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REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.36

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RADIATION DOSE TO THE EMBRYO/FETUS

A. INTRODUCTION

Section 20.1208 of 10 CFR Part 20, "Standards for Protection Against Radiation," requires that each licensee ensure that the dose to an embryo/fetus during the entire pregnancy, from occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv). Paragraph 20.1208(b) requires the licensee to make efforts to avoid substantial variation above a uniform monthly exposure rate to a declared pregnant woman that would satisfy the 0.5 rem (5 mSv) limit. The dose to the embryo/fetus is to be the sum of (1) the deep-dose equivalent to the declared pregnant woman (10 CFR 20.1208(c)(1)) and (2) the dose to the embryo/fetus from radionuclides in the embryo/fetus and radionuclides in the declared pregnant woman (10 CFR 20.1208(c)(2)).

This guide is being developed to provide guidance on calculating the radiation dose to the embryo/fetus. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," provides instructions concerning the risks associated with prenatal radiation exposure.

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

Calculating the radiation dose to the embryo/fetus from internally deposited radionuclides requires quantitative information about maternal radionuclide intake, placental transfer and kinetics, and resulting embryo/fetus radionuclide concentrations. Intakes of radioactive material occurring prior to the pregnancy may also be important if these materials remain in the pregnant woman during all or part of the gestation period. Transfer kinetics from the mother to the embryo/fetus are modeled as a function of stage of pregnancy, route of intake by the pregnant woman, and time after intake. The stage of gestation (or fetal development) is an important parameter in estimating radionuclide concentrations in the embryo/fetus. The geometry of the embryo/fetus (i.e., size and weight) affects the radionuclide dosimetry.

It is recognized that calculation of prenatal radiation doses from internally deposited radionuclides has many associated difficulties, including a lack of quantitative information about prenatal radionuclide concentrations and transfer across the placenta. The International Commission on Radiological Protection (ICRP) in Publication 56 (Ref. 1) states that, for most radionuclides, preliminary estimates from dosimetric and biokinetic models indicate that the dose to the embryo can be approximated by the dose to the uterus. The dose to the fetus is dependent upon the activity present in both fetal and maternal tissues. ICRP Publication 56 (Ref. 1) also states that, for most radionuclides, the dose to fetal tissue will be similar to

similar to or less than the dose to the corresponding maternal tissues.

The current methods available for assessing the radiation dose to the human embryo/fetus from internally deposited radioactive materials in the pregnant woman are subject to a number of uncertainties. Revison 1 to NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses—Interim Recommendations" (Ref. 2), provides recommendations and methods for estimating the radiation doses to the embryo/fetus from internal radionuclides. In Revision 1 to NUREG/CR-5631, a number of radionuclides were evaluated. To expedite efforts, the initial evaluation was directed to those radionuclides that were expected to be of greatest significance for prenatal exposure in the work environment. The radionuclides that were identified and included were ^{3}H, ^{14}C, ^{57}Co, ^{58}Co, ^{60}Co, ^{89}Sr, ^{90}Sr, ^{106}Ru, ^{125}I, ^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I, ^{134}Cs, ^{137}Cs, ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, and ^{241}Am. The methods of Revision 1 to NUREG/CR-5631 are considered interim as efforts continue to further develop the bases and calculational methods for estimating prenatal radiation doses. Revision 1 to NUREG/CR-5631 provides details of the data and bases for the dosimetric features that were used for the radionuclides listed above.

It is expected that the embryo/fetus dose assessment methods will evolve over the next several years as more research is conducted in this area. As additional research is conducted, better estimates of actual embryo/fetus doses resulting from the exposure of the declared pregnant woman will be possible. For internal doses, research that categorizes the degree of placental transfer, the resulting embryo/fetus/placenta concentrations, and the potential radiation exposures of the embryo/fetus from radionuclides in their more usual chemical forms should simplify assessment of the dose to the embryo/fetus based on the maternal exposure. The ICRP is considering the formulation of dose assessment methods specific for the embryo/fetus.

This regulatory guide provides acceptable methods that may be used in determining the dose to the embryo/fetus. For internal exposure, a simplified approach and a more detailed methodology are presented for conducting dose evaluations. The regulatory position specified in Section 1 provides guidance on the threshold criteria for use in determining when the dose to the embryo/fetus needs to be evaluated. The regulatory position specified in Section 2 presents a simplified approach for estimating the dose to the embryo/fetus from intakes by the declared pregnant woman. The regulatory position specified in Section 3 provides an alternative, more detailed methodology for a limited number of radionuclides, using the gestation-time dependent dosimetric data from Revision 1 to NUREG/CR-5631 (Ref. 2).

A graded approach for determining when to evaluate, with both a simple and more detailed dose assessment methodology, is provided. Both methods are acceptable for evaluating the dose to the embryo/fetus. It is recognized that some licensees will only need to demonstrate that the dose to the embryo/fetus is not likely to exceed the 0.05 rem (0.5 mSv) monitoring threshold of 10 CFR 20.1502, while other licensees may need to determine an embryo/fetus dose for demonstrating compliance with the dose limit of 10 CFR 20.1208 and the recordkeeping requirements of 10 CFR 20.2106(e).

Appendix A provides information on and a table of dose equivalent factors for use in approximating the embryo/fetus dose from radionuclides in maternal blood. Appendix B is a table of blood uptake fractions for ingested activity. Appendix C contains tables of gestation-time dependent doses to the embryo/fetus following introduction of specified radionuclides and chemical forms into maternal blood. Examples of the use of dose assessment methods are provided in Appendix D.

The total radiation dose to the embryo/fetus is the sum of the deep-dose equivalent to the declared pregnant worker and the dose to the embryo/fetus from intakes of the declared pregnant worker. If multiple dosimetric devices are used to measure the deep-dose equivalent to the declared pregnant worker, the results of monitoring that are most representative of the deep dose to the embryo/fetus may be used. The licensee need not use the deep dose to the maximally exposed portion of the whole body of the mother as the deep dose to the embryo/fetus. The licensee may employ temporary or permanent shielding to reduce the deep dose to the embryo/fetus. Alternatively, deep dose to the embryo/fetus may be limited by placing more stringent restrictions on the exposure of the declared pregnant woman than on other members of the occupational work force.

As specified in 10 CFR 20.1208(a), the dose to the embryo/fetus from occupational exposure of the declared pregnant woman during the entire gestation period is not to exceed 0.5 rem (5 mSv). In addition, the licensee is required to make efforts to avoid substantial variation in the monthly exposure throughout the period of gestation. If the dose to the embryo/fetus is found to have exceeded 0.5 rem (5 mSv) or is within 0.05 rem (0.5 mSv) of this dose by the time the woman declares the pregnancy to the licensee, the licensee is required to limit the additional dose to the embryo/fetus to 0.05 rem (0.5 mSv) during the remainder of the pregnancy.

The tables in the appendices to this guide were prepared directly from the computer outputs, which led to the values generally being expressed to three significant figures. This indicates greater accuracy than is warranted by the dosimetry model, but the results are presented in this form to avoid roundoff errors in calculations. In general, final results should be rounded to the nearest thousandth of a rem.

C. REGULATORY POSITION

1. CRITERIA FOR DETERMINING DOSE TO THE EMBRYO/FETUS

1.1 Monitoring

The dose equivalent to the embryo/fetus should be determined based on the monitoring of the declared pregnant woman as required by 10 CFR 20.1502. Specifically, 10 CFR 20.1502(a)(2) requires monitoring the exposure of a declared pregnant woman when the dose to the embryo/fetus is likely to exceed, in 1 year, a dose from external sources in excess of 10% of the limit of 10 CFR 20.1208 (i.e., 0.05 rem). According to 10 CFR 20.1502(b)(2), the licensee must monitor the occupational intakes of radioactive material for the declared pregnant woman if her intake is likely to exceed, in 1 year, a committed effective dose equivalent in excess of 0.05 rem (0.5 mSv). Based on this 0.05 rem (0.5 mSv) threshold, the dose to the embryo/fetus should be determined if the intake is likely to exceed 1% of ALI (stochastic) during the entire period of gestation.

These monitoring thresholds will ensure that any potentially significant exposures to the embryo/fetus are evaluated and, as appropriate, doses are determined. The conditions specified in 10 CFR 20.1502(a) and (b) are based on a 1-year period. Prior to declaration of pregnancy, the woman may not have been subject to monitoring based on conditions specified in 10 CFR 20.1502(a)(1) and 10 CFR 20.1502(b)(1). In this case, the licensee should estimate the exposure during the period monitoring was not provided, using any combination of surveys or other available data (for example, air monitoring, area monitoring, bioassay).

The monitoring criteria contained in 10 CFR 20.1502 do not establish required levels of detection sensitivity. For some radionuclides it may not be feasible to actually confirm by bioassay measurements an intake of 1% of their stochastic ALI. Workplace monitoring, occupancy factors, and access control should be considered as appropriate in evaluating potential exposures and monitoring requirements.

1.2 Evaluation of Dose to the Embryo/Fetus

The appropriate dose to be evaluated for the embryo/fetus is the dose equivalent for the duration of the pregnancy. An assessment of the 50-year committed dose is not appropriate. Also, it is not appropriate to use effective dose equivalent or committed effective dose equivalent. (Note: the committed dose equivalent to the uterus may be applied to the embryo/fetus under certain conditions as a simplified approach as described in the regulatory position specified in Section 2.)

1.3 External Dose to the Embryo/Fetus

According to 10 CFR 20.1208(c)(1), the deep-dose equivalent to the declared pregnant woman will be taken as the external dose component to the embryo/fetus. The determination of external dose should consider all occupational exposures of the declared pregnant woman since the estimated date of conception. The deep-dose equivalent that should be assigned is that dose that would be most representative of the exposure of the embryo/fetus (i.e., in the mother's lower torso region). If multiple measurements have been made, assignment of the highest deep-dose equivalent for the declared pregnant woman to the embryo/fetus is not required unless that dose is also the most representative deep-dose equivalent for the region of the embryo/fetus.

1.4 Internal Dose to the Embryo/Fetus

The internal dose to the embryo/fetus should consider the exposure to the embryo/fetus from radionuclides in the declared pregnant woman and in the embryo/fetus. The dose to the embryo/fetus should include the contribution from any radionuclides in the declared pregnant woman (body burden) from occupational intakes occurring prior to conception. The intake for the declared pregnant woman should be determined using air sample data, bioassay data, or a combination of the two. Guidance on bioassay measurements used to quantify intake is being developed and has been issued for public comment as Draft Regulatory Guide DG-8009, "Interpretation of Bioassay Measurements." Specific guidance on workplace air sampling is in Revision 1 to Regulatory Guide 8.25, "Air Sampling in the Workplace."

1.5 Evaluating Continuous Exposure

For continuous or near-continuous exposure to radioactive material that may be inhaled or ingested, the cumulative intake should be quantified and the dose determined at least every 30 days. If significant variation in the exposure levels may have occurred, the time interval for quantifying the intake should be reduced. More frequent evaluations should be considered as the potential dose to the embryo/fetus approaches the limit.

1.6 Existing Maternal Body Burdens

Maternal body burdens resulting from internal occupational exposures prior to conception should be included in determining the embryo/fetus dose. The contribution to the embryo/fetus dose from a maternal burden existing at the time of conception should be evaluated if the maternal burden at the time of pregnancy exceeds 1% of the radionuclide's stochastic ALI value for the appropriate mode of intake and class (for inhalation intakes). For multiple radionuclide burdens, the dose should be evaluated if the sum of the quotients of each burden divided by its stochastic ALI exceeds 0.01. Only body burdens existing at the time of conception need to be considered

in evaluating this threshold; radioactive material already eliminated from the body should not be included.

This threshold of 1% ALI provides a simplified approach for determining when pre-existing body burdens should be evaluated. At this threshold, it is unlikely that any resultant dose to the embryo/fetus would be significant (i.e., greater than 10% of the 0.5 rem (5 mSv) limit). As an alternative, the dose assessment methods presented in the regulatory position specified in Section 3 of this guide may be used for determining whether a pre-existing body burden represents a potentially significant dose (i.e., greater than 0.05 rem (0.5 mSv)).

2. SIMPLIFIED METHOD FOR DETERMINING EMBRYO/FETUS DOSE FROM MATERNAL INTAKES

The determination of the dose to the embryo/fetus from the intake of radioactive material by the pregnant woman should be based on the best available scientific data. At present, the NRC staff considers Revision 1 to NUREG/CR-5631 (Ref. 2) to provide such data. For most radionuclides, the dose to the embryo/fetus will be similar to or less than the dose to the maternal uterus (Ref. 1). However, the data in Revision 1 to NUREG/CR-5631 indicate that for some radionuclides the embryo/fetus dose may be significantly different, either greater than or less than the dose to the uterus.

Based on these premises (uterus dose similar to fetal dose and the data in Revision 1 to NUREG/CR-5631 (Ref. 2)), a set of dose factors has been developed for use in calculating an embryo/fetus dose. Except for those radionuclides addressed in Revision 1 to NUREG/CR-5631 (Ref. 2), the dose factors presented in Appendix A to this guide represent the committed dose equivalent to the uterus per introduction of unit activity into the first transfer compartment (i.e., blood) of the woman.[1] For the radionuclides in Revision 1 to NUREG/CR-5631, the dose factors in Appendix A represent the maximum dose equivalent to the embryo/fetus for the gestation period from the introduction of unit activity into the first transfer compartment of the woman at any time during the gestation period.

The dose limit for the embryo/fetus is expressed as a 9-month gestation dose equivalent. Particularly for certain radionuclides with both long radiological half-lives and long-term biological retention, the committed dose equivalent to the uterus may be significantly different from a 9-month gestation dose equivalent to the embryo/fetus. Several radionuclides of this type have been evaluated in Revision 1 to NUREG/CR-5631 (Ref. 2), and data have been developed for calculating an embryo/fetus gestation dose instead of using the committed dose equivalent to the uterus.

For demonstrating compliance with the dose limits of 10 CFR 20.1208, the dose factors in Appendix A may be used for approximating the embryo/fetus dose equivalent for the entire gestation period.

The steps for determining the embryo/fetus dose, using the simplified method, are as follows:

2.1 Include all the intakes by the declared pregnant woman at any time during the gestation period in the calculation of the embryo/fetus dose.

2.2 For ingested radionuclides, determine the activity uptake by the first transfer compartment (blood) by multiplying the intake (I) by the appropriate uptake factor (f_1) from Appendix B (adapted from Federal Guidance Report No. 11, Table 3 (Ref. 4)). The uptake factor, f_1, is the fraction of an ingested compound of a radionuclide that is transferred into the first transfer compartment (i.e., blood uptake fraction).

2.3 For inhaled radionuclides, determining the fraction of initial intake that is transferred to the blood involves an evaluation of the deposition in the three compartments of the lung and the subsequent time-dependent transfer to the body fluids and to the GI tract. Unless it is known otherwise, it should be assumed that the transfer from the lung to body fluids and from lung to GI tract to body fluids follows the ICRP 30 (Ref. 3) modeling (which is the basis for this guide).

2.4 For simplicity and conservatism in the modeling, the total uptake into the blood from the maternal intake is assumed to be instantaneous. However, for radionuclides with lung clearance class of W (10- to 100-day half-life clearance) or Y (greater than 100-day half-life clearance), the actual translocation from the lung and uptake in the blood may occur over a time period that exceeds the gestation period. Clearance from the lung may take up to several years. All the initially deposited material is not immediately available for uptake by the first transfer compartment (blood). However, an incremental transfer from the lung to the blood may be assessed based on the lung model as described in ICRP Publications 30 and 19 (Refs. 3 and 5).[2]

Table 1, adapted from the data in Figure 5.2 of ICRP 30 (Ref. 3), may be used for determining the total transfer from the lung to the first transfer

[1]The committed dose equivalent factors for the uterus presented in Appendix A were calculated based on the modeling employed during the development of the ICRP 30 (Ref. 3) data. It is recognized that the metabolism of the pregnant woman may not be adequately represented by the standard metabolic model. However, partly because of the lack of more definitive data, this modeling has been used for determining the dose commitment factors for the uterus that may be used for evaluating compliance with the embryo/fetus dose limit.

[2]As modeled in ICRP Publications 19 and 30, the clearance from the different lung compartments is assumed to follow first-order kinetics. This approach is complex, involving interlinking differential equations, and is considered outside the scope of a routine operational health physics program.

compartment (i.e., blood), where f_1 is the blood uptake fraction from Appendix B.[3] The lung clearance class (D, W, or Y) for a particular chemical form of a particular radionuclide may be obtained from Appendix B to 10 CFR 20.1001–20.2401.

Table 1	
Transfer Fraction of Inhaled Activity to First Transfer Compartment	
Class	**Transfer Fraction (TF)**
D	$0.48 + 0.15\ f_1$
W	$0.12 + 0.51\ f_1$
Y	$0.05 + 0.58\ f_1$

2.5 Based on the determination of the maternal intake, the dose to the embryo/fetus for the entire gestation period should be calculated using the following equations:

For ingestion intakes:

$$DE = \Sigma\ I_i \times f_{1,i} \times DF_i \qquad \text{(Equation 1)}$$

For inhalation intakes:

$$DE = \Sigma\ I_i \times TF_i \times DF_i \qquad \text{(Equation 2)}$$

here:

DE = dose equivalent to the embryo/fetus for the entire gestation period from the acute intakes of all radionuclides during the gestation period (rem)

I_i = intake of radionuclide i by the declared pregnant woman at any time during the gestation period (μCi)

DF_i = dose factor for use in approximating the dose equivalent to the embryo/fetus for the entire gestation period from the introduction of unit activity (1 μCi) into the maternal blood at any time during the gestation period, from tabular data presented in Appendix A to this guide (rem/μCi in maternal blood)

$f_{1,i}$ = the fraction of radionuclide i reaching the body fluids following ingestion (i.e., the fraction of ingested activity of radionuclide i that enters the blood), from data presented in Appendix B to this guide

TF_i = transfer fraction of inhaled activity to the first transfer compartment (i.e., the fraction of inhaled activity of radionuclide i that enters the blood, see Table 1 of this guide)

2.6 For pre-existing body burdens, the total burden determined to exist at time of pregnancy should be assumed to be available for uptake in the blood of the woman. The dose should be assigned to the embryo/fetus as if the maternal blood uptake occurs within the first month of pregnancy. The embryo/fetus dose is calculated by multiplying the maternal burden of the radionuclide by its dose factor from Appendix A using the equation:

$$DE = \Sigma\ A_i \times DF_i \qquad \text{(Equation 3)}$$

where:

DE = dose equivalent to the embryo/fetus

A_i = maternal burden existing at time of pregnancy (μCi)

DF_i = dose conversion factor (Appendix A)

This method provides a simplified and conservative approach for evaluating the significance of pre-existing conditions. If the embryo/fetus is likely to receive a dose in excess of 25% of the limit from pre-existing burdens (i.e., greater than 0.125 rem (1.25 mSv)), more detailed modeling should be considered.[4]

2.7 Doses from multiple nuclides or multiple intakes should be evaluated on a frequency corresponding to the determination of the intake. Multiple dose determinations should be added to determine the total dose. Doses may need to be reevaluated if better estimates of intakes are provided by followup bioassay measurements.

3. DETERMINING GESTATION-TIME DEPENDENT DOSE TO THE EMBRYO/FETUS USING REVISION 1 TO NUREG/CR-5631 METHODS

As an alternative to the simplified methods presented above, a gestation-time dependent dose to the embryo/fetus may be calculated for the radionuclides addressed in Revision 1 to NUREG/CR-5631 (Ref. 2). Revision 1 to NUREG/CR-5631 presents dosimetric methods for calculating the dose to the

[3]The coefficients for the transfer fraction equations in Table 1 re applicable to particles with a 1-micrometer activity median aerodynamic diameter (AMAD). As a default, these equations may be used for all particle sizes. However, if the actual particle size distribution is known, transfer fractions for other AMAD particle sizes may be derived from data in Figure 5.2 of ICRP 30 (Ref. 3).

[4]This approach for evaluating pre-existing body burdens does not specifically address time-dependent releases as could occur for certain radionuclides with both a long biological retention and radiological half-life. However, the assumption of blood uptake of the total burden in the first month of the gestation period provides a simple method with reasonable assurance that any actual dose to the embryo/fetus will not be significantly underestimated. More detailed evaluations may be needed for unusual circumstances in which a pre-existing body burden could present a significant source of exposure to the embryo/fetus. An evaluation of this nature should be conducted by individuals knowledgeable in the area of internal dosimetry. Such a detailed evaluation could consider the element retention functions as presented in ICRP Publications 30 and 54 (Refs. 3 and 6). Also, the modeling presented in Revision 1 to NUREG/CR-5631 (Ref. 2) could be applied. The details of this type of an evaluation are beyond the types of analyses that are considered routinely required and, as such, are outside the scope of this guide.

embryo/fetus following the instantaneous introduction of unit activity into the first transfer compartment (blood) of the pregnant woman at successive stages of gestation. These methods include the contribution to the embryo/fetus dose from the resultant body burdens of the declared pregnant woman and from activity in the embryo/fetus resulting from transfer across the placenta. Refer to Revision 1 to NUREG/CR-5631 (Ref. 2) for a detailed description of the modeling.

The methods and data of Revision 1 to NUREG/CR-5631 (Ref. 2) may be used for determining the dose to the embryo/fetus from maternal intakes at successive stages of gestation for the radionuclides ^{3}H, ^{14}C, ^{57}Co, ^{58}Co, ^{60}Co, ^{89}Sr, ^{90}Sr, ^{106}Ru, ^{125}I, ^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I, ^{134}Cs, ^{137}Cs, ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, and ^{241}Am.

The steps for determining the embryo/fetus dose using the Revision 1 to NUREG/CR-5631 (Ref. 2) methods are as follows:

3.1 The methods presented in the regulatory position in Sections 2.1 through 2.4 should be used for determining the uptake in the first transfer compartment (blood) of the declared pregnant woman.

3.2 Equations 1 and 2 of the regulatory position specified in Section 2.5 may be used for determining the embryo/fetus dose with the following clarifications:

3.2.1 For Equations 1 and 2, in place of the dose factor parameter, DF_i, the dose values should be taken from Appendix C to this guide for the time period representing the time of intake relative to stage of gestation. The data in Appendix C to this guide are for an absorbed dose (in rads) from the introduction of 1 μCi of the radionuclide into the first transfer compartment (blood) of the woman at the beginning of the specified month of gestation. To convert from an absorbed dose (rad) to a dose equivalent (rem), the data in Appendix C should be multiplied by the appropriate quality factor from Table 1004(b).1 of 10 CFR Part 20. For ^{3}H, ^{14}C, ^{57}Co, ^{58}Co, ^{60}Co, ^{89}Sr, ^{90}Sr, ^{106}Ru, ^{125}I, ^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I, ^{134}Cs, and ^{137}Cs, a quality factor of 1 should be applied. For ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, and ^{241}Am, a quality factor of 20 should be applied, recognizing that most of the embryo/fetus dose results from alpha decay.

For some radionuclides (e.g., ^{235}U), a blood uptake at the beginning of the gestation period results in a negligible dose contribution to the embryo/fetus. These radionuclides are identified in the tables in Appendix C to this guide by an "N" entry in the row for the 0-day of gestation at radionuclide introduction (i.e., the first row of dose factor data). For an intake of these radionuclides within the first month of gestation, a time-weighted dose factor using the second month data (31-day row) should be used. The 31-day dose factor should be multiplied by the quotient of the days-to-date in the first gestation month at time of intake divided by 30 days. For example, assuming a maternal intake of ^{14}C resulting in a 1-μCi blood uptake on the 20th day of the pregnancy, the embryo/fetus dose should be determined by multiplying the cumulated dose from an intake at day 31 (i.e., Table C3, Cumulated Dose column, 1.89E-04 rads) by the ratio of 20 days to 30 days (i.e., 20 divided by 30).

3.2.2 For using the tabular dose data in calculating the embryo/fetus dose, it may be assumed that all intakes occurring within any of the 30-day periods of gestation occur at the beginning of that period.[5] The cumulated dose column should be used in order to determine the total dose for the remainder of the gestation period.

3.2.3 For pre-existing body burdens from occupational exposure, the total burden determined to exist at time of pregnancy should be assumed to be available for uptake in the blood of the woman. The dose should be assigned to the embryo/fetus as if the maternal blood uptake occurs within the first month of pregnancy. The embryo/fetus dose is calculated by multiplying the maternal burden of the radionuclide by its dose factor (Equation 3). The dose factor to be used from the Appendix C tables is that factor corresponding to the cumulated dose for a 0-day of gestation at radionuclide introduction (i.e., right-most column, first data entry). However, for those radionuclides with an "N" for this 0-day entry, the entry for the second gestation month should be used (i.e., the right-most column, second data entry). Alternatively, time-dependent release kinetics may be used for calculating that fraction of the body burden that is translocated to the blood through the duration of the pregnancy. The time-dependent release is described in ICRP Publications 30 and 54 (Refs. 3 and 6). This approach is complex, involving interlinking differential equations, and is considered outside the scope of a routine health physics program.

3.3 Doses from multiple nuclides and multiple intakes should be evaluated with a frequency corresponding to the intake (i.e., at least once every 30 days). Multiple dose determinations should be added to determine the total dose. Doses may need to be reevaluated if better estimates of intakes are provided by followup bioassay 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 an acceptable alternative method of complying with specified portions of the Commission's regulations, the methods described in this guide will be used

[5]The correlation of intake to actual stage of gestation can only be roughly estimated. For this reason, it is believed that the correlation should be limited to the best estimate of the month of gestation.

in the evaluation of applications for new licenses, license renewals, and license amendments and for evaluating compliance with 10 CFR 20.1001–0.2401.

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6. International Commission on Radiological Protection, "Individual Monitoring for Intake of Radionuclides by Workers: Design and Interpretation," ICRP No. 54, Annals of the ICRP, Volume 19, No. 1–3, Pergamon Press Inc., 1988.

APPENDIX A

DOSE EQUIVALENT FACTORS FOR USE IN APPROXIMATING THE EMBRYO/FETUS DOSE FROM RADIONUCLIDES IN MATERNAL BLOOD

Except as noted, the dose factors (DF_i) presented in Table A–1 represent the committed dose equivalent to the uterus per introduction of unit activity into the first transfer compartment (i.e., blood) of the woman. These entries were calculated from tabulated values of uterine committed dose equivalent per unit intake and fractional absorption (f_1) from the gastrointestinal tract using ICRP–30 (Ref. A1) methodology. The DF_i dose factors were derived by dividing the committed dose equivalent per unit intake by the fractional absorption factor (f_1). These dose factors are based on unit activity in the blood. The most conservative f_1 (i.e., largest fraction) for each radionuclide has been used for deriving the data in Table A–1.

For the radionuclides ^{3}H, ^{14}C, ^{57}Co, ^{58}Co, ^{60}Co, ^{89}Sr, ^{90}Sr, ^{106}Ru, ^{125}I, ^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I, ^{134}Cs, ^{137}Cs, ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, and ^{241}Am, the dose factors in Table A–1 represent the maximum dose equivalent to the embryo/fetus for the gestation period from the introduction of unit activity into the first transfer compartment of the woman at any time during the gestation period. These entries are based on the modeling of Revision 1 to NUREG/CR–5631 (Ref. A2) and are derived from the data tables presented in Appendix C to this guide. The maximum calculated embryo/fetus dose (as presented in the Appendix C tables) from intake by the declared pregnant woman during the gestation period has been used for inclusion in Table A–1.

The dose factor data presented in Revision 1 to NUREG/CR–5631 (Ref. A2) are for an absorbed dose expressed in units of rads. To adapt these data as presented in Appendix C to this guide for inclusion in Table A–1, appropriate quality factors have been applied to convert from rads to dose equivalent, expressed in units of rem. For beta- and gamma-emitting radionuclides, a quality factor of 1 has been applied. For ^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, and ^{241}Am, a quality factor of 20 has been applied, recognizing that most of the embryo/fetus dose results from the alpha decay.

TABLE A-1

Dose Equivalent Factors for Use in Approximating the Embryo/Fetus Dose from Radionuclides in Maternal Blood

Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)
H-3	5.87E-05*	Cr-51	6.96E-04	Ga-68	5.66E-02
Be-7	1.67E-02	Mn-51	3.65E-04	Ga-70	8.99E-05
Be-10	1.79E-02	Mn-52	4.70E-02	Ga-72	1.53E+00
C-11	1.21E-05	Mn-52m	2.80E-04	Ga-73	9.36E-02
C-14	1.29E-03*	Mn-53	5.77E-05	Ge-66	1.42E-04
F-18	1.32E-05	Mn-54	1.86E-02	Ge-67	1.11E-05
Na-22	1.06E-02	Mn-56	2.18E-03	Ge-68	8.81E-04
Na-24	1.21E-03	Fe-52	1.30E-02	Ge-69	3.02E-04
Mg-28	3.83E-03	Fe-55	3.88E-03	Ge-71	6.99E-06
Al-26	5.33E-01	Fe-59	4.63E-02	Ge-75	1.61E-05
Si-31	3.85E-05	Fe-60	1.47E+00	Ge-77	3.40E-04
Si-32	4.33E-02	Co-55	4.01E-03	Ge-78	1.08E-04
P-32	3.03E-03	Co-56	3.43E-02	As-69	2.46E-05
P-33	4.33E-04	Co-57	2.20E-03*	As-70	2.90E-04
S-35	3.53E-04	Co-58	9.17E-03*	As-71	1.21E-03
Cl-36	2.96E-03	Co-58m	5.17E-05	As-72	2.70E-03
Cl-38	3.17E-05	Co-60	4.18E-02*	As-73	3.02E-04
Cl-39	3.89E-05	Co-60m	4.12E-07	As-74	2.90E-03
K-40	1.84E-02	Co-61	4.50E-05	As-76	1.11E-03
K-42	7.73E-04	Co-62m	5.33E-05	As-77	1.88E-04
K-43	7.10E-04	Ni-56	5.39E-02	As-78	1.85E-04
K-44	1.94E-05	Ni-57	3.60E-02	Se-70	1.61E-04
K-45	1.21E-05	Ni-59	2.71E-03	Se-73	3.66E-04
Ca-41	3.21E-05	Ni-63	6.29E-03	Se-73m	3.21E-05
Ca-45	6.61E-04	Ni-65	1.43E-03	Se-75	8.79E-03
Ca-47	5.18E-03	Ni-66	2.81E-03	Se-79	4.19E-03
Sc-43	2.48E+00	Cu-60	9.32E-05	Se-81	1.00E-06
Sc-44	4.59E+00	Cu-61	2.69E-04	Se-81m	1.46E-05
Sc-44m	2.56E+01	Cu-64	2.09E-04	Se-83	3.62E-05
Sc-46	3.15E+01	Cu-67	6.50E-04	Br-74	3.33E-05
Sc-47	1.86E+00	Zn-62	1.38E-03	Br-74m	6.18E-05
Sc-48	3.52E+01	Zn-63	5.92E-05	Br-75	6.07E-05
Sc-49	4.18E-04	Zn-65	3.49E-02	Br-76	1.20E-03
Ti-44	1.36E+00	Zn-69	3.09E-06	Br-77	3.27E-04
Ti-45	1.54E-02	Zn-69m	5.54E-04	Br-80	3.01E-06
V-47	2.29E-03	Zn-71m	5.75E-04	Br-80m	1.46E-04
V-48	4.37E-01	Zn-72	5.28E-03	Br-82	1.87E-03
V-49	8.36E-05	Ga-65	9.18E-03	Br-83	2.72E-05
Cr-48	5.77E-03	Ga-66	9.95E-01	Br-84	2.56E-05
Cr-49	3.51E-04	Ga-67	2.50E-01	Rb-79	1.15E-05

*Dose equivalent factor based on data presented in Revision 1 to NUREG/CR-5631 (Ref. A2). All other factors represent the committed dose equivalent to the uterus.

TABLE A-1 (continued)

Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)
Rb-81	8.18E-05	Nb-90	2.39E-01	Rh-105	1.93E-03
Rb-81m	1.08E-05	Nb-93m	9.29E-04	Rh-106m	6.86E-03
Rb-82m	3.49E-04	Nb-94	3.04E-01	Rh-107	8.51E-05
Rb-83	7.07E-03	Nb-95	1.24E-01	Pd-100	3.94E-01
Rb-84	1.05E-02	Nb-95m	1.27E-02	Pd-101	3.33E-02
Rb-86	8.14E-03	Nb-96	2.03E-01	Pd-103	1.39E-03
Rb-87	4.22E-03	Nb-97	4.11E-03	Pd-107	7.33E-06
Rb-88	1.02E-05	Nb-98	9.66E-03	Pd-109	1.27E-03
Rb-89	1.20E-05	Mo-90	7.77E-04	Ag-102	3.76E-04
Sr-80	3.96E-04	Mo-93	4.36E-04	Ag-103	8.58E-04
Sr-81	1.22E-04	Mo-93m	4.76E-04	Ag-104	3.05E-03
Sr-82	1.25E-02	Mo-99	9.39E-04	Ag-104m	1.09E-03
Sr-83	2.31E-03	Mo-101	1.48E-05	Ag-105	1.94E-02
Sr-85	4.03E-03	Tc-93	1.33E-04	Ag-106	2.12E-04
Sr-85m	4.81E-05	Tc-93m	4.67E-05	Ag-106m	8.21E-02
Sr-87m	1.62E-04	Tc-94	4.56E-04	Ag-108m	6.59E-02
Sr-89	1.84E-02*	Tc-94m	7.08E-05	Ag-110m	1.04E-01
Sr-90	5.22E-02*	Tc-95	3.86E-04	Ag-111	1.41E-03
Sr-91	1.49E-03	Tc-95m	1.23E-03	Ag-112	2.18E-03
Sr-92	7.79E-04	Tc-96	2.62E-03	Ag-115	1.98E-04
Y-86	2.18E+01	Tc-96m	2.29E-05	Cd-104	3.30E-03
Y-86m	1.26E+00	Tc-97	4.67E-05	Cd-107	1.95E-04
Y-87	1.01E+01	Tc-97m	2.42E-04	Cd-109	2.12E-02
Y-88	3.96E+01	Tc-98	2.97E-03	Cd-113	2.77E-01
Y-90	4.66E-04	Tc-99	2.79E-04	Cd-113m	2.55E-01
Y-90m	1.21E+00	Tc-99m	3.32E-05	Cd-115	9.47E-03
Y-91	6.03E-02	Tc-101	2.96E-06	Cd-115m	1.27E-02
Y-91m	2.13E-01	Tc-104	2.07E-05	Cd-117	4.23E-03
Y-92	4.81E-01	Ru-94	2.32E-03	Cd-117m	9.62E-03
Y-93	4.18E-01	Ru-97	6.89E-03	In-109	7.95E-03
Y-94	1.10E-01	Ru-103	1.97E-02	In-110	4.01E-02
Y-95	3.56E-02	Ru-105	4.09E-03	In-110	4.50E-03
Zr-86	8.62E-01	Ru-106	7.23E-03*	In-111	3.05E-02
Zr-88	3.87E-01	Rh-99	2.19E-02	In-112	9.47E-05
Zr-89	7.31E-01	Rh-99m	3.51E-03	In-113m	1.24E-03
Zr-93	8.79E-05	Rh-100	3.86E-02	In-114m	3.05E-02
Zr-95	6.16E-01	Rh-101	3.33E-02	In-115	8.99E-01
Zr-97	5.24E-01	Rh-101m	9.40E-03	In-115m	2.16E-03
Nb-88	1.17E-03	Rh-102	1.93E-01	In-116m	4.92E-03
Nb-89	1.83E-02	Rh-102m	3.48E-02	In-117	1.22E-03
Nb-89	1.30E-02	Rh-103m	1.18E-06	In-117m	2.61E-03

*Dose equivalent factor based on data presented in Revision 1 to NUREG/CR-5631 (Ref. A2). All other factors represent the committed dose equivalent to the uterus.

