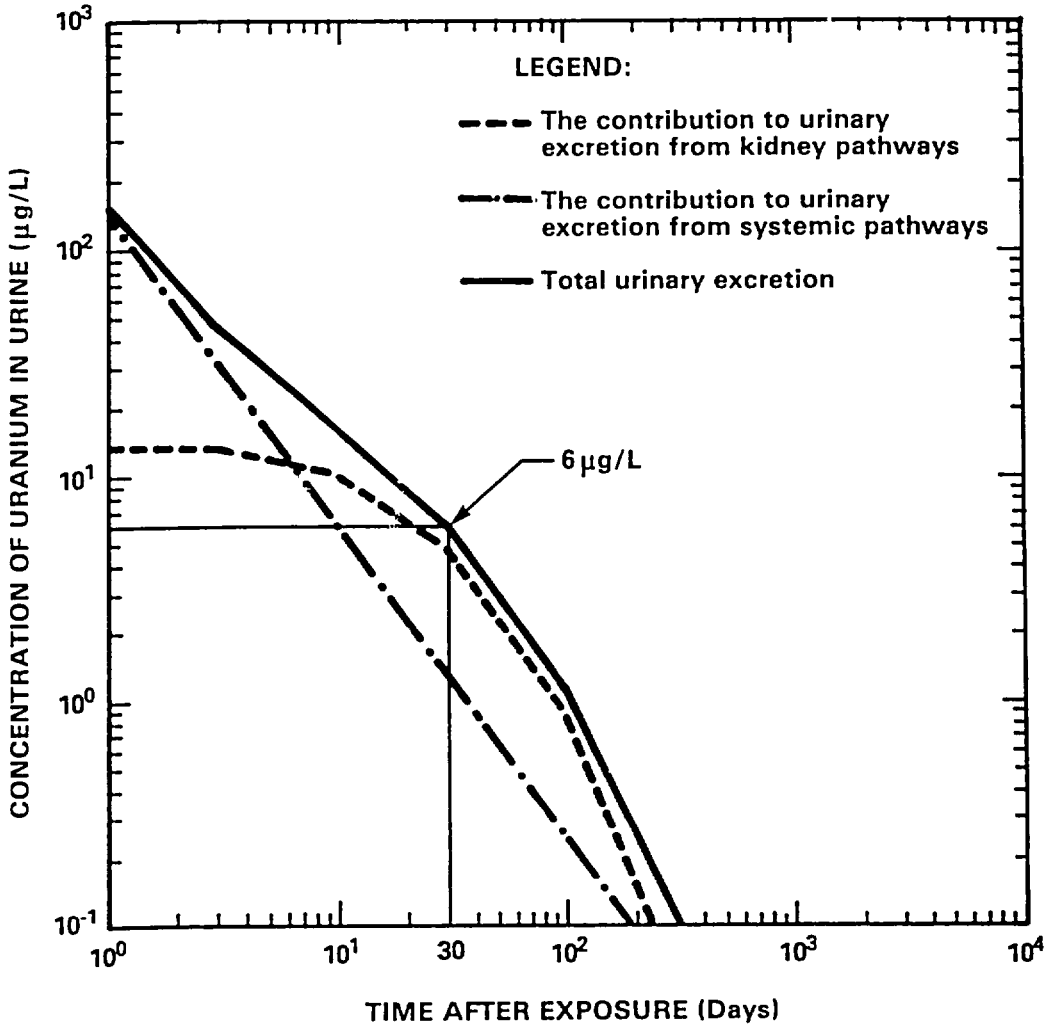

Figure 3 Uranium Concentration in Urine Following Exposure to Ore Dust (from NUREG-0874, Ref. 1)

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*Copies may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082; or the National Technical Information Service, 5285 Port Royal Road, Springfield, VA 22161.

**ICRP publications are available from Pergamon Press, Fairview Park, Elmsford, NY 10523.

***Available from the Health Physics Society, 1340 Old Chain Bridge Road, Suite 300, McLean, VA 22101.

APPENDIX A

LOWER LIMIT OF DETECTION OF URANIUM

For the purposes of this guide, the lower limit of detection (LLD) is defined as the smallest concentration of radioactive material in urine that has a 95% probability (chance) of being detected when measurement procedures are set so that the concentration level at which detection is considered significant produces only a 5% chance of calling a background reading a positive sample.* Radioactive material is then called "detected" when the value obtained from an instrument reading is above the LLD and is thus high enough to permit a conclusion that activity above the system background is determined to be present. Thus, for a fluorometric measurement that may include a radiochemical separation in which the "blank" urines fluctuate with a standard deviation S_b, the LLD corresponds to an activity that is defined as

$$LLD = \frac{4.65S_b}{KEvYe^{-\lambda t}}$$

Where

LLD = the lower limit of detection (μg/L or μCi/L),

S_b = the standard deviation of fluctuations in fluorometer blank measurements or count rate (counts per second) for a specific time of measurement and specific aliquot volume,

K = conversion or calibration factor to convert units of S_b from instrument scale reading units to mass or activity units; units of K may be μA/μg or d/sec-μCi if activity is counted to obtain the final result (this term is omitted if S_b is given in microcuries directly by use of a calibration standard),

E = the counting efficiency (counts per disintegration); it is 1 when a fluorometric standard is measured in the same geometry as the sample,

v = volume (in liters) of aliquot taken from the urine sample and added to the flux in the fusion dish. Note: As long as the concentration of uranium in the aliquot is the same as the concentration in the original urine sample, the volume of the original urine sample does not affect this calculation.

Y = the fractional radiochemical yield or recovery (if applicable),

λ = the decay constant for the particular radionuclide, and

t = the elapsed time between sample collection and counting for correcting for radioactive decay when decay during time t is significant, but decay is negligible during the fluorometric measurement.

*This definition of LLD was chosen to be consistent with the NRC position previously stated in Tables 1 and 3 of Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants." The definition is also used in other regulatory guides, among them 4.14, "Radiological Effluent and Environmental Monitoring at Uranium Mills", 8.14, "Personnel Neutron Dosimeters"; and 8.30, "Health Physics Surveys in Uranium Mills."

EXAMPLE: LLD FOR URANIUM WHEN FLUOROMETRIC ANALYSIS IS USED

This example is worked in terms of micrograms of natural uranium per liter of urine. The LLD could just as well be calculated in terms of microcuries or becquerels of uranium per liter. A conversion factor of 6.77×10^{-7} μCi/μg (0.025 Bq/μg) for natural uranium can be used if the uranium quantity is known in micrograms. The quantity of uranium added to the fusion dish will be determined, and then it will be divided by the volume of urine in the aliquot taken from the total collected sample.

First, determine the standard deviation of the background measurement (blank urine) (which will approximate an estimate of the standard error of the average of a triplicate measurement if calculated as shown below). In this example, urine samples were taken from 12 individuals who worked in areas of the plant where no uranium exposure could have occurred. For each of these "blank" urines, three (triplicate) measurements were made, each measurement consisted of taking 0.2 mL from an individual urine sample and pipetting it into a platinum dish containing a NaF pellet, which was then fused and placed into a fluorometer for measurement. The readings (in microamperes in this case) of the three 0.2 mL aliquots of each individual "blank" urine were then averaged.

The 12 triplicate averages for the blank urines were:

Sample Number, i	Average Fluorometer Readings (X_i) (microamperes)
1	0
2	0.07
3	0.07
4	0.07
5	0
6	0
7	0.13
8	0.13
9	0.17
10	0.10
11	0.13
12	0

The standard deviation S_b (same as an estimate of the standard error of the triplicate average) may be calculated by the following equation (or a computer or calculator programmed for this equation).

$$S_b = \left(\frac{1}{n-1} \sum_{i}^{n} (X_i - \bar{X})^2 \right)^{1/2}$$

n = the number of samples

X_i = the average reading for triplicate i from sample i

$\bar{X}$ = the average of all triplicate averages

For the data above, the standard deviation is:

$S_b = \pm 0.0612$ µA and $\bar{X} = 0.0725$ µA

Convert S_b to micrograms of uranium. On this fluorometer, samples of pure U_3O_8 averaging 0.012 µg added to the fusion dish gave readings in the fluorometer averaging 3.44 µA. The fluorometer will thus have a calibration factor of 287 µA/µg U_3O_8. The U_3O_8 compound is 85% uranium by weight (238 x 3 = 714, 16 x 8 = 128, 714/842 = 0.85). Therefore, the fluorometer will read 338 µA/µg of elemental uranium (287/0.85 = 338)

Now, the standard deviation in micrograms of uranium is calculated:

$$S_b = \frac{0.0612 \text{ µA}}{338 \text{ µA/µg}} = 0.000181 \text{ µg of uranium.}$$

If this is converted to microcuries using the conversion factor given before, then

$$S_b = 0.000181 \text{ µg} \times 6.77 \times 10^{-7} \text{ µCi/µg}$$

$$= 1.23 \times 10^{-10} \text{ µCi } (4.55 \times 10^{-6} \text{ Bq})$$

In the equation for LLD, the counting efficiency will be 1. (The term E is not applicable to a fluorometric analysis.) The aliquot volume of 0.2 mL is used in the LLD equation since the numerical value for each fluorescence reading is related to this volume of urine. Also, for a fluorometric reading compared against a calibration factor, the radiochemical yield is not applicable, and Y should be set equal to 1. The exponential term for radioactive decay, exp(-λt), will also be equal to 1 since the half-life of uranium is so long that the amount of decay between collection and analysis will be negligible. Therefore, the LLDs in mass and activity concentration units become:

$$LLD_m = \frac{4.65 \times 0.000181}{0.0002} = 4.21 \text{ µg/L}$$

$$LLD_a = \frac{4.65 \times 1.23 \times 10^{-10}}{0.0002}$$

$$= 2.86 \times 10^{-6} \text{ µCi/L} \quad (0.106 \text{ Bq})$$

VALUE/IMPACT STATEMENT

A draft value/impact statement was published with Proposed Revision 1 to Regulatory Guide 8.22 (Task OP 013-4) when the draft revised guide was published for public comment in January 1987. No significant changes were necessary, so a separate value/impact statement for the final guide has not been prepared. A copy of the draft value/impact statement is available for inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, DC, under Task OP 013-4.

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
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REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.25

(Draft was issued as DG-8003)

AIR SAMPLING IN THE WORKPLACE

A. INTRODUCTION

Air sampling in the workplace is an acceptable method for meeting certain of the survey and dose assessment requirements of 10 CFR Part 20, "Standards for Protection Against Radiation." For example, 10 CFR 20.1204 allows estimates of worker intakes of radioactive materials based on air sampling and allows adjustments of derived air concentrations (DACs) and annual limits on intake (ALIs) based on the particle size distribution; 10 CFR 20.1501 requires radiation surveys necessary to comply with the regulations and to evaluate potential radiological hazards; 10 CFR 20.1703 requires assessment of airborne radioactive material concentrations when respirators are used; 10 CFR 20.1902 requires posting of airborne radioactivity areas; 10 CFR 20.2103 requires records of radiation surveys; and 10 CFR 20.2202 and 10 CFR 20.2203 require reporting of excessive concentrations of or exposure to airborne radioactive materials.

This guide provides guidance on air sampling in restricted areas (as defined in 10 CFR Part 20) of the workplace. In this guide, the term "air sampling" includes the collection of samples for later analysis as well as real-time monitoring in which samples are analyzed as they are collected. The guide does not cover environmental or effluent sampling or the analysis of samples.

In addition, this guide does not apply to activities conducted under 10 CFR Part 50 at reactor facilities. Although the provisions of 10 CFR Part 20 apply equally to nuclear reactors and to other facilities, the air sampling programs of reactor licensees are well established, and the NRC is satisfied that the quality of air sampling at nuclear reactors is adequate. Therefore, no further guidance on air sampling is needed at this time for reactor licensees.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 20, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 20 have been cleared under OMB Clearance No. 3150-0014.

B. DISCUSSION

Air sampling can be used to determine whether the confinement of radioactive materials is effective, to measure airborne radioactive material concentrations in the workplace, to estimate worker intakes, to determine posting requirements, to determine what protective equipment and measures are appropriate, and to warn of significantly elevated levels of airborne radioactive materials. If bioassay measurements are used to determine worker doses of record, air sampling may be used to determine time of intake and to

determine which workers should have bioassay measurements.

General guidance on air sampling for specific types of facilities is also discussed in several other regulatory guides, including:

- Regulatory Guide 8.21, "Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants"
- Regulatory Guide 8.23, "Radiation Safety Surveys at Medical Institutions"
- Regulatory Guide 8.24, "Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication"
- Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills"

These facility-specific guides cover air sampling in general terms, while this guide discusses air sampling in more depth. Thus, the guides are complementary.

This guide provides recommendations on air sampling to meet the requirements of 10 CFR Part 20. Draft NUREG-1400, "Air Sampling in the Workplace,"[1] provides examples, methods, and techniques that the licensee may find useful for implementing the recommendations in this guide. However, NUREG-1400 does not establish regulatory positions or recommendations and should not be used as a compliance document to establish the adequacy of licensee programs.

C. REGULATORY POSITION

1. EVALUATING THE NEED FOR AIR SAMPLING

The implementation of some sections in 10 CFR Part 20 may require air sampling. This section of the guide provides recommendations on when and what type of air sampling is acceptable to meet the Part 20 requirements.

1.1 When To Evaluate the Need for Air Sampling

As a general rule, any licensee who handles or processes unsealed or loose radioactive materials in quantities that during a year will total more than 10,000 times the ALI for inhalation should evaluate the need for air sampling. (If the same material is used repeatedly, multiply the quantity used by the number of times used.) If more than one radioactive material is used, the need for air sampling should be determined by whether the sum of the quantities of each divided by each respective ALI exceeds 10,000. When quantities handled in a year are less than 10,000 times the ALI, air sampling generally is not needed. (The basis for this value is that experience has shown that worker intakes are unlikely to exceed one one-millionth of the material being handled or processed, as discussed in NUREG-1400.)