TABLE A-1 (continued)

Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)
In-119m	1.39E-05	Te-127m	1.82E-03	Ba-131m	1.32E-05
Sn-110	2.11E-02	Te-129	2.35E-05	Ba-133	1.27E-02
Sn-111	8.81E-04	Te-129m	3.39E-03	Ba-133m	8.77E-04
Sn-113	2.63E-02	Te-131	2.18E-04	Ba-135m	7.03E-04
Sn-117m	1.57E-02	Te-131m	6.64E-03	Ba-139	4.55E-05
Sn-119m	2.29E-03	Te-132	8.57E-03	Ba-140	1.54E-02
Sn-121	3.70E-05	Te-133	3.26E-05	Ba-141	9.47E-05
Sn-121m	5.70E-03	Te-133m	5.48E-04	Ba-142	2.74E-04
Sn-123	6.35E-03	Te-134	3.98E-04	La-131	3.77E-02
Sn-123m	2.48E-04	I-120	9.36E-05	La-132	5.07E-01
Sn-125	2.37E-02	I-120m	8.73E-05	La-135	3.43E-02
Sn-126	2.35E-01	I-121	1.79E-05	La-137	7.55E-02
Sn-127	1.14E-02	I-123	2.27E-05	La-138	2.84E+00
Sn-128	7.14E-03	I-124	2.16E-04	La-140	2.32E+00
Sb-115	2.00E-04	I-125	1.38E-03*	La-141	9.43E-03
Sb-116	1.59E-04	I-126	2.23E-04	La-142	1.91E-01
Sb-116m	1.49E-03	I-128	5.25E-06	La-143	2.85E-03
Sb-117	3.34E-04	I-129	5.11E-04	Ce-134	3.13E+00
Sb-118m	6.59E-03	I-130	2.29E-04	Ce-135	4.44E+00
Sb-119	2.08E-04	I-131	3.64E-03*	Ce-137	7.13E-02
Sb-120	3.70E-05	I-132	1.56E-04*	Ce-137m	3.31E-01
Sb-120	3.42E-02	I-132m	6.14E-05	Ce-139	1.15E+00
Sb-122	5.85E-03	I-133	9.04E-04*	Ce-141	5.56E-01
Sb-124	2.98E-02	I-134	4.83E-05*	Ce-143	1.05E+00
Sb-124m	4.88E-05	I-135	3.72E-04*	Ce-144	3.79E-01
Sb-125	8.51E-03	Cs-125	1.33E-05	Pr-136	4.12E-02
Sb-126	4.37E-02	Cs-127	5.96E-05	Pr-137	1.26E-01
Sb-126m	1.69E-04	Cs-129	2.13E-04	Pr-138m	9.61E-01
Sb-127	9.66E-03	Cs-130	6.99E-06	Pr-139	1.16E-01
Sb-128	1.33E-04	Cs-131	2.27E-04	Pr-142	1.36E-01
Sb-128	8.73E-03	Cs-132	2.10E-03	Pr-142m	1.73E-03
Sb-129	3.36E-03	Cs-134	1.11E-01*	Pr-143	4.53E-08
Sb-130	9.40E-04	Cs-134m	2.66E-05	Pr-144	8.44E-04
Sb-131	3.36E-04	Cs-135	7.07E-03	Pr-145	1.41E-02
Te-116	1.45E-03	Cs-135m	2.42E-05	Pr-147	1.95E-02
Te-121	4.87E-03	Cs-136	1.42E-02	Nd-136	3.59E-01
Te-121m	7.90E-03	Cs-137	5.94E-02*	Nd-138	8.26E-01
Te-123	3.09E-05	Cs-138	2.95E-05	Nd-139	4.11E-02
Te-123m	2.94E-03	Ba-126	1.14E-03	Nd-139m	1.74E+00
Te-125m	9.75E-04	Ba-128	1.17E-02	Nd-141	4.33E-02
Te-127	6.31E-05	Ba-131	7.40E-03	Nd-147	8.45E-01

*Dose equivalent factor based on data presented in Revision 1 to NUREG/CR-5631 (Ref. A2). All other factors represent the committed dose equivalent to the uterus.

TABLE A-1 (continued)

Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)
Nd-149	1.37E-01	Gd-149	2.47E+00	Tm-166	2.37E+00
Nd-151	2.53E-02	Gd-151	4.99E-01	Tm-167	1.03E+00
Pm-141	3.63E-02	Gd-152	0.00E-01	Tm-170	5.38E-02
Pm-143	1.79E+00	Gd-153	8.92E-01	Tm-171	8.13E-03
Pm-144	8.68E+00	Gd-159	1.52E-01	Tm-172	1.89E+00
Pm-145	2.58E-01	Tb-147	6.76E-01	Tm-173	5.88E-01
Pm-146	4.34E+00	Tb-149	1.27E+00	Tm-175	2.70E-02
Pm-147	3.49E-05	Tb-150	1.01E+00	Yb-162	8.97E-02
Pm-148	2.60E+00	Tb-151	2.33E+00	Yb-166	6.08E+00
Pm-148m	1.08E+01	Tb-153	1.16E+00	Yb-167	1.23E-02
Pm-149	4.70E-02	Tb-154	5.65E+00	Yb-169	2.47E+00
Pm-150	6.86E-01	Tb-155	9.52E-01	Yb-175	2.10E-01
Pm-151	1.11E+00	Tb-156	8.65E+00	Yb-177	6.98E-02
Sm-141	4.11E-02	Tb-156m	9.32E-01	Yb-178	4.11E-02
Sm-141m	1.42E-01	Tb-156m	2.89E-01	Lu-169	3.60E+00
Sm-142	2.11E-01	Tb-157	2.39E-02	Lu-170	8.42E+00
Sm-145	5.56E-01	Tb-158	4.79E+00	Lu-171	3.72E+00
Sm-146	0.00E-01	Tb-160	6.08E+00	Lu-172	9.20E+00
Sm-147	0.00E-01	Tb-161	2.64E-01	Lu-173	1.10E+00
Sm-151	1.26E-05	Dy-155	1.08E+00	Lu-174	8.93E-01
Sm-153	3.54E-01	Dy-157	5.81E-01	Lu-174m	5.54E-01
Sm-155	5.65E-03	Dy-159	4.19E-01	Lu-176	3.45E+00
Sm-156	3.55E-01	Dy-165	1.38E-02	Lu-176m	1.53E-0
Eu-145	2.00E+00	Dy-166	3.56E-01	Lu-177	2.24E-01
Eu-146	3.38E+00	Ho-155	1.41E-01	Lu-177m	6.80E+00
Eu-147	8.51E-01	Ho-157	2.57E-02	Lu-178	8.18E-03
Eu-148	3.53E+00	Ho-159	3.47E-02	Lu-178m	5.54E-02
Eu-149	1.40E-01	Ho-161	4.70E-02	Lu-179	3.03E-02
Eu-150	2.92E-02	Ho-162	4.66E-03	Hf-170	4.74E-01
Eu-150	3.02E+00	Ho-162m	1.43E-01	Hf-172	4.63E-01
Eu-152	2.20E+00	Ho-164	3.10E-03	Hf-173	2.26E-01
Eu-152m	1.38E-01	Ho-164m	1.32E-02	Hf-175	3.70E-01
Eu-154	2.28E+00	Ho-166	1.04E-01	Hf-177m	5.22E-02
Eu-155	1.60E-01	Ho-166m	1.07E+01	Hf-178m	2.94E+00
Eu-156	1.90E+00	Ho-167	2.38E-01	Hf-179m	8.51E-01
Eu-157	2.01E-01	Er-161	6.29E-01	Hf-180m	.71E-01
Eu-158	3.56E-02	Er-165	1.12E-01	Hf-181	4.96E-01
Gd-145	1.09E-01	Er-169	1.34E-04	Hf-182	1.16E+00
Gd-146	4.11E+00	Er-171	5.88E-01	Hf-182m	2.61E-02
Gd-147	4.91E+00	Er-172	2.59E+00	Hf-183	2.33E-02
Gd-148	0.00E-01	Tm-162	6.87E-02	Hf-184	1.94E-01

TABLE A-1 (continued)

Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)
Ta-172	4.07E-02	Os-189m	5.11E-06	Hg-193m	3.23E-04
Ta-173	1.94E-01	Os-191	1.99E-02	Hg-194	1.81E-01
Ta-174	4.25E-02	Os-191m	1.12E-03	Hg-195	7.47E-05
Ta-175	4.96E-01	Os-193	8.55E-03	Hg-195m	5.48E-04
Ta-176	8.25E-01	Os-194	8.69E-02	Hg-197	2.38E-04
Ta-177	1.30E-01	Ir-182	2.23E-03	Hg-197m	2.97E-04
Ta-178	1.47E-01	Ir-184	3.24E-02	Hg-199m	7.55E-06
Ta-179	9.40E-02	Ir-185	3.85E-02	Hg-203	5.33E-03
Ta-180	1.16E+00	Ir-186	1.12E-01	Tl-194	6.44E-06
Ta-180m	3.47E-02	Ir-187	2.08E-02	Tl-194m	2.16E-05
Ta-182	2.15E+00	Ir-188	1.60E-01	Tl-195	3.49E-05
Ta-182m	2.65E-03	Ir-189	1.96E-02	Tl-197	3.85E-05
Ta-183	5.44E-01	Ir-190	2.52E-01	Tl-198	1.94E-04
Ta-184	7.40E-01	Ir-190m	1.01E-03	Tl-198m	8.36E-05
Ta-185	9.25E-03	Ir-192	1.63E-01	Tl-199	5.55E-05
Ta-186	7.03E-03	Ir-192m	8.99E-02	Tl-200	6.55E-04
W-176	6.55E-04	Ir-194	7.55E-03	Tl-201	2.48E-04
W-177	3.66E-04	Ir-194m	4.55E-01	Tl-202	1.38E-03
W-178	6.43E-04	Ir-195	1.24E-03	Tl-204	2.43E-03
W-179	8.12E-06	Ir-195m	1.03E-02	Pb-195m	1.65E-04
W-181	2.80E-04	Pt-186	2.06E-02	Pb-198	3.92E-04
W-185	3.51E-07	Pt-188	1.21E-01	Pb-199	6.51E-04
W-187	1.04E-03	Pt-189	2.08E-02	Pb-200	3.37E-03
W-188	1.68E-04	Pt-191	4.88E-02	Pb-201	1.78E-03
Re-177	1.49E-05	Pt-193	1.07E-04	Pb-202	6.77E-02
Re-178	8.37E-06	Pt-193m	2.71E-03	Pb-202m	1.91E-03
Re-181	4.61E-04	Pt-195m	1.58E-02	Pb-203	2.02E-03
Re-182	4.56E-04	Pt-197	2.64E-03	Pb-205	3.63E-04
Re-182	1.92E-03	Pt-197m	1.12E-03	Pb-209	9.93E-06
Re-184	1.64E-03	Pt-199	5.40E-04	Pb-210	2.31E+00
Re-184m	1.31E-03	Pt-200	2.04E-02	Pb-211	3.63E-04
Re-186	4.53E-04	Au-193	1.63E-03	Pb-212	3.29E-02
Re-186m	9.43E-04	Au-194	1.10E-02	Pb-214	5.64E-04
Re-187	1.82E-06	Au-195	2.35E-03	Bi-200	1.66E-03
Re-188	3.73E-04	Au-198	5.66E-03	Bi-201	4.07E-03
Re-188m	8.19E-06	Au-198m	1.05E-02	Bi-202	4.83E-03
Re-189	2.46E-04	Au-199	1.68E-03	Bi-203	2.54E-02
Os-180	1.78E-03	Au-200	1.01E-04	Bi-205	4.82E-02
Os-181	1.75E-02	Au-200m	1.61E-02	Bi-206	9.03E-02
Os-182	1.07E-01	Au-201	1.15E-05	Bi-207	4.88E-02
Os-185	1.33E-01	Hg-193	4.88E-05	Bi-210	1.46E-03

TABLE A-1 (continued)

Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)	Nuclide	DF_i (rem/μCi)
Bi-210m	8.66E-02	U-233	5.84E-01*	Am-245	2.68E-04
Bi-212	1.70E-03	U-234	5.84E-01*	Am-246m	1.51E-02
Bi-213	4.36E-04	U-235	5.34E-01*	Am-246	2.03E-02
Bi-214	3.52E-04	U-236	1.81E-01	Cm-238	1.31E-01
Po-203	1.07E-03	U-237	5.42E-03	Cm-240	3.50E-02
Po-205	1.64E-03	U-238	5.10E-01*	Cm-241	8.69E-01
Po-207	4.03E-03	U-239	5.52E-05	Cm-242	3.30E-02
Po-210	3.05E+00	U-240	4.17E-03	Cm-243	3.74E-01
At-207	8.32E-04	Np-232	8.69E-03	Cm-244	3.19E-02
At-211	3.92E-02	Np-233	2.85E-03	Cm-245	3.11E-01
Fr-222	2.13E-03	Np-234	1.45E+00	Cm-246	1.27E-01
Fr-223	8.58E-03	Np-235	2.99E-03	Cm-247	9.51E-01
Ra-223	7.84E-01	Np-236	4.29E-01	Cm-248	3.49E+01
Ra-224	3.85E-01	Np-236	5.25E-02	Cm-249	1.07E-03
Ra-225	6.23E-01	Np-237	3.59E-01	Cm-250	2.76E+02
Ra-226	1.69E+00	Np-238	6.07E-01	Bk-245	4.11E-01
Ra-227	6.10E-05	Np-239	2.55E-01	Bk-246	1.04E+00
Ra-228	2.90E+00	Np-240	7.07E-02	Bk-247	2.83E-01
Ac-224	9.47E-02	Pu-234	1.24E-01	Bk-249	8.40E-04
Ac-225	3.68E-01	Pu-235	1.72E-03	Bk-250	1.54E-01
Ac-226	1.66E-01	Pu-236	6.81E-02	Cf-244	9.25E-05
Ac-227	2.60E-01	Pu-237	1.07E-01	Cf-246	2.88E-02
Ac-228	3.12E-01	Pu-238	1.11E+00*	Cf-248	4.18E-02
Th-226	3.02E-03	Pu-239	1.04E+00*	Cf-249	9.80E-01
Th-227	3.52E+00	Pu-240	2.80E-02	Cf-250	3.30E-01
Th-228	4.40E+01	Pu-241	2.96E-04	Cf-251	4.26E-01
Th-229	8.51E+01	Pu-242	2.81E-02	Cf-252	1.15E+01
Th-230	1.26E+01	Pu-243	9.62E-03	Cf-253	8.55E-04
Th-231	8.97E-02	Pu-244	1.07E+00	Cf-254	3.70E+02
Th-232	2.26E+01	Pu-245	2.22E-01	Es-250	4.77E-02
Th-234	2.33E-01	Pu-246	1.34E+00	Es-251	1.24E-01
Pa-227	2.42E-03	Am-237	2.60E-02	Es-253	3.58E-02
Pa-228	9.58E-01	Am-238	7.81E-02	Es-254m	5.22E-01
Pa-230	1.04E+00	Am-239	1.63E-01	Es-254	1.33E+00
Pa-231	2.25E-01	Am-240	1.16E+00	Fm-252	2.61E-02
Pa-232	8.95E-01	Am-241	2.22E-01*	Fm-253	1.38E-01
Pa-233	3.81E-01	Am-242m	3.64E-02	Fm-254	6.11E-03
Pa-234	6.77E-01	Am-242	1.32E-02	Fm-255	2.85E-02
U-230	6.13E-01	Am-243	4.74E-01	Fm-257	2.60E-01
U-231	2.63E-03	Am-244m	1.05E-05	Md-257	3.69E-02
U-232	6.02E-01	Am-244	3.92E-01	Md-258	5.96E-02

*Dose equivalent factor based on data presented in Revision 1 to NUREG/CR-5631 (Ref. A2). All other factors represent the committed dose equivalent to the uterus.

REFERENCES

A1. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP No. 30, Parts 1 through 4, including supplements, Annals of the ICRP, Volume 2, No. 3/4, Pergamon Press Inc., 1979.

A2. M. R. Sikov et al., "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Dose—Interim Recommendations," NUREG/CR-5631, Revision 1 (PNL-7445), U.S. Nuclear Regulatory Commission, March 1992.

APPENDIX B

BLOOD UPTAKE FRACTIONS FOR INGESTED ACTIVITY

Element	f_1	Element	f_1
Actinium (Ac)	1E-3	Einsteinium (Es)	1E-3
Aluminum (Al)	1E-2	Erbium (Er)	3E-4
Americium (Am)	1E-3	Europium (Eu)	1E-3
Antimony (Sb)	1E-1	Fermium (Fm)	1E-3
Arsenic (As)	5E-1	Fluorine (F)	1E0
Astatine (At)	1E0	Francium (Fr)	1E0
Barium (Ba)	1E-1	Gadolinium (Gd)	3E-4
Berkelium (Bk)	1E-3	Gallium (Ga)	1E-3
Beryllium (Be)	5E-3	Germanium (Ge)	1E0
Bismuth (Bi)	5E-2	Gold (Au)	1E-1
Bromine (Br)	1E0	Hafnium (Hf)	2E-3
Cadmium (Cd)	5E-2	Holmium (Ho)	3E-4
Calcium (Ca)	3E-1	Hydrogen (H)	1E0
Californium (Cf)	1E-3	Indium (In)	2E-2
Carbon (C)	1E0	Iodine (I)	1E0
Cerium (Ce)	3E-4	Iridium (Ir)	1E-2
Cesium (Cs)	1E0	Iron (Fe)	1E-1
Chlorine (Cl)	1E0	Lanthanum (La)	1E-3
Chromium (Cr)	1E-1	Lead (Pb)	2E-1
Cobalt (Co)	3E-1	Lutetium (Lu)	3E-4
Copper (Cu)	5E-1	Magnesium (Mg)	5E-1
Curium (Cm)	1E-3	Manganese (Mn)	1E-1
Dysprosium (Dy)	3E-4	Mendelevium (Md)	1E-3

Element	f_1	Element	f_1
Mercury (Hg)	1E0	Selenium (Se)	8E–1
Molybdenum (Mo)	8E–1	Silicon (Si)	1E–2
Neodymium (Nd)	3E–4	Silver (Ag)	5E–2
Neptunium (Np)	1E–3	Sodium (Na)	1E0
Nickel (Ni)	5E–2	Strontium (Sr)	3E–1
Niobium (Nb)	1E–2	Sulfur (S)	8E–1
Osmium (Os)	1E–2	Tantalum (Ta)	1E–3
Palladium (Pd)	5E–3	Technetium (Tc)	8E–1
Phosphorus (P)	8E–1	Tellurium (Te)	2E–1
Platinum (Pt)	1E–2	Terbium (Tb)	3E–4
Plutonium (Pu)	1E–3	Thallium (Tl)	1E0
Polonium (Po)	1E–1	Thorium (Th)	2E–4
Potassium (K)	1E0	Thulium (Tm)	3E–4
Praseodymium (Pr)	3E–4	Tin (Sn)	2E–2
Promethium (Pm)	3E–4	Titanium (Ti)	1E–2
Protactinium (Pa)	1E–3	Tungsten (W)	3E–1
Radium (Ra)	2E–1	Uranium (U)	5E–2
Rhenium (Re)	8E–1	Vanadium (V)	1E–2
Rhodium (Rh)	5E–2	Ytterbium (Yb)	3E–4
Rubidium (Rb)	1E0	Yttrium (Y)	1E–4
Ruthenium (Ru)	5E–2	Zinc (Zn)	5E–1
Samarium (Sm)	3E–4	Zirconium (Zr)	2E–3
Scandium (Sc)	1E–4		

APPENDIX C

RADIATION ABSORBED DOSE TO THE EMBRYO/FETUS FOLLOWING INTRODUCTION OF SPECIFIED RADIONUCLIDES AND CHEMICAL FORMS INTO THE MATERNAL TRANSFER COMPARTMENT (BLOOD)

The entries for selected radionuclides and chemical forms in the tables in this appendix have been calculated from the modeling presented in Revision 1 to NUREG/CR-5631 (Ref. C1). It has been assumed that 1 μCi of activity is introduced into the maternal transfer compartment (blood). Pregnancy is assumed to begin at the time of fertilization, roughly 2 weeks after menses, and gestation is considered to consist of nine 30-day months.

Radiation dose rates were calculated from the initial fraction that was present after a single administration at the start of each of these months or on the assumed final day (day 270) of gestation. Monthly doses were determined by integrating under the curve relating the fraction of the activity in the embryo/fetus at the start of each month after administration and the fraction at the beginning of the subsequent month of gestation. Monthly doses are shown for the inclusive periods, expressed in days. Doses to the embryo/fetus from radionuclides in maternal organs were calculated; when appropriate, these are included to provide total radiation absorbed doses. The tabulated values of cumulated doses were determined as the sum of the monthly doses.

As noted in Revision 1 to NUREG/CR-5631 (Ref. C1), ICRP Publication 30 (Ref. C2) employs a metabolic model in which a fraction of activity in the first transfer compartment (blood) often is assumed to go immediately to excretion. Because of the minuscule mass of the embryo/fetus immediately following fertilization, for some materials the biokinetic model thus predicts that there would be negligible initial activity in the embryo after administration at that time, and that there would be minimal activity at later times. As a consequence, the dose rate and doses also would be negligible, which is indicated by N in the table. For these nuclides, an approximation of the cumulative dose for an intake occurring during the first 30 days should be made based on a time-weighted average of the 31-day intake data. The cumulative dose from an intake in the first 30 days of pregnancy may be estimated by multiplying the 31-day cumulated dose value by the ratio of the days-to-date in the first month to a 30-day period. For example, assuming a maternal intake of ^{14}C resulting in a 1-μCi blood uptake on the 20th day of the pregnancy, the gestation dose should be determined by multiplying the cumulated dose from an intake at day 31 (i.e., Table C3, Cumulated Dose column, 1.89E-04 rads) by the ratio of 20 days to 30 days (i.e., 20 divided by 30).

Table C1
Radiation Doses to the Embryo/Fetus from 1 μCi of ^3H, as Tritiated Water, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	9.03E–06	3.96E–11	7.67E–14	2.00E–15	5.31E–17	2.63E–18	1.72E–19	1.34E–20	1.18E–21	9.03E–06
31		1.77E–05	2.64E–08	7.50E–10	1.94E–11	9.70E–13	6.30E–14	4.94E–15	4.33E–16	1.77E–05
61			3.93E–05	8.96E–07	2.47E–08	1.21E–09	7.91E–11	6.17E–12	5.41E–13	4.02E–05
91				3.82E–05	1.06E–06	5.19E–08	3.39E–09	2.64E–10	2.32E–11	3.93E–05
121					4.50E–05	2.14E–06	1.41E–07	1.10E–08	9.63E–10	4.73E–05
151						4.98E–05	3.22E–06	2.53E–07	2.21E–08	5.33E–05
181							5.28E–05	4.08E–06	3.57E–07	5.72E–05
211								5.40E–05	4.70E–06	5.87E–05
241									5.28E–05	5.28E–05

Table C2
Radiation Doses to the Embryo/Fetus from 1 μCi of ^3H, as a Hexose or Amino Acid, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		2.21E–05	2.14E–07	4.68E–08	1.04E–08	4.37E–09	2.35E–09	1.50E–09	1.06E–09	2.24E–05
61			6.00E–05	7.27E–06	1.67E–06	6.81E–07	3.68E–07	2.34E–07	1.66E–07	7.04E–05
91				5.82E–05	9.25E–06	3.69E–06	1.97E–06	1.26E–06	8.92E–07	7.53E–05
121					7.24E–05	1.97E–05	1.03E–05	6.50E–06	4.62E–06	1.14E–04
151						8.29E–05	3.05E–05	1.89E–05	1.33E–05	1.46E–04
181							8.96E–05	3.93E–05	2.72E–05	1.56E–04
211								9.31E–05	4.58E–05	1.39E–04
241									1.05E–04	1.05E–04

*N indicates that the metabolic pattern is such that the dose rates and doses would be negligible throughout gestation when activity is administered immediately after fertilization. Approximations of doses resulting from administration during the first month are described on page C–1.

Table C3
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{14}C, as a Bicarbonate, Hexose, Amino Acid, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		1.87E-04	1.72E-06	4.12E-07	9.18E-08	3.88E-08	2.09E-08	1.34E-08	9.56E-09	1.89E-04
61			4.96E-04	5.83E-05	1.46E-05	6.02E-06	3.26E-06	2.09E-06	1.49E-06	5.82E-04
91				4.81E-04	7.48E-05	3.24E-05	1.74E-05	1.11E-05	7.95E-06	6.25E-04
121					5.96E-04	1.59E-04	9.09E-05	5.74E-05	4.11E-05	9.44E-04
151						6.80E-04	2.47E-04	1.66E-04	1.17E-04	1.21E-03
181							7.33E-04	3.19E-04	2.39E-04	1.29E-03
211								7.61E-04	3.70E-04	1.13E-03
241									8.88E-04	8.88E-04

Table C4
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{57}Co Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	7.30E-04	2.76E-04	2.36E-04	1.97E-04	1.75E-04	1.56E-04	1.39E-04	1.23E-04	1.09E-04	2.14E-03
31		8.66E-04	2.74E-04	2.45E-04	2.07E-04	1.82E-04	1.60E-04	1.41E-04	1.24E-04	2.20E-03
61			8.71E-04	2.82E-04	2.56E-04	2.15E-04	1.88E-04	1.63E-04	1.42E-04	2.12E-03
91				8.96E-04	2.96E-04	2.67E-04	2.22E-04	1.91E-04	1.64E-04	2.04E-03
121					9.37E-04	3.08E-04	2.75E-04	2.25E-04	1.92E-04	1.94E-03
151						9.78E-04	3.18E-04	2.79E-04	2.27E-04	1.80E-03
181							1.01E-03	3.22E-04	2.83E-04	1.61E-03
211								1.03E-03	3.19E-04	1.35E-03
241									1.04E-03	1.04E-03

*N indicates that the metabolic pattern is such that the dose rates and doses would be negligible throughout gestation when activity is administered immediately after fertilization. Approximations of doses resulting from administration during the first month are described on page C-1.

Table C5
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{58}Co, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	4.81E-03	1.27E-03	9.03E-04	6.03E-04	4.25E-04	3.00E-04	2.13E-04	1.52E-04	1.09E-04	8.79E-03
31		5.12E-03	1.30E-03	9.30E-04	6.24E-04	4.37E-04	3.06E-04	2.15E-04	1.53E-04	9.08E-03
61			5.26E-03	1.34E-03	9.62E-04	6.41E-04	4.45E-04	3.09E-04	2.17E-04	9.17E-03
91				5.39E-03	1.38E-03	9.88E-04	6.54E-04	4.49E-04	3.11E-04	9.17E-03
121					5.59E-03	1.42E-03	1.01E-03	6.59E-04	4.53E-04	9.13E-03
151						5.75E-03	1.45E-03	1.02E-03	6.64E-04	8.88E-03
181							5.87E-03	1.46E-03	1.03E-03	8.36E-03
211								5.95E-03	1.45E-03	7.40E-03
241									6.00E-03	6.00E-03

Table C6
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{60}Co, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	1.28E-02	4.73E-03	4.37E-03	3.79E-03	3.60E-03	3.40E-03	3.22E-03	3.05E-03	2.88E-03	4.18E-02
31		1.38E-02	4.73E-03	4.40E-03	3.98E-03	3.73E-03	3.48E-03	3.26E-03	3.06E-03	4.04E-02
61			1.39E-02	4.76E-03	4.62E-03	4.12E-03	3.81E-03	3.52E-03	3.27E-03	3.80E-02
91				1.40E-02	4.99E-03	4.79E-03	4.22E-03	3.86E-03	3.54E-03	3.54E-02
121					1.46E-02	5.17E-03	4.90E-03	4.27E-03	3.88E-03	3.28E-02
151						1.52E-02	5.29E-03	4.96E-03	4.29E-03	2.97E-02
181							1.56E-02	5.35E-03	5.01E-03	2.60E-02
211								1.59E-02	5.29E-03	2.12E-02
241									1.60E-02	1.60E-02

Table C7
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{57}Co, as Vitamin B-12, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0-30	31-60	61-90	91-120	121-150	151-180	181-210	211-240	241-270	0-270
0	1.47E-03	1.11E-03	7.18E-04	4.88E-04	3.34E-04	2.28E-04	1.54E-04	1.02E-04	6.74E-05	4.67E-03
31		1.67E-03	1.10E-03	7.44E-04	5.10E-04	3.48E-04	2.35E-04	1.56E-04	1.03E-04	4.87E-03
61			1.68E-03	1.14E-03	7.80E-04	5.31E-04	3.59E-04	2.38E-04	1.57E-04	4.89E-03
91				1.74E-03	1.19E-03	8.13E-04	5.49E-04	3.64E-04	2.40E-04	4.90E-03
121					1.82E-03	1.24E-03	8.38E-04	5.56E-04	3.67E-04	4.82E-03
151						1.89E-03	1.28E-03	8.48E-04	5.60E-04	4.58E-03
181							1.95E-03	1.30E-03	8.55E-04	4.10E-03
211								1.98E-03	1.31E-03	3.29E-03
241									1.99E-03	1.99E-03

Table C8
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{60}Co, as Vitamin B-12, Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0-30	31-60	61-90	91-120	121-150	151-180	181-210	211-240	241-270	0-270
0	2.54E-02	1.90E-02	1.33E-02	9.38E-03	6.88E-03	4.98E-03	3.56E-03	2.52E-03	1.77E-03	8.68E-02
31		2.71E-02	1.90E-02	1.34E-02	9.82E-03	7.10E-03	5.09E-03	3.61E-03	2.53E-03	8.76E-02
61			2.70E-02	1.91E-02	1.40E-02	1.02E-02	7.28E-03	5.16E-03	3.62E-03	8.64E-02
91				2.74E-02	2.00E-02	1.45E-02	1.04E-02	7.38E-03	5.18E-03	8.49E-02
121					2.86E-02	2.08E-02	1.49E-02	1.05E-02	7.41E-03	8.22E-02
151						2.97E-02	2.13E-02	1.51E-02	1.06E-02	7.67E-02
181							3.04E-02	2.15E-02	1.51E-02	6.70E-02
211								3.08E-02	2.16E-02	5.24E-02
241									3.10E-02	3.10E-02

Table C9
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{89}Sr Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	4.09E-03	5.66E-04	2.92E-04	1.37E-04	6.64E-05	3.59E-05	2.10E-05	1.23E-05	7.01E-06	5.23E-03
31		5.35E-03	5.74E-04	2.95E-04	1.36E-04	6.57E-05	3.53E-05	2.05E-05	1.20E-05	6.49E-03
61			9.01E-03	1.20E-03	3.84E-04	1.63E-04	7.45E-05	3.86E-05	2.18E-05	1.09E-02
91				9.09E-03	1.36E-03	5.06E-04	2.12E-04	9.67E-05	4.93E-05	1.13E-02
121					1.07E-02	2.24E-03	8.99E-04	3.90E-04	1.84E-04	1.44E-02
151						1.19E-02	3.15E-03	1.40E-03	6.55E-04	1.71E-02
181							1.26E-02	3.87E-03	1.89E-03	1.84E-02
211								1.29E-02	4.38E-03	1.73E-02
241									1.31E-02	1.31E-02

Table C10
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{90}Sr (in Equilibrium with ^{90}Y) Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	9.07E-03	2.01E-03	1.57E-03	1.10E-03	8.07E-04	6.58E-04	5.81E-04	5.15E-04	4.43E-04	1.68E-02
31		1.13E-02	2.04E-03	1.57E-03	1.09E-03	7.99E-04	6.49E-04	5.69E-04	5.00E-04	1.85E-02
61			2.03E-02	3.60E-03	1.72E-03	1.33E-03	8.94E-04	7.10E-04	6.04E-04	2.92E-02
91				1.50E-02	3.31E-03	2.80E-03	1.67E-03	1.17E-03	8.98E-04	2.48E-02
121					1.90E-02	7.93E-03	4.71E-03	3.11E-03	2.22E-03	3.70E-02
151						2.69E-02	1.10E-02	7.41E-03	5.23E-03	5.05E-02
181							2.86E-02	1.36E-02	1.00E-02	5.22E-02
211								2.95E-02	1.54E-02	4.49E-02
241									2.93E-02	2.93E-02

Table C11
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{106}Ru (in Equilibrium with ^{106}Rh) Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	1.56E-03	1.00E-03	9.36E-04	7.68E-04	6.67E-04	5.94E-04	5.35E-04	4.85E-04	4.41E-04	6.99E-03
31		2.02E-03	1.21E-03	9.48E-04	7.77E-04	6.72E-04	5.94E-04	5.32E-04	4.80E-04	7.23E-03
61			2.42E-03	1.23E-03	9.56E-04	7.80E-04	6.70E-04	5.90E-04	5.27E-04	7.17E-03
91				2.50E-03	1.24E-03	9.68E-04	7.84E-04	6.68E-04	5.85E-04	6.74E-03
121					2.53E-03	1.25E-03	9.63E-04	7.77E-04	6.62E-04	6.18E-03
151						2.55E-03	1.26E-03	9.59E-04	7.69E-04	5.54E-03
181							2.55E-03	1.25E-03	9.55E-04	4.75E-03
211								2.54E-03	1.23E-03	3.77E-03
241									2.53E-03	2.53E-03

Table C12
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{125}I Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	2.08E-05	1.12E-05	7.34E-06	1.34E-05	1.46E-05	6.07E-06	4.65E-06	3.01E-06	2.07E-06	8.31E-05
31		2.72E-05	1.05E-05	1.27E-05	1.40E-05	1.04E-05	7.27E-06	4.83E-06	3.31E-06	9.02E-05
61			2.74E-05	1.70E-05	2.23E-05	1.63E-05	1.15E-05	7.66E-06	5.28E-06	1.07E-04
91				1.64E-04	5.21E-05	3.23E-05	2.05E-05	1.31E-05	8.84E-06	2.91E-04
121					8.79E-04	2.88E-04	1.22E-04	5.70E-05	3.05E-05	1.38E-03
151						7.81E-04	3.12E-04	1.40E-04	7.08E-05	1.30E-03
181							6.78E-04	2.99E-04	1.48E-04	1.12E-03
211								5.97E-04	2.98E-04	8.95E-04
241									5.33E-04	5.33E-04

Table C13
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{131}I Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	5.93E-05	2.58E-06	1.88E-07	2.20E-08	3.39E-09	2.10E-10	2.29E-11	1.32E-12	6.35E-14	6.21E-05
31		9.73E-05	2.31E-06	3.38E-07	5.05E-08	3.22E-09	3.47E-10	2.01E-11	9.66E-13	1.00E-04
61			9.44E-05	4.14E-06	7.60E-07	4.75E-08	5.23E-09	3.02E-10	1.46E-11	9.94E-05
91				6.52E-04	2.11E-05	9.30E-07	9.12E-08	5.01E-09	2.33E-10	6.74E-04
121					3.54E-03	8.90E-05	6.03E-06	2.33E-07	7.82E-09	3.64E-03
151						2.35E-03	1.49E-04	5.56E-06	1.75E-07	2.50E-03
181							2.88E-03	1.15E-04	3.48E-06	3.00E-03
211								1.98E-03	6.80E-05	2.05E-03
241									1.00E-03	1.00E-03

Table C14
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{132}I Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	8.43E-05	0	0	0	0	0	0	0	0	8.43E-05
31		1.06E-04	0	0	0	0	0	0	0	1.06E-04
61			1.27E-04	0	0	0	0	0	0	1.27E-04
91				1.30E-04	0	0	0	0	0	1.30E-04
121					1.51E-04	0	0	0	0	1.51E-04
151						1.53E-04	0	0	0	1.53E-04
181							1.56E-04	0	0	1.56E-04
211								1.56E-04	0	1.56E-04
241									1.56E-04	1.56E-04

Table C15
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{133}I Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	2.81E-04	0	0	0	0	0	0	0	0	2.81E-04
31		5.32E-04	0	0	0	0	0	0	0	5.32E-04
61			6.85E-04	0	0	0	0	0	0	6.85E-04
91				7.04E-04	0	0	0	0	0	7.04E-04
121					9.04E-04	0	0	0	0	9.04E-04
151						8.59E-04	0	0	0	8.59E-04
181							8.49E-04	0	0	8.49E-04
211								8.27E-04	0	8.27E-04
241									8.11E-04	8.11E-04

Table C16
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{134}I Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	2.22E-05	0	0	0	0	0	0	0	0	2.22E-05
31		2.79E-05	0	0	0	0	0	0	0	2.79E-05
61			3.44E-05	0	0	0	0	0	0	3.44E-05
91				3.50E-05	0	0	0	0	0	3.50E-05
121					3.81E-05	0	0	0	0	3.81E-05
151						3.91E-05	0	0	0	3.91E-05
181							4.03E-05	0	0	4.03E-05
211								4.83E-05	0	4.83E-05
241									4.06E-05	4.06E-05

Table C17
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{135}I Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	1.95E–04	0	0	0	0	0	0	0	0	1.95E–04
31		2.63E–04	0	0	0	0	0	0	0	2.63E–04
61			3.07E–04	0	0	0	0	0	0	3.07E–04
91				3.04E–04	0	0	0	0	0	3.04E–04
121					3.65E–04	0	0	0	0	3.65E–04
151						3.66E–04	0	0	0	3.66E–04
181							3.72E–04	0	0	3.72E–04
211								3.69E–04	0	3.69E–04
241									3.70E–04	3.70E–04

Table C18
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{134}Cs Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	2.55E–02	2.15E–02	1.69E–02	1.33E–02	1.05E–02	8.29E–03	6.35E–03	4.37E–03	2.81E–03	1.10E–01
31		2.82E–02	2.23E–02	1.75E–02	1.38E–02	1.09E–02	8.38E–03	5.75E–03	3.71E–03	1.11E–01
61			2.92E–02	2.30E–02	1.82E–02	1.44E–02	1.10E–02	7.59E–03	4.88E–03	1.08E–01
91				3.03E–02	2.40E–02	1.89E–02	1.45E–02	9.98E–03	6.43E–03	1.04E–01
121					3.16E–02	2.49E–02	1.91E–02	1.31E–02	8.46E–03	9.72E–02
151						3.28E–02	2.51E–02	1.73E–02	1.12E–02	8.64E–02
181							3.30E–02	2.28E–02	1.46E–02	7.04E–02
211								3.14E–02	2.03E–02	5.17E–02
241									3.24E–02	3.24E–02

Table C19
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{137}Cs Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	1.18E-02	1.13E-02	9.13E-03	7.36E-03	5.91E-03	4.74E-03	3.70E-03	2.60E-03	1.71E-03	5.83E-02
31		1.43E-02	1.17E-02	9.43E-03	7.59E-03	6.08E-03	4.74E-03	3.33E-03	2.19E-03	5.94E-02
61			1.50E-02	1.21E-02	9.72E-03	7.80E-03	6.09E-03	4.27E-03	2.81E-03	5.78E-02
91				1.55E-02	1.25E-02	1.00E-02	7.79E-03	5.48E-03	3.60E-03	5.49E-02
121					1.60E-02	1.29E-02	1.00E-02	7.02E-03	4.63E-03	5.05E-02
151						1.65E-02	1.29E-02	9.05E-03	5.96E-03	4.44E-02
181							1.65E-02	1.16E-02	7.60E-03	3.57E-02
211								1.56E-02	1.03E-02	2.59E-02
241									1.60E-02	1.60E-02

Table C20
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{233}U Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		1.41E-03	2.31E-05	5.30E-06	1.19E-06	5.01E-07	2.71E-07	1.74E-07	1.24E-07	1.44E-03
61			4.30E-03	7.86E-04	1.89E-04	7.84E-05	4.25E-05	2.72E-05	1.94E-05	5.44E-03
91				6.29E-03	1.52E-03	6.29E-04	3.42E-04	2.19E-04	1.56E-04	9.16E-03
121					8.10E-03	3.25E-03	1.78E-03	1.13E-03	8.09E-04	1.51E-02
151						9.51E-03	5.11E-03	3.28E-03	2.34E-03	2.02E-02
181							1.40E-02	8.88E-03	6.36E-03	2.92E-02
211								1.49E-02	1.06E-02	2.55E-02
241									2.38E-02	2.38E-02

*N indicates that the metabolic pattern is such that the dose rates and doses would be negligible throughout gestation when activity is administered immediately after fertilization. Approximations of doses resulting from administration during the first month are described on page C-1.

Table C21
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{234}U Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		1.40E-03	2.30E-05	5.26E-06	1.18E-06	5.00E-07	2.70E-07	1.73E-07	1.23E-07	1.43E-03
61			4.27E-03	7.82E-04	1.87E-04	7.79E-05	4.22E-05	2.70E-05	1.93E-05	5.41E-03
91				6.25E-03	1.51E-03	6.28E-04	3.39E-04	2.17E-04	1.55E-04	9.10E-03
121					8.05E-03	3.23E-03	1.77E-03	1.13E-03	8.07E-04	1.50E-02
151						9.46E-03	5.07E-03	3.26E-03	2.32E-03	2.01E-02
181							1.40E-02	8.88E-03	6.34E-03	2.92E-02
211								1.48E-02	1.05E-02	2.53E-02
241									2.36E-02	2.36E-02

Table C22
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{235}U Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		1.29E-03	2.11E-05	4.84E-06	1.09E-06	4.60E-07	2.48E-07	1.59E-07	1.13E-07	1.32E-03
61			3.93E-03	7.19E-04	1.73E-04	7.18E-05	3.88E-05	2.49E-05	1.77E-05	4.98E-03
91				5.75E-03	1.39E-03	5.78E-04	3.12E-04	2.00E-04	1.43E-04	8.37E-03
121					7.40E-03	2.97E-03	1.62E-03	1.04E-03	7.41E-04	1.38E-02
151						8.70E-03	4.67E-03	3.00E-03	2.14E-03	1.85E-02
181							1.28E-02	8.12E-03	5.82E-03	2.67E-02
211								1.36E-02	9.69E-03	2.33E-02
241									2.17E-02	2.17E-02

*N indicates that the metabolic pattern is such that the dose rates and doses would be negligible throughout gestation when activity is administered immediately after fertilization. Approximations of doses resulting from administration during the first month are described on page C-1.

Table C23
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{238}U Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		1.23E-03	2.01E-05	4.59E-06	1.04E-06	4.38E-07	2.36E-07	1.51E-07	1.08E-07	1.26E-03
61			3.75E-03	6.86E-04	1.64E-04	6.83E-05	3.70E-05	2.37E-05	1.69E-05	4.75E-03
91				5.49E-03	1.32E-03	5.49E-04	2.98E-04	1.90E-04	1.36E-04	7.98E-03
121					7.06E-03	2.83E-03	1.55E-03	9.91E-04	7.08E-04	1.31E-02
151						8.30E-03	4.45E-03	2.86E-03	2.04E-03	1.77E-02
181							1.22E-02	7.76E-03	5.54E-03	2.55E-02
211								1.30E-02	9.23E-03	2.22E-02
241									2.07E-02	2.07E-02

Table C24
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{238}Pu Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		2.68E-03	4.38E-05	1.00E-05	2.26E-06	9.55E-07	5.14E-07	3.30E-07	2.35E-07	2.74E-03
61			8.19E-03	1.50E-03	3.58E-04	1.49E-04	8.05E-05	5.16E-05	3.67E-05	1.04E-02
91				1.20E-02	2.89E-03	1.20E-03	6.50E-04	4.15E-04	2.96E-04	1.75E-02
121					1.54E-02	6.18E-03	3.37E-03	2.15E-03	1.54E-03	2.86E-02
151						1.81E-02	9.70E-03	6.24E-03	4.43E-03	3.85E-02
181							2.66E-02	1.69E-02	1.21E-02	5.56E-02
211								2.84E-02	2.01E-02	4.85E-02
241									4.51E-02	4.51E-02

*N indicates that the metabolic pattern is such that the dose rates and doses would be negligible throughout gestation when activity is administered immediately after fertilization. Approximations of doses resulting from administration during the first month are described on page C-1.

Table C25
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{239}Pu Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		2.52E–03	4.12E–05	9.40E–06	2.12E–06	8.97E–07	4.83E–07	3.10E–07	2.21E–07	2.57E–03
61			7.68E–03	1.40E–03	3.36E–04	1.40E–04	7.56E–05	4.85E–05	3.46E–05	9.71E–03
91				1.12E–02	2.71E–03	1.12E–03	6.07E–04	3.90E–04	2.78E–04	1.63E–02
121					1.45E–02	5.80E–03	3.17E–03	2.02E–03	1.44E–03	2.69E–02
151						1.70E–02	9.09E–03	5.85E–03	4.17E–03	3.61E–02
181							2.50E–02	1.59E–02	1.13E–02	5.22E–02
211								2.66E–02	1.88E–02	4.54E–02
241									4.23E–02	4.23E–02

Table C26
Radiation Doses to the Embryo/Fetus from 1 μCi of ^{241}Am Introduced into the Maternal Transfer Compartment (Blood)

Days of Gestation at Introduction	Dose (rad) to Embryo/Fetus During Indicated Gestation Periods (days)									Cumulated Dose
	0–30	31–60	61–90	91–120	121–150	151–180	181–210	211–240	241–270	0–270
0	N*	N	N	N	N	N	N	N	N	N
31		5.36E–04	8.76E–06	2.00E–06	4.52E–07	1.91E–07	1.03E–07	6.60E–08	4.71E–08	5.48E–04
61			1.64E–03	2.99E–04	7.16E–05	2.97E–05	1.61E–05	1.03E–05	7.35E–06	2.07E–03
91				2.39E–03	5.76E–04	2.39E–04	1.30E–04	8.30E–05	5.92E–05	3.48E–03
121					3.08E–03	1.23E–03	6.75E–04	4.31E–04	3.08E–04	5.72E–03
151						3.61E–03	1.94E–03	1.24E–03	8.89E–04	7.68E–03
181							5.32E–03	3.38E–03	2.41E–03	1.11E–02
211								5.67E–03	4.02E–03	9.69E–03
241									9.04E–03	9.04E–03

*N indicates that the metabolic pattern is such that the dose rates and doses would be negligible throughout gestation when activity is administered immediately after fertilization. Approximations of doses resulting from administration during the first month are described on page C-1.

REFERENCES

C1. M. R. Sikov et al., "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Dose—Interim Recommendations," NUREG/CR-5631, Revision 1 (PNL-7445), U.S. Nuclear Regulatory Commission, March 1992.

C2. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP No. 30, Parts 1 through 4, including supplements, Annals of the ICRP, Volume 2, No. 3/4, Pergamon Press Inc., 1979.

APPENDIX D

EXAMPLES OF EMBRYO/FETUS DOSE CALCULATIONS

The purpose of this appendix is to present examples of the methods of the guide for calculating the dose equivalent to the embryo/fetus. The examples have been developed to demonstrate the calculational methods; the methods for evaluating and determining maternal exposures, body burdens, and intakes are not included. These examples are not intended to describe all the measures that would be required for determining the maternal exposure. Instead, the examples are presented to concisely demonstrate the calculational methods once data on maternal exposure have been obtained. It is important to keep in mind that an evaluation is no better than the quality of the data. In applying the methods of this guide, a primary concern has to be the reliability of the maternal exposure data. The calculation of the embryo/fetus dose consists of a two-step process. First, the content of a radionuclide in maternal blood has to be determined. This is accomplished by multiplying the intake by the appropriate transfer fraction. The second step involves the determination of the embryo/fetus dose based on the maternal radionuclide blood content.

Six example calculations are provided. Cases 1 and 2 address ingestion intakes by the declared pregnant woman. Cases 3 and 4 address inhalation intakes. Case 5 evaluates a pre-existing body burden and determines the embryo/fetus dose equivalent based on the maternal burden existing at time of pregnancy. Case 6 presents an example of summing external and internal doses and instituting worker controls to ensure the dose limit is not exceeded.

The two methods in the guide for calculating the embryo/fetus dose equivalent are presented: the simplified method as presented in the regulatory position in Section 2 of this guide and the Revision 1 to NUREG/CR-5631 gestation-time dependent method as presented in the regulatory position in Section 3.

CASE 1

EMBRYO/FETUS DOSE FOLLOWING ACUTE INGESTION INTAKE BY DECLARED PREGNANT WOMAN

1.1 Exposure Scenario

A declared pregnant woman unknowingly ingests a substance that contains trace amounts of ^{58}Co. The licensee determines that the woman ingested 22 μCi of ^{58}Co over a 4-day period.* The intake is confined to a short time period (relative to the effective biological retention half-life of ^{58}Co) within the first month of the pregnancy. Because the intake is assumed to have occurred within a single 30-day gestation period interval (i.e., a 30-day period as used for calculating intakes and doses), the ingestion may be treated as a single, acute intake.

1.2 Determining Blood Uptake

The calculation of the dose to the embryo/fetus is based on the amount of the intake that is available for uptake within the first transfer compartment (i.e., blood). Applying the guidance of the regulatory position in Section 2.2 of the guide, the blood uptake for an ingestion intake may be calculated by multiplying the intake by the gut-to-blood transfer factor (f_1):

$$\text{Blood Uptake} = f_1 \times \text{Ingestion Intake}$$

For cobalt, the f_1 value from Appendix B to the guide is 0.3. For this example, the predetermined ingestion intake is 22 μCi. Inserting these values into the above equation results in the following calculation of the maternal blood content:

$$\text{Blood Uptake} = 0.3 \times 22\ \mu\text{Ci} = 6.6\ \mu\text{Ci}$$

1.3 Calculating the Embryo/Fetus Dose Equivalent

The calculation of the embryo/fetus dose equivalent is based on the activity uptake into the first transfer compartment (i.e., maternal blood). First, the dose will be calculated using the Simplified Method as presented in the regulatory position specified in Section 2 of the guide. Next, the gestation-time dependent method for calculating the dose equivalent will be presented.

1.3.1 Simplified Method

The regulatory position in Section 2 of the guide presents the Simplified Method for calculating the embryo/fetus dose equivalent. From Appendix A to the guide, the ^{58}Co dose equivalent factor is 9.17E-03 rem/μCi (in blood). The dose equivalent is calculated using Equation 1 from the regulatory position in Section 2.5 of the guide. Substituting the values for intake, the gut-to-blood transfer factor (f_1) and dose factor into this equation yields the following dose equivalent calculation:

$$\begin{aligned}\text{Dose Equivalent} &= \text{Intake} \times f_1 \times \text{Dose Factor} \\ &= 22\ \mu\text{Ci} \times 0.3 \times 9.17\text{E-}03\ \text{rem}/\mu\text{Ci} \\ &= 0.061\ \text{rem}\end{aligned}$$

1.3.2 Method Using Revision 1 to NUREG/CR-5631

The regulatory position specified in Section 3 of the guide presents the method for calculating the embryo/fetus dose using the gestation-time dependent methodology of Revision 1 to NUREG/CR-5631. Table C5 of Appendix C to the guide presents the gestation-time dependent dose factors for ^{58}Co. From this table, the column under the heading "Cumulated Dose" presents the dose to the embryo/fetus for the remainder of the gestation period resulting from the introduction of unit activity (i.e., 1 μCi) into the blood of the woman at the beginning of the specified monthly gestation period interval. The cumulated dose factor for a ^{58}Co intake during the first month of gestation is 8.79E-03 rads per microcurie in maternal blood. The regulatory position specified in Section 3.2.2 of the guide states that it should be assumed that all intakes occurring within any of the 30-day time periods of gestation occur at the beginning of that period. As discussed in the regulatory position in Section 3.2.1, a radiation quality factor of 1.0 should be used for ^{58}Co in converting from an absorbed dose in rads to an equivalent dose expressed as rems. Applying the method of the regulatory position specified in Section 3.2, the dose equivalent to the embryo/fetus is calculated as follows:

$$\begin{aligned}\text{Dose Equivalent} &= \text{Intake} \times f_1 \times \text{Dose Factor} \times 1.0\ \text{rem/rad} \\ &= 22\ \mu\text{Ci} \times 0.3 \times 8.79\text{E-}03\ \text{rad}/\mu\text{Ci} \times 1.0\ \text{rem/rad} \\ &= 0.058\ \text{rem}\end{aligned}$$

*Acceptable methods for determining intake using bioassay measurements are presented in Proposed Revision 1 to Regulatory Guide 8.9 (DG-8009), "Interpretation of Bioassay Measurements."

CASE 2

EMBRYO/FETUS DOSE FOLLOWING CHRONIC INGESTION INTAKE BY DECLARED PREGNANT WOMAN

2.1 Exposure Scenario

Over an extended period of time, a declared pregnant woman unknowingly consumes water that contains low levels of tritium contamination. The licensee discovers the tritium contaminated water in the third month of the woman's pregnancy. A thorough evaluation of the situation and associated personnel exposures is conducted, including bioassay measurements and contaminated water sample analysis. It is determined that the source did not exist prior to the woman's pregnancy. In keeping with the regulatory positions specified in Sections 2.7 and 3.3 of the guide, multiple intakes should be evaluated on at least a 30-day frequency. The licensee determines that the declared pregnant woman ingested the following amounts of tritium over the 3–month period:

Stage of Gestation at Time of Intake (days)	Intake (μCi)
0 – 30	156
31 – 60	248
61 – 90	185

2.2 Determining Blood Uptake

The amount of tritium that is available for uptake by the blood is calculated by multiplying the intake by the f_1 value for the radionuclide. For tritium, the value of f_1 is 1.0 (refer to the hydrogen entry in Appendix B to this guide). Therefore, the amount of tritium that is absorbed into the blood (as evaluated for calculating the embryo/fetus dose) is the same as the intake quantities presented above.

2.3 Calculating Embryo/Fetus Dose Equivalent

2.3.1 Simplified Method

Equation 1 from the regulatory position specified in Section 2.5 of the guide may be used for calculating the dose equivalent for the entire gestation period from each monthly intake. The tritium dose factor from Appendix A is 5.87E–05 rem per microcurie in maternal blood. The dose contribution to the embryo/fetus for each monthly intake may be calculated as follows:

Dose Equivalent = Intake x f_1 x Dose Factor

First-month intake

156 μCi x 1.0 x 5.87E–05 rem/μCi = 0.009 rem

Second-month intake

248 μCi x 1.0 x 5.87E–05 rem/μCi = 0.015 rem

Third-month intake

185 μCi x 1.0 x 5.87E–05 rem/μCi = 0.011 rem

TOTAL = 0.035 rem

2.3.2 Method Using Revision 1 to NUREG/CR–5631

Using the methods of Revision 1 to NUREG/CR–5631, the dose to the embryo/fetus is calculated in a manner similar to that of the Simplified Method, as presented above. However, as discussed in the regulatory position specified in Section 3.2.1, the dose factor should be taken from Appendix C for the time period representing the time of intake relative to stage of gestation. Table C1 in Appendix C presents the ^3H dose factors. The first column of Table C1 presents the gestation time (e.g., 0, 30, 60 days), and the last column presents the cumulated dose to the embryo/fetus for the remainder of the gestation period following the introduction of unit activity into maternal blood at the specified gestation time. As specified in the regulatory position in Section 3.2.2 of the guide, an intake at any time within a specific monthly gestation period (i.e., a 30–day period) may be assumed to have occurred at the beginning of the monthly period for the purpose of determining the appropriate dose factor to be used. For example, for intakes occurring during the first month of pregnancy, the dose factor under the "Cumulated Dose" column corresponding to 0 days of gestation (as designated in the left–most column of the table) should be used. Cumulated dose factors taken from Table C1 for intakes in the respective months of gestation are presented below:

Stage of Gestation at Time of Intake	Cumulated Dose Factor for Remainder of Gestation Period (rad/μCi, blood)
1st Month (0 – 30 days)	9.03E–06
2nd Month (31 – 60 days)	1.77E–05
3rd Month (61 – 90 days)	4.02E–05

Using these gestation-time dependent dose factors, the dose equivalent to the embryo/fetus is calculated using the regulatory position specified in Section 3.2 of the guide. The radiation quality factor for 3H is 1.0. The dose to the embryo/fetus for the remainder of the gestation period resulting from intakes occurring within each month is calculated as follows:

Dose Equivalent = Intake x f_1 x DF_i

First-month intake

156 μCi x 1.0 x 9.03E–06 rad/μCi x 1.0 rem/rad = 0.001 rem

Second-month intake

248 μCi x 1.0 x 1.77E–05 rad/μCi x 1.0 rem/rad = 0.004 rem

Third-month intake

185 μCi x 1.0 x 4.02E–05 rad/μCi x 1.0 rem/rad = 0.007 rem

TOTAL = 0.013 rem*

*The difference between the sum of the monthly doses and the total (i.e., 0.012 rem versus 0.013 rem) is caused by rounding. In keeping with the recommendation contained in the Discussion section of this guide, final results should be rounded to the nearest thousandth of a rem.

CASE 3

EMBRYO/FETUS DOSE FOLLOWING ACUTE INHALATION INTAKE BY DECLARED PREGNANT WOMAN

3.1 Exposure Scenario

During the performance of a medical administration, a woman worker accidentally receives a single, acute inhalation intake of 100 μCi of ^{131}I. At the time of the exposure, the woman was in the third month of pregnancy but had not declared her pregnancy to her employer (the licensee). Shortly thereafter, she declares her pregnancy in writing.

3.2 Determining Blood Uptake

The calculation of the dose to the embryo/fetus is based on the amount of the intake that is available for uptake within the first transfer compartment (i.e., blood). Also, the transfer to the blood is a function of the lung clearance class. The lung clearance class for all chemical compounds of iodine is Class D, denoting a 0- to 10-day lung clearance half-life. (Appendix B to 10 CFR 20.1001–20.2401 provides the lung clearance classes for the different chemical compounds of the specified radionuclides.) Applying the guidance of the regulatory positions specified in Sections 2.3 and 2.4 of the guide, the transfer fraction of inhaled activity to the blood for a Class D radionuclide may be calculated as follows:

$$TF_i \text{ (Class D)} = 0.48 + 0.15 \times f_{1,i}$$

where:

TF_i = transfer fraction of inhaled activity to the first transfer compartment (blood)

$f_{1,i}$ = gut-to-blood transfer factor for radionuclide i (from Appendix B to the guide)

0.48 = fraction of inhalation intake that is cleared directly from the lung to the blood for Class D compounds

0.15 = fraction of inhaled radionuclide that is cleared from the lung to the GI tract for Class D compounds

For iodine, the f_1 value from Appendix B to the guide is 1.0. Inserting these values into the above equation results in the following calculation of the transfer fraction:

$$TF_i = 0.48 + 0.15 \times 1.0 = 0.63$$

The resultant blood uptake may be calculated by multiplying the transfer fraction by the total intake:

$$\text{Blood Uptake} = TF_i \times \text{Inhalation Intake} = 0.63 \times 100\ \mu Ci = 63\ \mu Ci$$

3.3 Calculating Embryo/Fetus Dose Equivalent

3.3.1 Simplified Method

For this example, the predetermined inhalation intake is 100 μCi. From Appendix A to the guide, the dose factor for ^{131}I is 3.64E–03 rem/μCi (in blood). The dose equivalent to the embryo/fetus may be calculated using Equation 2 from the regulatory position specified in Section 2.5 of the guide:

$$\text{Dose Equivalent} = I_i \times TF_i \times DF_i = 100\ \mu Ci \times 0.63 \times 3.64E\text{–}03\ rem/\mu Ci = 0.229\ rem$$

3.3.2 Method Using Revision 1 to NUREG/CR–5631

The regulatory position specified in Section 3 of the guide presents the method for calculating the embryo/fetus dose using the methodology of Revision 1 to NUREG/CR–5631. The inhalation intake is determined to have occurred during the third month of the gestation period. Table C13 of Appendix C to the guide presents the gestation-time dependent dose factors for ^{131}I. In this table, the left-most column specifies the beginning time for each monthly gestation period (e.g., 0 for 0–30 days, 31 for 31–60 days). The right-most column presents the corresponding cumulated dose to the embryo/fetus for the remainder of the gestation period for unit activity introduced into the maternal blood. From this table, the cumulated dose factor for an ^{131}I intake during the third month of gestation is 9.94E–05 rad/μCi uptake into blood. As discussed in the regulatory position specified in Section 3.2.1, a radiation quality factor of 1.0 should be used for ^{131}I. Applying the methods of the regulatory position specified in Section 3.2, the dose equivalent to the embryo/fetus may be calculated. The value for the transfer fraction (TF_i) is the same as calculated above (i.e., 0.63). Using these parameter values along with Equation 2 from the guide, the embryo/fetus dose is calculated as follows:

$$\text{Dose Equivalent} = I_i \times TF_i \times DF_i \times 1.0\ rem/rad = 100\ Ci \times 0.63 \times 9.94E\text{–}05\ rad/\mu Ci \times 1.0\ rem/rad = 0.006\ rem$$

This example illustrates the difference that can occur by using the gestation-time dependent dose

factors for the calculation of the embryo/fetus dose equivalent. The Simplified Method, as presented above, for this example yields an embryo/fetus dose of 0.229 rem; using the gestation-time dependent dose factors results in a calculated embryo/fetus dose equivalent of 0.006 rem—a factor of almost 40 less. This difference reflects the fact that during early embryonic development there is no preferential uptake of iodine by the embryo; the thyroid has not yet developed. It is not until approximately the beginning of the fourth month of the gestation period that the fetal thyroid develops to a point that thyroid iodine uptake is thought to occur. Therefore, any maternal intakes during the second and third trimesters will result in a significantly larger dose to the embryo/fetus than will result from the same intake during the first trimester.

CASE 4

EMBRYO/FETUS DOSE FOR CHRONIC INHALATION INTAKE BY DECLARED PREGNANT WOMAN

4.1 Exposure Scenario

During the third through fifth month of her pregnancy, a declared pregnant woman is exposed to airborne levels of ^{238}U. Extensive air sampling and followup bioassay measurements are conducted to closely monitor the woman's intake. From these measurements, it is determined that the ^{238}U consists of a mixture of 30% Class D and 70% Class Y compounds. In keeping with the regulatory positions specified in Sections 2.7 and 3.3 of the guide, intakes over an extended time should be evaluated on at least a 30-day frequency. The licensee determines that the woman inhaled the following amounts of ^{238}U over the 3-month period:

Stage of Gestation at Time of Intake (days)	Class D Intake (μCi)	Class Y Intake (μCi)
61 – 90	0.038	0.089
91 – 120	0.061	0.14
121 – 150	0.15	0.35

4.2 Determining Blood Uptake

The calculation of the dose to the embryo/fetus is based on the amount of intake that is available for uptake within the first transfer compartment (i.e., blood). Also, the transfer to the blood is a function of the lung clearance class. Applying the guidance of the regulatory positions specified in Sections 2.3 and 2.4 of the guide, the transfer fraction (TF_i) of inhaled activity to the first transfer compartment for a Class D compound may be calculated as follows:

$$TF_i \text{ (Class D)} = 0.48 + 0.15 \times f_{1,i}$$

where:

TF_i = transfer fraction of inhaled activity to the first transfer compartment

$f_{1,i}$ = gut-to-blood transfer factor for radionuclide i (from Appendix B to the guide)

0.48 = fraction of inhalation intake that is cleared directly from the lung to the blood for Class D compounds

0.15 = fraction of inhaled radionuclide that is cleared from the lung to the GI tract for Class D compounds

The resultant total blood uptake is calculated by multiplying the TF_i value by the inhalation intake:

$$\text{Blood Uptake} = TF_i \times \text{Inhalation Intake}$$

For a Class Y compound, the transfer fraction is calculated as follows:

$$TF_i \text{ (Class Y)} = 0.05 + 0.58 \times f_{1,i}$$

where:

0.05 = fraction of inhalation intake that is cleared directly from the lung to the blood for Class Y compounds

0.58 = fraction of inhaled radionuclide that is cleared from the lung to the GI tract for Class Y compounds

The total blood uptake can be calculated in the same manner as discussed above for the Class D compound.

For uranium, the f_1 value from Appendix B to the guide is 0.05. Applying the above equations, the amounts of ^{238}U transferred to the blood as a function of gestation period are presented in the following table:

Stage of Gestation at Time of Intake (days)	Transfer Fraction and Blood Uptake (Class D)		Transfer Fraction and Blood Uptake (Class Y)	
	Transfer Fraction (TF_i)	Blood Uptake (μCi)	Transfer Fraction (TF_i)	Blood Uptake (μCi)
61 – 90	0.49	0.0186	0.079	0.00703
91 – 120	0.49	0.0299	0.079	0.0111
121 – 150	0.49	0.0735	0.079	0.0276

4.3 Calculating Embryo/Fetus Dose Equivalent

4.3.1 Simplified Method

The dose to the embryo/fetus is calculated by using Equation 2 from the regulatory position in Section 2.5 of the guide. From Appendix A, the dose factor for ^{238}U is 5.10E–01 rem/μCi (in blood). Applying this dose factor along with the monthly transfer fractions (as calculated above) results in the following dose calculations:

Class D Inhalation Intake

Dose Equivalent = Intake x TF_i x DF_i

Third-month intake

0.038 μCi x 0.49 x 5.10E–01 rem/μCi
= 0.009 rem

Fourth-month intake

0.061 μCi x 0.49 x 5.10E–01 rem/μCi
= 0.015 rem

Fifth-month intake

0.15 μCi x 0.49 x 5.10E–05 rem/μCi
= 0.037 rem

TOTAL = 0.061 rem

Class Y Inhalation Intake

Dose Equivalent = Intake x TF_i x DF_i

Third-month intake

0.089 μCi x 0.079 x 5.10E–01 rem/μCi
= 0.004 rem

Fourth-month intake

0.14 μCi x 0.079 x 5.10E–01 rem/μCi
= 0.006 rem

Fifth-month intake

0.35 μCi x 0.079 x 5.10E–01 rem/μCi
= 0.014 rem

TOTAL = 0.024 rem

The dose to the embryo/fetus resulting from each single-month intake should be determined by adding the Class D component with the Class Y component. The total gestation period dose is the sum of the cumulated dose resulting from each monthly intake.