1.2 Air Sampling Based on Potential Intakes and Concentrations

The extent of air sampling may be based on estimates of worker intakes and on estimated airborne concentrations of radioactive materials as shown in Table 1. Estimates of potential intakes and concentrations should be based on historical air sampling or bioassay data if these data are available. If the data are not available, potential intakes and concentrations should be estimated. Estimates of intakes and concentrations should be based on a consideration of (1) the quantity of radioactive material being handled, (2) the ALI of the material, (3) the release fraction for the radioactive material based on its physical form and use, (4) the type of confinement for the material, and (5) other factors appropriate for the specific facility. The estimated prospective intake provides only a guide to the appropriate types of air sampling. The radiation safety officer should use professional judgment and experience to perform air sampling appropriate for the specific situation.

1.3 Grab vs. Continuous Air Sampling

Air sampling may be continuous during work hours or intermittent (grab samples taken during part of the work). When continuous sampling during the work day is performed for continuous processes, a weekly sample exchange period is generally acceptable (except for very short-lived radionuclides). Longer sample exchange periods may be appropriate if airborne radioactive material concentrations and nuisance dust concentrations are both relatively low. When grab sampling is performed for continuous processes, a weekly sampling frequency is generally acceptable; however, monthly or quarterly sampling may be acceptable for areas in which concentrations of airborne radioactive material are expected to average below a few percent of the DAC. Grab sampling would also be appropriate when operations are conducted on an intermittent basis.

1.4 Air Sampling When Respiratory Protective Equipment Is Used

Air sampling is required by 10 CFR 20.1703(a)(3)(i) to evaluate airborne hazards whenever respiratory protective equipment is used to limit intakes pursuant to 10 CFR 20.1702. Air samplers that are located to determine worker intake are

[1]Single copies of draft NUREG-1400 are available free, to the extent of the supply. Submit a written request to the Office of Administration, Distribution and Mail Services Section, U.S Nuclear Regulatory Commission, Washington, DC 20555. A final version of NUREG-1400 is being developed and should be published in 1993.

Table 1
Air Sampling Recommendations Based on Estimated Intakes and Airborne Concentrations

Worker's estimated annual intake as a fraction of ALI	Estimated airborne concentrations as a fraction of DAC	Air sampling recommendations
< 0.1	< 0.01	Air sampling is generally not necessary. However, monthly or quarterly grab samples or some other measurement may be appropriate to confirm that airborne levels are indeed low.
	> 0.01	Some air sampling is appropriate. Intermittent or grab samples are appropriate near the lower end of the range. Continuous sampling is appropriate if concentrations are likely to exceed 0.1 DAC averaged over 40 hours or longer.
> 0.1	< 0.3	Monitoring of intake by air sampling or bioassay is required by 10 CFR 20.1502(b).
	> 0.3	A demonstration that the air samples are representative of the breathing zone air is appropriate if (1) intakes of record will be based on air sampling and (2) concentrations are likely to exceed 0.3 DAC averaged over 40 hours (i.e., intake more than 12 DAC-hours in a week).
Any annual intake	> 1	Air samples should be analyzed before work resumes the next day when potential intakes may exceed 40 DAC-hours in 1 week. When work is done in shifts, results should be available before the next shift ends. (Credit may be taken for protection factors if a respiratory protection program is in place.)
	> 5	Continuous air monitoring should be provided if there is a potential for intakes to exceed 40 DAC-hours in 1 day. (Credit may be taken for protection factors if a respiratory protection program is in place.)

acceptable for this purpose. If the worker's job activity will be the main source of airborne radioactive material, the sampling should be done during the activity, not prior to the activity.

1.5 Prompt Analysis of Certain Samples

In situations in which there is a potential for intakes to exceed 40 DAC-hours in a week, air samples should be analyzed promptly on a daily basis. (In evaluating the need for prompt analysis, credit may be taken for respirator protection factors if a respiratory protection program is in place.) Sample results should be available before work resumes the following day. When work is done in shifts, results should be available before the next shift ends, preferably during the first half of the next shift. For special or nonroutine operations, an attempt should be made to have analysis results available within one hour.

1.6 Continuous Air Monitoring

In situations in which there is a potential for accidents to cause intakes exceeding 40 DAC-hours in a day, continuous air monitoring should be done. When continuous air monitors with automatic alarms are used, the alarm set points should be set as low as practical for the work being conducted without causing excessive false alarms (e.g., more than once per quarter).If continuous air monitors with automatic alarms are used, check sources should be used weekly to check that the monitor responds and causes an alarm. Continuous check sources may also be used, provided there is no interference with the radionuclide of interest. If the response is not within

± 20 percent of the normal response, the monitor should be repaired or recalibrated.

1.7 Establishing Airborne Radioactivity Areas

Air sampling with samplers located to determine worker intake may be used to determine whether an area is an airborne radioactivity area. Any room, enclosure, or area must be posted as an airborne radioactivity area if (1) concentrations of airborne radioactive materials are in excess of the DAC or (2) a worker in the area would be exposed to more than 12 DAC-hours in a week (10 CFR 20.1902 and 20.1003). To determine whether the concentration exceeds the DAC over the short term, the sample collection time should not exceed 1 hour. Shorter sample collection times may be used if desired, but they are not required.

Areas should not be posted as airborne radioactivity areas on the basis of unlikely accidents that might cause the DAC to be exceeded. An airborne radioactivity area should be established based on the radioactivity levels normally encountered or on levels that can reasonably be expected to occur when work is being performed.

1.8 Air Sampling vs. Bioassay for Determining Intakes

If sufficient data to determine a worker's intake are available from both air sampling and bioassay measurements and the results are significantly different, the licensee should base the worker's intake estimate on the data considered by the radiation protection staff to be the most accurate.

1.9 Substitutes for Air Sampling

If experience indicates that worker intakes are generally low, it may be acceptable to substitute other techniques in place of air sampling. For example, when working with tritium, iodine, or other materials that are easily and effectively detected by bioassay, it could be appropriate to eliminate all air sampling and rely completely on bioassays to measure intakes and verify confinement.

2. LOCATION OF AIR SAMPLERS

Concentrations of airborne radioactive materials in a room are generally not uniform. Concentrations usually vary greatly from one location to another, sometimes by orders of magnitude even for locations that are relatively close. Therefore, the location of air samplers is important because inappropriately placed samplers can give misleading results.

This section applies only to fixed-location and portable samplers. It does not apply to personal (lapel) samplers.

2.1 Purpose of the Measurement

Before selecting a sampling location, the licensee should decide on the purpose of the measurement. Examples of purposes are (1) estimating worker intakes, (2) verifying that the confinement of radioactive materials is effective, (3) providing warning of abnormally high concentrations, (4) determining whether there is any leakage of radioactive materials from a sealed confinement system, and (5) determining whether an airborne radioactivity area exists.

2.2 Determination of Airflow Patterns

Airflow patterns should be determined in order to locate air samplers appropriately. The locations of ventilation air inlets and exhausts and of sources of airborne radioactive materials should be noted in order to determine the predominant airflow patterns and likely radioactive material transport routes. When sampling air in rooms with complex airflow patterns, it may be useful to use smoke tubes or neutrally buoyant markers to determine airflow patterns.

When sampling air in an airborne radioactivity area to determine the intakes of workers whose intake must be monitored under 10 CFR 20.1502(b), smoke tubes or neutrally buoyant markers should be used to determine airflow patterns from the source to the worker's breathing zone. In some instances, the use of larger smoke sources or neutrally buoyant marker sources to observe airflow patterns is desirable. However, observations of airflow patterns should be omitted in areas of high external radiation exposure if making the observations would result in total worker doses (internal plus external) that are not as low as is reasonably achievable.

The airflow pattern determinations should be repeated if there are changes at the facility, including changes in locations of the individual work locations and seasonal variations that might change airflow patterns, or if there is a reason to suspect problems. The radiation protection staff should be aware of facility characteristics, operations, and changes that might change airflow patterns. In addition, the location of at least 10 percent of the fixed-location samplers should be evaluated annually to confirm that their locations are still appropriate.

2.3 Selecting Sampler Locations

Air samples should be collected in airflow pathways downstream of sources of airborne radioactive material.

When the purpose of the sample is to verify the effectiveness of confinement or to provide warning of elevated concentrations, the sampling point should be located in the airflow pathway near the release point. These samplers do not have to be placed near the worker's breathing zone, and thus concentrations

might be considerably different from the concentrations in the breathing zone. If the room has several widely spaced sources of airborne radioactive material, more than one sampling point may be needed.

When the purpose of sampling is to determine worker intakes, each frequently occupied work location should have its own sampler. The air samplers should be placed as close to the breathing zone of the worker as practical without interfering with the work or the worker. In addition, air flow patterns in the area should be considered in placing samplers so that the sampler is likely to be in the airflow downstream of the source and prior to or coincident with the location of the worker. An estimate should be made of the time the worker spends at the work location (unless personal air samplers are being used).

For hoods, glove boxes, and other similar enclosures used to contain radioactive material, air samplers may be installed slightly above head height and in front of the worker or they may be installed on the front face of the enclosure.

Normally, air samplers intended to measure workplace concentrations should not be located in or near exhaust ducts, because concentrations there will usually be diluted compared to concentrations in work areas. However, samplers may be located in ducts if their purpose is to detect leakage from systems that do not leak during normal operation and if quantitative measurements of workplace airborne concentrations are not needed.

3. DEMONSTRATION THAT AIR SAMPLING IS REPRESENTATIVE OF INHALED AIR

Section 20.1502(b) of 10 CFR Part 20 requires monitoring of the intake of any worker whose intake is likely to exceed 0.1 ALI. Section 20.1204 allows the use of air sampling, bioassay, or a combination of both to determine a worker's intake.

3.1 Need To Demonstrate that Air Sampling Is Representative of Breathing Zone Air

It should be demonstrated that the air sampled is representative of breathing zone air if all four of the following conditions are met: (1) monitoring of intake is required by 10 CFR 20.1502(b) because annual intake is likely to exceed 0.1 ALI, (2) the intake of record will be based on air sampling rather than bioassay, and (3) the exposure will occur in an airborne radioactivity area where airborne concentrations are likely to exceed 12 DAC-hours in a week, and (4) lapel samplers or samplers located within about 1 foot of the worker's head are not used. (The results from lapel samplers or samplers that are located within about 1 foot of the worker's head may be accepted as representative without further demonstration that the results are representative.)

3.2 Demonstration that Air Sampling Is Representative

Four methods may be used to demonstrate representativeness of the results from samplers that are not located within about 1 foot of the worker's head: (1) comparison with lapel sampler results (for this comparison, lapel samplers may be equipped with cyclones with an efficiency of at least 50 percent for particles with an aerodynamic equivalent diameter of 4 micrometers if the particles sampled are solubility class W or Y),[2] (2) comparison with bioassay results, (3) comparison using multiple measurements near the breathing zone, and (4) comparison with quantitative airflow tests.

Table 2 describes the application of each of the methods and includes acceptance criteria for determining whether sampling results may be considered representative.

3.3 Corrective Actions if Sampling Results Are Not Representative

If the method used to demonstrate representativeness does not show that the sampling results are representative, the licensee should analyze the situation, determine the likely cause of the problem, and fix the problem. The licensee should also correct intake estimates made within the last year and subsequent to the previous demonstration of representativeness. To fix the problem, it may be appropriate to relocate samplers to be more representative, apply correction factors to correct sampling results, switch to lapel sampling, or use bioassay measurements to determine intakes.

4. ADJUSTMENTS TO DERIVED AIR CONCENTRATIONS

NRC regulations in 10 CFR 20.1204(c) permit, upon prior approval of the NRC, the adjustment of DACs to reflect the actual physical and chemical characteristics of airborne radioactive materials.

4.1 Adjusting DACs Based on Measurements of Particle Size

If the licensee elects to request approval to adjust DACs based on measured activity median aerodynamic diameters of airborne particles, the following information should be submitted:

1. The need for the adjustment.
2. The radioactive materials involved and either their chemical form (if the chemical

[2]American Conference of Governmental Industrial Hygienists, *Threshold Limit Values for Chemical Substances and Physical Agents and Biological Exposure Indices*, Notice of Intended Changes: Appendix D—Particle Size Selective Sampling Criteria for Airborne Particulate Matter, 1991. The 4-micrometer criterion is also in the process of being adopted by the International Standards Organization (ISO) and the European Standardization Committee (CEN).

Table 2
Methods To Demonstrate the Representativeness of Air Sampling

Method	Description
1. Comparison with lapel samplers	*Include:* Workers whose annual intakes must be monitored under 10 CFR 20.1502(b) because intakes are likely to exceed 10% of an ALI and whose dose of record will be based primarily on air sampling. *Comparison:* Compare intakes measured by air sampling with intakes measured by lapel samplers for at least 1 week for continuous operations or for several operations for repeated short-duration operations. *Acceptance criteria:* The ratio of the intakes calculated from air sampling divided by the intakes calculated from lapel samplers should exceed 0.7 when averaged for all workers included in the comparison. The ratio for each individual worker should exceed 0.5. (The values of 0.7 and 0.5 were selected so that the accuracy of intakes based on air sampling would be compatible with the accuracy expected of external radiation dosimeters.)
2. Comparison with bioassay results	*Include:* Workers whose annual intakes must be monitored under 10 CFR 20.1502(b) because intakes are likely to exceed 10% of an ALI and whose dose of record will be based primarily on air sampling. *Comparison:* Compare the sum of the intakes determined from air sampling with the sum of the intakes calculated from those bioassay measurements. *Acceptance criteria:* The ratio of the sum of the intakes calculated from air sampling divided by the sum of the intakes calculated from bioassay measurements should exceed 0.7 when averaged for all workers included in the comparison. The ratio for each individual worker should exceed 0.5 for each individual worker.
3. Comparison with multiple samplers	*Include:* Work locations at which airborne concentrations are likely to exceed 0.3 DAC and that are generally occupied by workers whose intakes must be monitored and whose dose of record will be based on air sampling. *Comparison:* Use multiple samplers to take measurements at four or more locations around the worker's head. *Acceptance criteria:* The concentration determined by the fixed-location sampler divided by the concentration averaged for all the multiple samplers should exceed 0.7 for the work location.
4. Comparison with quantitative airflow measurements	*Include:* Work locations at which airborne concentrations are likely to exceed 0.3 DAC that are generally occupied by workers whose intakes must be monitored and whose dose of record will be based on air sampling. *Comparison:* Release a tracer material near the source release point. Measure its concentration with the fixed-location sampler and with another sampler placed closed to the worker's head. *Acceptance criteria:* The concentration measured by fixed-location sampler divided by the concentration of the sampler placed close to the worker's head should exceed 0.7.

compounds are listed in Appendix B of Part 20) or their solubility classes (D, W, or Y). Describe how the chemical forms or solubility classes were determined.