Gestation Month	Class D Dose (rem)	Class Y Dose (rem)	Total Dose (rem)
3rd Month (61 – 90 days)	0.009	0.004	0.013
4th Month (91 – 120 days)	0.015	0.006	0.021
5th Month (121 – 150 days)	0.037	0.014	0.051
TOTAL	0.085 rem		

4.3.2 Method Using Revision 1 to NUREG/CR-5631

Using the methods of Revision 1 to NUREG/CR–5631, the dose to the embryo/fetus is calculated in a manner similar to the Simplified Method above. However, as discussed in the regulatory position specified in Section 3.2, the dose factor should be taken from Appendix C for the period representing

the time of intake relative to stage of gestation. Table C23 of Appendix C presents the gestation-time dependent dose factors for ^{238}U. In this table, the left-most column specifies the beginning time for each monthly gestation period (e.g., 0 for 0–30 days, 31 for 31–60 days). The right-most column presents the corresponding cumulated dose to the embryo/fetus for the remainder of the gestation period per unit activity introduced into the maternal blood. From Table C23, the ^{238}U cumulated dose factors for intakes in the respective month of gestation are presented below:

Stage of Gestation at Time of Intake	Cumulated Dose Factor for Remainder of Gestation Period (rad/μCi, blood)
3rd Month (61 – 90 days)	4.75E–03
4th Month (91 –120 days)	7.98E–03
5th Month (121 –150 days)	1.31E–02

Using these gestation-time dependent dose factors, the dose equivalent to the embryo/fetus is calculated using the regulatory position specified in Section 3.2 of the guide. A radiation quality factor of 20 should be used for ^{238}U as specified in the regulatory position in Section 3.2.1. The dose equivalent is calculated on a monthly basis as follows:

Class D Inhalation Intake

Dose Equivalent = Intake x TF_1 x DF_i x 20 rem/rad

Third-month intake

0.038 μCi x 0.49 x 4.75E–03 rad/μCi x 20 rem/rad = 0.002 rem

Fourth-month intake

0.061 μCi x 0.49 x 7.98E–03 rad/μCi x 20 rem/rad = 0.005 rem

Fifth-month intake

0.15 μCi x 0.49 x 1.31E–02 rad/μCi x 20 rem/rad = 0.019 rem

TOTAL = 0.026 rem

Class Y Inhalation Intake

Dose Equivalent = Intake x TF_1 x DF_i x 20 rem/rad

Third-month intake

0.089 μCi x 0.079 x 4.75E–03 rad/μCi x 20 rem/rad = 0.001 rem

Fourth-month intake

0.14 μCi x 0.079 x 7.98E–03 rad/μCi x 20 rem/rad = 0.002 rem

Fifth-month intake

0.35 μCi x 0.079 x 1.31E–02 rad/μCi x 20 rem/rad = 0.007 rem

TOTAL = 0.010 rem

The dose to the embryo/fetus resulting from each single-month intake should be determined by adding the Class D component with the Class Y component. The total gestation period dose is the sum of the cumulated dose resulting from each monthly intake.

Gestation Month	Class D Dose (rem)	Class Y Dose (rem)	Total Dose (rem)
3rd Month (61 – 90 days)	0.002	0.001	0.003
4th Month (91 –120 days)	0.005	0.002	0.007
5th Month (121 –150 days)	0.019	0.007	0.026
TOTAL	0.036 rem		

CASE 5

PRE-EXISTING MATERNAL BODY BURDEN AT TIME OF PREGNANCY

5.1 Exposure Scenario

A declared pregnant woman is determined to have an existing body burden of ^{137}Cs at the time of pregnancy. The burden is a result of an acute inhalation intake that occurred around 2 months prior to the pregnancy. Extrapolating from bioassay measurements, the body burden at the time of pregnancy is estimated to be 2.8 μCi.

5.2 Evaluating the 1% ALI Threshold

The regulatory position specified in Section 1.6 of the guide states that if a body burden existing at time of pregnancy exceeds 1% of the stochastic ALI for the appropriate mode of intake (ingestion or inhalation), the dose to the embryo/fetus from this burden should be evaluated. From Appendix B to 10 CFR 20.1001–20.2401, the inhalation stochastic ALI value for ^{137}Cs is 200 μCi (Column 2 entry under Table 1 of the appendix). Since the existing burden of 2.8 μCi is larger than 1% of this ALI value, the dose to the embryo/fetus should be evaluated.

5.3 Determining Blood Uptake

The regulatory position specified in Section 2.6 of the guide states that the total burden determined to exist at the time of pregnancy should be assumed to be available for uptake in the blood of the woman. Therefore, for this example, blood uptake should be assumed to be the same as the existing body burden of 2.8 μCi.

5.4 Calculating the Embryo/Fetus Dose Equivalent

5.4.1 Simplified Method

With the assumption that the blood uptake equates to the body burden existing at the time of pregnancy, the dose to the embryo/fetus is calculated simply by multiplying the burden by the radionuclide dose factor. From Appendix A to the guide, the dose factor for ^{137}Cs is 5.94E–02 rem/μCi (in blood); therefore, the dose is calculated as follows:

$$\begin{aligned}\text{Dose Equivalent} &= A_i \text{ (pre-existing burden)} \\ &\quad \times DF_i \\ &= 2.8\ \mu\text{Ci} \times 5.94\text{E–}02 \\ &\quad \text{rem}/\mu\text{Ci} \\ &= 0.166 \text{ rem}\end{aligned}$$

5.4.2 Method Using Revision 1 to NUREG/CR–5631

Similar to the calculation above, the dose to the embryo/fetus is calculated by multiplying the body burden existing at time of pregnancy by the appropriate gestation-time dependent dose factor. Table C19 of Appendix C to this guide presents the gestation-time dependent dose factors for ^{137}Cs. In this table, the left-most column specifies the beginning time for each monthly gestation period (e.g., 0 for 0–30 days, 31 for 31–60 days). The right-most column presents the corresponding cumulated dose to the embryo/fetus for the remainder of the gestation period for unit activity introduced into the maternal blood. As stated in the regulatory position specified in Section 3.2.3 of the guide, the uptake in the blood for burdens existing at time of pregnancy should be assumed to occur during the first month of pregnancy.* From this table, the cumulated dose factor for a ^{137}Cs intake during the first month of gestation is 5.83E–02 rad/μCi uptake into blood. As discussed in Section 3.2.1 of the guide, a radiation quality factor of 1.0 should be used for ^{137}Cs. The dose equivalent to the embryo/fetus is calculated as follows:

$$\begin{aligned}\text{Dose Equivalent} &= A_i \text{ (pre-existing burden)} \\ &\quad \times DF_i \times 1.0 \text{ rem/rad} \\ &= 2.8\ \mu\text{Ci} \times 5.83\text{E–}02 \\ &\quad \text{rad}/\mu\text{Ci} \times 1.0 \text{ rem/rad} \\ &= 0.163 \text{ rem}\end{aligned}$$

*The regulatory position specified in Section 3.2.3 of the guide allows the use of time-dependent release kinetics for estimating the uptake in the maternal blood. This in-depth evaluation may be warranted for unusual exposure situations; however, for this example, the simplifying assumption of total uptake during the first month will be used. Also, note that for certain radionuclides a blood uptake at the beginning of the gestation period results in a negligible dose contribution to the embryo/fetus. For these radionuclides, per guidance of the regulatory position specified in Section 3.2.3 and Appendix C, the cumulated dose value for the second month of the gestation period (i.e., the 31-day gestation time) should be used.

CASE 6

MATERNAL CHRONIC EXTERNAL EXPOSURE AND INHALATION INTAKE

.1 Exposure Scenario

During the processing of byproduct material specimens, a woman receives periodic exposure to airborne levels of ^{137}Cs and ^{144}Ce. The lung clearance class for all compounds of cesium is Class D; and for cerium the chemical compound is determined to be an oxide, thereby representing a "Y" lung clearance class. The woman becomes pregnant. However, she does not inform her employer (the licensee) until the third month of the gestation period. At this time, she becomes a declared pregnant woman and the more restrictive dose limits of 10 CFR 20.1208 for the embryo/fetus become applicable. Once declared, past exposures incurred during the gestation period and any burdens existing at time of pregnancy should be evaluated.

The licensee evaluates the dosimetry records for the declared pregnant woman, including air sample data and bioassay measurements. It is determined that at the time of pregnancy the woman had an existing body burden of 1.14 μCi of ^{137}Cs and 0.12 μCi of ^{144}Ce. Intakes during the first, second, and third months of the gestation period are determined and are presented in the following table:

Stage of Gestation at Time of Intake (days)	Total Intake (μCi)	
	^{137}Cs (Class D)	^{144}Ce (Class Y)
Pre-Existing	1.14	0.12
0 - 30	0.48	0.078
31 - 60	0.76	0.14
61 - 90	0.23	0.093

The declared pregnant woman's external exposure is evaluated and is determined to be 0.285 rem from the time of pregnancy to the time of declaration. After declaration, the licensee imposes radiological controls to ensure that additional exposures are kept to a minimum, pending a thorough evaluation of the woman's exposures and the resultant embryo/fetus dose equivalent.

6.2 Evaluating Embryo/Fetus Dose Equivalent from Pre-Existing Body Burden

6.2.1 Evaluating the 1% ALI Threshold

The regulatory position specified in Section 1.6 of the guide states that if a body burden existing at time of pregnancy exceeds 1% of the stochastic ALI for the appropriate mode of intake (ingestion or inhalation), the dose to the embryo/fetus from this burden should be evaluated. From Appendix B to 10 CFR 20.1001-20.2401, the inhalation stochastic ALI value for ^{137}Cs is 200 μCi, and for Class Y ^{144}Ce is 10 μCi (Column 2 entry under Table 1 of the appendix). Since the sum of the existing burdens of 1.14 μCi of ^{137}Cs and 0.12 μCi of ^{144}Ce divided by their respective ALI values is larger than 0.01 (i.e., Σ (burden$_i$ ÷ ALI$_i$) > 0.01), the dose to the embryo/fetus resulting from the maternal pre-existing burden should be evaluated.

6.2.2 Determining Blood Uptake

The regulatory position specified in Section 2.6 of the guide states that the total burden determined to exist at the time of pregnancy should be assumed to be available for uptake in the blood of the woman. Therefore, for this example, blood uptake should be assumed to be the same as the existing body burdens of 1.14 μCi of ^{137}Cs and 0.12 μCi of ^{144}Ce.

6.2.3 Calculating the Embryo/Fetus Dose Equivalent from Pre-Existing Burden

Only the Simplified Method will be used in this example for calculating the embryo/fetus doses. For ^{137}Cs, the approach for using the gestation-time dependent method (Revision 1 to NUREG/CR-5631 method) would be similar to the calculations presented in Case 5, Section 5.4.2. For ^{144}Ce, gestation-time dependent dose factors have not been developed.

With the assumption that the blood uptake equates to the body burden existing at the time of pregnancy, the dose to the embryo/fetus is calculated simply by multiplying the burden by the radionuclide dose factor. From Appendix A to the guide, the dose factor for ^{137}Cs is 5.94E-02 rem/μCi (in blood) and for ^{144}Ce is 3.79E-01 rem/μCi (in blood). The dose is calculated as follows:

Dose Equivalent = Σ A_i (pre-existing burden
x DF_i
= (1.14 μCi
x 5.94E-02 rem/μCi)
+ (0.12 μCi
x 3.79E-01 rem/μCi)
= 0.068 + 0.045 rem
= 0.113 rem

6.3 Calculating the Embryo/Fetus Dose Equivalent from Intakes During Pregnancy

6.3.1 Evaluating 1% ALI Threshold

Based on the requirements of 10 CFR 20.1502(b)(2) and the regulatory position specified in Section 1.1 of this guide, the dose to the embryo/fetus is to be evaluated if intakes during the year by the declared pregnant woman are likely to exceed 1% of the stochastic ALIs. Without having to consider other intakes by the woman during the year, the 1% threshold is exceeded based on the intakes by the declared pregnant woman during the first 3 months of the pregnancy. Therefore, an evaluation of the embryo/fetus dose is required.

With multiple intakes occurring during a single monthly period, the intakes may be modeled as cumulative intakes within each specified gestational monthly period.

6.3.2 Determining Blood Uptake

The calculation of the dose to the embryo/fetus is based on the amount of the intake that is available for uptake within the first transfer compartment (i.e., blood). Also, the transfer to the blood is a function of the lung clearance class. Applying the guidance of the regulatory positions specified in Sections 2.3 and 2.4, the transfer fraction (TF_i) of inhaled activity to the first transfer compartment for a Class D compound may be calculated as follows:

$$TF_i \text{ (Class D)} = 0.48 + 0.15 \times f_{1,i}$$

where:

TF_i = transfer fraction of inhaled activity to the first transfer compartment

$f_{1,i}$ = gut-to-blood transfer factor for radionuclide i (from Appendix B to this guide)

0.48 = fraction of inhalation intake that is cleared directly from the lung to the blood for Class D compounds

0.15 = fraction of inhaled radionuclide that is cleared from the lung to the GI tract for Class D compounds

The resultant total blood uptake is calculated by multiplying the TF_i value by the inhalation intake:

$$\text{Blood Uptake} = TF_i \times \text{Inhalation Intake}$$

For a Class Y compound, the transfer fraction is calculated as follows:

$$TF_i \text{ (Class Y)} = 0.05 + 0.58 \times f_{1,i}$$

where:

0.05 = fraction of inhalation intake that is cleared directly from the lung to the blood for Class Y compounds

0.58 = fraction of inhaled radionuclide that is cleared from the lung to the GI tract for Class Y compounds

The total blood uptake can be calculated in the same manner as discussed above for the Class D compound.

For cesium, the f_1 value from Appendix B to this guide is 1.0; for cerium, the value is 3E-04. Applying the above equations, the amounts of ^{137}Cs and ^{144}Ce that are transferred to the blood as a function of gestation period are presented in the following table:

Stage of Gestation at Time of Intake (days)	Transfer Fraction and Blood Uptake of ^{137}Cs (Class D)		Transfer Fraction and Blood Uptake of ^{144}Ce (Class Y)	
	Transfer Fraction (TF_i)	Blood Uptake (μCi)	Transfer Fraction (TF_i)	Blood Uptake (μCi)
0 - 30	0.63	0.30	0.050	0.0039
31 - 60	0.63	0.48	0.050	0.0070
61 - 90	0.63	0.14	0.050	0.0046

6.3.3 Calculating Embryo/Fetus Dose Equivalent from Maternal Intakes

Only the Simplified Method will be used in this example for calculating the embryo/fetus doses. For ^{137}Cs, the approach of the gestation-time dependent method (the method in Revision 1 to NUREG/CR-5631) would be similar to the calculations presented in Case 4, Section 4.3.2, of this Appendix D. For ^{144}Ce, gestation-time dependent dose factors have not been developed. The dose to the embryo/fetus is calculated by using Equation 2 from the regulatory position specified in Section 2.5 of this guide. From Appendix A, the dose factor for ^{137}Cs is 5.94E–02 rem/μCi (in blood) and for ^{144}Ce is 3.79E–01 rem/μCi (in blood). Applying these dose factors along with the monthly transfer fractions (as calculated above) results in the following dose calculations:

Class D Inhalation Intake—^{137}Cs

Dose Equivalent = Intake x TF_1 x DF_i

First-month intake

0.48 μCi x 0.63 x 5.94E–02 rem/μCi
= 0.018 rem

Second-month intake

0.76 μCi x 0.63 x 5.94E–02 rem/μCi
= 0.028 rem

Third-month intake

0.23 μCi x 0.63 x 5.94E–02 rem/μCi
= 0.009 rem

TOTAL = 0.055 rem

Class Y Inhalation Intake—^{144}Ce

Dose Equivalent = Intake x TF_1 x DF_i

First-month intake

0.078 μCi x 0.050 x 3.79E–01 rem/μCi
= 0.001 rem

Second-month intake

0.14 μCi x 0.050 x 3.79E–01 rem/μCi
= 0.003 rem

Third-month intake

0.093 μCi x 0.050 x 3.79E–01 rem/μCi
= 0.002 rem

TOTAL = 0.006 rem

6.4 Summing Internal and External Doses

The doses to the embryo/fetus for the existing maternal burden, the maternal inhalation intakes, and the deep-dose equivalent to the declared pregnant woman are summarized in the following table:

Exposure Pathway and Stage of Gestation	Embryo/Fetus Dose Equivalent (rem)		
	^{137}Cs	^{144}Cs	Total
Pre-Existing Body Burden	0.068	0.045	0.113
Inhalation Intakes (0 – 30 days)	0.018	0.001	0.019
Inhalation Intakes (31 – 60 days)	0.028	0.003	0.031
Inhalation Intakes (61 – 90 days)	0.009	0.002	0.011
Deep-Dose Equivalent (0 – 90 days)	0.285		
Total	0.459		

The sum of the deep-dose equivalent to the declared pregnant woman and the embryo/fetus dose resulting from the inhalation intakes of the declared pregnant woman represents the total dose equivalent to the embryo/fetus (i.e., 0.285 rem deep-dose equivalent, plus 0.174 rem dose equivalent from internal exposures). This total of 0.459 rem is within 0.05 rem of the 0.5 rem limit for the embryo/fetus. Therefore, the dose limit for the embryo/fetus for the remainder of the gestation period is an additional dose of 0.05 rem from the date of the declared pregnancy (refer to 10 CFR 20.1208(d)).

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., Washington, DC, as an enclosure to Part 20 (56 FR 23360).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

U.S. NUCLEAR REGULATORY COMMISSION

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REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

(This page reissued
May 1977)

REGULATORY GUIDE 8.10

OPERATING PHILOSOPHY FOR MAINTAINING OCCUPATIONAL RADIATION EXPOSURES AS LOW AS IS REASONABLY ACHIEVABLE

A. INTRODUCTION

Paragraph 20.1(c) of 10 CFR Part 20, "Standards for Protection Against Radiation," states, in part, that licensees should make every reasonable effort to maintain radiation exposures as far below the limits specified in that part as practicable. This guide describes to licensees a general operating philosophy acceptable to the NRC staff as a necessary basis for a program of maintaining occupational exposures to radiation as low as is reasonably achievable.

Both this guide and Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as is Reasonably Achievable (Nuclear Power Reactors)," deal with the concept of "as low as is reasonably achievable" occupational exposures to radiation. The main difference between the two guides, aside from the fact that Regulatory Guide 8.8 applies only to nuclear power reactors and this guide applies to all specific licensees, is that Regulatory Guide 8.8 is addressed to applicants for a license and tells them what information relevant to "as low as is reasonably achievable" should be included in their license applications. This guide, on the other hand, describes an operating philosophy that the NRC staff believes all specific licensees should follow to keep occupational exposures to radiation as low as is reasonably achievable.

B. DISCUSSION

Even though current occupational exposure limits provide a very low risk of injury, it is prudent to avoid unnecessary exposure to radiation. The objective is thus to reduce occupational exposures as far below the specified limits as is reasonably achievable by means of good radiation protection planning and practice, as well as by management commitment to policies that foster vigilance against departures from good practice.

In addition to maintaining doses to individuals as far below the limits as is reasonably achievable, the sum of the doses received by all exposed individuals should also be maintained at the lowest practicable level. It would not be desirable, for example, to hold the highest doses to individuals to some fraction of the applicable limit if this involved exposing additional people and significantly increasing the sum of radiation doses received by all involved individuals.

C. REGULATORY POSITION

Two basic conditions are considered necessary in any program for keeping occupational exposures as far below the specified limits as is reasonably achievable. The management of the licensed facility should be committed to maintaining exposures as low as is reasonably achievable, and the personnel responsible for radiation protection should be continually vigilant for means to reduce exposures.

1. Management Commitment

The commitment made by licensee management to minimize exposures should provide clearly defined radiation protection responsibilities and an environment in which the radiation protection staff can do its job properly. There are several aspects to this commitment:

a. Plant personnel should be made aware of management's commitment to keep occupational exposures as low as is reasonably achievable. The commitment should appear in policy statements, instructions to personnel, and similar documents. As a minimum, workers should be sufficiently familiar with this commitment that they can explain what the management commitment is, what "as low as is reasonably achievable exposure to radiation" means, why it is recommended, and how they have been advised to implement it on their jobs.

The first page of this guide is being reissued with the words "For Comment" deleted. The staff's consideration of comments received during the initial public comment period has resulted in the determination that there is no need for a revision at this time.

It is suggested that you attach this page to the first page of the complete guide. No changes have been made to the text of either this page or the remainder of the guide.

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

Revision 1
September 1975

REGULATORY GUIDE 8.10

OPERATING PHILOSOPHY FOR MAINTAINING OCCUPATIONAL RADIATION EXPOSURES AS LOW AS IS REASONABLY ACHIEVABLE

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b. **Management should periodically perform a formal audit to determine how exposures might be lowered.** This should include reviews of operating procedures and past exposure records, plant inspections, and consultations with the radiation protection staff or outside consultants. As a minimum, management should be able to discuss which operating procedures were reviewed, in which locations most exposures are being received, what groups of workers are receiving the highest exposures, what discussions they have had with the radiation protection staff or outside consultants, and what steps they have taken to reduce exposures.

* c. **The management should ensure that there is a well-supervised radiation protection capability with well-defined responsibilities.** The qualifications for the Radiation Protection Manager for a nuclear power reactor facility are presented in Regulatory Guides 1.8 and 8.8. Applicants submitting applications for any specific license other than a nuclear power reactor license should select and state the qualifications for the lead individual who will be responsible for implementing the radiation protection program for the facility, i.e., the Radiation Safety Officer (RSO).[1] The qualifications selected should be commensurate with the potential problems anticipated to be encountered in a facility of the type subject to the license.

d. **The management should see that plant workers receive sufficient training.** Section 19.12 of 10 CFR Part 19 requires instruction of personnel on radiation protection. The radiation worker should understand how radiation protection relates to his job and should be tested on this understanding at least once per year. He should have frequent opportunities to discuss radiation safety with the radiation protection staff whenever the need arises. Management should be committed to a review of radiation protection at least once every three years. Training should be sufficient to ensure that the workers can correctly answer questions on radiation protection as it relates to their jobs.

e. **The RSO should be given sufficient authority to enforce safe plant operation.** The RSO should have the authority to prevent unsafe practices and to communicate promptly with an appropriate level of management about halting an operation he deems unsafe. Operating procedures related to radiation safety should be reviewed and approved by radiation protection personnel. This authority should be demonstrable by written policy statements.

f. **Modifications to operating and maintenance procedures and to plant equipment and facilities should be made where they will substantially reduce exposures at a reasonable cost.** The management should be able to demonstrate that improvements have been *sought*, that modifications have been considered, and that they have been implemented where practicable. Where modifications have been considered but not implemented, the licensee should be prepared to describe the reasons for not implementing them.

2. Vigilance by the RSO and the Radiation Protection Staff

It should be the responsibility of the RSO and the radiation protection staff to conduct surveillance programs and investigations to ensure that occupational exposures are as far below the specified limits as is reasonably achievable. Additionally, they should be vigilant in searching out new and better ways to perform all radiation jobs with less exposure. There are several aspects to this responsibility.

a. **The RSO and the radiation protection staff should know the origins of radiation exposures in the plant.** They should know these by location, operation, and job category and should be aware of trends in exposures. Where radiation work permits are used, exposures received should be recorded on the permits. The RSO and the radiation protection staff should be able to describe which locations, operations, and jobs are associated with the highest exposures and why exposures are increasing or decreasing.

b. **The RSO and the radiation protection staff should look for ways to reduce exposures.** When unusual exposures have occurred, the radiation protection staff should direct and participate in an investigation of the circumstances of such exposures to determine the causes and take steps to reduce the likelihood of similar future occurrences. For each such occurrence, the RSO should be able to demonstrate that such an investigation has been carried out, that conclusions were reached as a result of the investigation, and that corrective action was taken, as appropriate.

The RSO and the radiation protection staff should periodically review operating procedures that may affect radiation safety and survey plant operations to identify situations in which exposures can be reduced. Indicated changes should be promptly implemented. Procedures for receiving and evaluating suggestions relating to radiation protection from employees should be established. Workers should be knowledgeable of the procedures for making suggestions on radiation protection.

c. **Adequate equipment and supplies for radiation protection work should be provided.** The RSO should be responsible for ensuring that proper equipment and supplies are available, are maintained in good working order, and are used properly. Written procedures for the use of the equipment should be available and followed.

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*Lines indicate substantive changes from previous issue.

[1] The term "Radiation Safety Officer" is used by many licensees; other terms are equally acceptable.

D. IMPLEMENTATION

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

Except in those cases in which the applicant or licensee proposes an alternative method for complying with the specified portions of the Commission's regulations, the methods described herein will be used in the evaluation of submittals in connection with applications for a specific license.

Regulatory Guides 1.8 and 8.8 address nuclear power reactor facilities specifically and will be used by the NRC staff in evaluating submittals in connection with licensing actions for nuclear power reactors.

U.S. NUCLEAR REGULATORY COMMISSION

Revision 3
June 1999

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.13

(Draft was issued as DG-8014)

INSTRUCTION CONCERNING PRENATAL RADIATION EXPOSURE

A. INTRODUCTION

The Code of Federal Regulations in 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," in Section 19.12, "Instructions to Workers," requires instruction in "the health protection problems associated with exposure to radiation and/or radioactive material, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed." The instructions must be "commensurate with potential radiological health protection problems present in the work place."

The Nuclear Regulatory Commission's (NRC's) regulations on radiation protection are specified in 10 CFR Part 20, "Standards for Protection Against Radiation"; and 10 CFR 20.1208, "Dose to an Embryo/Fetus," requires licensees to "ensure that the dose to an embryo/fetus during the entire pregnancy, due to occupational exposure of a declared pregnant woman, does not exceed 0.5 rem (5 mSv)." Section 20.1208 also requires licensees to "make efforts to avoid substantial variation above a uniform monthly exposure rate to a declared pregnant woman." A declared pregnant woman is defined in 10 CFR 20.1003 as a woman who has voluntarily informed her employer, in writing, of her pregnancy and the estimated date of conception.

This regulatory guide is intended to provide information to pregnant women, and other personnel, to help them make decisions regarding radiation exposure during pregnancy. This Regulatory Guide 8.13 supplements Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure" (Ref. 1), which contains a broad discussion of the risks from exposure to ionizing radiation.

Other sections of the NRC's regulations also specify requirements for monitoring external and internal occupational dose to a declared pregnant woman. In 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose," licensees are required to monitor the occupational dose to a declared pregnant woman, using an individual monitoring device, if it is likely that the declared pregnant woman will receive, from external sources, a deep dose equivalent in excess of 0.1 rem (1 mSv). According to Paragraph (e) of 10 CFR 20.2106, "Records of Individual Monitoring Results," the licensee must maintain records of dose to an embryo/fetus if monitoring was required, and the records of dose to the embryo/fetus must be kept with the records of dose to the declared pregnant woman. The declaration of pregnancy must be kept on file, but may be maintained separately from the dose records. The licensee must retain the re-

quired form or record until the Commission terminates each pertinent license requiring the record.

The information collections in this regulatory guide are covered by the requirements of 10 CFR Parts 19 or 20, which were approved by the Office of Management and Budget, approval numbers 3150-0044 and 3150-0014, respectively. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

B. DISCUSSION

As discussed in Regulatory Guide 8.29 (Ref. 1), exposure to any level of radiation is assumed to carry with it a certain amount of risk. In the absence of scientific certainty regarding the relationship between low dose exposure and health effects, and as a conservative assumption for radiation protection purposes, the scientific community generally assumes that any exposure to ionizing radiation may cause undesirable biological effects and that the likelihood of these effects increases as the dose increases. At the occupational dose limit for the whole body of 5 rem (50 mSv) per year, the risk is believed to be very low.

The magnitude of risk of childhood cancer following in utero exposure is uncertain in that both negative and positive studies have been reported. The data from these studies "are consistent with a lifetime cancer risk resulting from exposure during gestation which is two to three times that for the adult" (NCRP Report No. 116, Ref. 2). The NRC has reviewed the available scientific literature and has concluded that the 0.5 rem (5 mSv) limit specified in 10 CFR 20.1208 provides an adequate margin of protection for the embryo/fetus. This dose limit reflects the desire to limit the total lifetime risk of leukemia and other cancers associated with radiation exposure during pregnancy.

In order for a pregnant worker to take advantage of the lower exposure limit and dose monitoring provisions specified in 10 CFR Part 20, the woman must declare her pregnancy in writing to the licensee. A form letter for declaring pregnancy is provided in this guide or the licensee may use its own form letter for declaring pregnancy. A separate written declaration should be submitted for each pregnancy.

C. REGULATORY POSITION

1. Who Should Receive Instruction

Female workers who require training under 10 CFR 19.12 should be provided with the information contained in this guide. In addition to the information contained in Regulatory Guide 8.29 (Ref. 1), this information may be included as part of the training required under 10 CFR 19.12.

2. Providing Instruction

The occupational worker may be given a copy of this guide with its Appendix, an explanation of the contents of the guide, and an opportunity to ask questions and request additional information. The information in this guide and Appendix should also be provided to any worker or supervisor who may be affected by a declaration of pregnancy or who may have to take some action in response to such a declaration.

Classroom instruction may supplement the written information. If the licensee provides classroom instruction, the instructor should have some knowledge of the biological effects of radiation to be able to answer questions that may go beyond the information provided in this guide. Videotaped presentations may be used for classroom instruction. Regardless of whether the licensee provides classroom training, the licensee should give workers the opportunity to ask questions about information contained in this Regulatory Guide 8.13. The licensee may take credit for instruction that the worker has received within the past year at other licensed facilities or in other courses or training.

3. Licensee's Policy on Declared Pregnant Women

The instruction provided should describe the licensee's specific policy on declared pregnant women, including how those policies may affect a woman's work situation. In particular, the instruction should include a description of the licensee's policies, if any, that may affect the declared pregnant woman's work situation after she has filed a written declaration of pregnancy consistent with 10 CFR 20.1208.

The instruction should also identify who to contact for additional information as well as identify who should receive the written declaration of pregnancy. The recipient of the woman's declaration may be identified by name (e.g., John Smith), position (e.g., immediate supervisor, the radiation safety officer), or department (e.g., the personnel department).

4. Duration of Lower Dose Limits for the Embryo/Fetus

The lower dose limit for the embryo/fetus should remain in effect until the woman withdraws the declaration in writing or the woman is no longer pregnant. If a declaration of pregnancy is withdrawn, the dose limit for the embryo/fetus would apply only to the time from the estimated date of conception until the time the declaration is withdrawn. If the declaration is

not withdrawn, the written declaration may be considered expired one year after submission.

5. Substantial Variations Above a Uniform Monthly Dose Rate

According to 10 CFR 20.1208(b), "The licensee shall make efforts to avoid substantial variation above a uniform monthly exposure rate to a declared pregnant woman so as to satisfy the limit in paragraph (a) of this section," that is, 0.5 rem (5 mSv) to the embryo/fetus. The National Council on Radiation Protection and Measurements (NCRP) recommends a monthly equivalent dose limit of 0.05 rem (0.5 mSv) to the embryo/fetus once the pregnancy is known (Ref. 2). In view of the NCRP recommendation, any monthly dose of less than 0.1 rem (1 mSv) may be considered as not a substantial variation above a uniform monthly dose rate and as such will not require licensee justification. However, a monthly dose greater than 0.1 rem (1 mSv) should be justified by the licensee.

D. IMPLEMENTATION

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

Unless a licensee or an applicant proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used by the NRC staff in the evaluation of instructions to workers on the radiation exposure of pregnant women.

REFERENCES

1. USNRC, "Instruction Concerning Risks from Occupational Radiation Exposure," Regulatory Guide 8.29, Revision 1, February 1996.

2. National Council on Radiation Protection and Measurements, *Limitation of Exposure to Ionizing Radiation*, NCRP Report No. 116, Bethesda, MD, 1993.

APPENDIX

QUESTIONS AND ANSWERS CONCERNING PRENATAL RADIATION EXPOSURE

1. Why am I receiving this information?

The NRC's regulations (in 10 CFR 19.12, "Instructions to Workers") require that licensees instruct individuals working with licensed radioactive materials in radiation protection as appropriate for the situation. The instruction below describes information that occupational workers and their supervisors should know about the radiation exposure of the embryo/fetus of pregnant women.

The regulations allow a pregnant woman to decide whether she wants to formally declare her pregnancy to take advantage of lower dose limits for the embryo/fetus. This instruction provides information to help women make an informed decision whether to declare a pregnancy.

2. If I become pregnant, am I required to declare my pregnancy?

No. The choice whether to declare your pregnancy is completely voluntary. If you choose to declare your pregnancy, you must do so in writing and a lower radiation dose limit will apply to your embryo/fetus. If you choose not to declare your pregnancy, you and your embryo/fetus will continue to be subject to the same radiation dose limits that apply to other occupational workers.

3. If I declare my pregnancy in writing, what happens?

If you choose to declare your pregnancy in writing, the licensee must take measures to limit the dose to your embryo/fetus to 0.5 rem (5 millisievert) during the entire pregnancy. This is one-tenth of the dose that an occupational worker may receive in a year. If you have already received a dose exceeding 0.5 rem (5 mSv) in the period between conception and the declaration of your pregnancy, an additional dose of 0.05 rem (0.5 mSv) is allowed during the remainder of the pregnancy. In addition, 10 CFR 20.1208, "Dose to an Embryo/Fetus," requires licensees to make efforts to avoid substantial variation above a uniform monthly dose rate so that all the 0.5 rem (5 mSv) allowed dose does not occur in a short period during the pregnancy.

This may mean that, if you declare your pregnancy, the licensee may not permit you to do some of your normal job functions if those functions would have allowed you to receive more than 0.5 rem, and you may not be able to have some emergency response responsibilities.

4. Why do the regulations have a lower dose limit for the embryo/fetus of a declared pregnant woman than for a pregnant worker who has not declared?

A lower dose limit for the embryo/fetus of a declared pregnant woman is based on a consideration of greater sensitivity to radiation of the embryo/fetus and the involuntary nature of the exposure. Several scientific advisory groups have recommended (References 1 and 2) that the dose to the embryo/fetus be limited to a fraction of the occupational dose limit.

5. What are the potentially harmful effects of radiation exposure to my embryo/fetus?

The occurrence and severity of health effects caused by ionizing radiation are dependent upon the type and total dose of radiation received, as well as the time period over which the exposure was received. See Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Exposure" (Ref. 3), for more information. The main concern is embryo/fetal susceptibility to the harmful effects of radiation such as cancer.

6. Are there any risks of genetic defects?

Although radiation injury has been induced experimentally in rodents and insects, and in the experiments was transmitted and became manifest as hereditary disorders in their offspring, radiation has not been identified as a cause of such effect in humans. Therefore, the risk of genetic effects attributable to radiation exposure is speculative. For example, no genetic effects have been documented in any of the Japanese atomic bomb survivors, their children, or their grandchildren.

7. What if I decide that I do not want any radiation exposure at all during my pregnancy?

You may ask your employer for a job that does not involve any exposure at all to occupational radiation dose, but your employer is not obligated to provide you with a job involving no radiation exposure. Even if you receive no occupational exposure at all, your embryo/fetus will receive some radiation dose (on average 75 mrem (0.75 mSv)) during your pregnancy from natural background radiation.

The NRC has reviewed the available scientific literature and concluded that the 0.5 rem (5 mSv) limit

provides an adequate margin of protection for the embryo/fetus. This dose limit reflects the desire to limit the total lifetime risk of leukemia and other cancers. If this dose limit is exceeded, the total lifetime risk of cancer to the embryo/fetus may increase incrementally. However, the decision on what level of risk to accept is yours. More detailed information on potential risk to the embryo/fetus from radiation exposure can be found in References 2-10.

8. What effect will formally declaring my pregnancy have on my job status?

Only the licensee can tell you what effect a written declaration of pregnancy will have on your job status. As part of your radiation safety training, the licensee should tell you the company's policies with respect to the job status of declared pregnant women. In addition, before you declare your pregnancy, you may want to talk to your supervisor or your radiation safety officer and ask what a declaration of pregnancy would mean specifically for you and your job status.

In many cases you can continue in your present job with no change and still meet the dose limit for the embryo/fetus. For example, most commercial power reactor workers (approximately 93%) receive, in 12 months, occupational radiation doses that are less than 0.5 rem (5 mSv) (Ref. 11). The licensee may also consider the likelihood of increased radiation exposures from accidents and abnormal events before making a decision to allow you to continue in your present job.

If your current work might cause the dose to your embryo/fetus to exceed 0.5 rem (5 mSv), the licensee has various options. It is possible that the licensee can and will make a reasonable accommodation that will allow you to continue performing your current job, for example, by having another qualified employee do a small part of the job that accounts for some of your radiation exposure.

9. What information must I provide in my written declaration of pregnancy?

You should provide, in writing, your name, a declaration that you are pregnant, the estimated date of conception (only the month and year need be given), and the date that you give the letter to the licensee. A form letter that you can use is included at the end of these questions and answers. You may use that letter, use a form letter the licensee has provided to you, or write your own letter.

10. To declare my pregnancy, do I have to have documented medical proof that I am pregnant?

NRC regulations do not require that you provide medical proof of your pregnancy. However, NRC regulations do not preclude the licensee from requesting medical documentation of your pregnancy, especially if a change in your duties is necessary in order to comply with the 0.5 rem (5 mSv) dose limit.

11. Can I tell the licensee orally rather than in writing that I am pregnant?

No. The regulations require that the declaration must be in writing.

12. If I have not declared my pregnancy in writing, but the licensee suspects that I am pregnant, do the lower dose limits apply?

No. The lower dose limits for pregnant women apply only if you have declared your pregnancy in writing. The United States Supreme Court has ruled (in *United Automobile Workers International Union v. Johnson Controls, Inc.*, 1991) that "Decisions about the welfare of future children must be left to the parents who conceive, bear, support, and raise them rather than to the employers who hire those parents" (Reference 7). The Supreme Court also ruled that your employer may not restrict you from a specific job "because of concerns about the next generation." Thus, the lower limits apply only if you choose to declare your pregnancy in writing.

13. If I am planning to become pregnant but am not yet pregnant and I inform the licensee of that in writing, do the lower dose limits apply?

No. The requirement for lower limits applies only if you declare in writing that you are already pregnant.

14. What if I have a miscarriage or find out that I am not pregnant?

If you have declared your pregnancy in writing, you should promptly inform the licensee in writing that you are no longer pregnant. However, if you have not formally declared your pregnancy in writing, you need not inform the licensee of your nonpregnant status.

15. How long is the lower dose limit in effect?

The dose to the embryo/fetus must be limited until you withdraw your declaration in writing or you inform the licensee in writing that you are no longer pregnant. If the declaration is not withdrawn, the written declaration may be considered expired one year after submission.

16. If I have declared my pregnancy in writing, can I revoke my declaration of pregnancy even if I am still pregnant?

Yes, you may. The choice is entirely yours. If you revoke your declaration of pregnancy, the lower dose limit for the embryo/fetus no longer applies.

17. What if I work under contract at a licensed facility?

The regulations state that you should formally declare your pregnancy to the licensee in writing. The licensee has the responsibility to limit the dose to the embryo/fetus.

18. Where can I get additional information?

The references to this Appendix contain helpful information, especially Reference 3, NRC's Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure," for general information on radiation risks. The licensee should be able to give this document to you.

For information on legal aspects, see Reference 7, "The Rock and the Hard Place: Employer Liability to Fertile or Pregnant Employees and Their Unborn Children—What Can the Employer Do?" which is an article in the journal *Radiation Protection Management*.

You may telephone the NRC Headquarters at (301) 415-7000. Legal questions should be directed to the Office of the General Counsel, and technical questions should be directed to the Division of Industrial and Medical Nuclear Safety.

You may also telephone the NRC Regional Offices at the following numbers: Region I, (610) 337-5000; Region II, (404) 562-4400; Region III, (630) 829-9500; and Region IV, (817) 860-8100. Legal questions should be directed to the Regional Counsel, and technical questions should be directed to the Division of Nuclear Materials Safety.

REFERENCES FOR APPENDIX

1. National Council on Radiation Protection and Measurements, *Limitation of Exposure to Ionizing Radiation*, NCRP Report No. 116, Bethesda, MD, 1993.

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4. Committee on the Biological Effects of Ionizing Radiations, National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation* (BEIR V), National Academy Press, Washington, DC, 1990.

5. United Nations Scientific Committee on the Effects of Atomic Radiation, *Sources and Effects of Ionizing Radiation*, United Nations, New York, 1993.

6. R. Doll and R. Wakeford, "Risk of Childhood Cancer from Fetal Irradiation," *The British Journal of Radiology*, 70, 130-139, 1997.

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8. National Council on Radiation Protection and Measurements, *Considerations Regarding the Unintended Radiation Exposure of the Embryo, Fetus, or Nursing Child*, NCRP Commentary No. 9, Bethesda, MD, 1994.

9. National Council on Radiation Protection and Measurements, *Risk Estimates for Radiation Protection*, NCRP Report No. 115, Bethesda, MD, 1993.

10. National Radiological Protection Board, *Advice on Exposure to Ionising Radiation During Pregnancy*, National Radiological Protection Board, Chilton, Didcot, UK, 1998.

11. M.L. Thomas and D. Hagemeyer, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1996," Twenty-Ninth Annual Report, NUREG-0713, Vol. 18, USNRC, 1998.[2]

[1]Single copies of regulatory guides, both active and draft, and draft NUREG documents may be obtained free of charge by writing the Reproduction and Distribution Services Section, OCIO, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to <DISTRIBUTION@NRC.GOV>. Active guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161. Copies of active and draft guides are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

[2]Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202)634-3273; fax (202)634-3343.

FORM LETTER FOR DECLARING PREGNANCY

This form letter is provided for your convenience. To make your written declaration of pregnancy, you may fill in the blanks in this form letter, you may use a form letter the licensee has provided to you, or you may write your own letter.

DECLARATION OF PREGNANCY

To:_____

In accordance with the NRC's regulations at 10 CFR 20.1208, "Dose to an Embryo/Fetus," I am declaring that I am pregnant. I believe I became pregnant in_____ (only the month and year need be provided).

I understand the radiation dose to my embryo/fetus during my entire pregnancy will not be allowed to exceed 0.5 rem (5 millisievert) (unless that dose has already been exceeded between the time of conception and submitting this letter). I also understand that meeting the lower dose limit may require a change in job or job responsibilities during my pregnancy.

_____

(Your signature)

_____

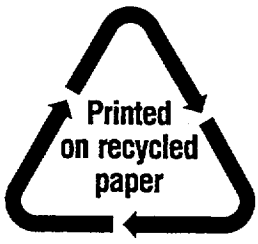
(Your name printed)

_____

(Date)

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this regulatory guide. A 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, Washington, DC, as an enclosure to Part 20 (56 FR 23360).



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OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.22
(Task OP 013-4)

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 (3.7×10^{-6} Bq/mL) for a 40-hour workweek or to an average ore-dust concentration of 10^{-10} μCi/mL (3.7×10^{-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 (3.7×10^{-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 (3.7×10^{-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 (3.7×10^{-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 1×10^{-10} μCi/mL (3.7×10^{-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 1×10^{-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 (6.77×10^{-7} Ci/g or 2.5×10^{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.

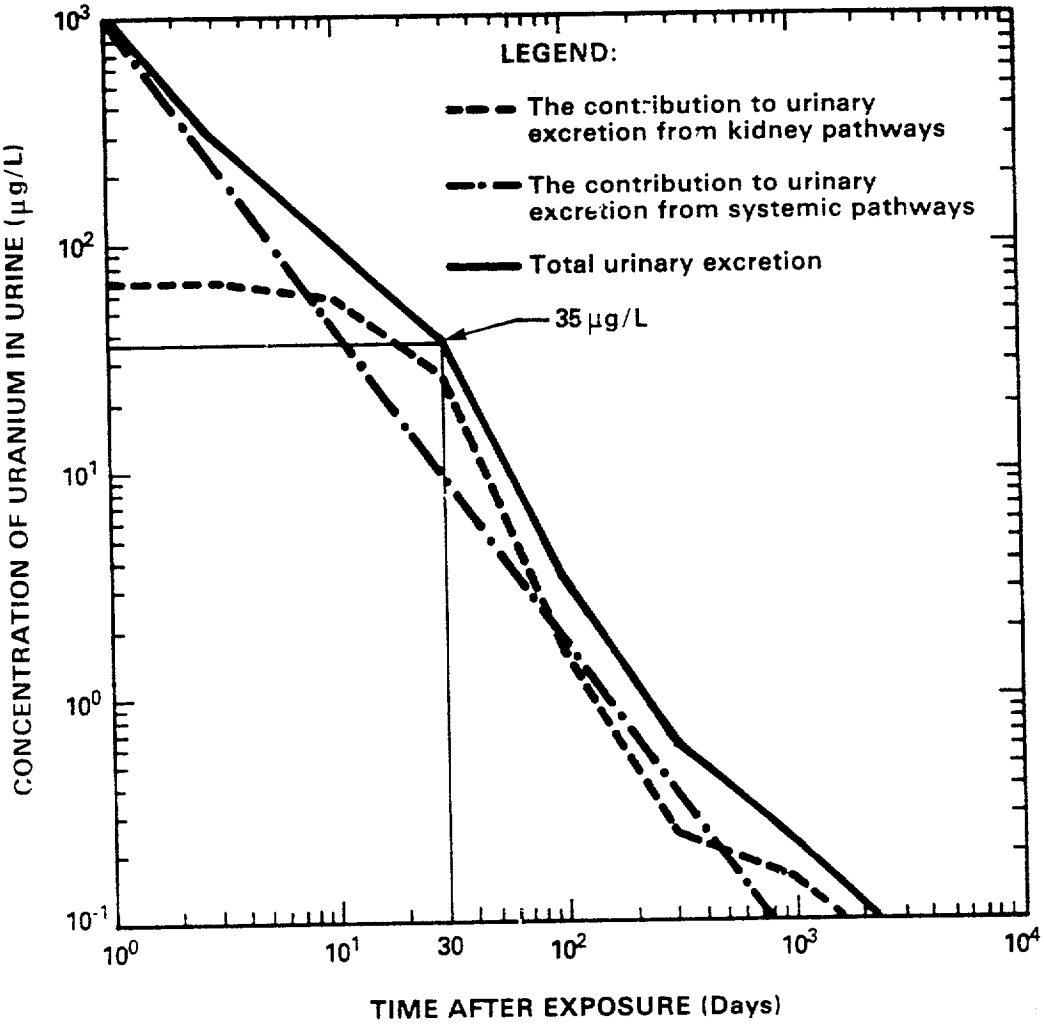

Figure 1 Uranium Concentration in Urine Following Single Exposure to High-Fired Yellowcake (Intake = 160,000 µg U = 1 ALI) (from NUREG-0874, Ref. 1)

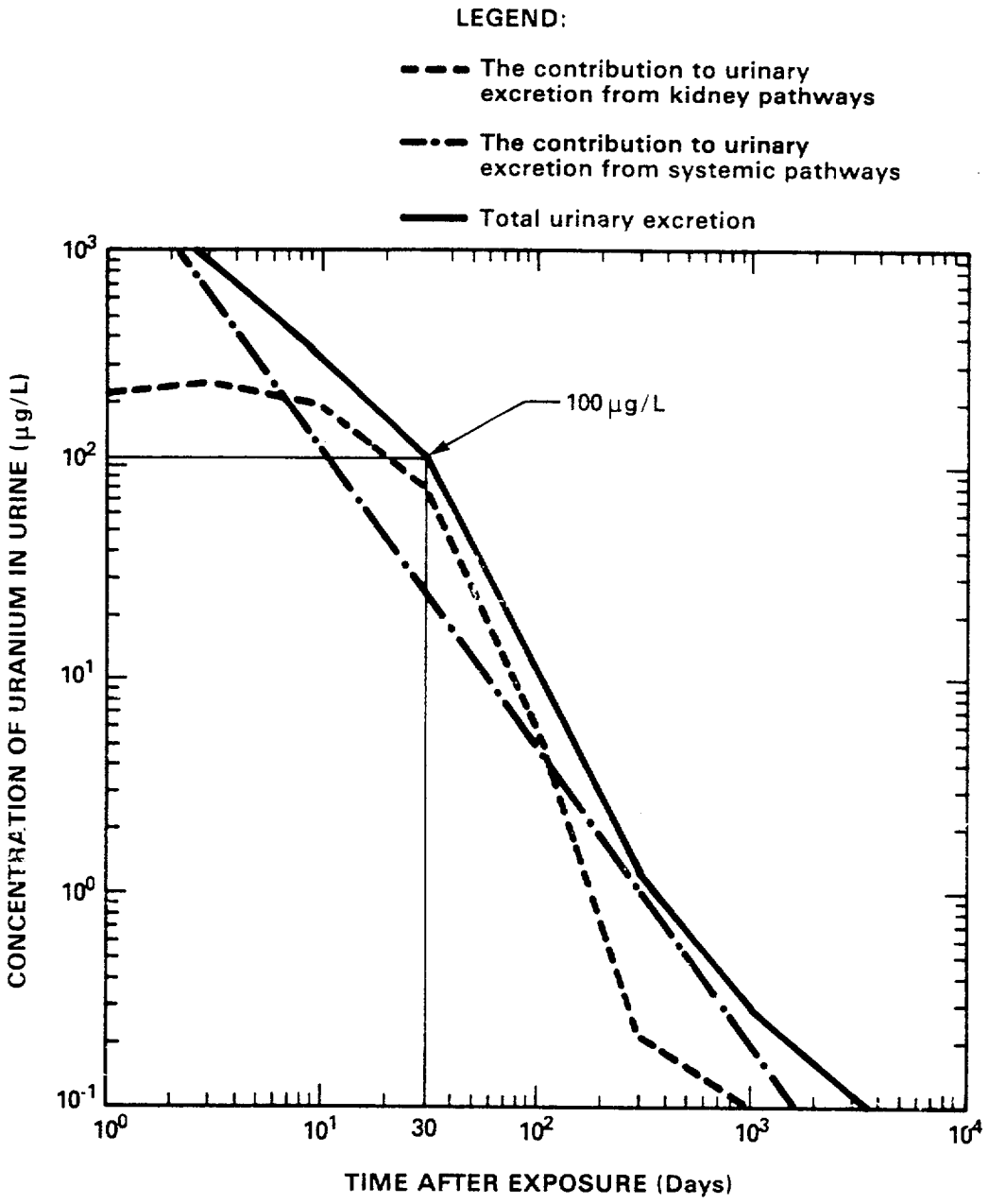

Figure 2 Uranium Concentration in Urine Following Single Exposure to Low-Fired Yellowcake (Intake = 260,000 µg U = 1 ALI) (from NUREG-0874, Ref. 1)

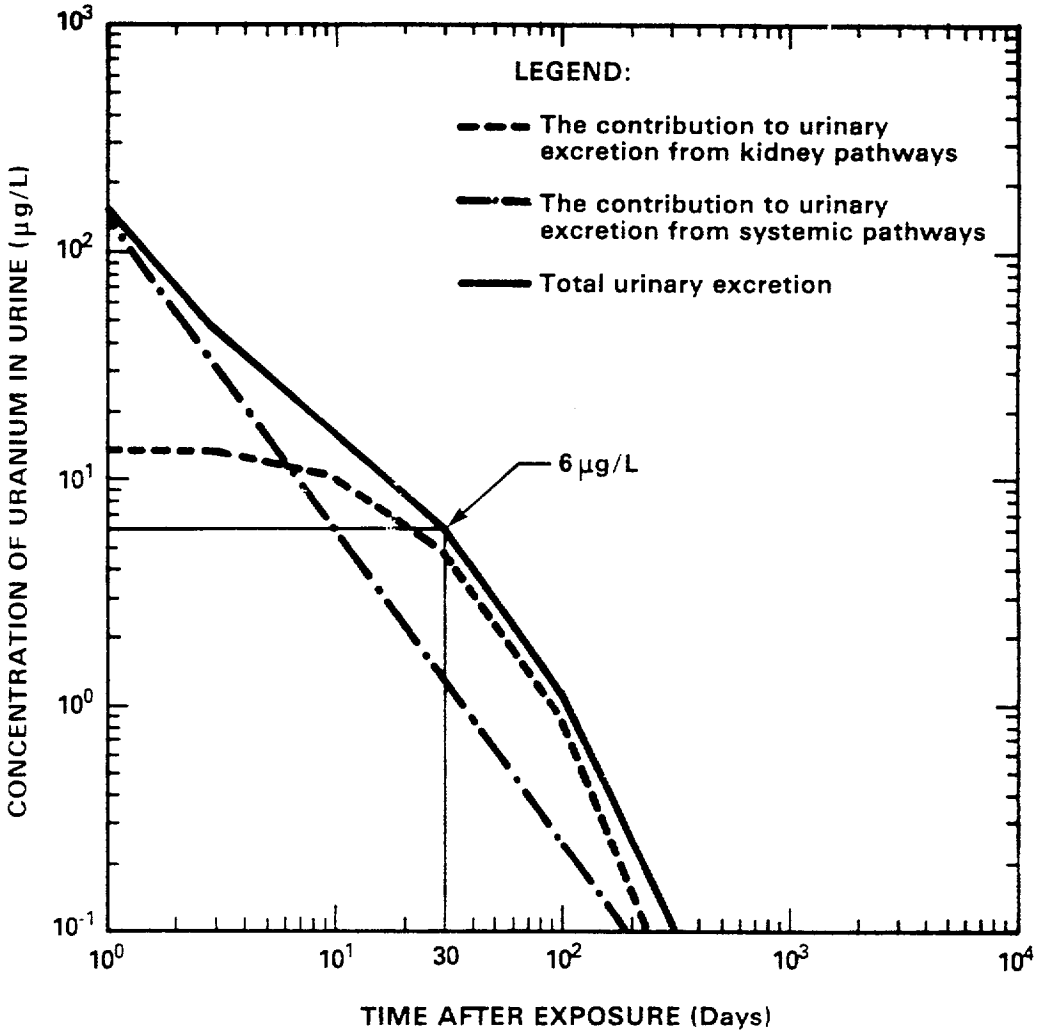

Figure 3 Uranium Concentration in Urine Following Exposure to Ore Dust (from NUREG-0874, Ref. 1)

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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, $\exp(-\lambda t)$, 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.

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

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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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

U.S. NUCLEAR REGULATORY COMMISSION September 1980

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.26

(Draft was issued as OH 714-4, dated August 1979)

APPLICATIONS OF BIOASSAY FOR FISSION AND ACTIVATION PRODUCTS

A. INTRODUCTION

Section 20.108, "Orders Requiring Furnishing of Bioassay Services," of 10 CFR Part 20, "Standards for Protection Against Radiation," states that the Nuclear Regulatory Commission may incorporate in any license certain provisions requiring bioassay measurements as necessary or desirable to aid in determining the extent of an individual's exposure to concentrations of radioactive material. As used by the Commission, the term bioassay includes *in vivo* measurements as well as measurements of radioactive material in excreta.

This guide identifies the bases that will be used by the NRC staff in evaluating the need for license provisions to require bioassay programs in installations where employees may be subject to internal radiation exposure from the inhalation or ingestion of fission or neutron activation products. The guide also describes methods acceptable to the NRC staff for determining the persons to be included in a bioassay program, the sampling and measurement techniques to be used, the frequency of bioassay measurements to be made, actions to be taken based on designated levels of internal radioactivity, estimations of internal dose to be calculated from bioassay measurements, and record systems to be maintained appropriate to such bioassay programs.

The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

This guide was issued after consideration of comments received from the public.

Regulatory guides are issued in 10 broad divisions: 1, Power Reactors; 2, Research and Test Reactors; 3, Fuels and Materials Facilities; 4, Environmental and Siting; 5, Materials and Plant Protection; 6, Products; 7, Transportation; 8, Occupational Health; 9, Antitrust and Financial Review; and 10, General.

Electronic copies of this guide and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML090090115.

B. DISCUSSION

Working Group N343, a subcommittee of the Health Physics Society Standards Committee, developed a standard[1] for the American National Standards Institute (ANSI) presenting requirements and recommended practices for the surveillance and protection of employees of licensee installations where fission or activation products may be processed or handled in unencapsulated form. This standard was approved by ANSI in January 1979, and NRC staff review has indicated that the standard's provisions in these areas are adequate as modified or supplemented by the regulatory position of this guide.

C. REGULATORY POSITION

Bioassay programs that meet the requirements and recommendations of ANSI Standard N343-1978 are acceptable for complying with license provisions pursuant to § 20.108 of 10 CFR Part 20 that may require bioassay for any fission or activation product radionuclides listed in this standard. However, for compliance with NRC requirements, paragraph 6.2.2 of the standard dealing with the selection of individuals to be included in the bioassay program should be interpreted as follows:

> "All facility personnel who routinely enter bioassay areas for routine operations or for maintenance work are to be scheduled for *in vivo* measurements in accordance with the minimum bioassay program. For nonroutine entries the health physicist or radiation protection manager[2] shall determine the need on a case basis."

The ANSI standard recommends in Sections 11, "Calculational Methods" and 12, "Interpretation of Results for Diagnostic Purposes," that "As more representative morphological and metabolic parameters become available, these should be substituted for the ones suggested here" and that "The organ burdens, retention functions, dose rates, and dose commitments shall be based on ICRP models when specific data are unavailable." Since the International Commission on Radiological Protection (ICRP)[3] methods presented in the ANSI standard were developed, more recent data and methods of calculation (Refs. 1-17) have been published by the scientists involved in the continued development of methods of internal dosimetry, including some new calculations for the ICRP and the Medical Internal Radiation Dose (MIRD) Committee of the Society of Nuclear Medicine.[3]

In cases where any direct or indirect bioassay measurements indicate that an individual may receive more than 10 percent of any permissible annual intake derived from concentrations specified in NRC regulations, the additional references listed in this guide, as well as the methods and references of the ANSI standard, should be consulted to determine the most accurate methods of internal dose assessment for the radionuclides and conditions of exposure involved. In some cases, more than one method of evaluation may be required to properly assess internal exposures. All methods of internal dose assessment, as well as all data used in the assessments, should be clearly referenced and recorded as part of the records systems recommended in Section 16, "Records," of the ANSI standard. Calculations for

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1 Copies of ANSI N343-1978, "Internal Dosimetry for Mixed Fission and Activation Products," are available from the American National Standards Institute, 1430 Broadway, New York, N.Y. 10018.

2 The title "Radiation Protection Manager" is used synonymously with radiation safety officer by many licensees; other titles are equally acceptable.

3 Publications of the International Commission on Radiological Protection (ICRP) listed in this guide, in the ANSI standard, or to be published in the future may be ordered from Pergamon Press, Inc., Maxwell House, Elmsford, N.Y. 10523 or through bookstores in the United States. Publications of the MIRD Committee may be obtained from Medical Internal Radiation Dose (MIRD) Committee, Society of Nuclear Medicine, 475 Park Avenue South, New York, N.Y. 10016.

each individual may be recorded together with references to the standard model, where a number of individuals may have been subject to similar exposure conditions.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants 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, the NRC staff will use the methods described herein after December 1, 1980, in the evaluation of bioassay programs included in license applications.

If an applicant or licensee wishes to use the methods described in this regulatory guide on or before December 1, 1980, the pertinent portions of the application or the licensee's performance will be evaluated on the basis of this guide.

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10. G. G. Killough, D. E. Dunning, Jr., S. R. Bernard, and J. C. Pleasant, *Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities,* Vol. 1, ORNL/NUREG/TM-190, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830, June 1978.

11. G. G. Killough, D. E. Dunning, Jr., and J. C. Pleasant, *INREM II: A Computer Implementation of Recent Models for Estimating the Dose Equivalent to Organs of Man from an Inhaled or Ingested Radionuclide,* ORNL/NUREG/TM-84, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830, 1978.

[4] The ICRP-10 series of reports describing methods of calculating internal dose from bioassay data is being revised and updated. Information on published ICRP reports may be obtained from Pergamon Press, Inc., Maxwell House, Elmsford, N.Y. 10523 or through bookstores in the United States.

12. D. E. Dunning, Jr., J. C. Pleasant, and G. G. Killough, *S FACTOR: A Computer Code for Calculating Dose Equivalent to a Target Organ per Microcurie-Day Residence of a Radionuclide in a Source Organ,* ORNL/NUREG/TM-85, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 37830, 1977.

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VALUE/IMPACT STATEMENT[5]

ANSI Standard N343-1978, "Internal Dosimetry for Mixed Fission and Activation Products," was developed by the Health Physics Society's Standards Committee (HPSSC) on a high-priority basis and was approved by the American National Standards Institute (ANSI) for publication in 1979.

This guidance is needed to facilitate the licensing process, since different methods of measurement and interpretation for these nuclides are carried out in different licensee facilities. NRC staff members have participated in the work performed by the HPSSC working group and have collected NRC staff comments on the draft standards for ANSI. This endorsement of ANSI N343-1978 by a regulatory guide was determined to be the only viable option for alleviating present uncertainties and conflicts in judgment between various licensees and various professionals in establishing bioassay requirements and interpreting compliance for exposures to different fission and activation product radionuclides. This guide replaces interim informal guidance provided by the NRC staff prior to the guide's issuance.

[5] No change in the Draft Value/Impact Statement (published in August 1979 with Draft Guide OH 714-4) was suggested by the public comments or other information received by the NRC staff. These drafts are available for inspection at the NRC Public Document Room, 1717 H Street NW., Washington, D.C.

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
February 1996

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.29

(Draft was issued as DG-8012)

INSTRUCTION CONCERNING RISKS FROM OCCUPATIONAL RADIATION EXPOSURE

A. INTRODUCTION

Section 19.12 of 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," requires that all individuals who in the course of their employment are likely to receive in a year an occupational dose in excess of 100 mrem (1 mSv) be instructed in the health protection issues associated with exposure to radioactive materials or radiation. Section 20.1206 of 10 CFR Part 20, "Standards for Protection Against Radiation," requires that before a planned special exposure occurs the individuals involved are, among other things, to be informed of the estimated doses and associated risks.

This regulatory guide describes the information that should be provided to workers by licensees about health risks from occupational exposure. This revision conforms to the revision of 10 CFR Part 20 that became effective on June 20, 1991, to be implemented by licensees no later than January 1, 1994. The revision of 10 CFR Part 20 establishes new dose limits based on the effective dose equivalent (EDE), requires the summing of internal and external dose, establishes a requirement that licensees use procedures and engineering controls to the extent practicable to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA), provides for planned special exposures, establishes a dose limit for the embryo/fetus of an occupationally exposed declared pregnant woman, and explicitly states that Part 20 is not to be construed as limiting action that may be necessary to protect health and safety during emergencies.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 19 or 10 CFR Part 20. These regulations provide the regulatory bases for this guide. The information collection requirements in 10 CFR Parts 19 and 20 have been cleared under OMB Clearance Nos. 3150-0044 and 3150-0014, respectively.

B. DISCUSSION

It is important to qualify the material presented in this guide with the following considerations.

The coefficient used in this guide for occupational radiation risk estimates, 4×10^{-4} health effects per rem, is based on data obtained at much higher doses and dose rates than those encountered by workers. The risk coefficient obtained at high doses and dose rates was reduced to account for the reduced effectiveness of lower doses and dose rates in producing the stochastic effects observed in studies of exposed humans.

The assumption of a linear extrapolation from the lowest doses at which effects are observable down to

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public such information as methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and data needed by the NRC staff in its review of applications for permits and licenses. 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.

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the occupational range has considerable uncertainty. The report of the Committee on the Biological Effects of Ionizing Radiation (Ref. 1) states that

> "... departure from linearity cannot be excluded at low doses below the range of observation. Such departures could be in the direction of either an increased or decreased risk. Moreover, epidemiologic data cannot rigorously exclude the existence of a threshold in the 100 mrem dose range. Thus, the possibility that there may be no risk from exposures comparable to external natural background radiation cannot be ruled out. At such low doses and dose rates, it must be acknowledged that the lower limit of the range of uncertainty in the risk estimates extends to zero."

The issue of beneficial effects from low doses, or hormesis, in cellular systems is addressed by the United Nations Scientific Committee on the Effects of Atomic Radiation (Ref. 2). UNSCEAR states that "... it would be premature to conclude that cellular adaptive responses could convey possible beneficial effects to the organism that would outweigh the detrimental effects of exposures to low doses of low-LET radiation."

In the absence of scientific certainty regarding the relationship between low doses and health effects, and as a conservative assumption for radiation protection purposes, the scientific community generally assumes that any exposure to ionizing radiation can cause biological effects that may be harmful to the exposed person and that the magnitude or probability of these effects is directly proportional to the dose. These effects may be classified into three categories:

Somatic Effects: Physical effects occurring in the exposed person. These effects may be observable after a large or acute dose (e.g., 100 rems[1] (1 Sv) or more to the whole body in a few hours); or they may be effects such as cancer that may occur years after exposure to radiation.

Genetic Effects: Abnormalities that may occur in the future children of exposed individuals and in subsequent generations (genetic effects exceeding normal incidence have not been observed in any of the studies of human populations).

Teratogenic Effects: Effects such as cancer or congenital malformation that may be observed in children who were exposed during the fetal and embryonic stages of development (these effects have been observed from high, i.e., above 20 rems (0.2 Sv), acute exposures).

The normal incidence of effects from natural and manmade causes is significant. For example, approximately 20% of people die from various forms of cancer whether or not they ever receive occupational exposure to radiation. To avoid increasing the incidence of such biological effects, regulatory controls are imposed on occupational doses to adults and minors and on doses to the embryo/fetus from occupational exposures of declared pregnant women.

Radiation protection training for workers who are occupationally exposed to ionizing radiation is an essential component of any program designed to ensure compliance with NRC regulations. A clear understanding of what is presently known about the biological risks associated with exposure to radiation will result in more effective radiation protection training and should generate more interest on the part of the workers in complying with radiation protection standards. In addition, pregnant women and other occupationally exposed workers should have available to them relevant information on radiation risks to enable them to make informed decisions regarding the acceptance of these risks. It is intended that workers who receive this instruction will develop respect for the risks involved, rather than excessive fear or indifference.

C. REGULATORY POSITION

Instruction to workers performed in compliance with 10 CFR 19.12 should be given prior to occupational exposure and periodically thereafter. The frequency of retraining might range from annually for licensees with complex operations such as nuclear power plants, to every three years for licensees who possess, for example, only low-activity sealed sources. If a worker is to participate in a planned special exposure, the worker should be informed of the associated risks in compliance with 10 CFR 20.1206.

In providing instruction concerning health protection problems associated with exposure to radiation, all occupationally exposed workers and their supervisors should be given specific instruction on the risk of biological effects resulting from exposure to radiation. The extent of these instructions should be commensurate with the radiological risks present in the workplace.

The instruction should be presented orally, in printed form, or in any other effective communication media to workers and supervisors. The appendix to this guide provides useful information for demonstrating compliance with the training requirements in 10 CFR Parts 19 and 20. Individuals should be given an opportunity to discuss the information and to ask questions. Testing is recommended, and each trainee should be asked to acknowledge in writing that the instruction has been received and understood.

[1] In the International System of Units (SI), the rem is replaced by the sievert; 100 rems is equal to 1 sievert (Sv).

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 acceptable alternative methods for complying with specified portions of the Commission's regulations, the guidance and instructional materials 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 19.12 and 10 CFR Part 20.

REFERENCES

1. National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation*, Report of the Committee on the Biological Effects of Ionizing Radiation (BEIR V), National Academy Press, Washington, DC, 1990.

2. United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), *Sources and Effects of Ionizing Radiation*, United Nations, New York, 1993.

APPENDIX

INSTRUCTION CONCERNING RISKS FROM OCCUPATIONAL RADIATION EXPOSURE

This instructional material is intended to provide the user with the best available information about the health risks from occupational exposure to ionizing radiation. Ionizing radiation consists of energy or small particles, such as gamma rays and beta and alpha particles, emitted from radioactive materials, which can cause chemical or physical damage when they deposit energy in living tissue. A question and answer format is used. Many of the questions or subjects were developed by the NRC staff in consultation with workers, union representatives, and licensee representatives experienced in radiation protection training.