3. A graph of the adjusted DAC vs. activity median aerodynamic diameter.

4. The method by which the activity median aerodynamic diameter will be measured.

5. The locations at which the measurements will be made.

6. The frequency of measurements.

7. Methods or techniques that will be used to average results by location or time.

The following locations and frequency of measurements are acceptable to the NRC. For an initial determination of the adjustment, the licensee should take the average of three measurements of the activity median aerodynamic diameter at or near each work location or process. The licensee should then determine whether the entire area or room can be represented by a single activity median aerodynamic diameter or whether the area or room should be divided into areas with different particle sizes. After the initial determination of median diameter in each area of the workplace has been made, the licensee should reassess the median diameters by making another measurement at approximately one-quarter of the work locations at 6-month intervals, selecting different locations each time. However, if two consecutive reassessments do not show a substantial change in the median diameter, reassessments may be annual. Reassessments should also be done after there have been process changes likely to affect the size distribution of particles. If the activity median aerodynamic diameter has changed, the median diameter for the area should either be reassessed or replaced with a default value of 1 micrometer.

If the licensee elects to adjust the DAC based on the size distribution for short-duration operations, such as special maintenance jobs, at least one measurement should be made each time the job is done. In the event of abnormal or accident conditions, the median diameter for normal operating conditions may be assumed for intake assessments.

4.2 Using Cyclones To Adjust Measured Airborne Concentrations

If the licensee elects to request approval to use cyclones or other particle size discrimination samplers to adjust the measured airborne concentrations, the following information should be submitted:

1. The need for the adjustment.

2. The radioactive materials involved and their chemical form (relative to the chemical forms listed in Appendix B to Part 20) or solubility class (D, W, or Y).

3. A description of how the chemical form or solubility class was determined.

4. The type of cyclone, the type of sampler, the air flow rate, and the collection efficiency of 4 micrometer particles at the flow rate that will be used.

5. A list of locations or worker areas that will be sampled using cyclones.

In general, this method is suitable for solubility class W and Y compounds but not solubility class D compounds. Cyclones should have an efficiency of at least 50 percent for particles with an aerodynamic diameter of 4 micrometers.[2]

4.3 Adjusting DACs for Solubility

NRC regulations in 10 CFR 20.1204(c) permit, upon prior approval of the NRC, the adjustment of the DAC based on chemical characteristics. If the licensee elects to request approval to adjust DACs based on particle solubility in the human body, the following information should be submitted:

1. The need for adjustment.

2. A description of how the solubility of the material was determined.

3. A description of how the adjusted DAC was determined.

4. The number and frequency of measurements. (A frequency of at least annually is recommended.)

5. MEASURING THE VOLUME OF AIR SAMPLED

The accuracy of air sampling measurements and the calibration of air sampling instruments is not explicitly dealt with in Part 20. However, it is implied that measurements required by Part 20 must be suitably accurate. This section of the guide describes acceptable methods to determine the volume of air to be sampled to ensure suitable accuracy.

5.1 Means To Determine Volume of Air Sampled

All air samplers to be used for quantitative measurements should have a means to determine the volume of air sampled. This recommendation applies to fixed-location samplers, portable samplers, and lapel samplers.

5.2 Calibration Frequency and Methods

The licensee should calibrate airflow meters at least annually. Additional calibrations should be

performed after repairs or modifications to the meter or if the meter is believed to have been damaged. The methods described in Section F of "Air Sampling Instruments"[3] to calibrate airflow meters are acceptable to the NRC staff.

5.3 Uncertainty

The uncertainty in the volume of air sampled should be less than 20 percent. The uncertainty, U_v, in percent may be calculated from the equation:

$$U_v = [U_s^2 + U_c^2 + U_t^2]^{1/2}$$

where: U_s = the percent uncertainty in reading the meter scale

U_c = the percent uncertainty in determining the calibration factor

U_t = the percent uncertainty in the measurement of the sampling time.

5.4 Inleakage

Air samplers and associated sampling lines should be checked for leakage of air into the sampling line upstream of the flow measurement device when they are calibrated for volume of air sampled.

5.5 Change in Flow Rate

If the flow rate changes by more than ± 10 percent during collection of a sample, a correction should be made by averaging the initial and the final flow rates.

6. EVALUATION OF SAMPLING RESULTS

6.1 Detecting Changes in Air Concentrations Over Time

For fixed-location sampling whose purpose is to confirm confinement of radioactive materials for routine or repeated operations, the results should either (1) be analyzed for trends (for example, by control charts) to determine whether airborne concentrations are within the normal range and administrative and engineering controls are thus operating properly to maintain occupational doses as low as is reasonably achievable or (2) be compared with administrative action levels that serve as a basis for determining when confinement is satisfactory.

6.2 Efficiency of Collection Media

If the efficiency of the collection media (such as filters) for an air sample is less than 95 percent for the material being collected, the sample result should be corrected to account for radioactive material not collected by the collection media. If penetration of radioactive material into the collection media or self-absorption of radiation by the material collected would reduce the count rate by more than 5 percent, a correction factor should be used.

6.3 Detection Sensitivity

The 10 CFR Part 20 monitoring criteria (i.e., 10 percent of the limit) do not establish required levels of detection sensitivity (lower level of detection, minimum detectable activity, minimum detectable concentration, etc.). For example, lapel samplers may not be able to detect uranium concentrations of 10 percent of the DAC, but lapel samplers are still acceptable for measuring the uranium intake of workers. The monitoring criteria should not be considered requirements on the sensitivity of a particular measurement because when the results of multiple measurements are summed, the sum will have a greater statistical power than the individual measurements. However, to achieve the greater statistical power, the licensee should record all numerical values measured, even values below "minimum detectable amounts" and values that are negative because the measured count rate is below the background. Results should not be recorded as "below MDA" or similar statements.

If the licensee desires to calculate the minimum detectable activity of a single sample (MDA), it may be calculated by use of the following equation:

$$MDA = \frac{2.71 + 3.29[R_b T_s(1 + T_s/T_b)]^{1/2}}{EKT_s}$$

where: R_b = the background count rate

T_s = the sample counting time

T_b = the background (or blank) counting time

E = the filter efficiency

K = a calibration factor to convert counts per minute into activity (e.g., counts per minute per microcurie)

(The derivation of this equation is described in NUREG-1400.)

If the proportion of the total activity of a sample that is due to a specific radionuclide in a mixture is known, the MDA for that radionuclide should be reduced proportionally:

$$MDA_i = A_i/A \times MDA$$

where:

A_i/A = the proportion of the total sample activity from radionuclide i.

[3] 7th Edition, American Conference of Governmental Industrial Hygienists, 1989. Copies are available for purchase from the ACGIH, 6500 Glenway Avenue, Building D-7, Cincinnati, Ohio 45211.

6.4 Deposition of Particulates in Sampling Lines

If sampling lines are used for collecting airborne particulates, the lines should be as short as possible and should be made of a material not subject to significant static charge effects (e.g., grounded metal). However, up to several feet of flexible plastic tubing, such as tygon, may be used to connect the sampling line to the sample collector. The penetration of particles with an aerodynamic equivalent diameter of 10 micrometers should be at least 50 percent. DEPOSITION[4] software is an acceptable means of calculating penetration.

6.5 Annual Review of Air Sampling Measurements

Section 20.1101(c) of Part 20 requires that the licensee periodically (at least annually) review the radiation protection program content and implementation. The review of the air sampling component of the program should determine (1) whether the measurements are accurate and reliable and (2) whether changes should be made to improve the measurements. The review should be done annually and should cover the prior year's activities. The annual review of air sampling measurements may be combined with reviews of other aspects of the radiation protection program.

The annual review should include but not necessarily be limited to:

1. *Purposes and amount of air sampling:* Was the air sampling appropriate for the intended purposes? Was there too much or too little air sampling done?

2. *Location of Sampling:* Were fixed-location air samplers located properly? Were grab samples taken with proper regard to airflow patterns?

3. *Trends:* Do trends in air sampling results and worker intakes indicate that confinement of radioactive materials remains adequate? Were prospective estimates of intake reasonably accurate?

4. *Posting:* Is the posting of airborne radioactivity areas appropriate?

5. *Procedures:* Are written procedures still suitable and up to date?

6. *Adjustment of DACs:* Were DACs adjusted for particle size or solubility? If so, are the original adjustment factors still valid?

7. *Correction factors:* Were correction factors applied to air samples to determine worker intakes? If so, are the correction factors still valid?

8. *False alarms:* Was continuous air monitoring done? If so, did excessive false alarms occur?

9. *Representativeness:* For air sampling done to determine significant intakes, was the representativeness demonstrated to be adequate?

10. *Changes:* Have changes in air sampling procedures or equipment occurred that could affect the quality of the measurements? Have changes in the facility operation or equipment occurred that could affect the quality of air sampling measurements?

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 in those cases in which an applicant proposes acceptable alternative methods for complying with specified portions of the Commission's regulations, the methods described in this guide will be used in the evaluation of applications for new licenses, license renewals, and license amendments and for evaluating compliance with 10 CFR 20.1001–20.2401.

[4]N.K. Anand and A. R. McFarland, "DEPOSITION: Software for Characterizing Aerosol Particle Deposition in Sampling Lines," Draft NUREG/GR-0006, October 1991. Single copies are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555. A final version of NUREG/GR-0006 is being developed. For information on DEPOSITION software contact. Aerosol Technology Laboratory, Department of Mechanical Engineering, Texas A&M University, College Station, TX 77843, Attention: Dr. Andrew R. McFarland. Telephone (409) 845-2204.

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this regulatory guide. The regulatory analysis prepared for 10 CFR Part 20, "Standards for Protection Against Radiation" (56 FR 23360), provides the regulatory basis for this guide and examines the costs 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 at the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC, as an enclosure to Part 20.

U.S. NUCLEAR REGULATORY COMMISSION

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REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.30
(Task OH 710-4)

HEALTH PHYSICS SURVEYS IN URANIUM MILLS

A. INTRODUCTION

Section 40.32, "General Requirements for Issuance of Specific Licenses," of 10 CFR Part 40, "Domestic Licensing of Source Material," states that the Commission will approve an application to operate a uranium mill 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 10 CFR Part 20, "Standards for Protection Against Radiation," of the Commission's regulations deal with the protection of mill workers: §20.201 requires adequate surveys, §20.101 limits worker exposure to external radiation, §20.103 limits exposure to airborne radioactive material in restricted areas, §20.202 requires personnel radiation dosimeters in certain instances, §20.203 requires posting of warning signs and controlling access to areas with high radiation levels, §20.401 requires records of radiation surveys and personnel monitoring reports, and §20.405 requires reports of overexposures.

This guide describes health physics surveys acceptable to the NRC staff for protecting uranium mill workers from radiation and the chemical toxicity of uranium while on the job. The guidance can also be applied, in part, to other types of uranium recovery facilities and portions of conversion facilities since some of the processes used in these facilities are similar to those in uranium mills.

The 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 mill and surveys of equipment and packages leaving the mill.

Any guidance in this document related to information collection activities has been cleared under OMB Clearance No. 3150-0019 and No. 3150-0013.

B. DISCUSSION

Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills," outlines the type of information that applicants for a uranium mill 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," in Regulatory Guide 3.5.

The contents of this guide are based to a significant extent on 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. 1). The NRC is also developing a report on occupational radiological monitoring at uranium mills that will describe how many of the surveys in this guide can be performed properly. That report will be available in late 1983.

The subjects of 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," Regulatory Guide 8.22, "Bioassay at Uranium Mills," and Regulatory Guide 8.31, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Are As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

1. SURVEYS

1.1 Surveys for Airborne Uranium Ore Dust

Surveys for airborne uranium ore dust are necessary (1) to demonstrate compliance with the quarterly intake

limits for workers specified in §20.103(a) of 10 CFR Part 20, (2) to meet the posting requirements for airborne radioactivity areas in §20.203(d), (3) to determine whether precautionary procedures such as process or other engineering controls, increased surveillance, limitation on working times, provision of respiratory protective equipment, or other precautions should be considered to meet §§20.103(b)(1) and (b)(2), and (4) to determine whether exposures to radioactive materials are being maintained as low as is reasonably achievable as stated in §§20.1(c) and 20.103(b)(2).

The concentration applicable to limiting exposure to airborne uranium ore dust in restricted areas is given in paragraph 4 of the Note to Appendix B, "Concentrations in Air and Water Above Natural Background," of Part 20. If gross alpha counting of the air sample is performed, concentration is 1×10^{-10} microcuries (μCi) of alpha activity per milliliter (ml) of air. This concentration applies to the alpha emissions of uranium-238, uranium-235 (negligible), 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 concentration is 5×10^{-11} μCi of uranium per ml of air. In mass units the concentration is 75 micrograms (μg) of natural uranium per cubic meter of air.* 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 1×10^{-10} 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 mill 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 1×10^{-10} μCi of natural uranium per ml of air (or 200 μg of soluble natural uranium/m^3**) may be applicable throughout the plant. To know when uranium ore dust concentrations are sufficiently low to allow use of this limit for natural uranium, paragraph 5 of the Note to Appendix B to Part 20 should be consulted. If uranium ore dust concentrations are below 10% of the applicable concentration value in Appendix B of Part 20 (i.e., below 5×10^{-12} μCi/ml), uranium ore dust may be considered to be not present, and the appropriate value for natural uranium (1×10^{-10} 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 5×10^{-11} μCi/ml applies) or a new concentration value for the mixture, MPC_m, may be calculated using the following equation:

$$MPC_m = \left[\frac{f_{nu}}{MPC_{nu}} + \frac{f_{od}}{MPC_{od}} \right]^{-1}$$

where:

MPC_{nu} = regulatory concentration value for natural uranium

MPC_{od} = regulatory concentration value (in radiometric units) for natural uranium in ore dust

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

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

Since this equation would only be used with the 5×10^{-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, 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 5×10^{-12} μCi of natural uranium per ml of air (or 7.5 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 §20.203(d), 30-minute samples should be taken weekly if workers occupy the area. Outdoor areas such as the ore pad should be sampled 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 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.