This Revision 1 to Regulatory Guide 8.29 updates the material in the original guide on biological effects and risks and on typical occupational exposure. Additionally, it conforms to the revised 10 CFR Part 20, "Standards for Protection Against Radiation," which was required to be implemented by licensees no later than January 1, 1994. The information in this appendix is intended to help develop respect by workers for the risks associated with radiation, rather than unjustified fear or lack of concern. Additional guidance concerning other topics in radiation protection training is provided in other NRC regulatory guides.

1. What is meant by health risk?

A health risk is generally thought of as something that may endanger health. Scientists consider health risk to be the statistical probability or mathematical chance that personal injury, illness, or death may result from some action. Most people do not think about health risks in terms of mathematics. Instead, most of us consider the health risk of a particular action in terms of whether we believe that particular action will, or will not, cause us some harm. The intent of this appendix is to provide estimates of, and explain the bases for, the risk of injury, illness, or death from occupational radiation exposure. Risk can be quantified in terms of the probability of a health effect per unit of dose received.

When x-rays, gamma rays, and ionizing particles interact with living materials such as our bodies, they may deposit enough energy to cause biological damage. Radiation can cause several different types of events such as the very small physical displacement of molecules, changing a molecule to a different form, or ionization, which is the removal of electrons from atoms and molecules. When the quantity of radiation energy deposited in living tissue is high enough, biological damage can occur as a result of chemical bonds being broken and cells being damaged or killed. These effects can result in observable clinical symptoms.

The basic unit for measuring absorbed radiation is the rad. One rad (0.01 gray in the International System of units) equals the absorption of 100 ergs (a small but measurable amount of energy) in a gram of material such as tissue exposed to radiation. To reflect biological risk, rads must be converted to rems. The new international unit is the sievert (100 rems = 1 Sv). This conversion accounts for the differences in the effectiveness of different types of radiation in causing damage. The rem is used to estimate biological risk. For beta and gamma radiation, a rem is considered equal to a rad.

2. What are the possible health effects of exposure to radiation?

Health effects from exposure to radiation range from no effect at all to death, including diseases such as leukemia or bone, breast, and lung cancer. Very high (100s of rads), short-term doses of radiation have been known to cause prompt (or early) effects, such as vomiting and diarrhea,[1] skin burns, cataracts, and even death. It is suspected that radiation exposure may be linked to the potential for genetic effects in the children of exposed parents. Also, children who were exposed to high doses (20 or more rads) of radiation prior to birth (as an embryo/fetus) have shown an increased risk of mental retardation and other congenital malformations. These effects (with the exception of genetic effects) have been observed in various studies of medical radiologists, uranium miners, radium workers, radiotherapy patients, and the people exposed to radiation from atomic bombs dropped on Japan. In addition, radiation effects studies with laboratory animals, in which the animals were given relatively high doses, have provided extensive data on radiation-induced health effects, including genetic effects.

It is important to note that these kinds of health effects result from high doses, compared to occupational levels, delivered over a relatively short period of time.

Although studies have not shown a consistent cause-and-effect relationship between current levels of occupational radiation exposure and biological effects, it is prudent from a worker protection perspective to assume that some effects may occur.

[1]These symptoms are early indicators of what is referred to as the acute radiation syndrome, caused by high doses delivered over a short time period, which includes damage to the blood-forming organs such as bone marrow, damage to the gastrointestinal system, and, at very high doses, can include damage to the central nervous system.

3. What is meant by early effects and delayed or late effects?

EARLY EFFECTS

Early effects, which are also called immediate or prompt effects, are those that occur shortly after a large exposure that is delivered within hours to a few days. They are observable after receiving a very large dose in a short period of time, for example, 300 rads (3 Gy) received within a few minutes to a few days. Early effects are not caused at the levels of radiation exposure allowed under the NRC's occupational limits.

Early effects occur when the radiation dose is large enough to cause extensive biological damage to cells so that large numbers of cells are killed. For early effects to occur, this radiation dose must be received within a short time period. This type of dose is called an acute dose or acute exposure. The same dose received over a long time period would not cause the same effect. Our body's natural biological processes are constantly repairing damaged cells and replacing dead cells; if the cell damage is spread over time, our body is capable of repairing or replacing some of the damaged cells, reducing the observable adverse conditions.

For example, a dose to the whole body of about 300–500 rads (3–5 Gy), more than 60 times the annual occupational dose limit, if received within a short time period (e.g., a few hours) will cause vomiting and diarrhea within a few hours; loss of hair, fever, and weight loss within a few weeks; and about a 50 percent chance of death if medical treatment is not provided. These effects would not occur if the same dose were accumulated gradually over many weeks or months (Refs. 1 and 2). Thus, one of the justifications for establishing annual dose limits is to ensure that occupational dose is spread out in time.

It is important to distinguish between whole body and partial body exposure. A localized dose to a small volume of the body would not produce the same effect as a whole body dose of the same magnitude. For example, if only the hand were exposed, the effect would mainly be limited to the skin and underlying tissue of the hand. An acute dose of 400 to 600 rads (4–6 Gy) to the hand would cause skin reddening; recovery would occur over the following months and no long-term damage would be expected. An acute dose of this magnitude to the whole body could cause death within a short time without medical treatment. Medical treatment would lessen the magnitude of the effects and the chance of death; however, it would not totally eliminate the effects or the chance of death.

DELAYED EFFECTS

Delayed effects may occur years after exposure. These effects are caused indirectly when the radiation changes parts of the cells in the body, which causes the normal function of the cell to change, for example, normal healthy cells turn into cancer cells. The potential for these delayed health effects is one of the main concerns addressed when setting limits on occupational doses.

A delayed effect of special interest is genetic effects. Genetic effects may occur if there is radiation damage to the cells of the gonads (sperm or eggs). These effects may show up as genetic defects in the children of the exposed individual and succeeding generations. However, if any genetic effects (i.e., effects in addition to the normal expected number) have been caused by radiation, the numbers are too small to have been observed in human populations exposed to radiation. For example, the atomic bomb survivors (from Hiroshima and Nagasaki) have not shown any significant radiation-related increases in genetic defects (Ref. 3). Effects have been observed in animal studies conducted at very high levels of exposure and it is known that radiation can cause changes in the genes in cells of the human body. However, it is believed that by maintaining worker exposures below the NRC limits and consistent with ALARA, a margin of safety is provided such that the risk of genetic effects is almost eliminated.

4. What is the difference between acute and chronic radiation dose?

Acute radiation dose usually refers to a large dose of radiation received in a short period of time. Chronic dose refers to the sum of small doses received repeatedly over long time periods, for example, 20 mrem (or millirem, which is 1-thousandth of a rem) (0.2 mSv) per week every week for several years. It is assumed for radiation protection purposes that any radiation dose, either acute or chronic, may cause delayed effects. However, only large acute doses cause early effects; chronic doses within the occupational dose limits do not cause early effects. Since the NRC limits do not permit large acute doses, concern with occupational radiation risk is primarily focused on controlling chronic exposure for which possible delayed effects, such as cancer, are of concern.

The difference between acute and chronic radiation exposure can be shown by using exposure to the sun's rays as an example. An intense exposure to the sun can result in painful burning, peeling, and growing of new skin. However, repeated short exposures provide time for the skin to be repaired between exposures. Whether exposure to the sun's rays is long term or spread over short periods, some of the injury may not be repaired and may eventually result in skin cancer.

Cataracts are an interesting case because they can be caused by both acute and chronic radiation. A certain threshold level of dose to the lens of the eye is required before there is any observable visual impairment, and the impairment remains after the exposure is stopped. The threshold for cataract development

from acute exposure is an acute dose on the order of 100 rads (1 Gy). Further, a cumulative dose of 800 rads (8 Gy) from protracted exposures over many years to the lens of the eye has been linked to some level of visual impairment (Refs. 1 and 4). These doses exceed the amount that may be accumulated by the lens from normal occupational exposure under the current regulations.

5. What is meant by external and internal exposure?

A worker's occupational dose may be caused by exposure to radiation that originates outside the body, called "external exposure," or by exposure to radiation from radioactive material that has been taken into the body, called "internal exposure." Most NRC-licensed activities involve little, if any, internal exposure. It is the current scientific consensus that a rem of radiation dose has the same biological risk regardless of whether it is from an external or an internal source. The NRC requires that dose from external exposure and dose from internal exposure be added together, if each exceeds 10% of the annual limit, and that the total be within occupational limits. The sum of external and internal dose is called the total effective dose equivalent (TEDE) and is expressed in units of rems (Sv).

Although unlikely, radioactive materials may enter the body through breathing, eating, drinking, or open wounds, or they may be absorbed through the skin. The intake of radioactive materials by workers is generally due to breathing contaminated air. Radioactive materials may be present as fine dust or gases in the workplace atmosphere. The surfaces of equipment and workbenches may be contaminated, and these materials can be resuspended in air during work activities.

If any radioactive material enters the body, the material goes to various organs or is excreted, depending on the biochemistry of the material. Most radioisotopes are excreted from the body in a few days. For example, a fraction of any uranium taken into the body will deposit in the bones, where it remains for a longer time. Uranium is slowly eliminated from the body, mostly by way of the kidneys. Most workers are not exposed to uranium. Radioactive iodine is preferentially deposited in the thyroid gland, which is located in the neck.

To limit risk to specific organs and the total body, an annual limit on intake (ALI) has been established for each radionuclide. When more than one radionuclide is involved, the intake amount of each radionuclide is reduced proportionally. NRC regulations specify the concentrations of radioactive material in the air to which a worker may be exposed for 2,000 working hours in a year. These concentrations are termed the derived air concentrations (DACs). These limits are the total amounts allowed if no external radiation is received. The resulting dose from the internal radiation sources (from breathing air at 1 DAC) is the maximum allowed to an organ or to the worker's whole body.

6. How does radiation cause cancer?

The mechanisms of radiation-induced cancer are not completely understood. When radiation interacts with the cells of our bodies, a number of events can occur. The damaged cells can repair themselves and permanent damage is not caused. The cells can die, much like the large numbers of cells that die every day in our bodies, and be replaced through the normal biological processes. Or a change can occur in the cell's reproductive structure, the cells can mutate and subsequently be repaired without effect, or they can form precancerous cells, which may become cancerous. Radiation is only one of many agents with the potential for causing cancer, and cancer caused by radiation cannot be distinguished from cancer attributable to any other cause.

Radiobiologists have studied the relationship between large doses of radiation and cancer (Refs. 5 and 6). These studies indicate that damage or change to genes in the cell nucleus is the main cause of radiation-induced cancer. This damage may occur directly through the interaction of the ionizing radiation in the cell or indirectly through the actions of chemical products produced by radiation interactions within cells. Cells are able to repair most damage within hours; however, some cells may not be repaired properly. Such misrepaired damage is thought to be the origin of cancer, but misrepair does not always cause cancer. Some cell changes are benign or the cell may die; these changes do not lead to cancer.

Many factors such as age, general health, inherited traits, sex, as well as exposure to other cancer-causing agents such as cigarette smoke can affect susceptibility to the cancer-causing effects of radiation. Many diseases are caused by the interaction of several factors, and these interactions appear to increase the susceptibility to cancer.

7. Who developed radiation risk estimates?

Radiation risk estimates were developed by several national and international scientific organizations over the last 40 years. These organizations include the National Academy of Sciences (which has issued several reports from the Committee on the Biological Effects of Ionizing Radiations, BEIR), the National Council on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Each of these organizations continues to review new research findings on radiation health risks.

Several reports from these organizations present new findings on radiation risks based upon revised estimates of radiation dose to survivors of the atomic bombing at Hiroshima and Nagasaki. For example, UNSCEAR published risk estimates in 1988 and 1993 (Refs. 5 and 6). The NCRP also published a report in 1988, "New Dosimetry at Hiroshima and Nagasaki and Its Implications for Risk Estimates" (Ref. 7). In January 1990, the National Academy of Sciences released the fifth report of the BEIR Committee, "Health Effects of Exposure to Low Levels of Ionizing Radiation" (Ref. 4). Each of these publications also provides extensive bibliographies on other published studies concerning radiation health effects for those who may wish to read further on this subject.

8. What are the estimates of the risk of fatal cancer from radiation exposure?

We don't know exactly what the chances are of getting cancer from a low-level radiation dose, primarily because the few effects that may occur cannot be distinguished from normally occurring cancers. However, we can make estimates based on extrapolation from extensive knowledge from scientific research on high dose effects. The estimates of radiation effects at high doses are better known than are those of most chemical carcinogens (Ref. 8).

From currently available data, the NRC has adopted a risk value for an occupational dose of 1 rem (0.01 Sv) Total Effective Dose Equivalent (TEDE) of 4 in 10,000 of developing a fatal cancer, or approximately 1 chance in 2,500 of fatal cancer per rem of TEDE received. The uncertainty associated with this risk estimate does not rule out the possibility of higher risk, or the possibility that the risk may even be zero at low occupational doses and dose rates.

The radiation risk incurred by a worker depends on the amount of dose received. Under the linear model explained above, a worker who receives 5 rems (0.05 Sv) in a year incurs 10 times as much risk as another worker who receives only 0.5 rem (0.005 Sv). Only a very few workers receive doses near 5 rems (0.05 Sv) per year (Ref. 9).

According to the BEIR V report (Ref. 4), approximately one in five adults normally will die from cancer from all possible causes such as smoking, food, alcohol, drugs, air pollutants, natural background radiation, and inherited traits. Thus, in any group of 10,000 workers, we can estimate that about 2,000 (20%) will die from cancer without any occupational radiation exposure.

To explain the significance of these estimates, we will use as an example a group of 10,000 people, each exposed to 1 rem (0.01 Sv) of ionizing radiation. Using the risk factor of 4 effects per 10,000 rem of dose, we estimate that 4 of the 10,000 people might die from delayed cancer because of that 1-rem dose (although the actual number could be more or less than 4) in addition to the 2,000 normal cancer fatalities expected to occur in that group from all other causes. This means that a 1-rem (0.01 Sv) dose may increase an individual worker's chances of dying from cancer from 20 percent to 20.04 percent. If one's lifetime occupational dose is 10 rems, we could raise the estimate to 20.4 percent. A lifetime dose of 100 rems may increase chances of dying from cancer from 20 to 24 percent. The average measurable dose for radiation workers reported to the NRC was 0.31 rem (0.0031 Sv) for 1993 (Ref. 9). Today, very few workers ever accumulate 100 rems (1 Sv) in a working lifetime, and the average career dose of workers at NRC-licensed facilities is 1.5 rems (0.015 Sv), which represents an estimated increase from 20 to about 20.06 percent in the risk of dying from cancer.

It is important to understand the probability factors here. A similar question would be, "If you select one card from a full deck of cards, will you get the ace of spades?" This question cannot be answered with a simple yes or no. The best answer is that your chance is 1 in 52. However, if 1000 people each select one card from full decks, we can predict that about 20 of them will get an ace of spades. Each person will have 1 chance in 52 of drawing the ace of spades, but there is no way we can predict which persons will get that card. The issue is further complicated by the fact that in a drawing by 1000 people, we might get only 15 successes, and in another, perhaps 25 correct cards in 1000 draws. We can say that if you receive a radiation dose, you will have increased your chances of eventually developing cancer. It is assumed that the more radiation exposure you get, the more you increase your chances of cancer.

The normal chance of dying from cancer is about one in five for persons who have not received any occupational radiation dose. The additional chance of developing fatal cancer from an occupational exposure of 1 rem (0.01 Sv) is about the same as the chance of drawing any ace from a full deck of cards three times in a row. The additional chance of dying from cancer from an occupational exposure of 10 rem (0.1 Sv) is about equal to your chance of drawing two aces successively on the first two draws from a full deck of cards.

It is important to realize that these risk numbers are only estimates based on data for people and research animals exposed to high levels of radiation in short periods of time. There is still uncertainty with regard to estimates of radiation risk from low levels of exposure. Many difficulties are involved in designing research studies that can accurately measure the projected small increases in cancer cases that might be caused by low exposures to radiation as compared to the normal rate of cancer.

These estimates are considered by the NRC staff to be the best available for the worker to use to make an informed decision concerning acceptance of the risks associated with exposure to radiation. A worker who decides to accept this risk should try to keep exposure to radiation as low as is reasonably achievable (ALARA) to avoid unnecessary risk.

9. If I receive a radiation dose that is within occupational limits, will it cause me to get cancer?

Probably not. Based on the risk estimates previously discussed, the risk of cancer from doses below the occupational limits is believed to be small. Assessment of the cancer risks that may be associated with low doses of radiation are projected from data available at doses larger than 10 rems (0.1 Sv) (Ref. 3). For radiation protection purposes, these estimates are made using the straight line portion of the linear quadratic model (Curve 2 in Figure 1). We have data on cancer probabilities only for high doses, as shown by the solid line in Figure 1. Only in studies involving radiation doses above occupational limits are there dependable determinations of the risk of cancer, primarily because below the limits the effect is small compared to differences in the normal cancer incidence from year to year and place to place. The ICRP, NCRP, and other standards-setting organizations assume for radiation protection purposes that there is some risk, no matter how small the dose (Curves 1 and 2). Some scientists believe that the risk drops off to zero at some low dose (Curve 3), the threshold effect. The ICRP and NCRP endorse the linear quadratic model as a conservative means of assuring safety (Curve 2).

For regulatory purposes, the NRC uses the straight line portion of Curve 2, which shows the number of effects decreasing linearly as the dose decreases. Because the scientific evidence does not conclusively demonstrate whether there is or is not an effect at low doses, the NRC assumes for radiation protection purposes, that even small doses have some chance of causing cancer. Thus, a principle of radiation protection is to do more than merely meet the allowed regulatory limits; doses should be kept as low as is reasonably achievable (ALARA). This is as true for natural carcinogens such as sunlight and natural radiation as it is for those that are manmade, such as cigarette smoke, smog, and x-rays.

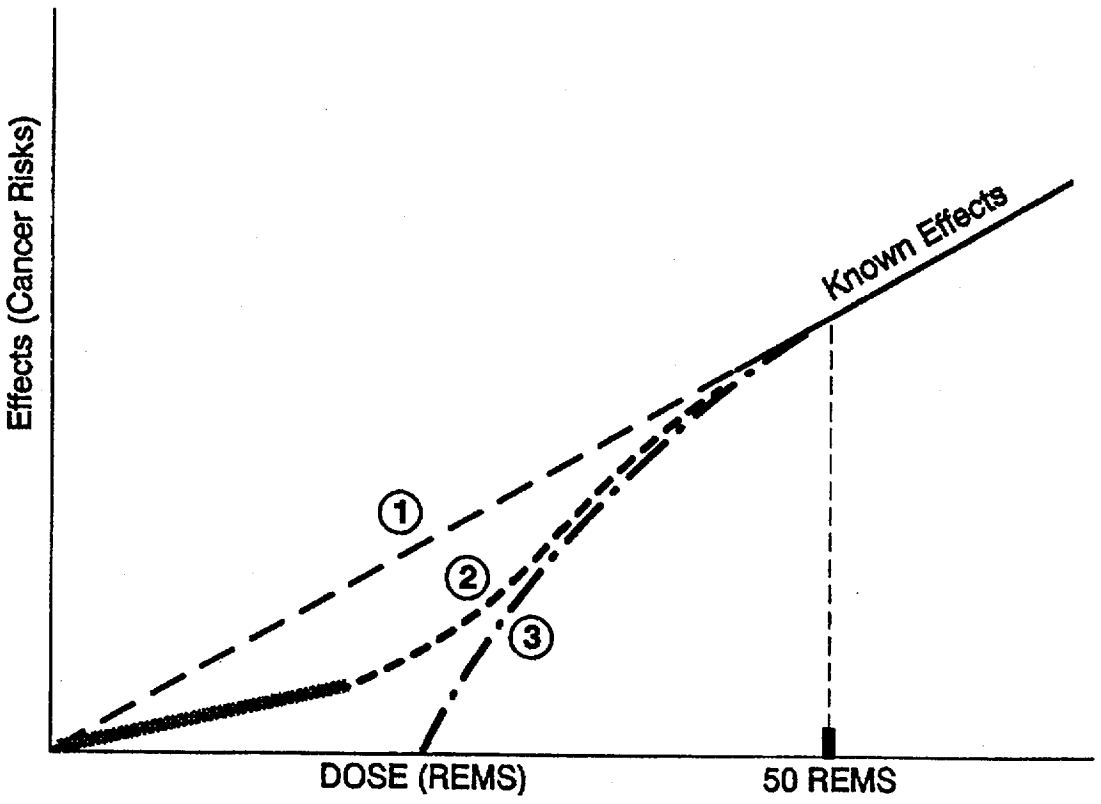

Figure 1. Some Proposed Models for How the Effects of Radiation Vary With Doses at Low Levels

10. How can we compare the risk of cancer from radiation to other kinds of health risks?

One way to make these comparisons is to compare the average number of days of life expectancy lost because of the effects associated with each particular health risk. Estimates are calculated by looking at a large number of persons, recording the age when death occurs from specific causes, and estimating the average number of days of life lost as a result of these early deaths. The total number of days of life lost is then averaged over the total observed group.

Several studies have compared the average days of life lost from exposure to radiation with the number of days lost as a result of being exposed to other health risks. The word "average" is important because an individual who gets cancer loses about 15 years of life expectancy, while his or her coworkers do not suffer any loss.

Some representative numbers are presented in Table 1. For categories of NRC-regulated industries with larger doses, the average measurable occupational dose in 1993 was 0.31 rem (0.0031 Sv). A simple calculation based on the article by Cohen and Lee (Ref. 10) shows that 0.3 rem (0.003 Sv) per year from age 18 to 65 results in an average loss of 15 days. These estimates indicate that the health risks from occupational radiation exposure are smaller than the risks associated with many other events or activities we encounter and accept in normal day-to-day activities.

It is also useful to compare the estimated average number of days of life lost from occupational exposure to radiation with the number of days lost as a result of working in several types of industries. Table 2 shows average days of life expectancy lost as a result of fatal work-related accidents. Table 2 does not include non-accident types of occupational risks such as occupational disease and stress because the data are not available.

These comparisons are not ideal because we are comparing the possible effects of chronic exposure to radiation to different kinds of risk such as accidental death, in which death is inevitable if the event occurs. This is the best we can do because good data are not available on chronic exposure to other workplace carcinogens. Also, the estimates of loss of life expectancy for workers from radiation-induced cancer do not take into consideration the competing effect on the life expectancy of the workers from industrial accidents.

11. What are the health risks from radiation exposure to the embryo/fetus?

During certain stages of development, the embryo/fetus is believed to be more sensitive to radiation damage than adults. Studies of atomic bomb survivors exposed to acute radiation doses exceeding 20 rads (0.2 Gy) during pregnancy show that children born after receiving these doses have a higher risk of mental retardation. Other studies suggest that an association exists between exposure to diagnostic x-rays before birth and carcinogenic effects in childhood and in adult life. Scientists are uncertain about the magnitude of the risk. Some studies show the embryo/fetus to be more sensitive to radiation-induced cancer than adults, but other studies do not. In recognition of the possibility of increased radiation sensitivity, and because dose to the

Table 1 Estimated Loss of Life Expectancy from Health Risks[a]

Health Risk	*Estimate of Life Expectancy Lost (average)*	
Smoking 20 cigarettes a day		6 years
Overweight (by 15%)		2 years
Alcohol consumption (U.S. average)		1 year
All accidents combined		1 year
Motor vehicle accidents	207 days	
Home accidents	74 days	
Drowning	24 days	
All natural hazards (earthquake, lightning, flood, etc.)		7 days
Medical radiation		6 days
Occupational Exposure		
0.3 rem/y from age 18 to 65	15 days	
1 rem/y from age 18 to 65	51 days	

[a]Adapted from Reference 10.

Table 2 Estimated Loss of Life Expectancy from Industrial Accidents[a]

Industry Type	*Estimated Days of Life Expectancy Lost (Average)*
All industries	60
Agriculture	320
Construction	227
Mining and Quarrying	167
Transportation and Public Utilities	160
Government	60
Manufacturing	40
Trade	27
Services	27

[a]Adapted from Reference 10.

embryo/fetus is involuntary on the part of the embryo/fetus, a more restrictive dose limit has been established for the embryo/fetus of a declared pregnant radiation worker. See Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

If an occupationally exposed woman declares her pregnancy in writing, she is subject to the more restrictive dose limits for the embryo/fetus during the remainder of the pregnancy. The dose limit of 500 mrems (5 mSv) for the total gestation period applies to the embryo/fetus and is controlled by restricting the exposure to the declared pregnant woman. Restricting the woman's occupational exposure, if she declares her pregnancy, raises questions about individual privacy rights, equal employment opportunities, and the possible loss of income. Because of these concerns, the declaration of pregnancy by a female radiation worker is voluntary. Also, the declaration of pregnancy can be withdrawn for any reason, for example, if the woman believes that her benefits from receiving the occupational exposure would outweigh the risk to her embryo/fetus from the radiation exposure.

12. Can a worker become sterile or impotent from normal occupational radiation exposure?

No. Temporary or permanent sterility cannot be caused by radiation at the levels allowed under NRC's occupational limits. There is a threshold below which these effects do not occur. Acute doses on the order of 10 rems (0.1 Sv) to the testes can result in a measurable but temporary reduction in sperm count. Temporary sterility (suppression of ovulation) has been observed in women who have received acute doses of 150 rads (1.5 Gy). The estimated threshold (acute) radiation dose for induction of permanent sterility is about 200 rads (2 Gy) for men and about 350 rads (3.5 Gy) for women (Refs. 1 and 4). These doses are far greater than the NRC s occupational dose limits for workers.

Although acute doses can affect fertility by reducing sperm count or suppressing ovulation, they do not have any direct effect on one's ability to function sexually. No evidence exists to suggest that exposures within the NRC's occupational limits have any effect on the ability to function sexually.

13. What are the NRC occupational dose limits?

For adults, an annual limit that does not exceed:

- 5 rems (0.05 Sv) for the total effective dose equivalent (TEDE), which is the sum of the deep dose equivalent (DDE) from external exposure to the whole body and the committed effective dose equivalent (CEDE) from intakes of radioactive material.
- 50 rems (0.5 Sv) for the total organ dose equivalent (TODE), which is the sum of the DDE from external exposure to the whole body and the committed dose equivalent (CDE) from intakes of radioactive material to any individual organ or tissue, other than the lens of the eye.
- 15 rems (0.15 Sv) for the lens dose equivalent (LDE), which is the external dose to the lens of the eye.
- 50 rems (0.5 Sv) for the shallow dose equivalent (SDE), which is the external dose to the skin or to any extremity.

For minor workers, the annual occupational dose limits are 10 percent of the dose limits for adult workers.

For protection of the embryo/fetus of a declared pregnant woman, the dose limit is 0.5 rem (5 mSv) during the entire pregnancy.

The occupational dose limit for adult workers of 5 rems (0.05 Sv) TEDE is based on consideration of the potential for delayed biological effects. The 5-rem (0.05 Sv) limit, together with application of the concept of keeping occupational doses ALARA, provides a level of risk of delayed effects considered acceptable by the NRC. The limits for individual organs are below the dose levels at which early biological effects are observed in the individual organs.

The dose limit for the embryo/fetus of a declared pregnant woman is based on a consideration of the possibility of greater sensitivity to radiation of the embryo/fetus and the involuntary nature of the exposure.

14. What is meant by ALARA?

ALARA means "as low as is reasonably achievable." In addition to providing an upper limit on an individual's permissible radiation dose, the NRC requires that its licensees establish radiation protection

programs and use procedures and engineering controls to achieve occupational doses, and doses to the public, as far below the limits as is reasonably achievable. "Reasonably achievable" also means "to the extent practicable." What is practicable depends on the purpose of the job, the state of technology, the costs for averting doses, and the benefits. Although implementation of the ALARA principle is a required integral part of each licensee's radiation protection program, it does not mean that each radiation exposure must be kept to an absolute minimum, but rather that "reasonable" efforts must be made to avert dose. In practice, ALARA includes planning tasks involving radiation exposure so as to reduce dose to individual workers and the work group.

There are several ways to control radiation doses, e.g., limiting the time in radiation areas, maintaining distance from sources of radiation, and providing shielding of radiation sources to reduce dose. The use of engineering controls, from the design of facilities and equipment to the actual set-up and conduct of work activities, is also an important element of the ALARA concept.

An ALARA analysis should be used in determining whether the use of respiratory protection is advisable. In evaluating whether or not to use respirators, the goal should be to achieve the optimal sum of external and internal doses. For example, the use of respirators can lead to increased work time within radiation areas, which increases external dose. The advantage of using respirators to reduce internal exposure must be evaluated against the increased external exposure and related stresses caused by the use of respirators. Heat stress, reduced visibility, and reduced communication associated with the use of respirators could expose a worker to far greater risks than are associated with the internal dose avoided by use of the respirator. To the extent practical, engineering controls, such as containments and ventilation systems, should be used to reduce workplace airborne radioactive materials.

15. What are background radiation exposures?

The average person is constantly exposed to ionizing radiation from several sources. Our environment and even the human body contain naturally occurring radioactive materials (e.g., potassium-40) that contribute to the radiation dose that we receive. The largest source of natural background radiation exposure is terrestrial radon, a colorless, odorless, chemically inert gas, which causes about 55 percent of our average, nonoccupational exposure. Cosmic radiation originating in space contributes additional exposure. The use of x-rays and radioactive materials in medicine and dentistry adds to our population exposure. As shown below in Table 3, the average person receives an annual radiation dose of about 0.36 rem (3.6 mSv). By age 20, the average person will accumulate over 7 rems (70 mSv) of dose. By age 50, the total dose is up to 18 rems (180 mSv). After 70 years of exposure this dose is up to 25 rems (250 mSv).

Table 3 Average Annual Effective Dose Equivalent to Individuals in the U.S.[a]

Source	*Effective Dose Equivalent (mrems)*	
Natural		
Radon	200	
Other than Radon	100	
Total		300
Nuclear Fuel Cycle		0.05
Consumer Products[b]		9
Medical		
Diagnostic X-rays	39	
Nuclear Medicine	14	
Total		53
Total		about 360 mrems/year

[a]Adapted from Table 8.1, NCRP 93 (Ref. 11).

[b]Includes building material, television receivers, luminous watches, smoke detectors, etc. (from Table 5.1, NCRP 93, Ref. 11).

16. What are the typical radiation doses received by workers?

For 1993, the NRC received reports on about a quarter of a million people who were monitored for occupational exposure to radiation. Almost half of those monitored had no measurable doses. The other half had an average dose of about 310 mrem (3.1 mSv) for the year. Of these, 93 percent received an annual dose of less than 1 rem (10 mSv); 98.7 percent received less than 2 rems (20 mSv); and the highest reported dose was for two individuals who each received between 5 and 6 rems (50 and 60 mSv).

Table 4 lists average occupational doses for workers (persons who had measurable doses) in various occupations based on 1993 data. It is important to note that beginning in 1994, licensees have been required to sum external and internal doses and certain licensees are required to submit annual reports. Certain types of licensees such as nuclear fuel fabricators may report a significant increase in worker doses because of the exposure to long-lived airborne radionuclides and the requirement to add the resultant internal dose to the calculation of occupational doses.

Table 4 Reported Occupational Doses for 1993[a]

Occupational Subgroup	Average Measurable Dose per Worker (millirems)
Industrial Radiography	540
Commercial Nuclear Power Reactors	310
Manufacturing and Distribution of Radioactive Materials	300
Low-Level Radioactive Waste Disposal	270
Independent Spent Nuclear Fuel Storage	260
Nuclear Fuel Fabrication	130

[a]From Table 3.1 in NUREG–0713 (Ref. 9).

17. How do I know how much my occupational dose (exposure) is?

If you are likely to receive more than 10 percent of the annual dose limits, the NRC requires your employer, the NRC licensee, to monitor your dose, to maintain records of your dose, and, at least on an annual basis for the types of licensees listed in 10 CFR 20.2206, "Reports of Individual Monitoring," to inform both you and the NRC of your dose. The purpose of this monitoring and reporting is so that the NRC can be sure that licensees are complying with the occupational dose limits and the ALARA principle.

External exposures are monitored by using individual monitoring devices. These devices are required to be used if it appears likely that external exposure will exceed 10 percent of the allowed annual dose, i.e., 0.5 rem (5 mSv). The most commonly used monitoring devices are film badges, thermoluminescence dosimeters (TLDs), electronic dosimeters, and direct reading pocket dosimeters.

With respect to internal exposure, your employer is required to monitor your occupational intake of radioactive material and assess the resulting dose if it appears likely that you will receive greater than 10 percent of the annual limit on intake (ALI) from intakes in 1 year. Internal exposure can be estimated by measuring the radiation emitted from the body (for example, with a "whole body counter") or by measuring the radioactive materials contained in biological samples such as urine or feces. Dose estimates can also be made if one knows how much radioactive material was in the air and the length of time during which the air was breathed.

18. What happens if a worker exceeds the annual dose limit?

If a worker receives a dose in excess of any of the annual dose limits, the regulations prohibit any occupational exposure during the remainder of the year in which the limit is exceeded. The licensee is also required to file an overexposure report with the NRC and provide a copy to the individual who received the dose. The licensee may be subject to NRC enforcement action such as a fine (civil penalty), just as individuals are subject to a traffic fine for exceeding a speed limit. The fines and, in some serious or repetitive cases, suspension of a license are intended to encourage licensees to comply with the regulations.

Radiation protection limits do not define safe or unsafe levels of radiation exposure. Exceeding a limit does not mean that you will get cancer. For radiation protection purposes, it is assumed that risks are related to the size of the radiation dose. Therefore, when your dose is higher your risk is also considered to be higher. These limits are similar to highway speed limits. If you drive at 70 mph, your risk is higher than at 55 mph, even though you may not actually have an accident. Those who set speed limits have determined that the risks of driving in excess of the speed limit are not acceptable. In the same way, the revised 10 CFR Part 20 establishes a limit for normal occupational exposure of 5 rems (0.05 Sv) a year. Although you will not necessarily get cancer or some other radiation effect at doses above the limit, it does mean that the licensee's safety program has failed in some way. Investigation is warranted to determine the cause and correct the conditions leading to the dose in excess of the limit.

19. What is meant by a "planned special exposure"?

A "planned special exposure" (PSE) is an infrequent exposure to radiation, separate from and in addition to the radiation received under the annual occupational limits. The licensee can authorize additional dose in any one year that is equal to the annual occupational dose limit as long as the individual's total dose from PSEs does not exceed five times the annual dose limit during the individual's lifetime. For example, licensees may authorize PSEs for an adult radiation worker to receive doses up to an additional 5 rems (0.05 Sv) in a year above the 5-rem (0.05–Sv) annual TEDE occupational dose limit. Each worker is limited to no more than 25 rems (0.25 Sv) from planned special exposures in his or her lifetime. Such exposures are only allowed in exceptional situations when alternatives for avoiding the additional exposure are not available or are impractical.

Before the licensee authorizes a PSE, the licensee must ensure that the worker is informed of the purpose and circumstances of the planned operation, the estimated doses expected, and the procedures to keep the doses ALARA while considering other risks that may

be present. (See Regulatory Guide 8.35, "Planned Special Exposures.")

20. Why do some facilities establish administrative control levels that are below the NRC limits?

There are two reasons. First, the NRC regulations state that licensees must take steps to keep exposures to radiation ALARA. Specific approval from the licensee for workers to receive doses in excess of administrative limits usually results in more critical risk-benefit analyses as each additional increment of dose is approved for a worker. Secondly, an administrative control level that is set lower than the NRC limit provides a safety margin designed to help the licensee avoid doses to workers in excess of the limit.

21. Why aren't medical exposures considered as part of a worker's allowed dose?

NRC rules exempt medical exposure, but equal doses of medical and occupational radiation have equal risks. Medical exposure to radiation is justified for reasons that are quite different from the reasons for occupational exposure. A physician prescribing an x-ray, for example, makes a medical judgment that the benefit to the patient from the resulting medical information justifies the risk associated with the radiation. This judgment may or may not be accepted by the patient. Similarly, each worker must decide on the benefits and acceptability of occupational radiation risk, just as each worker must decide on the acceptability of any other occupational hazard.

Consider a worker who receives a dose of 3 rems (0.03 Sv) from a series of x-rays in connection with an injury or illness. This dose and any associated risk must be justified on medical grounds. If the worker had also received 2 rems (0.02 Sv) on the job, the combined dose of 5 rems (0.05 Sv) would in no way incapacitate the worker. Restricting the worker from additional job exposure during the remainder of the year would not have any effect on the risk from the 3 rems (0.03 Sv) already received from the medical exposure. If the individual worker accepts the risks associated with the x-rays on the basis of the medical benefits and accepts the risks associated with job-related exposure on the basis of employment benefits, it would be unreasonable to restrict the worker from employment involving exposure to radiation for the remainder of the year.

22. How should radiation risks be considered in an emergency?

Emergencies are "unplanned" events in which actions to save lives or property may warrant additional doses for which no particular limit applies. The revised 10 CFR Part 20 does not set any dose limits for emergency or lifesaving activities and states that nothing in Part 20 "shall be construed as limiting actions that may be necessary to protect health and safety."

Rare situations may occur in which a dose in excess of occupational limits would be unavoidable in order to carry out a lifesaving operation or to avoid a large dose to large populations. However, persons called upon to undertake any emergency operation should do so only on a voluntary basis and with full awareness of the risks involved.

For perspective, the Environmental Protection Agency (EPA) has published emergency dose guidelines (Ref. 2). These guidelines state that doses to all workers during emergencies should, to the extent practicable, be limited to 5 rems (0.05 Sv). The EPA further states that there are some emergency situations for which higher limits may be justified. The dose resulting from such emergency exposures should be limited to 10 rems (0.1 Sv) for protecting valuable property, and to 25 rems (0.25 Sv) for lifesaving activities and the protection of large populations. In the context of this guidance, the dose to workers that is incurred for the protection of large populations might be considered justified for situations in which the collective dose to others that is avoided as a result of the emergency operation is significantly larger than that incurred by the workers involved.

Table 5 presents the estimates of the fatal cancer risk for a group of 1,000 workers of various ages, assuming that each worker received an acute dose of 25 rems (0.25 Sv) in the course of assisting in an emergency. The estimates show that a 25-rem emergency dose might increase an individual's chances of developing fatal cancer from about 20% to about 21%.

Table 5
Risk of Premature Death from Exposure to 25-Rems (0.25-Sv) Acute Dose

Age at Exposure (years)	*Estimated Risk of Premature Death (Deaths per 1,000 Persons Exposed)*
20–30	9.1
30–40	7.2
40–50	5.3
50–60	3.5

Source: EPA–400–R–92–001 (Ref. 2).

23. How were radiation dose limits established?

The NRC radiation dose limits in 10 CFR Part 20 were established by the NRC based on the recommendations of the ICRP and NCRP as endorsed in Federal radiation protection guidance developed by the EPA

(Ref. 12). The limits were recommended by the ICRP and NCRP with the objective of ensuring that working in a radiation-related industry was as safe as working in other comparable industries. The dose limits and the principle of ALARA should ensure that risks to workers are maintained indistinguishable from risks from background radiation.

24. Several scientific reports have recommended that the NRC establish lower dose limits. Does the NRC plan to reduce the regulatory limits?

Since publication of the NRC's proposed rule in 1986, the ICRP in 1990 revised its recommendations for radiation protection based on newer studies of radiation risks (Ref. 13), and the NCRP followed with a revision to its recommendations in 1993. The ICRP recommended a limit of 10 rems (0.1 Sv) effective dose equivalent (from internal and external sources), over a 5-year period with no more than 5 rems (0.05 Sv) in 1 year (Ref. 13). The NCRP recommended a cumulative limit in rems, not to exceed the individual's age in years, with no more than 5 rems (0.05 Sv) in any year (Ref. 14).

The NRC does not believe that additional reductions in the dose limits are required at this time. Because of the practice of maintaining radiation exposures ALARA (as low as is reasonably achievable), the average radiation dose to occupationally exposed persons is well below the limits in the current Part 20 that became mandatory January 1, 1994, and the average doses to radiation workers are below the new limits recommended by the ICRP and the NCRP.

25. What are the options if a worker decides that the risks associated with occupational radiation exposure are too high?

If the risks from exposure to occupational radiation are unacceptable to a worker, he or she can request a transfer to a job that does not involve exposure to radiation. However, the risks associated with the exposure to radiation that workers, on the average, actually receive are comparable to risks in other industries and are considered acceptable by the scientific groups that have studied them. An employer is not obligated to guarantee a transfer if a worker decides not to accept an assignment that requires exposure to radiation.

Any worker has the option of seeking other employment in a nonradiation occupation. However, the studies that have compared occupational risks in the nuclear industry to those in other job areas indicate that nuclear work is relatively safe. Thus, a worker may find different kinds of risk but will not necessarily find significantly lower risks in another job.

26. Where can one get additional information on radiation risk?

The following list suggests sources of useful information on radiation risk:

- The employer—the radiation protection or health physics office where a worker is employed.
- Nuclear Regulatory Commission Regional Offices:

King of Prussia, Pennsylvania	(610) 337–5000
Atlanta, Georgia	(404) 331–4503
Lisle, Illinois	(708) 829–9500
Arlington, Texas	(817) 860–8100

- U.S. Nuclear Regulatory Commission
  Headquarters
  Radiation Protection & Health Effects Branch
  Office of Nuclear Regulatory Research
  Washington, DC 20555
  Telephone: (301) 415–6187
- Department of Health and Human Services
  Center for Devices and Radiological Health
  1390 Piccard Drive, MS HFZ–1
  Rockville, MD 20850
  Telephone: (301) 443–4690
- U.S. Environmental Protection Agency
  Office of Radiation and Indoor Air
  Criteria and Standards Division
  401 M Street NW.
  Washington, DC 20460
  Telephone: (202) 233–9290

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*Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202) 634-3273; fax (202) 634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202) 512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

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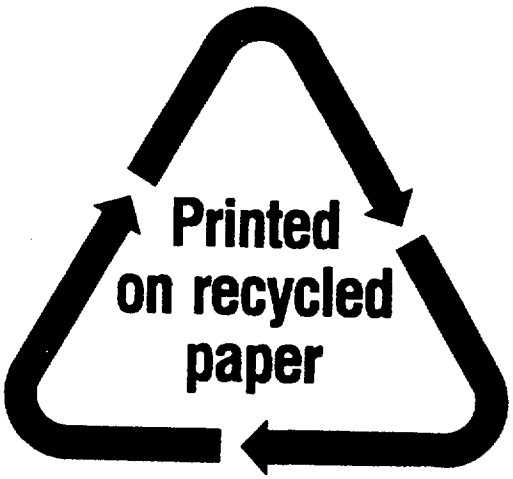
REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this Revision 1 to Regulatory Guide 8.29. A value/impact statement, which evaluated essentially the same subjects as are discussed in a regulatory analysis, accompanied Regulatory Guide 8.29 when it was issued in July 1981.

This Revision 1 to Regulatory Guide 8.29 is needed to conform with the Revised 10 CFR Part 20, "Standards for Protection Against Radiation," as published May 21, 1991 (56 FR 23360). The regulatory analysis prepared for 10 CFR Part 20 provides the regulatory basis for this Revision 1 of Regulatory Guide 8.29, and it 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 in the NRC's Public Document Room at 2120 L Street NW., Washington, DC 20555–0001.



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Revision 1
May 2002

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.30

(Draft was issued as DG-8026)

HEALTH PHYSICS SURVEYS IN URANIUM RECOVERY FACILITIES

A. INTRODUCTION

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

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

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

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

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

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

B. DISCUSSION

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

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

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

C. REGULATORY POSITION

1. OCCUPATIONAL DOSE LIMITS

In 10 CFR Part 20, "Standards for Protection Against Radiation," 10 CFR 20.1201 establishes radiation dose limits for occupationally exposed adults. These dose limits apply to the

sum of the dose received from external exposure and the dose from internally deposited radioactive material. These dose limits are summarized in Table 1. The occupational dose limits for minors according to 10 CFR 20.1207 are 10% of the dose limit for adults, and 10 CFR 20.1208 establishes a dose limit for the embryo/fetus of 0.5 rem (0.005 Sv) during the entire declared pregnancy.

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

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

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

TABLE 1
Dose Limits and Associated Terminology

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

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

2. SURVEYS

2.1 Surveys for Airborne Uranium Ore Dust

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

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

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

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

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

where

DAC_m = regulatory concentration value for natural uranium

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

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

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

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

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

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

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

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

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

2.2 Surveys for Airborne Yellowcake

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

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

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

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

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

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

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

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

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

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

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

2.3 Surveys for Radon-222 and Its Daughters

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

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

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

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

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

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

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

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

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

2.4 Surveys for External Radiation

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

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

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

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

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

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

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

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

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

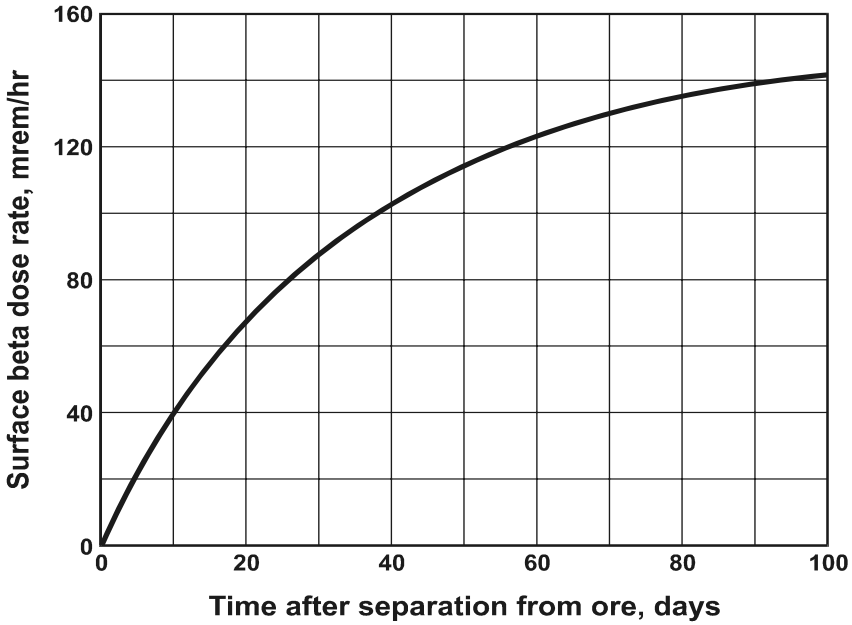

Figure 1. Beta Dose Rate on the Surface of Yellowcake

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

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

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

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

2.5 Surveys for Surface Contamination in Restricted Area

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

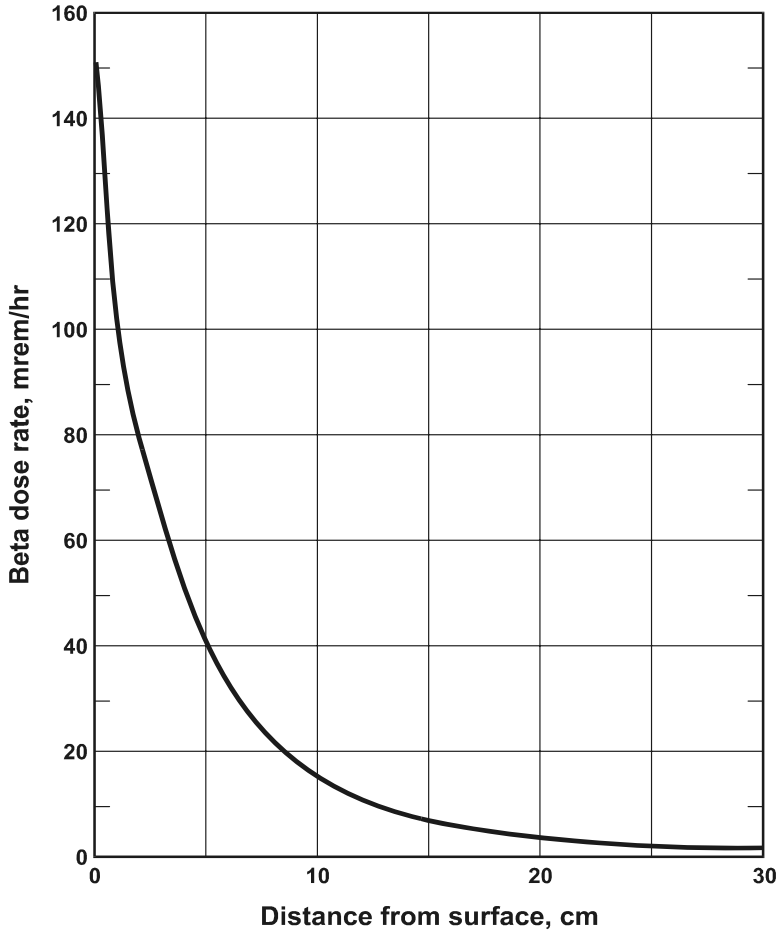

Figure 2. Beta Dose Rate from Yellowcake Separate from Ore for More than 100 Days (from Reference 10)

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

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

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

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

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

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

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

TABLE 2
Surface Contamination Levels for Uranium and Daughters on Equipment To Be Released for Unrestricted Use, on Clothing, and on Nonoperating Areas of UR Facilities*

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

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

** The value includes both fixed and removable contamination.

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

2.6 Surveys for Contamination of Skin and Personal Clothing

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

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

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

2.7 Surveys of Equipment Prior to Release to Unrestricted Areas

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

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

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

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

2.8 Surveys of Packages Prepared for Shipment

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

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

2.9 Ventilation Surveys

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

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

2.10 Surveys for Contamination on Respirators

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

2.11 Summary of Survey Frequencies

Table 3 summarizes the survey frequencies given in this guide.

3. INTAKE AND EXPOSURE CALCULATIONS

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

evaluated. The regulatory requirements for determining the internal dose are contained in 10 CFR 20.1204.

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

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

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

TABLE 3
SUMMARY OF SURVEY FREQUENCIES

Type of Survey	Type of Area	Survey Frequency	Lower Limit of Detection
1. Uranium ore dust	Airborne radioactivity areas Other indoor process areas Outdoor areas	Weekly grab samples Monthly grab samples Quarterly grab samples	5×10^{-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 Beta	Throughout UR facility Radiation areas Where workers are in close contact with yellowcake	Semiannually Quarterly Survey by operation done once plus whenever procedures change	0.1 mrad/hr 1 mrem/hr
5. Surface contamination	Yellowcake areas Eating rooms, change rooms, control rooms, offices	Daily Weekly	Visual 500 dpm alpha per 100 cm^2
6. Skin and personal clothing	Yellowcake workers who shower Yellowcake workers who do not shower	Quarterly Each day before leaving	500 dpm alpha per 100 cm^2
7. Equipment to be released	Equipment to be released that may be contaminated	Once before release	500 dpm alpha per 100 cm^2
8. Package containing yellowcake	Packages	Spot check before release	500 dpm alpha per 100 cm^2
9. Ventilation	All areas with airborne radioactivity	Daily	Not applicable
10. Respirators	Respirator face pieces and hoods	Before reuse	100 dpm alpha per 100 cm^2

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

where

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

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

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

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

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

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

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

Method 2: Use of DACs from 10 CFR Part 20

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

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

where

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

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

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

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

where

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

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

t = The duration of the exposure (hours)

2000 = The number of hours in a work year

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

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

4. ADMINISTRATIVE ACTION LEVELS

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

4.1 Uranium Ore[7] Dust

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

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

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

4.2 Yellowcake

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

4.3 Radon Daughters

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

4.4 Time-Weighted Exposure to Airborne Radioactivity

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

4.5 Gamma Dose Rates

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

4.6 Dosimeter Results

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

4.7 Contamination on Skin and Clothing

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

4.8 Low Airborne Radioactivity Concentrations

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

5. REPORTING REQUIREMENTS

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

6. ESTABLISHING "AIRBORNE RADIOACTIVITY AREAS"

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

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

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

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

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

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

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

8. CALIBRATION OF SURVEY INSTRUMENTS

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

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

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

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

calibrated with a gamma source alone. A correction factor must be applied to determine the beta dose rate.

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

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

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

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

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

9. PROTECTIVE CLOTHING

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

Protective clothing should be changed and discarded or laundered weekly or whenever yellowcake is visible on the clothing. Potentially contaminated clothing should not be sent to a laundry that is not specifically authorized by the NRC or an Agreement State to process clothing

contaminated with uranium unless the clothing has been surveyed and found to have less uranium contamination than the values in Table 2 of this guide.

10. QUALITY ASSURANCE PROGRAM

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

D. IMPLEMENTATION

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

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

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APPENDIX A

DERIVATION OF EQUATION FOR DAC_m

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

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

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

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

the equality in the first equation will apply. Therefore:

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

Solve for DAC_m

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

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

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

The term

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

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

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

APPENDIX B

LOWER LIMIT OF DETECTION

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

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

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

where:

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

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

$3.7 x10^4$ = the number of disintegrations/sec/μCi (this term is omitted if Sb is given in terms of microcuries)

E = the counting efficiency (counts per disintegration)

V = the sample volume (ml)

Y = the fractional radiochemical yield (if applicable)

λ = the decay constant for the particular radionuclide

t = the elapsed time between sample collection and counting

Example: LLD for Uranium when Fluorometric Analysis Is Used

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

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

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

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

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

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

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

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

where:

n = the number of samples

X_i = the reading for sample i

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

For the data above, the standard deviation is:

$S_b = 0.023$ 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:

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

$= 0.0029$ µg of uranium

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

$S_b = 0.0029$ µg $\times 6.77 \times 10^{-7}$ μCi/µg

$= 1.97 \times 10^{-9}$ μCi

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

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

V= 10 liters/min x 30 minutes

= 300 liters

= 300,000 ml

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

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

Therefore

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

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

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

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

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

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

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

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

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

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

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

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

LLD = 0.025 WL

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

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APPENDIX C

BETA CORRECTION FACTOR FOR SURVEY INSTRUMENT

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

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

Observed dose rate x CF = actual dose rate

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

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

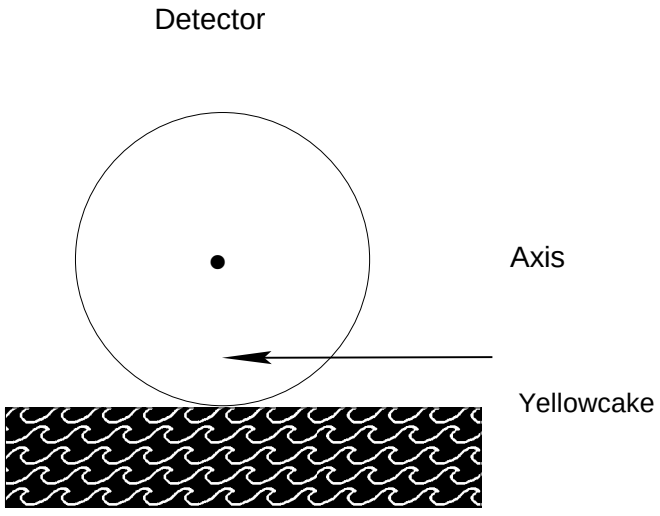
Cf_{sur} = 6 mrem/mR (at the surface)

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

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

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

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



REGULATORY ANALYSIS

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

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

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
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REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE

8.31

(Draft was issued as DG-8027)

INFORMATION RELEVANT TO ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES AT URANIUM RECOVERY FACILITIES WILL BE AS LOW AS IS REASONABLY ACHIEVABLE

A. INTRODUCTION

This revision of Regulatory Guide 8.31 has been developed to provide guidance on design criteria and administrative practices acceptable to the NRC staff for maintaining occupational exposures as low as is reasonably achievable (ALARA) in uranium recovery (UR) facilities (for example, uranium milling, in situ leach facilities, ion exchange facilities, heap leach facilities). This guidance can also be applied, in part, to other types of UR facilities and portions of conversion facilities since some of the processes used in these facilities are similar to those in UR facilities.

Section 20.1101 of 10 CFR Part 20, "Standards for Protection Against Radiation," states that licensees must use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable" (Ref. 1), sets forth the philosophy and general management policies and programs that licensees should follow to achieve this objective.

An existing NRC report, NUREG-0706, "Final Generic Environmental Impact Statement on Uranium Milling" (Ref. 2), also provides detailed information for controlling the radiation hazard and chemical toxicity of airborne uranium and its daughter products in UR facilities.

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 UR operation workers will be exposed to external radiation in addition to inhaled particulates. Therefore, ensuring protection of operation 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 UR facility ALARA (see Regulatory Guide 4.14, "Radiological Effluent and Environmental Monitoring At Uranium Mills" (Ref. 3).

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

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. 4). 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 (Ref. 3), as well as in recommendations of the National Academy of Sciences-National Research Council (Ref. 5), the Federal Radiation Council (Ref. 6), and other independent scientific and professional organizations (Ref. 7). Therefore, NRC has incorporated this basic radiation protection philosophy from these recommendations into its regulations and guides.

Regulatory Guide 8.10 (Ref. 1) lists the types of management commitments and radiation protection programs that would help to achieve the objective of maintaining occupational exposures ALARA for all specific licensees. This guide provides a detailed supplement of the basic philosophy of Regulatory Guide 8.10 for uranium recovery licensees.

Regulatory Guide 3.5, “Standard Format and Content of License Applications for Uranium Mills” (Ref. 8), 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 Regulatory Position 5, “Operations,” of Regulatory Guide 3.5. Also see Regulatory Guide 3.46, “Standard Format and Content of License Applications, Including Environmental Reports, for In Situ Uranium Solution Mining” (Ref. 9).

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 UR facility for appropriate sections of an application[1] 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 staff and are consistent with the philosophy of ALARA. Unique features not addressed here will be specifically reviewed by the NRC licensing staff. This guide could also be used by facilities that concentrate uranium as a secondary process to control hazards from uranium.

A licensee's program for occupational protection against uranium and its daughters will be considered consistent with the ALARA philosophy if the UR facility’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 UR facility 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,[2] the radiation safety officer (RSO),[3] and all workers in the UR facility.

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 and visitors.

Licensee management should provide the following:

[1] 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, DC 20555.

[2] “Management” is the persons authorized by the licensee of record to make policies and to direct activities of the recovery facility.

[3] 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.

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 radiation safety (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 employees in the facility 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 protection 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 Uranium Recovery Workers

Because a radiation protection and ALARA program is only as effective as the workers' adherence to the program, all workers at a UR facility 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;

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 a UR 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 NRC'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 and techniques 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 when the potential for exposure to radioactive material exists and for which no standard written operating procedure already exists, a radiation work permit (RWP)[4] 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

With sufficient management interest, exposure to hazardous materials is reduced. Frequent management audit and inspection of worker health protection practices at a UR facility 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 facility foreman should conduct a weekly inspection of all facility 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 facility 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 or other retrievable record format. 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 with 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, the RSO's staff, or the RSO's designee prior to initiation of work.

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

[4]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.

the resident manager and all department heads for their review a written summary of the month's significant worker protection activities that contains (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 UR facility. In addition, one member of the team should be experienced in the operational aspects of specialized UR facility 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 the 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:

- Trends in personnel exposures for identifiable categories of workers and types of operational activities.

- Whether equipment for exposure control is being properly used, maintained, and inspected.

- 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 UR facility 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 UR operations 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 recovery. In addition, the RSO should attend refresher training on UR facility 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 UR facility, 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 UR process and equipment used in the facility and how the hazards are generated and controlled during the UR 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 UR facility. 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 UR facilities;

   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 UR facility; 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 UR facilities;

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 UR facility, 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" (Ref. 10), and Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure" (Ref. 11). Additionally, the training should be commensurate with the risks and hazards of the task. This course of instruction should include the following topics:

1. Fundamentals of Health Protection
   - The radiologic and toxic hazards of exposure to uranium and its daughters,
   - How uranium and its daughters enter the body (inhalation, ingestion, and skin penetration),
   - Why exposures to uranium and its daughters should be kept ALARA.

2. Personal Hygiene at UR Facilities
   - Wearing protective clothing,
   - Using respiratory protective equipment correctly,
   - Eating, drinking, and smoking only in designated areas,
   - Using proper methods for decontamination (i.e., showers).

3. Facility-Provided Protection
   - Ventilation systems and effluent controls,
   - Cleanliness of the work place,
   - Features designed for radiation safety for process equipment,
   - Standard operating procedures,
   - Security and access control to designated areas,
   - Electronic data gathering and storage,
   - Automated processes.

4. Health Protection Measurements
   - Measurement of airborne radioactive materials,
   - Bioassays to detect uranium (urinalysis and in vivo counting),

- Surveys to detect contamination of personnel and equipment,
- Personnel dosimetry.

5. Radiation Protection Regulations
   - Regulatory authority of NRC, MSHA, and State,
   - Employee rights in 10 CFR Part 19,
   - Radiation protection requirements in 10 CFR Part 20.

6. Emergency Procedures.

A written or oral test with questions directly relevant to the principles of radiation safety and health protection in UR 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 and on the nonradiological hazards 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 meetings.

All visitors who have not received training should be escorted by someone properly trained and knowledgeable about the hazards of the facility. At a minimum, visitors should be instructed specifically on what they should do to avoid possible radiological and nonradiological hazards in the areas of the facility they will be visiting.

Contractors that have work assignments in a UR facility should also be given appropriate training and safety instruction. Contractor 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 facility 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 the Regulatory Position of Regulatory Guide 8.30, "Health Physics Surveys in Uranium Recovery Facilities" (Ref. 12).

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 protection 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 these individuals who will use respiratory protective equipment. If the licensee elects to take credit for protection factors, the respiratory protection program must meet, at a minimum, the requirements of 10 CFR 20.1703 and should follow the recommendations in Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection" (Ref.13), which are supported in NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials" (Ref. 14).

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" (Ref. 15).

3. FACILITY AND EQUIPMENT DESIGN

General considerations for the design of UR facilities 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

The 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 facility layout should provide for:

1. Safe access to process equipment for routine maintenance;
2. Adequate ventilation in all facility areas in which radioactive materials might be spilled, suspended, or volatilized, (e.g., engineered controls);
3. Isolation of yellowcake drying, packaging, and shipping areas from other accessible facility areas;

4. Controlling access to the UR facility and the ability to secure or restrict entry to any airborne radioactivity areas;
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 facility 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 procedures, or security locks when permitted by fire protection regulations.

3.3 Ventilation Systems

To the extent practicable, the facility ventilation system 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.[5] 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 Derived Air Concentration (DAC) given in Table 1 of Appendix B to 10 CFR Part 20.

2. Establish a facility-specific, operational ALARA goal for concentrations of natural uranium and its daughters at less than 25% of the DAC 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 facility areas.

4. Locate exhaust vents in a way that ensures compliance with the requirements of 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public,” for effluents to unrestricted areas, as well as ALARA exposure considerations for the worker, and 40 CFR, "Protection of Environment," Part 190, "Environmental Radiation Standards for Nuclear Power Operations,” and 10 CFR 20.1101(d) regarding constraint on air emissions of radioactive material.

[5]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 10 CFR 20.1003. According to 10 CFR 20.1701, “The licensee shall use, to the extent practical, process or other engineering controls (e.g., containment or ventilation) to control the concentrations of radioactive material in air.”

3.4 Fire Control

Because of the potential for loss of control of radioactive material in the event of a fire, a UR 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 UR 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 be as close to background as practicable but less than 200 dpm α/100 cm^2 of total surface contamination. All laboratories in the facility 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

UR facility plans should include the following:

1. Provisions for storage of raw ore[6] or other materials to be processed, fine ore bins, and yellowcake storage in areas so that the material does not cause unnecessary exposure to the facility's 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;

[6] Ore is a natural or native matter that may be mined and treated for the extraction of any of its constituents or any other matter from which source material is extracted in a licensed uranium or thorium mill.

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 10 CFR 20.1901, as discussed in more detail in Revision 1 of Regulatory Guide 8.30, "Health Physics Surveys in Uranium Recovery Facilities" (Ref. 12);

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 UR 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 UR facility. Some of these areas are applicable to mills only and others are applicable to all types of UR facilities. 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 Regulatory Position 4.4 of this guide. Appropriate design objectives for ventilation and dust control systems recommended for each of these generalized facility 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 conveyor 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 facility 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 DAC 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 facility 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 25% of the DAC for natural uranium. The use of wet scrubbers on general ventilation systems that service areas of the facility where grinding and leaching equipment are located is recommended. Scrubbers are not necessary on ventilation systems that service areas of the facility 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 25% of the DAC for natural uranium. 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 facility. 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 should be 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 UR facility that have the potential to contain hazardous levels of uranium and its daughters in air include maintenance shops, metallurgical and bioassay

laboratories, and general laundries, if they exist. Each of the above facility areas should be serviced by ventilation systems designed to maintain air concentration of natural uranium and its daughters to less than 25% of the DAC for natural uranium. Wet scrubbers are not necessary on these systems, but bag filters are recommended.

D. IMPLEMENTATION

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

Except when an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods in this guide reflecting public comments, along with Regulatory Guide 3.5, "Standard Format and Content of License Applications for Uranium Mills" (Ref. 8); Regulatory Guide 3.46, "Standard Format and Content of License Applications, Including Environmental Reports, for In Situ Uranium Solution Mining" (Ref. 9); Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection" (Ref. 13); Regulatory Guide 8.22, "Bioassay at Uranium Mills" (Ref. 15); and Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills" (Ref. 12), will be used as the basis for evaluating license applications and radiation safety and ALARA programs of NRC-licensed UR facilities.

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10. USNRC, “Instruction Concerning Prenatal Radiation Exposure,” Regulatory Guide 8.13, Revision 3, June 1999.[1]

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

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11. USNRC, “Instruction Concerning Risks from Occupational Radiation Exposure,” Regulatory Guide 8.29, Revision 1, February 1996.[1]

12. USNRC, “Health Physics Surveys in Uranium Recovery Facilities,” Regulatory Guide 8.30, Revision 1, May 2002.[1]

13. USNRC, “Acceptable Programs for Respiratory Protection,” Regulatory Guide 8.15, Revision 1, October 1999.[1]

14. USNRC, “Manual of Respiratory Protection Against Airborne Radioactive Materials,” NUREG-0041, October 1976.[2]

15. USNRC, “Bioassay at Uranium Mills,” Regulatory Guide 8.22, Revision 1, August 1988.[1]

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this Revision 1 to Regulatory Guide 8.31. A value/impact statement, which evaluated essentially the same subjects as are discussed in a regulatory analysis, accompanied Regulatory Guide 8.31 when it was issued in May 1983.

A Revision 1 to Regulatory Guide 8.31 is needed to conform with the revised 10 CFR Part 20, "Standards for Protection Against Radiation," as published May 21, 1991 (56 FR 23360). The regulatory analysis prepared for 10 CFR Part 20 provides the regulatory basis for this Revision 1 of Regulatory Guide 8.31, and it examines the cost and benefits of the rule as implemented by the guide. A copy of the "Regulatory Analysis for the Revision of 10 CFR Part 20" (PNL-6712, November 1988), is available for inspection and copying for a fee in the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or 1-(800)397-4209; fax (301)415-3548; e-mail <PDR@NRC.GOV>.

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REGULATORY GUIDE 8.34

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MONITORING CRITERIA AND METHODS TO CALCULATE OCCUPATIONAL RADIATION DOSES

A. INTRODUCTION

Monitoring of an individual's external radiation exposure is required by 10 CFR 20.1502(a) if the external occupational dose is likely to exceed 10% of the dose limit appropriate for the individual (i.e., adult, minor, or declared pregnant woman). External radiation monitoring is also required by 10 CFR 20.1502(a)(3) for any individual entering a high or very high radiation area.

Monitoring of the intake of radioactive material is required by 10 CFR 20.1502(b) if the intake is likely to exceed 0.1 ALI (annual limit on intake) during the year for an adult worker or the committed effective dose equivalent is likely to exceed 0.05 rem (0.5 mSv) for the occupationally exposed minor or declared pregnant woman.