*Micrograms of uranium can be converted to microcuries by using the specific activity of natural uranium: 6.77×10^{-7} μCi/μg.

**The primary standard for airborne soluble natural uranium is 200 $\mu g/m^3$. Multiplying that value by 6.77×10^{-7} μCi/μg gives 1.35×10^{-10} μCi/ml. This is rounded down to give the Appendix B concentration of 1×10^{-10} μCi/ml.

During the first year of operation, new mills will need a more extensive air sampling program than operating mills to determine what locations 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).*

Regulatory limits on the intake of ore dust are discussed in Section C.3 of this guide.

1.2 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 excreted or distributed to various body organs. The rate of dissolution for yellowcake appears to depend on its temperature history. Yellowcake dried at low temperature, which is predominantly composed of ammonium diuranate, dissolves more quickly than yellowcake dried at higher temperature; and a relatively large fraction is rapidly transferred to kidney tissues (Refs. 2-4). 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, which 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 primarily due to the large insoluble component. For compliance purposes, yellowcake dried at 400°C and above should be classified as insoluble (Refs. 5 and 6).

Solubility classification is important with respect to compliance with the Commission's weekly intake regulations for soluble uranium. Paragraph 20.103(a)(2), in connection with footnote 4 of Appendix B to Part 20, imposes a weekly intake limit of 0.0065 μCi (9.6 mg) for soluble uranium. If this limit is exceeded during a calendar week, an overexposure has occurred.** A weekly overexposure limit is imposed because hazardous conditions must be corrected quickly where chemical toxicity to the kidney may be involved.

Solubility classification is not an important consideration from the viewpoint of complying with the Commission's quarterly intake limits for natural uranium. Paragraph 20.103(a)(1), footnote 3, requires that every quarterly intake limit be calculated as the product of the Appendix B, Column 1 concentration and the constant 6.3×10^8 ml (which is the assumed number of milliliters of air inhaled by a worker, while on the job, during one calendar quarter). The concentration value for either soluble or insoluble natural uranium is 1×10^{-10} μCi/ml of air. Thus, the quarterly intake limit for any type of yellowcake is 0.063 μCi (approximately 93 mg) of uranium.* If this value is exceeded, an overexposure has occurred.

The regulations for insoluble uranium do not contain overexposure limits based on the weekly intake. However, a weekly control measure is specified in §20.103(b)(2), which is applicable to insoluble natural uranium, such as yellowcake dried at high temperature. It is not a violation of the NRC's regulations if a worker's intake of insoluble uranium exceeds the equivalent of 40 hours at a concentration of 1×10^{-10} μCi/ml in any period of seven consecutive days, for a single time. However, failure to make an evaluation of an occurrence, take appropriate actions to ensure against recurrence, and maintain the required records is a violation of §20.103(b)(2).

Thus, surveys for airborne yellowcake are necessary to demonstrate compliance with the weekly and quarterly intake limits in §§20.103(a)(1) and (a)(2). 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 §20.103(b).

The recommended survey program for yellowcake uses a combination of general air sampling and breathing zone sampling during operations that may involve considerable intake such as those that require a special work permit.

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. The increased duration of surveys in airborne radioactivity areas should be performed to meet the requirement in §20.103(b)(2) for increased surveillance in such areas.

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 10 MPC-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.

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*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.

**In connection with the 0.0065 μCi weekly limit and the 0.063-μCi quarterly limit, note that 0.0065 multiplied by 13 does not yield 0.063, as would normally be expected. The reason is as follows. The 0.0065 μCi weekly limit is derived from the 200-μg/m^3 value specified in footnote 4 of Appendix B. The 0.063-μCi quarterly limit is derived from the 1×10^{-10} μCi/ml value from Column 1, Appendix B. The 1×10^{-10} value contains a roundoff error that essentially accounts for the anomaly.

*1×10^{-10} μCi/ml $\times$ 6.3×10^8 ml/quarter = 0.063 μCi/quarter. 0.063 μCi $\div$ 6.77×10^{-7} μCi/μg = 9.3×10^4 μg = 93 mg.

**MPC is the acronym for maximum permissible concentration.

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 1×10^{-11} μCi/ml (10% of the Part 20, Appendix B concentration). 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.

1.3 Surveys for Radon-222 and Its Daughters

In uranium mills, significant concentrations in air 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* 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 uranium mills, 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 both 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. 7, p. 78, and Ref. 8).

Monthly measurements of radon daughter concentrations should be made where radon daughters routinely exceed 10% of the limit or 0.03 working level (i.e., the radon daughter concentrations are considered to be present according to paragraph 5 of the Note to Appendix B to Part 20). If radon daughter concentrations are normally greater than 0.08 working level (25% of limit) or radon concentrations are above 8×10^{-9} μCi/ml (8 pCi/l), 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 8×10^{-9} μCi/ml (8 pCi/l). 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.

Radon daughter samples should be representative of worker exposures. Samples should be taken near 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 lower limit of detection for radon daughter measurements should be 0.03 working level so that concentrations defined as being present in paragraph 5 of the Note to Appendix B to Part 20 can be detected. Appendix B of this guide shows how to calculate the LLD for a radon daughter measurement. Measured values less than the lower limit of detection, including negative values, should still be recorded on data sheets. The lower limit of detection 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 lower limit of detection.

The modified Kusnetz method for measuring radon daughter working levels is a suitable method for uranium mills. 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. 1). 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 1 and 9. Other acceptable methods discussed in Reference 1 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. Recently, "instant working level" meters have been developed, which have the advantage of speed. These are also acceptable if an LLD of 0.03 working level can be achieved.

1.4 Surveys for External Radiation

Most, but not all, mill workers receive external gamma radiation doses of less than 1 rem per year (Ref. 1). Gamma

*The term "radon" used in this guide means "radon-222."

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. 1). 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 semiannually throughout the mill at locations representative of where workers are exposed in order to allow determination of "radiation area" boundaries in accordance with §20.203(b) and to determine external radiation dosimetry requirements, in accordance with §20.202. At new mills, 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 5 mrem in any hour or a dose in excess of 100 mrem in any 5 consecutive days, the area must be designated a "radiation area," as defined in §20.202(b)(2). For example, if the maximum time any individual worker spends in a room in a 5-day period is 40 hours, the room will be a "radiation area" if the exposure rate exceeds 2.5 mR/hr. Few mills 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 12 inches from the surfaces.* Use of 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 the mill.

To determine the need for personnel monitoring, quarterly radiation exposures expected for each category of plant worker should be calculated from the measured radiation levels and predicted occupancy times. If the calculated quarterly gamma ray dose for any individual worker exceeds 0.31 rem, §20.202 of 10 CFR Part 20 requires that the worker wear a personnel radiation dosimeter (e.g., film badge or TLD). In addition, personnel monitoring should be used for at least a 1-year period to verify the survey results even if predicted levels are below 0.31 rem. When feasible, the personnel monitoring results should be correlated with the gamma survey results as a cross-check on each.

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 if 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 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. 10). Figure 2 shows the beta dose rate from aged yellowcake as a function of distance from the surface (Ref. 10). 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%. Extremity monitoring is required by §20.202(a) for any worker whose hand dose would exceed 4.68 rems in a quarter.

In the case of beta surveys, it is usually acceptable to substitute evaluations of beta doses based on Figures 1 and 2 in place of 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 uranium mill (see, for example, Tables A-11 and A-12 of Reference 11 and Tables 6 and 9 of Reference 12). Workers' beta doses should be estimated from the beta surveys described above rather than from personnel monitoring reports.

1.5 Surveys for Surface Contamination

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 §20.103(a).

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

*See § 20.204(a) 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."

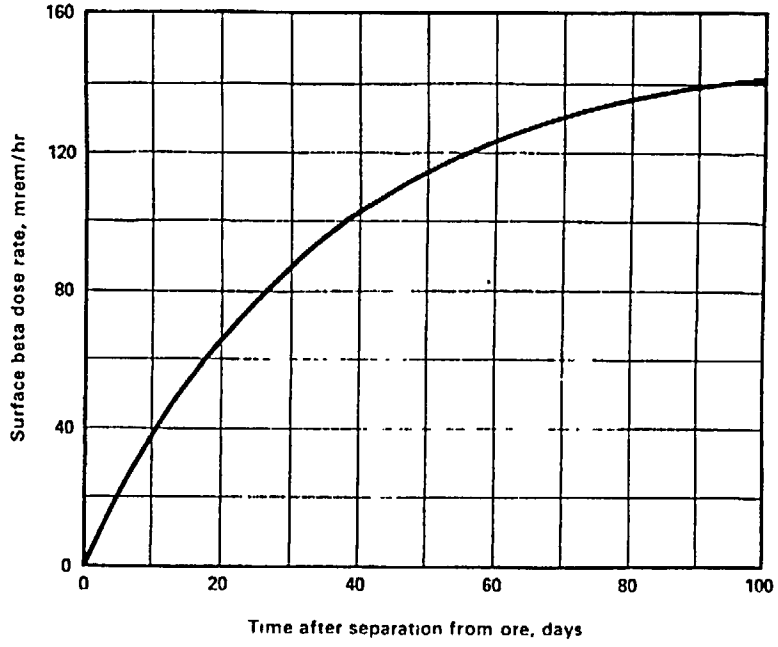

FIGURE 1. BETA DOSE RATE ON THE SURFACE OF YELLOWCAKE

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. 13).

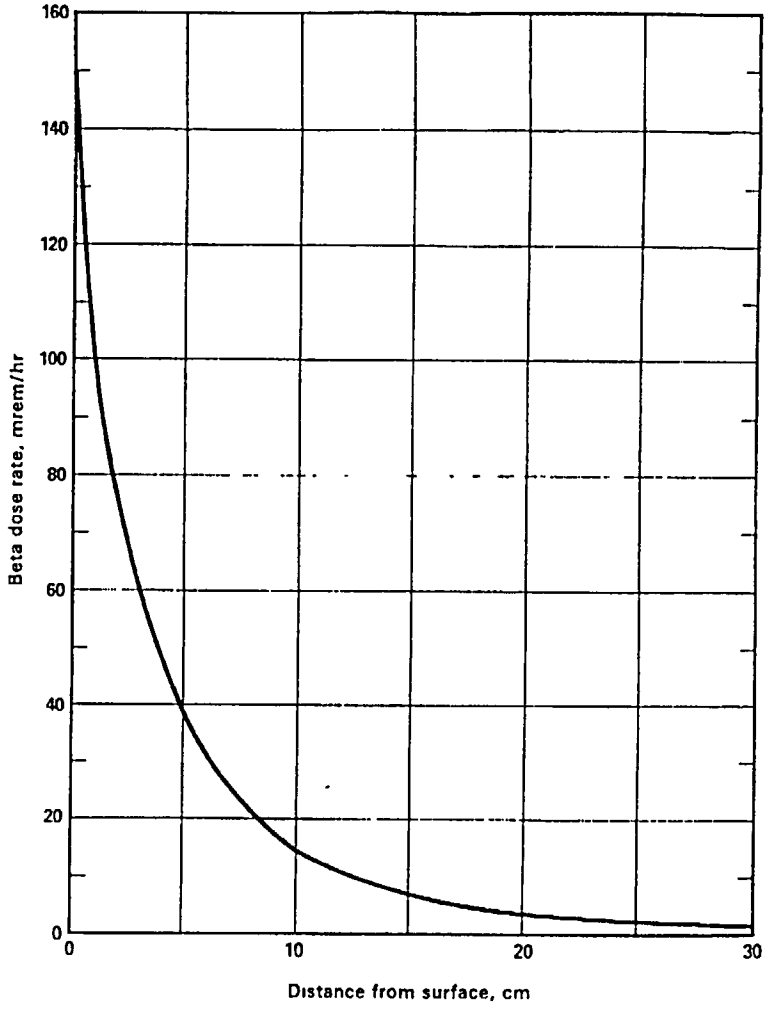

FIGURE 2. BETA DOSE RATE FROM YELLOWCAKE SEPARATED FROM ORE FOR MORE THAN 100 DAYS (from Reference 12)

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 barrelling areas, surface contamination can be a problem because of the concentrated nature of the yellowcake. The International Atomic Energy Agency (IAEA) recommends (Ref. 1) 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. 14), 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 uranium mills. 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. Surfaces painted prior to the implementation date of this guide need not be repainted merely to meet this recommendation. However, when such surfaces are repainted they should be painted in contrasting colors.

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 should be maintained. 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 1.

TABLE 1

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

Average	5,000 dpm alpha per 100 cm^2	Averaged over no more than 1 m^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

Note: The contamination levels 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 very roughly approximate 100 cm^2. However, there is no need to be very precise about the area to be smeared.

* These values are taken from: Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," 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," Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, November 1976. Available in NRC Public Document Room for inspection and copying for a fee.

1.6 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 the mill. Prior to 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 the mill should either be brushed or monitored before leaving the mill. An alpha survey instrument should be available at the exit of the employee change room. 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 the mill.

Limits on acceptable levels of alpha contamination of skin and clothing are those in Table 1, but 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.* Additional showering or washing should be done 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 1, the clothing should be laundered before leaving the site. If the soles of shoes exceed the value in Table 1, the shoes should be brushed or scrubbed until they are below the limit.