In the revised 10 CFR Part 20, "Standards for Protection Against Radiation," 10 CFR 20.1201 establishes radiation dose limits for occupationally exposed adults. These limits apply to the sum of the dose received from external exposure and the dose from internally deposited radioactive material. In 10 CFR 20.1201(a)(1), the annual limits for adults are (i) 5 rems (0.05 Sv) total effective dose equivalent or (ii) 50 rems (0.5 Sv) total organ dose equivalent to any single organ or tissue (other than the lens of the eye), whichever is more limiting. The occupational dose limits for minors in 10 CFR 20.1207 are 10% of the dose limit for adults, and 10 CFR 20.1208 establishes a dose limit for the embryo/fetus of 0.5 rem (0.005 Sv) during the entire pregnancy.

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

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

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 provides criteria acceptable to the NRC staff that may be used by licensees to determine when monitoring is required, and it describes methods acceptable to the NRC staff for calculating occupational doses when the intake is known. Guidance on calculating doses to the embryo/fetus is contained in Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus." Revision 1 to Regulatory Guide 8.9, "Interpretation of Bioassay Measurements," is under development and will provide guidance on determining intakes from bioassay results. Guidance on determining intakes from air sampling measurements is contained in Revision 1 to Regulatory Guide 8.25, "Air Sampling in the Workplace." Guidance on recording the calculated doses onto NRC Forms 4 and 5 is described in Revision 1 to Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data."

The appendix to this guide gives examples of the calculations of internal and external doses for entry onto NRC Form 5.

C. REGULATORY POSITIONS

1. MONITORING CRITERIA

The monitoring requirements in 10 CFR Part 20 are summarized in Table 1. For external dose monitoring, 10 CFR 20.1502(a) requires the use of individual monitoring devices. Individual monitoring devices are not required for monitoring the intake of radioactive material.

The monitoring requirements apply separately to each external dose type (i.e., deep-dose equivalent, shallow-dose equivalent to the skin, eye dose equivalent, and shallow-dose equivalent to the extremities).

1.1 Evaluation of Likely Annual Occupational Dose

Evaluation of the likelihood of doses exceeding 10% of the limit should be based on the potential occupational dose to the individual for the year. Doses that may have been received or will be received during the year from employment by another licensee are not included in the determination of monitoring requirements. The requirements in 10 CFR 20.1502 refer to *each* licensee. Each licensee makes the determination independently. It would not be appropriate to base the monitoring requirements at one licensee's facility on exposure conditions at a different licensee's facility. Rather, the need for monitoring at a facility should be based on the exposure conditions at that facility only.

Evaluations of previous dosimetric or bioassay data may be considered in projecting doses. The use of and credit for respiratory protective equipment may be considered in the evaluations, provided use of the equipment is in compliance with the requirements of 10 CFR 20.1703. Surveys of dose rates and estimates of occupancy times may be used to estimate expected external doses. Measurements and predictions of airborne radionuclide concentrations and the expected duration of exposure may be used to predict radionuclide intakes. The potential for unlikely exposures and accident conditions need not be considered because these events, by definition, are not likely.

1.2 Establishing Categories of Workers for Monitoring

If groups or categories of workers are exposed to similar radiological conditions, a single evaluation may be used to determine the need for monitoring. For simplicity, licensees may establish routine operational guidelines for categories of workers who will be monitored. For example, licensees may establish criteria or procedures for monitoring based on anticipated area access or work functions.

1.3 Change in Exposure Conditions

If an individual's radiation exposure conditions change during the year, the need to provide individual monitoring should be reevaluated.

For example, consider an unmonitored individual whose work assignment is changed from periodic delivery of supplies to a restricted area to performing maintenance activities within a radiation area. Under this new job assignment, if the licensee determines that the worker's dose is likely to exceed 10% of the limit, 10 CFR 20.1502 requires that monitoring be provided. When monitoring is required, 10 CFR 20.2106 requires that the monitored doses be recorded.

Similarly, if reevaluation of a monitored individual's anticipated annual occupational dose indicates that the dose is likely to be below 10% of the limits, monitoring may be terminated. Even when the doses are actually below 10% of the limit, the doses measured while monitoring was provided must be recorded pursuant to 10 CFR 20.2106 because the monitoring was provided to meet 10 CFR 20.1502.

1.4 Monitoring Performed But Not Required by 10 CFR 20.1502

Individual monitoring may be conducted for reasons other than those noted in 10 CFR 20.1502. While the results of required monitoring are subject to the dose recording requirements of 10 CFR 20.2106, the results of monitoring provided when not

Table 1

Summary of 10 CFR Part 20 Monitoring Requirements

The use of individual monitoring devices for external dose is required:

For adults who are likely to receive an annual dose in excess of any of the following (each evaluated separately):

- 0.5 rem (0.005 Sv) deep-dose equivalent.
- 1.5 rems (0.015 Sv) eye dose equivalent.
- 5 rems (0.05 Sv) shallow-dose equivalent to the skin.
- 5 rems (0.05 Sv) shallow-dose equivalent to any extremity.

For minors who are likely to receive an annual dose in excess of any of the following (each evaluated separately):

- 0.05 rem (0.5 mSv) deep-dose equivalent.
- 0.15 rem (1.5 mSv) eye dose equivalent.
- 0.5 rem (0.005 Sv) shallow-dose equivalent to the skin.
- 0.5 rem (0.005 Sv) shallow-dose equivalent to any extremity.

For declared pregnant women who are likely to receive an annual dose from occupational exposure in excess of 0.05 rem (0.5 mSv) deep-dose equivalent, although the dose limit applies to the entire gestation period.

Individuals entering a high or a very high radiation area.

Internal exposure monitoring (not necessarily individual monitoring devices) is required:

For adults likely to receive in 1 year an intake in excess of 10% of the applicable ALIs for ingestion and inhalation.

For minors and declared pregnant women likely to receive in 1 year a committed effective dose equivalent in excess of 0.05 rem (0.5 mSv).

required by 10 CFR 20.1502 are not subject to those dose recording requirements.

Surveys and monitoring results that serve as confirmatory measures are not subject to the individual dose recordkeeping requirements of 10 CFR 20.2106(a) provided such results confirm that actual individual doses are less than 10% of the limits. These surveys and monitoring results may be used to meet 10 CFR 20.1501 requirements. An example of confirmatory monitoring is an individual's annual bioassay measurement used as confirmation of the adequacy of airborne control measures. Another example is placing monitoring devices, such as thermoluminescence dosimeters (TLDs), on a sample of workers to provide a confirmation that doses are not above those anticipated.

1.5 Detection Sensitivity

The monitoring criteria contained in 10 CFR 20.1502 do not establish required levels of detection sensitivity, e.g., the lower limit of detection (LLD). For example, it may not be feasible to actually confirm intakes of 10% of the ALI, particularly for bioassay measurements of some alpha-emitting radionuclides. Therefore, monitoring thresholds should not be considered requirements on the sensitivity of a particular measurement. Workplace monitoring and occupancy factors should be considered, as appropriate, in evaluating potential exposures and monitoring requirements.

2. DETERMINATION OF EXTERNAL DOSES

There are three dose limits included in 10 CFR 20.1201 that apply to external exposure: deep dose to the whole body (5 rems or 0.05 Sv), shallow dose to the skin or extremities (50 rems or 0.5 Sv), and dose to the lens of the eye (15 rems or 0.15 Sv). According to the definitions in 10 CFR 20.1003, the deep-dose equivalent to the whole body is considered to be at a tissue depth of 1 cm (1000 mg/cm^2), shallow-dose equivalent to the skin or extremities at 0.007 cm (7 mg/cm^2), and eye dose equivalent at 0.3 cm (300 mg/cm^2). In evaluating the eye dose equivalent, it is acceptable to take credit for the shielding provided by protective lenses.

2.1 Placement of Individual Monitoring Devices

External dose is typically determined by the use of individual monitoring devices, such as film badges and thermoluminescence dosimeters (TLDs). The device for monitoring the whole body dose should be placed near the location expected to receive the highest dose during the year (10 CFR 20.1201(c)). When the whole body is exposed fairly uniformly, the individual monitoring device is typically worn on the front of the upper torso.

If the radiation dose is highly nonuniform, causing a specific part of the whole body (head, trunk, arms above the elbow, or legs above the knees) to receive a substantially higher dose than the rest of the whole body, the individual monitoring device should be placed near that part of the whole body expected to receive the highest dose. For example, if the dose rate to the head of an individual is expected to be higher than the dose rate to the trunk of the body, a monitoring device should be located on or close to the head so as to measure the dose received by the head.

If postexposure evaluations indicate that the maximum dose to a part of the whole body was substantially higher than the dose measured by the individual monitoring device, an evaluation should be conducted to estimate the actual maximum dose.

2.2 Use of More Than One Dosimeter

An acceptable alternative approach for highly nonuniform radiation fields is to use more than one dosimeter to separately track doses to different parts of the whole body. At the end of the year, each of the doses for each location would be summed. The deep- dose equivalent to be recorded would be that of the dosimeter location receiving the highest dose.

2.3 Extremity Monitoring

If the licensee determines that extremity monitoring is required, it may be appropriate to use an extremity dosimeter for some, but not all, radiation exposure. The licensee could supply an extremity dosimeter when exposure is nonuniform. When exposure is uniform, the shallow-dose equivalent measured by a torso dosimeter would be representative of the shallow-dose equivalent to the extremities, and separate extremity monitoring would not be needed.

If protective gloves are used, it is acceptable to place the extremity dosimeter under the gloves.

3. CALCULATION OF COMMITTED EFFECTIVE DOSE EQUIVALENT FROM INHALATION

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

Some noble gases in Appendix B to §§ 20.1001–20.2401 do not have inhalation ALI values listed and are listed "submersion" class. For these radionuclides, the internal dose is negligible compared to the external dose. These radionuclides may be excluded from the determination of the internal dose.

There are at least five methods acceptable to the NRC staff for calculating committed effective dose equivalent from inhaled radioactive materials. The five methods are described below.

3.1 Use of Federal Guidance Report No. 11

Federal Guidance Report No. 11 (Ref. 1) lists the committed effective dose equivalent per unit intake by inhalation in sieverts per becquerel in its Table 2.1. These values may be used directly after converting the units from sieverts per becquerel to rem per microcurie (Sv/Bq x 3.7 x 10^6 = rem/μCi).

3.2 Use of Stochastic Inhalation ALIs from 10 CFR Part 20

ALI values have been established for individual radionuclides and are presented in Table 1 in Appendix B to §§ 20.1001–20.2401. The ALI values for inhalation, presented in Column 2 in Table 1, correspond to a committed effective dose equivalent of 5 rems (0.05 Sv) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue, whichever is more limiting. If the ALI value presented in Table 1 is limited by the 50-rem committed dose equivalent, the controlling organ is listed directly below the ALI value, and the stochastic ALI value based on the 5-rem committed effective dose equivalent is listed in parentheses directly below the organ name. If a stochastic ALI is listed in parentheses, that value should be used to calculate the committed effective dose equivalent. The committed effective dose equivalent for each radionuclide may be calculated, using the estimated radionuclide intake, by Equation 1.

$$H_{i,E} = \frac{5\ I_i}{ALI_{i,E}} \qquad \text{Equation 1}$$

where

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

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

$ALI_{i,E}$ = Value of the stochastic inhalation ALI (based on the committed effective dose equivalent) from Column 2 of Table 1 in Appendix B to §§ 20.1001–20.2401 (μCi)

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

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

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

3.3 Use of DACs from 10 CFR Part 20

Committed effective dose equivalent may also be calculated from exposures expressed in terms of DAC-hours. If the DAC in Appendix B to §§ 20.1001–20.2401 for a radionuclide represents a stochastic value (i.e., the corresponding ALI does not have the name of an organ below it), the DAC may be used directly. If Appendix B to §§ 20.1001–20.2401 does not list a stochastic DAC, which will be the case any time there is a stochastic ALI value in parentheses, it is preferred (but not required) that the licensee calculate and use a stochastic DAC. The stochastic DAC can be calculated from the stochastic ALI (the ALI in parentheses) by the following equation:

$$DAC_{stoc,i} = \frac{ALI_{stoc,i}}{2.4 \times 10^9} \qquad \text{Equation 2}$$

where

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

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

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

Then

$$H_{i,E} = \frac{5\ C_i t}{2000\ DAC_{stoc,i}} \qquad \text{Equation 3}$$

where

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

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

t = The duration of the exposure (hours)

2000 = The number of hours in a workyear

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

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

3.4 Use of ICRP Publication 30

The supplements to ICRP Publication 30 (Ref. 2) list "weighted committed dose equivalent to target organs or tissues per intake of unit activity" for inhalation in sieverts per becquerel. The sum of the values given is the committed effective dose equivalent. ICRP Publication 30 (Ref. 2) does not give the sum, but the licensee can easily add the values given to calculate the sum. Then it is only necessary to convert from sieverts per becquerel to rems per microcurie (3.7×10^6 x Sv/Bq = rem/μCi).

3.5 Use of Individual or Material-Specific Information

NRC regulations (10 CFR 20.1204(c)) state that "When specific information on the physical and biochemical properties of the radionuclides taken into the body or the behavior of the material in an individual is known, the licensee may...use that information to calculate the committed effective dose equivalent...." No prior NRC approval is required for using this approach, but records must be kept.

This approach requires the licensee to do considerably more work and to have greater technical expertise than the other approaches. Thus, the approach is unlikely to be attractive to most licensees

for small routine intakes. On the other hand, it might be attractive in the case of accidental large exposures if more accurate information would lead to a better estimate of the actual dose.

When this approach is used, the dose to organs not "significantly irradiated" may be excluded from the calculation (10 CFR 20.1202(b)(3)).

4. CALCULATION OF COMMITTED EFFECTIVE DOSE EQUIVALENT DUE TO INGESTION

There are annual limits on intake (ALIs) for occupational ingestion of radioactive material. Only one ingestion ALI is given for each radionuclide, whereas for inhalation a different ALI was given for each solubility class. Solubility classes are not used for ingestion. The ingestion ALI given for each radionuclide is used for all chemical forms of that radionuclide.

If ingestion has occurred, the methods for determining the committed effective dose equivalent are similar to the methods used for estimating inhalation dose. Four acceptable methods are described here.

Some noble gas radionuclides in Appendix B to §§ 20.1001–20.2401 do not have ingestion ALI values listed because the ingestion pathway does not contribute significantly to the dose. These radionuclides may be excluded from the determination of the internal dose from ingestion.

4.1 Use of Federal Guidance Report No. 11

Federal Guidance Report No. 11 (Ref. 1) lists in its Table 2.2 the committed effective dose equivalent per unit of intake by ingestion in sieverts per becquerel. These values may be used directly after converting the units from sieverts per becquerel to rems per microcurie (by multiplying the Sv/Bq value by 3.7×10^6).

4.2 Use of Stochastic Ingestion ALIs from 10 CFR Part 20

If the amount of ingested radioactive material is known, the stochastic ingestion ALIs from Column 1 of Table 1 in Appendix B to §§ 20.1001–20.2401 may be used. Equation 4 may be used for this determination.

$$H_{i,E} = \frac{5\, I_i}{ALI_{i,E,oral}} \qquad \text{Equation 4}$$

where

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

I_i = Intake of radionuclide i by ingestion during the calendar year (μCi)

$ALI_{i,E,oral}$ = Value of the stochastic ingestion ALI for the committed effective dose equivalent from Column 1 of Table 1 in Appendix B to §§ 20.1001–20.2401 (μCi)

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

4.3 Use of ICRP Publication 30

The supplements to ICRP Publication 30 (Ref. 2) list "weighted committed dose equivalent to target organs or tissues per intake of unit activity" for oral intake in sieverts per becquerel. The sum of the values given is the committed effective dose equivalent. ICRP Publication 30 does not give the sum, but the licensee can easily add the values given to calculate the sum. Then it is only necessary to convert from sieverts per becquerel to rems per microcurie (by multiplying the Sv/Bq value by 3.7×10^6).

4.4 Use of Individual or Material-Specific Information

NRC regulations (10 CFR 20.1204(c)) allow the committed effective dose equivalent to be calculated based on specific information on the physical and biochemical properties of radionuclides taken into the body of a specific worker. The doses due to ingestion can be calculated using the specific information previously described for inhalation.

5. DETERMINATION OF ORGAN-SPECIFIC COMMITTED DOSE EQUIVALENTS

The internal dose component needed for demonstrating compliance with the dose limit specified in 10 CFR 20.1201(a)(1)(ii) is the organ-specific committed dose equivalent. The organ-specific committed dose equivalent is calculated for an individual organ. Tissue weighting factors are not used.

Organ-specific committed dose equivalents need be calculated only if the committed effective dose equivalent exceeds 1 rem or if an overexposure has occurred, because if the committed effective dose equivalent is less than 1 rem and no overexposure has occurred, the 50-rem nonstochastic organ limit cannot be exceeded.

Five acceptable methods to calculate the organ-specific committed dose equivalent are described here.

5.1 Use of Federal Guidance Report No. 11

One method for calculating the organ-specific committed dose equivalent is to use the factors in

Federal Guidance Report No. 11 (Ref. 1). The organ-specific exposure-to-dose conversion factors presented in Table 2.1 (for inhalation) and Table 2.2 (for ingestion) of Federal Guidance Report No. 11 (Ref. 1) provide acceptable data for calculating individual organ doses based on intakes as follows:

$$H_{i,T} = I_i \times DCF_i \times 3.7 \times 10^6 \qquad \text{Equation 5}$$

where

$H_{i,T}$ = Committed dose equivalent to the tissue or organ from radionuclide i (rems)

I_i = Intake of radionuclide i (μCi)

DCF_i = Dose conversion factor for radionuclide i from Table 2.1 or 2.2 in Federal Guidance Report No. 11 (Sv/Bq)

3.7×10^6 = Conversion factor to convert from Sv/Bq to rem/μCi

5.2 Use of Nonstochastic Inhalation ALIs from Part 20

It is possible to calculate organ-specific committed dose equivalents for those radioactive materials for which nonstochastic ALIs are given in 10 CFR Part 20. (Nonstochastic ALIs are those in which the organ is identified underneath the ALI in Appendix B to §§ 20.1001–20.2401.) The equation is:

$$H_{i,T} = \frac{50\ I_i}{ALI_{i,T}} \qquad \text{Equation 6}$$

where

$H_{i,T}$ = Committed dose equivalent to tissue or organ T from radionuclide i (rems)

I_i = Intake of radionuclide i by inhalation during the calendar year (μCi)

$ALI_{i,T}$ = Value of the nonstochastic inhalation ALI for radionuclide i (based on the organ-specific committed dose equivalent) from Column 2 of Table 1 in Appendix B to §§ 20.1001–20.2401 (μCi)

50 = Committed dose equivalent to maximum-exposed organ from inhalation of 2000 DAC-hours (rems)

5.3 Use of DACs from Part 20

If a radionuclide has an ALI based on a nonstochastic dose limit to an organ, the corresponding DAC may be used to calculate the organ-specific committed dose equivalent to the organ with the highest dose using the following equation:

$$H_{i,T} = \frac{50\ C_i t}{2000\ DAC_i} \qquad \text{Equation 7}$$

$H_{i,T}$ = Committed dose equivalent to tissue or organ T from radionuclide i (rems)

C_i = The concentration of the radionuclide i (microcuries/ml)

DAC_i = The nonstochastic DAC for radionuclide i (microcuries/ml)

t = The duration of the exposure (hours)

2000 = The number of hours in the workyear

50 = Committed dose equivalent to maximum-exposed organ from annual intake of 1 ALI or 2000 DAC-hours (rems)

If intakes during the monitoring period are from more than one radionuclide and the organs receiving the highest dose are different from each radionuclide, this method may substantially overestimate the maximum organ dose. In this situation, the licensee may wish to use one of the other methods.

5.4 Use of ICRP Publication 30

The supplements to ICRP Publication 30 (Ref. 2) list "committed dose equivalent in target organs or tissues per intake of unit activity," in sieverts per becquerel, to significantly exposed organs. These values may be used to calculate organ-specific committed dose equivalents after converting the units from Sv/Bq to rem/μCi.

5.5 Use of Individual or Material-Specific Information

NRC regulations (10 CFR 20.1204(c)) state that the committed effective dose equivalent may be calculated based on specific information on the physical and biochemical properties of radionuclides taken into the body. Although not explicitly stated, the organ-specific committed dose equivalent may also be calculated based on specific information.

In general, if specific information is used to calculate the committed effective dose equivalent, it should also be used to calculate the organ-specific dose equivalent so that both dose calculations have the same basis.

6. DOSES FROM INTAKES THROUGH WOUNDS OR ABSORPTION THROUGH SKIN

According to 10 CFR 20.1202(d), the licensee must evaluate and, to the extent practical, account for intakes through wounds or skin absorption. (Dose from tritium absorption through the skin is taken into account in the DAC value in Appendix B to §§ 20.1001–20.2401.) As a practical matter, the intake by skin absorption of airborne radioactive materials usually does not need to be considered because it will be negligible compared to the intake from inhalation. It may be necessary to consider absorption through the skin when solutions containing dissolved radioactive material come in contact with the skin.

7. RECORDING OF INDIVIDUAL MONITORING RESULTS

The requirements for recording individual monitoring results are contained in 10 CFR 20.2106, which requires that the recording be done on NRC Form 5 or equivalent. NRC Form 5 is used to record, on an annual basis, doses received. Thus, for workers who work for the same licensee for the entire year, the monitoring period will normally be January 1 to December 31. The monitoring year may be adjusted as necessary to permit a smooth transition from one monitoring year to another—so long as the year begins and ends within the month of January, the change is made at the beginning of the year, and no day is omitted or duplicated in consecutive years. A copy of NRC Form 5 and instructions for filling it out are contained in Revision 1 to Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Exposure Data."

7.1 Summation of External and Internal Doses

Summation of external and internal doses is required in 10 CFR 20.1202 when both external monitoring and internal monitoring of an individual are required to meet 10 CFR 20.1502(a) and (b). The requirement for summation applies to the occupationally exposed adult and minor and to the embryo/fetus of a declared pregnant woman.

The requirements for summation of external and internal doses specified in 10 CFR 20.1202(a) are not applicable to the shallow-dose equivalent to the skin or extremities or to the eye dose equivalent. Only external dose is considered in evaluating the shallow-dose equivalent to the skin and the extremities and the eye dose equivalent.

Total effective dose equivalent is calculated by summing the external component (deep-dose equivalent) and the internal component (committed effective dose equivalent). Likewise, the total organ dose equivalent is calculated by summing the external component (deep-dose equivalent) and the internal component to the organ or tissue (committed dose equivalent to any organ or tissue).

7.2 Preferred Units

The preferred unit for dose is the "rem." The use of "millirems" on NRC Form 5 is permitted but is discouraged. The preferred unit for intakes is the "microcurie." NRC regulations (10 CFR 20.2101(a)) do not permit the use of the units "sieverts" or "becquerels" on Part 20 records.

7.3 Roundoff of Doses

Licensees should avoid entering doses on NRC Form 5 with more significant figures than justified by the precision of the basic measured values. In general, it is appropriate to enter dose values with two significant figures on NRC Form 5 using the standard rules for roundoff. Thus, a computer-generated calculated dose of "1.726931 rems" should be entered on NRC Form 5 as "1.7 rems." However, licensees should generally carry at least three significant figures in calculations to avoid loss of accuracy due to multiple roundoffs.

In addition, licensees should not enter doses smaller than 0.001 rem on NRC Form 5 because smaller values are insignificant relative to the dose limits. Therefore, a calculated committed effective dose equivalent of "0.006192 rem" should be entered as "0.006 rem," and a value of "0.000291 rem" should be entered as "0 rem."

The rounding recommended in this section is illustrated in the appendix to this guide.

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, 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.

REFERENCES

1. K. F. Eckerman, A. B. Wolbarst, and A. C. B. Richardson, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Environmental Protection Agency, Federal Guidance Report No. 11 (EPA 520/1-8-020), September 1988. This report may be purchased from the National Technical Information Service, Springfield, VA 22161. For information and credit card sales, call (703) 487-4650.

2. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30 (7-volume set, including supplements), Pergamon Press Inc., 1982.

APPENDIX
EXAMPLE OF THE CALCULATION OF OCCUPATIONAL DOSES

This example illustrates the calculation of dose information needed for NRC Form 5, "Occupational Exposure Record for a Monitoring Period." An NRC Form 5 with the data and calculations in this example is provided to illustrate how to fill out the form. In this example, it is assumed that the individual was exposed to external radiation and received an intake by inhalation of five airborne radionuclides.

Deep-Dose Equivalent (Whole Body)

The licensee provided individual monitoring for the deep-dose equivalent (1-cm depth) based on the likelihood of exceeding 0.5 rem deep-dose equivalent. In this example, the sum of the dosimeter reading for the year is assumed to be 1.435 rems of low-LET radiation (gamma), which in the licensee's calculations is rounded to 1.44 rems, maintaining three significant figures for calculational purposes, but entered as 1.4 rems on NRC Form 5.

Eye Dose Equivalent

The licensee provided monitoring for eye dose equivalent because the dose to the eye was likely to exceed 1.5 rems. The total annual dose measured at a depth of 0.3 cm by a dosimeter worn on the trunk was 1.720 rems. The rounded value of 1.7 is entered on NRC Form 5.

Shallow-Dose Equivalent

The shallow-dose equivalent to the skin or extremities must be monitored if the shallow-dose equivalent is likely to exceed 5 rems in the year. In this example, the licensee concluded at the start of the year that the shallow-dose equivalent was not likely to exceed 5 rems, and, therefore, monitoring of the shallow-dose equivalent was not required by 10 CFR 20.1502. Nevertheless, the licensee provided shallow-dose equivalent monitoring because the dosimeter supplier automatically provided a shallow-dose equivalent reading on all badges. The annual monitored total of the shallow-dose equivalent was 1.85 rems, confirming that monitoring of the shallow-dose equivalent was not necessary. The licensee could enter "NR," meaning not required, on NRC Form 5 because monitoring the shallow-dose equivalent was not required by 10 CFR 20.1502. However, in this case the licensee decided, for the sake of completeness, to enter the rounded value of 1.9 rems as the shallow-dose equivalent, whole body column, but he entered "NR" under shallow-dose equivalent to the extremities because no extremity monitoring was required or provided. The licensee also could have entered 1.9 rems on the basis that the extremities received about the same dose as the dosimeters located on the trunk. Either of those entries is acceptable. A value of zero should not be entered if no monitoring was provided. Any numerical value, including zero, should signify a measured or estimated dose.

Radionuclide Intakes

The intake of each radionuclide must be entered separately. The solubility class of each radionuclide must be specified. The intake mode, inhalation (H) in this case, must also be entered. Based on air sampling data, worker stay times, and respirator protection factors when applicable, the licensee calculated the intakes from inhalation (H), which are shown in Table A.1 using this equation:

$$I_i = \frac{C_i\, B\, t}{APF} \qquad \text{Equation A.1}$$

where

I_i = intake from radionuclide i in microcuries

C_i = the concentration of radionuclide i in microcuries/ml

B = the worker's breathing rate of 20,000 ml/min

t = duration of the worker's exposure in minutes

APF = assigned respiratory protection factor, dimensionless

All the data in Table A.1 must be entered on NRC Form 5.

Committed Effective Dose Equivalent

The committed effective dose equivalent from each radionuclide is calculated by using Equation 1. The data used in Equation 1 are shown in Table A.2.

The sum (1.3 rems) in Table A.2 must be entered on NRC Form 5.

Total Effective Dose Equivalent

The total effective dose equivalent is the sum of the deep-dose equivalent and the sum of the committed effective dose equivalent from all radionuclides. In this case, the total effective dose equivalent is 1.44 + 1.30 rems = 2.74 rems, which is rounded to 2.7 rems for entry onto NRC Form 5.

Organ-Specific Committed Dose Equivalent

The organ-specific committed dose equivalents should be calculated because the committed effective

Table A.1
Worker Intakes

Radionuclide	Solubility Class	Intake Mode	Intake (microcuries)
U-238	D	H	0.022
U-235	D	H	0.0031
U-234	D	H	0.060
Cs-137	D	H	1.87
Ce-144	Y	H	2.07

Table A.2
Calculation of Committed Effective Dose Equivalent

Radionuclide and Class	Intake, I_i (microcuries)	$ALI_{i,E}$ (microcuries)	CEDE (rems)
U-238 (D)	0.022	2	0.055
U-235 (D)	0.0031	2	0.008
U-234 (D)	0.060	2	0.15
Cs-137 (D)	1.87	200	0.047
Ce-144 (Y)	2.07	10	1.04
Sum			1.30

dose equivalent exceeds 1 rem. The organ dose factors in Federal Guidance Report No. 11* may be used. The organ dose factors from Table 2.1 of that report are reproduced in Table A.3. The dose factor for the "remainder" listed in Federal Guidance Report No. 11 is not listed here or used to calculate organ-specific committed dose equivalents because it does not represent a dose to a particular individual organ.

To calculate the organ-specific committed dose equivalent, multiply the intake by the organ dose factor and a conversion factor to convert from Sv/Bq to rem/μCi. The equation is:

$$H_{T,i} = I_i \times DCF_{T,i} \times 3.7 \times 10^6 \qquad \text{Equation A.2}$$

where

$H_{i,T}$ = 50-year committed dose to organ or tissue T from radionuclide i, in rems

I_i = the intake of radionuclide i, in microcuries

$DCF_{T,i}$ = the dose conversion factor for organ or tissue T from radionuclide i, in Sv/Bq

The results are shown in Table A.4.

The doses in Table A.4 were calculated using the rounding method described in this guide.

Organ Dose

The organ dose to the most exposed organ is the sum of the deep-dose equivalent and the committed dose equivalent to the organ with the largest dose. In this case, the deep-dose equivalent is 1.44 rems. The lung is the organ with the highest committed dose equivalent (6.22 rems). The organ dose is the sum, 7.66 rems, which is rounded to 7.7 rems and entered on NRC Form 5.

*K.F. Eckerman, A.B. Wolbarst, and A.C.B. Richardson, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Environmental Protection Agency, Federal Guidance Report No. 11 (EPA 520/1-88-020), September 1988. This report may be purchased from the National Technical Information Service, Springfield, VA 22161. For information and credit card sales, call (703) 487-4650.

Table A.3
Organ Dose Factors From Federal Guidance Report No. 11

Radionuclide	Dose Per Unit Intake (Sv/Bq)					
	Gonad	Breast	Lung	R Marrow	B Surface	Thyroid
U-238 (D)	2.23E-8	2.23E-8	2.80E-7	6.58E-7	9.78E-6	2.22E-8
U-235 (D)	2.37E-8	2.38E-8	2.95E-7	6.58E-7	1.01E-5	2.37E-8
U-234 (D)	2.50E-8	2.50E-8	3.18E-7	6.98E-7	1.09E-5	2.50E-8
Cs-137 (D)	8.76E-9	7.84E-9	8.82E-9	8.30E-9	7.94E-9	7.93E-9
Ce-144 (Y)	2.39E-10	3.48E-10	7.91E-7	2.88E-9	4.72E-9	2.92E-10

Table A.4
Calculated Organ-Specific Committed Dose Equivalents

Radionuclide	Intake (μCi)	Organ-Specific Committed Dose Equivalent (rems)					
		Gonad	Breast	Lung	R Marrow	B Surface	Thyroid
U-238(D)	0.022	0.002	0.002	0.023	0.054	0.796	0.002
U-235(D)	0.0031	0	0	0.003	0.008	0.116	0
U-234(D)	0.060	0.006	0.006	0.071	0.155	2.42	0.006
Cs-137(D)	1.87	0.061	0.054	0.061	0.057	0.055	0.055
Ce-144(Y)	2.07	0.002	0.003	6.06	0.022	0.036	0.002
Sum		0.071	0.065	6.22	0.296	3.42	0.065

NRC FORM 5
(6-92)
10 CFR PART 20

U.S. NUCLEAR REGULATORY COMMISSION

OCCUPATIONAL EXPOSURE RECORD FOR A MONITORING PERIOD

APPROVED BY OMB NO. 3150-0006
EXPIRES:

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST MINUTES FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0006), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

1. NAME (LAST, FIRST, MIDDLE INITIAL)	2. IDENTIFICATION NUMBER	3. ID TYPE	4. SEX	5. DATE OF BIRTH
McGuire, Stephen A.	[redacted]	SSN	☒ MALE ☐ FEMALE	[redacted]

6. MONITORING PERIOD	7. LICENSEE NAME	8. LICENSE NUMBER(S)	9A	9B
1-1-94 to 12-31-94	XYZ Corp.	SNM-944	☒ RECORD ☐ ESTIMATE	☒ ROUTINE ☐ PSE

INTAKES

10A. RADIONUCLIDE	10B. CLASS	10C. MODE	10D. INTAKE IN μCi
U-238	D	H	0.022
U-235	D	H	0.0031
U-234	D	H	0.060
Cs-137	D	H	1.87
Ce-144	Y	H	2.07

DOSES (in rem)

DEEP DOSE EQUIVALENT	(DDE)	11 1.4
EYE DOSE EQUIVALENT TO THE LENS OF THE EYE	(LDE)	12 1.7
SHALLOW DOSE EQUIVALENT, WHOLE BODY	(SDE,WB)	13 1.9
SHALLOW DOSE EQUIVALENT, MAX EXTREMITY	(SDE,ME)	14 NR
COMMITTED EFFECTIVE DOSE EQUIVALENT	(CEDE)	15 1.3
COMMITTED DOSE EQUIVALENT, MAXIMALLY EXPOSED ORGAN	(CDE)	16 6.2
TOTAL EFFECTIVE DOSE EQUIVALENT *(BLOCKS 11 + 15)*	*(TEDE)*	17 2.7
TOTAL ORGAN DOSE EQUIVALENT, MAX ORGAN *(BLOCKS 11 + 16)*	*(TODE)*	18. 7.7

19. COMMENTS

Value in Box 18 is not equal to sum of Box 11 plus Box 16 because rounding to two significant figures was not done until the final step.

20. SIGNATURE - LICENSEE

[signature]

21. DATE PREPARED 1-31-95

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., Washington, DC, as an enclosure to Part 20 (56 FR 23360).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001