1.7 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 1 are recommended.** 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 for workers and visitors is outside the restricted area.

1.8 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 some, but not all barrels, should be surveyed to determine the effectiveness of the wash downs.

Contamination on packages should not exceed Department of Transportation limits in 49 CFR §173.397. 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 §§173.397(a) and (a)(1)) or 22,000 dpm/100 cm^2 if an exclusive-use vehicle is to be used (49 CFR §§173.397(b) and (a)(1)). Packages having higher contamination levels should be cleaned and resurveyed prior to shipment. Visible yellowcake should be cleaned off.

1.9 Ventilation Surveys

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

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

1.10 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. 16, Section 9.6).

1.11 Summary of Survey Frequencies

Table 2 summarizes the survey frequencies given in this guide.

2. INTAKE AND EXPOSURE CALCULATIONS

2.1 Uranium Ore Dust and Yellowcake

In 10 CFR Part 20, §20.103(a)(1) establishes a quarterly intake limit on airborne uranium in yellowcake and in ore dust, §20.103(a)(2) establishes a weekly intake limit on airborne soluble uranium (low-temperature dried yellowcake), and §20.103(b)(2) establishes a weekly control measure for ore dust and airborne insoluble uranium (high-temperature dried yellowcake).

This guide presents two equivalent methods for calculating worker intake. The first method expresses intake in terms of microcuries or micrograms. The second method expresses intake in terms of MPC-hours of exposure. The methods are equivalent and either may be used.

Method 1: The Intake Method (Microcuries or Micrograms)

The intake of uranium ore dust or yellowcake during the weekly or quarterly period being evaluated may be estimated using the following equation:

$$I_u = b \sum_{i=1}^{n} \frac{X_i t_i}{PF}$$

where:

I_u = uranium intake, μg or μCi

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*This value is comparable to the limit of 10^{-5} μCi/cm^2 or 2,200 dpm per 100 cm^2, recommended by the International Atomic Energy Agency on page 15 of Reference 1 and the United Kingdom Atomic Energy Authority in Reference 15.

**See Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," 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," Division of Fuel Cycle and Material Safety USNRC, Washington, D.C. 20555, November 1976. Available in NRC Public Document Room for inspection and copying for a fee.

TABLE 2

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	5×10^{-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	1×10^{-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	Throughout mill Radiation areas	Semiannually Quarterly	0.1 mR/hr
Beta	Where workers are in close contact with yellowcake	Survey by operation done once plus whenever procedures change	1 mrad/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	Quarterly	500 dpm alpha per 100 cm^2
	Yellowcake workers who do not shower	Each day before leaving	
7. Equipment to be released	Equipment to be released that may be contaminated	Once before release	500 dpm alpha per 100 cm^2
8. Packages 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

t_i = time of exposure to average concentration X_i (hr)

X_i = average concentration of uranium in breathing zone air during the time t_i, $\mu g/m^3$ or $\mu Ci/m^3$

b = breathing rate, 1.2 m^3/hr

PF = the respirator protection factor, if applicable*

n = the number of exposure periods during the week or quarter

Method 2: The MPC-hour Method

The intake of uranium ore dust or yellowcake during the weekly or quarterly period being evaluated may be estimated using the following equation:

$$I_u = \sum_{i=1}^{n} \frac{X_i t_i}{MPC \times PF}$$

where:

I_u = uranium intake, MPC-hours

t_i = time that the worker is exposed to concentrations X_i (hr)

X_i = average concentration of uranium in the air near the worker's breathing zone, $\mu Ci/ml$

MPC = the concentration value for the radioactive material from Appendix B of Part 20, $\mu Ci/ml$

X_i/MPC = the number of MPCs

PF = the respirator protection factor, if applicable*

n = the number of exposure periods during the week or quarter

2.2 Radon Daughters

In 10 CFR Part 20, §20.103(a)(1) establishes an annual limit on the intake of radon daughters. Radon daughter intake may be estimated using either of the two following equations:

Method 1: The Intake Method (Working-Level Months)

$$I_r = \frac{1}{170} \sum_{i=1}^{n} \frac{W_i t_i}{PF}$$

where:

I_r = radon daughter intake, working-level months

t_i = time of exposure to W_i (hr)

170 = number of hours in a working month

W_i = average number of working levels in breathing zone air during the time (t_i)

PF = the respirator protection factor, if applicable *

n = the number of exposure periods during the year

Method 2: The MPC-hour Method

$$I_r = \sum_{i=1}^{n} \frac{W_i t_i}{MPC \times PF}$$

where:

I_r = radon daughter intake, MPC-hours

t_i = time of exposure to W_i (hr)

W_i = average number of working levels in breathing zone air during the time (t_i)

MPC = the Appendix B (Part 20) concentration value for radon daughters (0.33 working levels)

W_i/MPC = the number of MPCs of radon daughters

PF = respirator protection factor, if applicable*

n = the number of exposure periods during the year

The values of t_i may be determined by actual timing and recording for each exposure, or t_i values may be derived from a time study of worker occupancy in the various mill areas. Such studies should be updated annually and after any significant change in mill equipment, procedures, or job functions. When nonroutine maintenance or cleanup operations are performed, accurate time records should be kept, and the results of special area or breathing zone samples taken over this period should be added to the calculations of employee exposures.

3. REPORTS OF OVEREXPOSURES TO AIRBORNE MATERIALS

Any overexposure of a person to airborne radioactivity must be reported to the NRC. Section 20.405 requires

*If the licensee's respiratory protection program is being conducted in conformance with Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," and the appropriate NRC Regional Office has been notified that the licensee plans to use respirators, the prescribed protection factor (PF) may be used in the calculation of I_u and I_r.

overexposure reports to the appropriate NRC regional office if the intake of uranium ore dust or yellowcake exceeds the quantities specified in §20.103 or if the exposure to radon daughters exceeds the working-level values specified in footnote 3 to Appendix B to 10 CFR Part 20. Many uranium mill workers are exposed to a combination of these materials. In such cases, Appendix B to 10 CFR Part 20 specifies the method for determining whether NRC exposure limits have been exceeded. Overexposure reports are also required for combined exposures that exceed NRC limits.

A listing of exposure limits follows:

1. Soluble uranium, weekly determination.

If during a period of 1 calendar week a worker has an intake of soluble uranium (yellowcake dried at a temperature below 400°C) exceeding 9.6 mg, an overexposure has occurred.*

2. Airborne radioactivity, quarterly determination.

For a worker exposed to uranium ore dust, yellowcake, or both, it is necessary to determine whether an overexposure has occurred during the quarter. Either one of the two following methods may be used for this purpose.

Method 1: The Intake Method (Microcuries or Milligrams). The ore dust uranium intake in microcuries (or milligrams) is divided by 0.03 μCi** (or 47 mg) to calculate the fraction of the limit that has been taken in. The yellowcake intake for the quarter in microcuries (or milligrams) is divided by 0.063 μCi (or 93 mg). Add the two fractions. If the sum exceeds unity, an overexposure has occurred.

Method 2: MPC-hour Method. Add the exposures, in MPC-hours, of uranium ore dust and yellowcake. If the total for any worker exceeds 520 MPC-hours*** an overexposure has occurred.

3. Radon daughters, annual determination.

Exposure to radon daughters is limited on an annual basis. If the intake method is used, an intake exceeding 4 working-level months in a calendar year is an overexposure. If the MPC-hour method is used, an exposure exceeding 2080 MPC-hours in a calendar year is an overexposure.

4. ACTION LEVELS

4.1 The 40-Hour Control Measure

The 40-hour control measure, specified in §20.103(b)(2), is an action level of concern to the uranium mill operator. If during a week a worker is subjected to an intake exceeding 40 MPC-hours, §20.103(b)(2) requires that the cause must be determined, corrective action to prevent another such occurrence must be taken, and a record of the corrective action must be maintained.

Use either of the two methods in Section C.2 of this guide to calculate a worker's weekly intake. If the microcurie (or milligram) method is used, a weekly intake of uranium ore dust plus yellowcake exceeding 1/13 of the quarterly limit given in Section C.3 of this guide exceeds the 40-hour control measure. Do not include radon daughters because these are considered only on an annual basis. If the sum of the two fractions for the weekly intake exceeds 1/13, the 40-hour control measure has been exceeded.

If the MPC-hour method is used, the MPC-hours from ore dust and yellowcake are added. If the sum exceeds 40 MPC-hours, the 40-hour control measure has been exceeded.

4.2 Administrative Action Levels

In addition, 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.

1. *Uranium ore 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 the 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.

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 MPC-hours. If any action level is exceeded, the RSO should find out why and should take corrective action if appropriate.

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. *Time-weighted exposure to airborne radioactivity.* If any worker's time-weighted exposure, calculated by either of the two methods in Section C.2 of this guide, exceeds 25% of the exposure limits, as listed in Section C.3 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 400°C), a quarterly basis for uranium ore dust and yellowcake combined, and an annual basis for radon daughters.

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* 40 hours at a concentration of 0.2 mg/m^3 and a breathing rate of 1.2 m^3/hr.

** If total alpha activity is measured instead of uranium activity, divide by 0.06 μCi.

*** 40 hours/week x 13 weeks = 520 hours.

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 find out the cause of the elevation and should take corrective action, if appropriate.

6. *Dosimeter results.* The RSO should establish action levels for the monthly dosimeter results. If the action level for any person is exceeded, the RSO should find out the cause and take corrective action, if appropriate.

7. *Contamination on skin and clothing.* If alpha contamination of the skin or clothing of workers leaving the mill 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.

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

5. ESTABLISHMENT OF "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 §20.203(d), if (1) at any time the uranium concentration exceeds 0.5×10^{-10} 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 Section C.1.3 of this guide), their concentrations divided by the appropriate Table 1 Appendix B value 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.

6. 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 §20.203.

The mill and tailings area should be fenced to restrict access, and the fence should be posted with "Caution, Radioactive Material" signs as required in §20.203(e)(2). If the fence and all entrances are posted and in addition contain the words "Any area within this mill may contain radioactive material," the entire area is posted adequately to meet the requirement in §20.203(e)(2). Additional posting of each room with "Radioactive Material" signs is not necessary.

"Radiation Areas" and "Airborne Radioactivity Areas" must be posted in accordance with §§20.203(b) and (d). 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 the mill with such signs defeats this purpose.

7. 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. 17, 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. 17, 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 Regulatory Guide 10.6, Appendix A).

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 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 calibration 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 calibration 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 mrem/hr, as shown in Figure 2. To determine the calibration 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 calibration factor that will make the difference between the open-window and closed-window readings equal to 75 mrem/hr, as shown in Figure 2. A sample calculation is shown in Appendix C.

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 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."

The fluorometric analysis system 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 detect contamination on skin and equipment should receive a constancy check each week and a calibration annually.

8. 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 to reduce the beta dose rate 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 1 of this guide.

9. 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," 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 in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, applications for new uranium mills and renewal applications submitted after July 1, 1983, should follow the recommendations in this guide.

APPENDIX A

DERIVATION OF EQUATION FOR MPC_m

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

$$\frac{C_a}{MPC_a} + \frac{C_b}{MPC_b} + \frac{C_c}{MPC_c} \leq 1$$

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 MPC for the mixture

$$C_{nu} + C_{od} = MPC_m$$

$$\frac{C_{nu} + C_{od}}{MPC_m} = 1$$

the equality in the first equation will apply.

Therefore:

$$\frac{C_{nu}}{MPC_{nu}} + \frac{C_{od}}{MPC_{od}} = \frac{C_{nu} + C_{od}}{MPC_m}$$

Solve for MPC_m

$$MPC_m = \frac{C_{nu} + C_{od}}{\frac{C_{nu}}{MPC_{nu}} + \frac{C_{od}}{MPC_{od}}}.$$

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

$$MPC_m = \frac{1}{\frac{C_{nu}}{(C_{nu} + C_{od})(MPC_{nu})} + \frac{C_{od}}{(C_{nu} + C_{od})(MPC_{od})}}$$

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:

$$MPC_m = \left[\frac{f_{nu}}{MPC_{nu}} + \frac{f_{od}}{MPC_{od}}\right]^{-1}$$

APPENDIX B

LOWER LIMIT OF DETECTION

For the purposes 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.* 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):

$$LLD = \frac{4.66 S_b}{3.7 \times 10^4 E V Y e^{-\lambda t}}$$

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 S_b 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 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 readings for 10 clean filter papers are as follows:

Sample number	Fluorometric reading (X_i) (microamperes)
1	0.062
2	0.072
3	0.050
4	0.050
5	0.050
6	0.040
7	0.086
8	0.088
9	0.088
10	0.018

Calculate the standard deviation S_b by the equation (or by pocket calculator):

$$S_b^2 = \frac{1}{n-1} \sum_{i=1}^{n} (X_i - \bar{X})^2$$

where:

n = the number of samples

X_i = the reading for sample i

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

For the data above, the standard deviation is:

$S_b = 0.023$ μA

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:

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

*This definition of LLD was chosen to be consistent with the NRC position previously stated in Tables 1 and 3 of Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants." The basis for the definition is given in References 18 and 19 of this guide. The definition is also used in other regulatory guides, among them 4.14, "Radiological Effluent and Environmental Monitoring at Uranium Mills," and 8.14, "Personnel Neutron Dosimeters."

= 0.0029 μg 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

$$LLD = \frac{4.66 \times 1.97 \times 10^{-9}\ \mu Ci}{300{,}000\ ml}$$

$= 3 \times 10^{-14}$ μCi of uranium/ml of air

This LLD is about 150 times more sensitive than recommended in the guide as an acceptable lower limit of detection.

Example: LLD for radon daughters 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 S_b 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:

$$LLD = \frac{4.66 \times 0.84\ counts/min}{0.27\ counts/dis \times 25\ liters \times 1 \times 0.21}$$

= 2.8 dpm/liter

To convert this LLD to working levels (WL), divide by the factor from Figure 1 in ANSI N13.8-1973 (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 this guide.

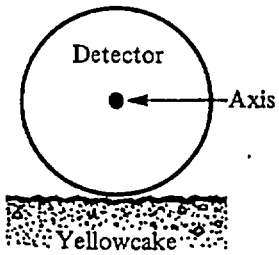
APPENDIX C

BETA CALIBRATION OF SURVEY INSTRUMENT

Here is an example for calibrating 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 calibration factor CF_{sur} can be calculated from the equation below:



Observed dose rate x CF = actual dose rate

$25 \text{ mR/hr} \times CF_{sur} = 150 \text{ mrem/hr}$

$$CF_{sur} = \frac{150 \text{ mrem/hr}}{25 \text{ mR/hr}}$$

$CF_{sur} = 6$ 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 calibration factor CF_{2cm} can be calculated:

$$CF_{2cm} = \frac{75 \text{ mrem/hr}}{20 \text{ mR/hr}}$$

$CF_{2cm} = 3.75$ 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.

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VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Applicants for a uranium milling license must submit a license application containing the information specified in Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills." The purpose of this proposed action is to describe health physics surveys that are acceptable to the NRC staff to protect workers. Information about health physics surveys is covered under Section C.5, "Operations," in Regulatory Guide 3.5.

1.2 Need

Licensees are now uncertain what the NRC staff will accept in the way of a health physics survey program to protect workers. As a consequence, a wide variety of programs are submitted. In order to meet minimum acceptable standards, much correspondence between the applicant and NRC is required. A guide will reduce the amount of correspondence needed, save manpower for both NRC and the applicant, show clearly how NRC regulations apply to uranium mills, and establish a uniform standard for an acceptable survey program for worker protection.

1.3 Value/Impact

1.3.1 NRC

The impact of the proposed guidance will be primarily to reduce licensing staff effort expended in reviewing applications and corresponding with applicants in areas where the application does not meet acceptable NRC licensing standards. One staff-year was required to develop the guide.

1.3.2 Other Government Agencies

The proposed guidance will impact on the Mine Safety and Health Administration (MSHA) because they also regulate occupational health protection at uranium mills and on Agreement State regulatory agencies that regulate mills, primarily agencies in New Mexico, Colorado, Texas, Washington, and Florida. A Memorandum of Understanding (MOU) signed by NRC and MSHA states that each agency will coordinate the development of standards with the other agency. The MOU was published in the *Federal Register* (45 FR 1315) on January 4, 1980.

1.3.3 Industry

Industry will benefit from having clear guidance on what constitutes NRC licensing policy. The total cost of the occupational health physics program (surveys plus other parts of the program) is estimated to be roughly 4 staff-years per year or about $300,000 per year per mill when the costs of overhead, supplies, equipment, and contracted services are included. This does not include the cost of the environmental and effluent monitoring program nor does it include amortization costs on equipment in the mill installed to limit occupational exposure. Equipment design is not covered in this guide, therefore, costs are not estimated here. However, the annual amortization and operating costs of equipment installed to protect workers is not negligible.

1.3.4 Workers

Workers' protection should improve from having clearly stated and consistent standards for health physics survey programs. Workers and workers' representatives will now have access to a clearly defined standard health physics survey program. This will help them understand whether their employer has an adequate program and why some things are done as they are.

1.3.5 Public

The guidance pertains to worker protection programs. It will not directly affect the public.

1.4 Decision

The NRC should develop guidance on standard health physics survey programs for worker protection that are acceptable to the NRC licensing staff.

2. TECHNICAL APPROACH

The technical approach in the guidance is based on (1) NRC licensing policy as expressed in Safety Evaluation Reports (SER) written by the NRC licensing staff, especially the recent SER for Minerals Exploration Company Sweetwater Uranium Project; (2) the IAEA Manual on Radiological Safety in Uranium and Thorium Mines and Mills, IAEA Safety Series No. 43, 1976; (3) public comments received on Draft Guide OH 710-4; and (4) other references cited in the guide.

The most important technical question raised by the public comments concerned the duration of grab samples for uranium ore dust and yellowcake. The draft guide recommended 60-minute samples.

Mr. William Shelley of Kerr-McGee, speaking for the American Mining Congress (AMC), wrote that sampling for uranium ore dust in non-airborne radioactivity areas should be weekly with 5-minute high-volume samples rather than monthly with 60-minute samples as in the guide. The AMC, in a subsequent letter intended to supplement Mr. Shelley's comments, stated that 60-minute samples at 20 to 25 operator-occupied sites would require 3 to 4 days for sample collection, which is excessive. The AMC recommended monthly 30-minute samples with a stipulation requiring additional sampling in the area if an action level were exceeded. The AMC said weekly 5-minute high-volume samples "are not

deemed preferable in this context." The AMC recommended weekly 15-minute high-volume samples with a flow rate of 30 cfm when more frequent sampling was needed and said such sampling would satisfy the LLD values in the guide. The AMC stated that filters could clog during long sampling times, thereby reducing the accuracy of the measurement.

Mr. Gerald Sinke of Kerr McGee, in a subsequent letter to clarify the AMC objection to 60-minute samples, stated that the Kerr-McGee mill sampled weekly at 36 locations in ore handling areas. Mr. Sinke said that 5-minute samples would be more accurate than 60-minute samples because the technician would be present during sample collection, whereas he would not be present during a 60-minute sample. Mr. Sinke showed by calculation that an LLD of 2.7×10^{-12} μCi/ml was obtained using a 5-minute sample with a flow rate of 760 liters/min. This meets the recommended LLD of 5×10^{-12} μCi/ml. Sinke's method is based on alpha counting after radon decay. Alpha counting will not work well for ore dust with long sampling times because the dust loading on the filter paper will cause self-absorption of the alpha particles. The State of New Mexico Environmental Improvement Agency said that 30-minute samples seemed excessively long.

The above comments claim that 60-minute samples are too long and state that the recommended LLD can be obtained with shorter samples. Based on NRC's calculations such as those shown in the new appendix to the guide, it is correct that an acceptable LLD can be met with samples of far less than 60-minute duration as long as the air flow is sufficient and the analysis background is low enough.

The NRC agrees that excessive dust loading is likely to be deposited on filters of high-volume samplers during a 60-minute sample. On the other hand, monthly 5-minute samples seem too short to account for short-term variations in air concentrations. A time longer than 5 minutes is believed to be necessary because the grab samples are taken at a fairly low frequency – weekly or monthly depending on the levels of airborne radioactivity present. The NRC accepts the fairly low weekly or monthly frequency because concentrations of ore dust are generally low in ore dust areas (typically 10% of the Appendix B values) and because the concentrations have been observed to fall within fairly narrow ranges, except for seasonal variations due to increased ventilation during warmer months. Concentrations of yellowcake when equipment is not operating are also low and fall within limited ranges. More extensive sampling is required for maintenance operations and in certain operations when yellowcake is actively handled.

In view of this, the recommended sample duration is lowered to 30 minutes at an adequate air flow rate to meet the recommended LLD of 5×10^{-12} μCi/ml. However, in areas that are not airborne radioactivity areas, weekly 5-minute samples are acceptable instead of monthly 30-minute samples.

The second most important technical question raised by the public comments concerned the recommended limits on surface contamination in work areas, namely the value for alpha activity of 0.001 μCi/cm^2. Mr. L. M. Cook of Chevron Resources Company said that the limit on contamination levels of 0.001 μCi/cm^2 may not keep ingestion low enough and that bioassays would routinely be high.

The NRC response is that surface alpha contamination levels of 0.001 μCi/cm^2 are generally recognized as being adequate to maintain the inhalation of resuspended particles to very low levels. Experimental work in a uranium facility showed that surface contamination of this magnitude contributed less than 1% of the exposures received by employees.[1] Experience in plants led the International Atomic Energy Agency to recommend this value for uranium mills.[2] Theoretical calculations based on resuspension factors led the British National Radiological Protection Board to recommend the same limit.[3] In the words of the International Commission on Radiological Protection (ICRP), "Experience has shown that there is not necessarily a correlation between surface contamination in the workplace and the exposure of workers."[4]

There are several physical factors that reduce the resuspension of small respirable particles. Fine dusts (<50 microns) are extremely resistant to resuspension by wind because these particles lie in the laminar layer next to the ground and do not protrude much into the turbulent air layers.[5] In addition, respirable particles (<10 microns) tend to agglomerate in a process called weathering and their resuspension depends on a mechanical impact to break the agglomerate.[6]

A more complete "Response to Public Comments on Health Physics Surveys in Uranium Mills" is available from the author of the guide: Dr. Stephen A. McGuire, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

3. PROCEDURAL APPROACH

In its preliminary value/impact assessment, the staff considered several procedural approaches for carrying out the proposed action and selected the publication of a regulatory guide.

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3.1 Decision on Procedural Approach

Developing a regulatory guide is the favored procedural approach.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

NRC authority for issuance of this guide derives from the Atomic Energy Act of 1954, as amended, through those portions of the Commission's regulations in Title 10 of the Code of Federal Regulations cited in the introduction to the guide.

4.2 Need for NEPA Assessment

The proposed action is not a major action significantly affecting the quality of the human environment as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. CONCLUSION

The regulatory guide on health physics survey programs for worker protection in uranium mills should be issued.

U.S. NUCLEAR REGULATORY COMMISSION

May 1983

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.31
(Task OH 941-4)

INFORMATION RELEVANT TO ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES AT URANIUM MILLS WILL BE AS LOW AS IS REASONABLY ACHIEVABLE

A. INTRODUCTION

Paragraph 20.1(c) of 10 CFR Part 20, "Standards for Protection Against Radiation," states that licensees should make every reasonable effort to keep radiation exposures, as well as releases of radioactive material to unrestricted areas, as far below the limits specified in Part 20 as is reasonably achievable. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," sets forth the philosophy and general management policies and programs that licensees should follow to achieve this objective.

This guide recommends design criteria and administrative practices acceptable to the NRC staff for maintaining occupational exposures as low as is reasonably achievable (ALARA) in uranium mills. However, some of the basic processes at other types of uranium recovery facilities have a similar potential for exposing workers to uranium and its daughters. Therefore, the guidance provided in this guide can be applied, as appropriate, to those facilities as well.

An existing NRC report, NUREG-0706, "Final Generic Environmental Impact Statement on Uranium Milling" (Ref. 1), also provides detailed information for controlling the radiation hazard and chemical toxicity of airborne uranium and its daughter products in uranium mills.

This guide is directed toward occupational health protection from radiologic and toxic hazards from airborne particulates of uranium and its daughters. However, it is also recognized that uranium mill workers will be exposed to external radiation in addition to inhaled particulates. Therefore, ensuring protection of mill workers from external radiation hazards is also addressed.

Specific guidance regarding protection of the public from radiologic and toxic hazards caused by materials in effluents to unrestricted areas is beyond the scope of this guide. This topic is mentioned only in connection with actions that influence both occupational exposure and effluent control. Some of the same controls that have been shown to keep occupational exposures to airborne uranium and its daughters ALARA also tend to keep releases of these materials from the mill ALARA (see Regulatory Guide 4.14, "Radiological Effluent and Environmental Monitoring At Uranium Mills").

Any guidance in this document related to information collection activities has been cleared under OMB Clearance No. 3150-0014.

B. DISCUSSION

The principle of maintaining occupational radiation exposures as low as is reasonably achievable is an extension of an original recommendation of the National Committee on Radiation Protection (NCRP) (now the National Council on Radiation Protection and Measurements) in its Report No. 17 (Ref. 2). In this early report, the NCRP introduced the philosophy of assuming that any radiation exposure may carry some risk and recommended that radiation exposure be kept at a level "as low as practicable" below the recommended maximum permissible dose equivalent. This philosophy is currently referred to as "as low as is reasonably achievable" (ALARA). Similar recommendations to keep exposures ALARA have been included in NCRP reports up to the present time (Ref. 3), as well as in recommendations of the National Academy of Sciences–National Research Council (Ref. 4), the Federal Radiation Council (Ref. 5), and other independent scientific and professional organizations (Ref. 6). Therefore, NRC has incorporated this basic radiation protection philosophy from these recommendations into its regulations and guides.

This guide provides a detailed supplement for uranium mill licensees of the basic philosophy of Regulatory Guide 8.10, which lists for all specific licensees the types of

management commitments and radiation protection programs that would help to achieve the objective of maintaining occupational exposures ALARA.

Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills," outlines the information that applicants should include in an application for a uranium mill license. This regulatory guide describes the details of an acceptable radiation protection and ALARA program that an applicant should describe as recommended in Section C.5, "Operations," of Regulatory Guide 3.5.

C. REGULATORY POSITION

The principles and practices presented in this guide should be used as guidance in developing the radiation protection and ALARA program for a uranium mill for appropriate sections of an application* for a new or renewal license. The recommendations of this guide are intended to assist applicants in preparing license applications that are acceptable to the NRC licensing staff and are consistent with the philosophy of ALARA. Unique features not addressed here will be specifically reviewed by the NRC licensing staff.

A licensee's program for occupational protection against uranium and its daughters will be considered consistent with the ALARA philosophy if the uranium mill's operating policies and programs satisfy the following major principles and practices.

1. ALARA PHILOSOPHY

A major purpose of the occupational radiation protection program at a uranium mill is to maintain radiation exposure ALARA for all employees, contractors, and visitors.

The implementation and effectiveness of a successful ALARA program is the responsibility of everyone involved in the processing of uranium ores. Responsibilities for conducting a radiation protection and ALARA program are shared by licensee management,** the radiation safety officer (RSO),*** and all mill workers.

1.1 Licensee Management

Licensee management is responsible for developing, implementing, and enforcing the rules, policies, and procedures necessary for an effective radiation protection and ALARA program to ensure the health and safety of workers.

Licensee management should provide the following:

1. A strong commitment to and continuing support for the development and implementation of the radiation protection and ALARA program;

2. Information and policy statements to employees, contractors, and visitors;

3. A periodic management audit program that reviews procedural and operational efforts to maintain exposures ALARA;

4. Continuing management evaluation of the health physics program, its staff, and its allocation of adequate space and money;

5. Appropriate briefings and training in radiation safety, including ALARA concepts for all uranium mill employees and, when appropriate, for contractors and visitors.

1.2 Radiation Safety Officer

The radiation safety officer (RSO) has primary responsibility for the technical adequacy and correctness of the radiation protection and ALARA program and has continuing responsibility for surveillance and supervisory action in the enforcement of the program.

The radiation safety officer should be assigned the following:

1. Major responsibility for the development and administration of the radiation protection and ALARA program;

2. Sufficient authority to enforce regulations and administrative policies that affect any aspect of the radiological safety program;

3. Responsibility to review and approve plans for new equipment, process changes, or changes in operating procedures to ensure that the plans do not adversely affect the protection program against uranium and its daughters;

4. Adequate equipment and laboratory facilities to monitor relative attainment of the ALARA objective.

1.3 Mill Workers

Because a radiation protection and ALARA program is only as effective as the workers' adherence to the program, all workers at the mill should be responsible for the following:

1. Adhering to all rules, notices, and operating procedures for radiation safety established by licensee management and the RSO;

*An application and a suggested format for its completion may be obtained from the licensing staff of the Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

**"Management" is defined here as those persons authorized by the licensee of record to make policies and to direct activities of the recovery facility.

***The title "radiation safety officer" is used synonymously with "radiation protection manager" by many licensees and will be used in this guide to designate the qualified individual who is responsible for developing and supervising the radiation safety program; other titles are equally acceptable.

2. Reporting promptly to the RSO and licensee management equipment malfunctions or violations of standard practices or procedures that could result in increased radiological hazard to any individual;

3. Suggesting improvements for the radiation protection and ALARA program.

2. HEALTH PHYSICS ORGANIZATION AND ADMINISTRATIVE PROCEDURES

2.1 Health Physics Authorities and Responsibilities

The radiation safety officer at the mill site should be responsible for conducting the health physics program and for assisting the resident manager in ensuring compliance with NRC's regulations and the license conditions applicable to worker health protection.

Generally, the RSO should report directly to the resident manager on matters of radiation safety. The RSO should be directly responsible for supervising the health physics technicians, for overseeing the day-to-day operation of the health physics program, and for ensuring that records required by the NRC are maintained. The RSO should have both the responsibility and the authority, through appropriate line management, to suspend, postpone, or modify any work activity that is unsafe or potentially a violation of the Commission's regulations or license conditions, including the ALARA program. It is recommended that management delegate this responsibility and authority directly to the RSO. The RSO may have other safety-related duties, such as responsibility for programs of industrial hygiene and fire safety, but should have no direct production-related responsibility.

2.2 Operating Procedures

Written standard operating procedures should be established for all activities that involve handling, processing, or storing radioactive materials. All such procedures should include consideration of pertinent radiation safety practices. Written procedures should also be established for such activities as health physics monitoring, sampling, analysis, and instrument calibration. An up-to-date copy of each written procedure, including accident response and radiological fire protection plans, should be kept accessible to all employees. All written procedures involving radioactive material control should be compiled in a manual that allows documentation of each revision and its date.

To ensure that proper radiation protection principles are being applied, written procedures for all activities should be reviewed and approved in writing by the RSO before being implemented and whenever a change in a procedure is proposed. In addition, the RSO should review all existing operating procedures at least annually to ensure the procedures do not violate any newly established radiation protection practices.

For work on nonroutine maintenance jobs where the potential for exposure to radioactive material exists and for which no standard written operating procedure already exists, a radiation work permit (RWP)* should be used. Such permits should describe the following:

1. The details of the job to be performed,

2. Any precautions necessary to reduce exposure to uranium and its daughters,

3. The radiological monitoring and sampling necessary before, during, and following completion of the job.

The RSO should indicate by signature the review of each RWP prior to the initiation of work, and the work should be carried out in strict adherence to the conditions of the RWP. The RSO should designate a member of the radiation safety office staff or a supervisory member of the production staff who has received specialized radiation protection training to review and sign RWPs when the RSO is not available, e.g., during off shifts.

2.3 Surveillance: Audits and Inspections

It has been observed repeatedly that, if sufficient management interest exists, exposure to hazardous materials is reduced. Frequent management audit and inspection of worker health protection practices at a uranium mill can serve to provide management with the information necessary to conduct an appropriate ALARA program.

2.3.1 Daily and Weekly Inspections

The RSO and the mill foreman should conduct a weekly inspection of all mill areas to observe general radiation control practices and review required changes in procedures and equipment. The RSO or designated health physics technician should conduct a daily walk-through (visual) inspection of all work and storage areas of the mill to ensure proper implementation of good radiation safety procedures, including good housekeeping and cleanup practices that would minimize unnecessary contamination. Problems observed during all inspections should be noted in writing in an inspection logbook. The entries should be dated, signed, and maintained on file for at least 1 year. The RSO should review all violations of radiation safety procedures or other potentially hazardous problems with the resident manager or other mill employees who have authority to correct the problem. Also, the RSO should review the daily work-order and shift logs on a regular basis to determine that all jobs and operations having a potential for exposing personnel to uranium, especially those RWP jobs that would require a radiation survey and monitoring, were approved in writing by the RSO, his staff, or designee prior to initiation of work.

*The term "radiation work permit" is used by many licensees and will be used throughout this guide; other terms such as "special work permit" are equally acceptable.

2.3.2 Monthly Reviews

At least monthly, the RSO should review the results of daily and weekly inspections, including a review of all monitoring and exposure data for the month. The RSO should provide to the resident manager and all department heads for their review a written summary of the month's significant worker protection activities containing (1) a summary of the most recent personnel exposure data, including bioassays and time-weighted calculations, and (2) a summary of all pertinent radiation survey records.

In addition, the monthly summary report should specifically address any trends or deviations from the radiation protection and ALARA program, including an evaluation of the adequacy of the implementation of license conditions regarding radiation protection and ALARA. The summary should provide a description of unresolved problems and the proposed corrective measures. Monthly summary reports should be maintained on file and readily accessible for at least 5 years.

2.3.3 Radiation Protection and ALARA Program Audit

Licensee management should have annual audits of the radiation protection and ALARA program performed and written reports on the audits submitted to corporate management. All members of the audit team should be knowledgeable concerning the radiation protection program at the mill. In addition, one member of the team should be experienced in the operational aspects of specialized uranium mill radiation protection practices. The RSO should accompany the audit team but should not be a member.

The audit report should summarize the following data:

1. Employee exposure records (external and time-weighted calculations),

2. Bioassay results,

3. Inspection log entries and summary reports of daily, weekly, and monthly inspections,

4. Documented training program activities,

5. Radiation safety meeting reports,

6. Radiological survey and sampling data,

7. Reports on overexposure of workers submitted to NRC, Mine Safety and Health Administration (MSHA), or States,

8. Operating procedures that were reviewed during this time period.

The report on the annual radiation protection and ALARA audit should specifically discuss the following:

1. Trends in personnel exposures for identifiable categories of workers and types of operational activities,

2. Whether equipment for exposure control is being properly used, maintained, and inspected,

3. Recommendations on ways to further reduce personnel exposures from uranium and its daughters.

2.4 Technical Qualifications of Health Physics Staff

2.4.1 Radiation Safety Officer

The RSO should have the following education, training, and experience:

1. Education: A bachelor's degree in the physical sciences, industrial hygiene, or engineering from an accredited college or university or an equivalent combination of training and relevant experience in uranium mill radiation protection. Two years of relevant experience are generally considered equivalent to 1 year of academic study.

2. Health physics experience: At least 1 year of work experience relevant to uranium mill operation in applied health physics, radiation protection, industrial hygiene, or similar work. This experience should involve actually working with radiation detection and measurement equipment, not strictly administrative or "desk" work.

3. Specialized training: At least 4 weeks of specialized classroom training in health physics specifically applicable to uranium milling. In addition, the RSO should attend refresher training on uranium mill health physics every 2 years.

4. Specialized knowledge: A thorough knowledge of the proper application and use of all health physics equipment used in the mill, the chemical and analytical procedures used for radiological sampling and monitoring, methodologies used to calculate personnel exposure to uranium and its daughters, and a thorough understanding of the uranium milling process and equipment used in the mill and how the hazards are generated and controlled during the milling process.

2.4.2 Health Physics Technicians

In addition to the RSO, there should be a minimum of one full-time health physics technician at any full-scale operating uranium mill. The health physics technician should have one of the following combinations of education, training, and experience:

1. Education: An associate degree or 2 or more years of study in the physical sciences, engineering, or a health-related field,

Training: At least a total of 4 weeks of generalized training (up to 2 weeks may be on-the-job training) in radiation health protection applicable to uranium mills,

Experience: One year of work experience using sampling and analytical laboratory procedures that involve health physics, industrial hygiene, or industrial safety measures to be applied in a uranium mill; or

2. Education: A high school diploma,

Training: A total of at least 3 months of specialized training (up to 1 month may be on-the-job training) in radiation health protection relevant to uranium mills,

Experience: Two years of relevant work experience in applied radiation protection.

The health physics technician should demonstrate a working knowledge of the proper operation of health physics instruments used in the mill, surveying and sampling techniques, and personnel dosimetry requirements.

2.5 Radiation Safety Training

All new employees should be instructed by means of an established course in the inherent risks of exposure to radiation and the fundamentals of protection against exposure to uranium and its daughters before beginning their jobs. Other guidance pertinent to this course is found in Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," and Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure." This course of instruction should include the following topics:

1. Fundamentals of Health Protection
   a. The radiologic and toxic hazards of exposure to uranium and its daughters,
   b. How uranium and its daughters enter the body (inhalation, ingestion, and skin penetration),
   c. Why exposures to uranium and its daughters should be kept as low as is reasonably achievable (ALARA).
2. Personal Hygiene at Uranium Mills
   a. Wearing protective clothing,
   b. Using respirators correctly,
   c. Eating, drinking, and smoking only in designated areas,
   d. Using proper methods for decontamination (i.e., showers).
3. Facility-Provided Protection
   a. Ventilation systems and effluent controls,
   b. Cleanliness of the work place,
   c. Features designed for radiation safety for process equipment,
   d. Standard operating procedures,
   e. Security and access control to designated areas.
4. Health Protection Measurements
   a. Measurement of airborne radioactive materials,
   b. Bioassays to detect uranium (urinalysis and in vivo counting),
   c. Surveys to detect contamination of personnel and equipment,
   d. Personnel dosimetry.
5. Radiation Protection Regulations
   a. Regulatory authority of NRC, MSHA, and State,
   b. Employee rights in 10 CFR Part 19,
   c. Radiation protection requirements in 10 CFR Part 20.
6. Mill Emergency Procedures.

A written or oral test with questions directly relevant to the principles of radiation safety and health protection in uranium milling covered in the training course should be given to each worker. The instructor should review the test results with each worker. The instructor should discuss any wrong answers to test questions with the worker until the worker understands the correct answer. Workers who fail the test should be retested after receiving additional training. These tests and results should be maintained on file.

Each permanent worker should be provided an abbreviated retraining course annually. Documented successful completion of the retraining course should also be maintained on file. Retraining should include relevant information that has become available during the past year, a review of safety problems that have arisen during the year, changes in regulations and license conditions, exposure trends, and other current topics.

In addition, all new workers, including supervisors, should be given specialized instruction on the health and radiation safety aspects of the specific jobs they will perform. This instruction should be in the form of individualized on-the-job training. Supervisors should be provided additional specialized training on their supervisory responsibilities in the area of worker radiation protection. Retraining should be conducted annually and documented. All employees should sign a statement that they received job-specific radiation safety training. The statement should indicate the dates the training was received and it should be cosigned by the instructor. Radiation safety matters of concern that arise during plant operation should be discussed

with all workers during regular monthly or bimonthly safety meetings.

All visitors who have not received training should be escorted by someone properly trained and knowledgeable about the hazards of the mill. At a minimum, visitors should be instructed specifically on what they should do to avoid possible hazards in the areas of the mill they will be visiting.

Contractors having work assignments in the mill should also be given appropriate training and safety instruction. Contract workers who will perform work on heavily contaminated equipment should receive the same training and radiation safety instruction normally required of all permanent workers. Only job-specific radiation safety instruction is necessary for contract workers who have previously received full training on prior work assignments at the mill or have evidence of recent and relevant radiation safety training elsewhere.

2.6 Surveys

The RSO and radiation safety office staff are responsible for performing all routine and special radiation surveys as required by license conditions and by 10 CFR Part 20. Acceptable survey methods are specified in Section C.1 of Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills."

2.7 Respiratory Protection

The RSO and the radiation safety office staff are responsible for the implementation of a respiratory protection program, if one is needed. There should be adequate supplies of respiratory devices to enable issuing a device to each individual who enters an airborne radioactivity area. Additional respiratory protection devices should be located near access points of airborne radioactivity areas. All airborne radioactivity areas should have controlled access. Routine physical (medical) evaluation should be required of those individuals who will use respirators. If the licensee elects to take credit for protection factors the respiratory protection program must meet, at a minimum, the requirements of § 20.103 of 10 CFR and should follow the recommendations in Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," which are supported in NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials" (Ref. 7).

2.8 Bioassay Procedures

The RSO is responsible for implementing a bioassay program. The frequency adopted and the type of analysis should meet the recommendations in Regulatory Guide 8.22, "Bioassay at Uranium Mills."

3. FACILITY AND EQUIPMENT DESIGN

General considerations for the design of uranium mills and uranium ore processing equipment should not be based solely on chemical process efficiency, but should also be based on the relative potential for radiologic and toxic hazards resulting from exposure of personnel to uranium and its daughters. Major aspects of planning and design that should be considered are discussed below.

3.1 Space Layout

Facility layout should be designed to maintain employee exposures ALARA while at the same time ensuring that exposure to other persons is not thereby increased. The mill layout should provide for:

1. Safe access to process equipment and for routine maintenance;

2. Adequate ventilation in all mill areas in which radioactive materials might be spilled, suspended, or volatilized;

3. Isolation of yellowcake drying, packaging, and shipping areas from other accessible mill areas;

4. Controlling access to the uranium mill proper and the ability to secure or restrict entry to any airborne radioactivity area;

5. Change rooms and shower facilities so that all workers can remove any possible radioactive contamination before leaving the site;

6. Dispersion control on radioactive materials moving from contamination areas (e.g., crushers) to relatively contamination-free areas (e.g., crusher control room);

7. Isolation of mill areas where there is a high potential for the dispersal of uranium as the result of a fire.

3.2 Access Control

Access to airborne radioactivity areas should be controlled or restricted by the use of caution signs and operational procedures, or security locks when permitted by fire protection regulations.

3.3 Ventilation Systems

To the extent practicable, the facility ventilation systems should accomplish the following:

1. As a minimum design objective, provide local exhaust ventilation (such as chemical hoods) or general area ventilation where concentrations of natural uranium and its daughters may be present in excess of 25% of the values given in Table 1 of Appendix B to 10 CFR Part 20.* The design ventilation rate (air exchange rate) should be sufficient to maintain airborne concentrations of natural uranium and its daughters to less than 25% of the maximum permissible concentration (MPC) given in Table 1 of Appendix B to 10 CFR Part 20.

*The figure 25% is used here to encourage the use of ventilation systems and other process controls in an effort to prevent the existence of airborne radioactivity areas as defined in § 20.203(d), and according to § 20.103(b)(1), "The licensee shall, as a precautionary procedure, use process or other engineering controls, to the extent practicable, to limit concentrations of radioactive materials in air to below those which delimit an airborne radioactivity area...."

2. In addition, establish a facility-specific, operational ALARA goal for concentrations of natural uranium and its daughters at less than 25% of the values given in Table 1 of Appendix B to 10 CFR Part 20.

3. Design exhaust stacks so that exhausted air will not enter air intakes that service any other mill areas.

4. Locate exhaust vents in a way that ensures compliance with the requirements of § 20.106, "Radioactivity in effluents to unrestricted areas," of 10 CFR Part 20, and 40 CFR, "Protection of Environment," Part 190, "Environmental Radiation Standards for Nuclear Power Operations," for effluents to unrestricted areas, as well as ALARA exposure considerations for the worker.

3.4 Fire Control

Because of the potential for loss of control of radioactive material in the event of a fire, a facility should have adequate firefighting equipment and workers should be trained in its proper use.

Provisions should be made for fire alarms, fire extinguishers, sprinkler systems, fire hydrants, water tanks, and other general firefighting equipment. Emergency procedures and training should include immediate fire control as a priority item. Design features should include automatic fire detection and suppression equipment in high fire-potential areas (e.g., solvent extraction area). In the event of fire, there should be provision for drainage of solvent to sumps or to outside lined ponds. Appropriate caution signs should be posted in areas of fire hazard. Fire detection systems should be checked weekly. Fire drills should be performed at least semiannually.

3.5 Laboratory Design Features

Consideration should be given to providing different laboratory facilities for metallurgical and bioassay analyses, if they are both performed at the mill site. Owing to the sensitivity required in performing bioassay analyses, provisions should be made to ensure against cross-contamination of uranium from mill ore samples. Laboratory equipment and surfaces should be constructed of materials that are easily decontaminated. Laboratory surfaces used for the preparation of bioassay samples should be decontaminated daily to less than 200 dpm $\alpha/100$ cm^2 of total surface contamination. All mill laboratories should provide adequate general ventilation and exhaust fume hoods. Special attention should be directed to the design of air exhaust systems that service ore sample pulverizing and grinding equipment. The design of the laboratory should provide for the safe handling, storage, and disposal of radioactive wastes resulting from sample analyses.

3.6 Ore and Product Storage

Uranium mill plans should include the following:

1. Provisions for raw ore storage, fine ore bins, and yellowcake storage in areas so that the material does not cause unnecessary exposure to mill personnel and so that material is not dispersed by wind and rain;

2. Adequate space in the yellowcake storage and packaging areas to conduct initial surveys and spot smear tests of yellowcake packages and to enable decontamination of drums to avoid transporting a contaminated package through other mill areas;

3. Locations for yellowcake storage and shipping areas that minimize the handling time required prior to shipment.

3.7 General Equipment Considerations

General features applicable to equipment that will be used for handling, containing, or contacting uranium and its daughters are as follows:

1. Equipment that contains large volumes of uranium bearing liquids should be designed with sumps or dikes to contain the liquids in the event of leaks or spills;

2. Equipment should be designed for optimum ease of carrying out procedures, especially routine maintenance, to minimize working time where personnel are exposed to radiation or radioactive material, and to maximize distances of personnel from the source of radiation with which they are working;

3. Appropriate caution signs and symbols should be provided to meet the requirements of § 20.203 of 10 CFR Part 20, as discussed in more detail in Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills";

4. The use of semiautogenous methods for grinding ore is recommended because of the significantly reduced generation of airborne dusts.

4. CONTROL OF AIRBORNE URANIUM AND ITS DAUGHTERS

One of the major inhalation hazards associated with uranium milling facilities results from the resuspension in air of uranium and its daughters. Therefore, properly designed ventilation and dust control systems are needed to ensure that exposure of workers is maintained ALARA. There are, in general, four areas that present radiologic and toxic hazards caused by airborne materials at a typical uranium mill. These areas encompass (1) ore storage, handling, and crushing; (2) ore grinding, leaching, and concentrating processes; (3) yellowcake precipitation, drying, and packaging; and (4) miscellaneous mill locations as specified in Section 4.4. Appropriate design objectives for ventilation and dust control systems recommended for each of these generalized mill areas are given below.

4.1 Ore Storage, Handling, and Crushing Areas

Where ore is handled in the open, the objective should be to minimize blowing of dust. Water sprinkling systems

are recommended for use on ore piles when the ore moisture content is less than 10%. If ore is crushed and transported in the dry state (i.e., moisture content less than 25%), the use of ventilation systems and dust collectors is recommended. As ore travels along convevor belts to the grinder, all drop points should have either hooded dust collectors or dust suppressant systems, such as sprinklers or foam ejectors. When crushers are used prior to grinding, it is recommended that a hooded ventilation system be installed over all external openings to the crusher. The use of wet scrubbers or dust collectors is recommended for ventilation systems that service ore storage, handling, and crushing areas of the mill to prevent recirculation of contaminated air.

4.2 Grinding, Leaching, and Concentrating Process Areas

General ventilation systems are recommended to service mill areas where any grinding method is performed to ensure against the buildup of radon-222 and its daughters and ore dust normally released in the grinding process. The ventilation rate should be adequate to maintain the concentrations of radon-222 or its daughters and natural uranium from ore dust to less than 25% of the value specified in Table 1 of Appendix B to 10 CFR Part 20 as modified by the note to Appendix B. It is recommended that all leaching and thickening tanks located in enclosed structures be covered and vented directly to the outside atmosphere. General ventilation systems for mill areas where leaching and thickening tanks are located should be designed to maintain natural uranium ore dust concentrations in air at less than 19.0 $\mu g/m^3$ of uranium. If the mill is so designed that the solvent extraction (SX) concentration process equipment is in enclosed structures, a general ventilation system is recommended and should be designed to maintain the airborne natural uranium concentration in air to less than 50 $\mu g/m^3$ of uranium or 2.5×10^{-11} $\mu Ci/cm^3$ (i.e., 25% of the MPC for natural uranium). The use of wet scrubbers on general ventilation systems that service areas of the mill where grinding and leaching equipment are located is recommended. Scrubbers are not necessary on ventilation systems that service areas of the mill where the clarification or solvent extraction equipment is located.

4.3 Precipitation, Drying, and Packaging Areas

General ventilation systems are required and should be designed to maintain the concentration in air of yellowcake near precipitation tanks, yellowcake thickeners, yellowcake filters, and yellowcake repulp equipment to less than 50 $\mu g/m^3$ of uranium in air or 2.5×10^{-11} $\mu Ci/cm^3$ (i.e., 25% of the maximum permissible concentration). The next step of the recovery process involves the drying and packaging of yellowcake. Since the potential for the release of airborne yellowcake is much greater in dry form, it is recommended that drying and packaging of yellowcake should be performed in an enclosure that is separated from other areas of the mill. Also, the drying and packaging enclosure should be maintained under negative pressure. A separate air suction ring system should also be used at each yellowcake drumming station. Individual suction ring systems need only be operated during periods when the drum at that location is being filled. The exhausts for the drying and packaging enclosure and the suction ring should be vented through a wet scrubber. To ensure proper operation, the scrubber system on the concentrate drying and packaging area should be checked every shift and documented, or automatic malfunction alarm or interlock systems installed. Manometer readings or operational and instrument checks should be recorded once per shift and subsequently documented.

4.4 Miscellaneous Locations

Other important areas of the mill that have the potential for containing hazardous levels of uranium and its daughters in air include maintenance shops, rubber shops, metallurgical and bioassay laboratories, and general laundries, if they exist. Each of the above mill areas should be serviced by ventilation systems designed to maintain air concentration of natural uranium and its daughters to less than 50 $\mu g/m^3$ or 2.5×10^{-11} $\mu Ci/cm^3$ of uranium. Wet scrubbers are not necessary on these systems, however, bag filters are recommended.

D. IMPLEMENTATION

Except in those cases in which an applicant or licensee proposes an acceptable alternative method, this guide and Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills"; Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"; Regulatory Guide 8.22, "Bioassay at Uranium Mills"; and Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills," will be used as the basis for evaluating license applications and radiation safety and ALARA programs of NRC-licensed uranium mills.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Final Generic Environmental Impact Statement on Uranium Milling," USNRC Report NUREG-0706, September 1980.*

2. National Bureau of Standards, "Permissible Dose from External Sources of Ionizing Radiation," *Handbook 59, Recommendations of the National Council on Radiation Protection* (NCRP Report No. 17), Washington, D.C., September 24, 1954.

3. National Council on Radiation Protection and Measurements, "Review of the Current State of Radiation Protection Philosophy," Report No. 43, Washington, D.C., January 15, 1975.

4. National Academy of Sciences – National Research Council, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Washington, D.C., 1972.

5. Federal Radiation Council, "Background Material for the Development of Radiation Protection Standards," Report No. 1, Washington, D.C., 1960.

6. International Commission on Radiological Protection, "Implications of Commission Recommendations That Doses Be Kept As Low As Readily Achievable," Report No. 22, Pergamon Press, Elmsford, New York, 1974.

7. U.S. Nuclear Regulatory Commission, "Manual of Respiratory Protection Against Airborne Radioactive Materials," USNRC Report NUREG-0041, October 1976.*

*Copies are available from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Applicants for a uranium milling license must submit a license application containing the information specified in Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills." The purpose of this action is to describe both administrative health physics programs and methods to achieve ALARA occupational exposure to workers that are acceptable to the NRC staff. Health physics programs are covered in Section C.5, "Operations," in Regulatory Guide 3.5.

1.2 Need for the Proposed Action

Currently, licensees are uncertain what the NRC staff will accept in the way of a health physics and ALARA program or procedures and design features needed to achieve ALARA exposures in a uranium mill. As a consequence, a wide variety of programs are submitted. To meet minimum standards, much correspondence between the applicant and NRC is required. A guide will reduce the amount of correspondence needed, save personnel resources for both NRC and the applicant, show clearly how NRC regulations apply to uranium mills, and establish a uniform standard for an acceptable health physics and ALARA program for worker protection.

1.3 Value/Impact of the Action

1.3.1 NRC

The impact of the guidance will be primarily to reduce licensing staff effort in reviewing applications and in corresponding with applicants about areas where the application does not meet current NRC licensing requirements. An estimated 0.75 staff-year is required to develop the guide.

1.3.2 Other Government Agencies

The guidance will impact on the Mine Safety and Health Administration (MSHA) because they also regulate occupational health protection at uranium mills and on Agreement State regulatory agencies that regulate mills, primarily New Mexico, Colorado, Texas, Washington, and Florida. A Memorandum of Understanding (MOU) signed by NRC and MSHA states that each agency will coordinate the development of standards with the other agency. The MOU was published in the *Federal Register* (45 FR 1315) on January 4, 1980.

1.3.3 Industry

Industry will benefit from having clear guidance on what constitutes NRC licensing policy. Some minor expense may be involved, however, in upgrading current health physics programs and in establishing an effective ALARA program where one does not currently exist to meet the recommendations in the guidance.

1.3.4 Workers

Workers' protection should improve from having clearly stated and consistent standards for health physics and ALARA programs. Workers and their representatives will now have access to a clearly defined standard ALARA program for uranium mills. This will help them understand whether their employer has an adequate program and why some things are done as they are.

1.3.5 Public

The guidance pertains to worker protection programs. It will not directly affect the public.

1.4 Decision

The NRC should publish guidance on a standard administrative health physics and ALARA program for worker protection that is acceptable to the NRC licensing staff.

2. TECHNICAL APPROACH

The technical approach in the guidance is based on (1) NRC licensing policy as expressed in Safety Evaluation Reports (SER) written by the NRC licensing staff, especially the recent SER for Minerals Exploration Company Sweetwater Uranium Project, and (2) other references to be cited in the guidance.

3. PROCEDURAL APPROACH

3.1 Procedural Alternatives

The three reasonable procedural alternatives are as follows:

a. Regulation,

b. Regulatory guide,

c. Continue to handle each licensing application on a case-by-case basis

3.2 Value/Impact of Procedural Alternatives

A regulation is not suitable for the type of guidance envisioned because some of the program must be tailored to the design and needs of the individual mill.

A regulatory guide is recommended since it provides the best mix of flexibility and clear statement of a uniform and consistent licensing policy.

3.3 Decision on Procedural Approach

The staff concludes that a regulatory guide should be published.

4. STATUTORY CONSIDERATIONS

Authority for this guide is derived from the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as amended, through the Commission's regulations.

5. CONCLUSION

In summary, it is proposed that a regulatory guide should be published concerning radiation protection and ALARA programs in uranium mills for worker protection.