

# U.S. NUCLEAR REGULATORY COMMISSION

## DRAFT REGULATORY GUIDE DG-8060



### *Proposed Revision 1 to Regulatory Guide 8.34*

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## MONITORING CRITERIA AND METHODS TO CALCULATE OCCUPATIONAL RADIATION DOSES

### A. INTRODUCTION

#### **Purpose**

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) to meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection Against Radiation” (Ref. 1), for monitoring and determining the radiation dose to occupationally exposed individuals.

#### **Applicability**

This RG applies to all NRC licensees (reactor and nonreactor) subject to 10 CFR Part 20.

#### **Applicable Regulations**

- 10 CFR Part 19, “Notices, Instructions, and Reports to Workers: Inspection and Investigation” (Ref. 2), establishes licensee requirements to provide notices, instructions, and reports to individuals. The regulations in 10 CFR Part 19 also establish the rights and responsibilities of the Commission and individuals during interviews compelled by subpoena or any matter within the Commission’s jurisdiction.
  - 10 CFR 19.12, “Instruction to workers,” establishes requirements that all individuals who are likely to receive an occupational dose in excess of 100 millirem in a year to be instructed in radiological safety information.
  - 10 CFR 19.13, “Notifications and reports to individuals,” establishes requirements for reporting individual radiological exposure data to workers.
- 10 CFR Part 20 provides standards for protection against ionizing radiation for occupationally exposed workers and members of the public.

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This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, <http://www.regulations.gov>, by searching for draft regulatory guide DG-8060. Alternatively, comments may be submitted to the Office of Administration, Mailstop: TWFN 7A-06M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC’s public Web site under the Regulatory Guides document collection of the NRC Library at <https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/>. The DG is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>, under Accession No. ML21068A160. The regulatory analysis may be found in ADAMS under Accession No. ML21068A161.

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- 10 CFR 20.1003, “Definitions,” defines terms used in 10 CFR Part 20.
- 10 CFR 20.1007, “Communications,” states that unless otherwise specified, communications or reports concerning the regulations in this part should be addressed to the NRC’s Executive Director for Operations.
- 10 CFR 20.1101, “Radiation protection programs,” establishes requirements to limit radiation exposures to individuals within the specified regulatory dose limits and “as low as is reasonably achievable” (ALARA).
- 10 CFR 20.1201, “Occupational dose limits for adults,” establishes limits on the annual occupational dose to individual adults.
- 10 CFR 20.1202, “Compliance with requirements for summation of external and internal doses,” requires that if both external and internal doses are monitored, compliance with dose limits is achieved by summing the external and internal doses.
- 10 CFR 20.1203, “Determination of external dose from airborne radioactive material,” requires that when determining the dose from airborne radioactive material, the dose must include the contribution from the airborne radioactive material to the deep-dose equivalent (DDE), lens-dose equivalent (LDE), and shallow-dose equivalent (SDE).
- 10 CFR 20.1204, “Determination of internal exposure,” requires that when monitoring internal dose, suitable and timely measurements must be taken of airborne radioactivity concentrations, or quantities of radionuclides in the body or excreta from the body, or combinations of these measurements.
- 10 CFR 20.1206, “Planned special exposures,” authorizes adult workers to receive doses in addition to, and accounted for separately, from the doses received under the limits specified in 10 CFR 20.1201.
- 10 CFR 20.1207, “Occupational dose limits for minors,” establishes occupational dose limits for minors.
- 10 CFR 20.1208, “Dose equivalent to an embryo/fetus,” establishes dose limits to the embryo/fetus of a declared pregnant woman (DPW).
- 10 CFR 20.1501, “General,” requires that radiological surveys be performed that are reasonable under the circumstances to evaluate radiation levels and concentrations, quantities of residual radioactivity, and potential radiological hazards.
- 10 CFR 20.1502, “Conditions requiring individual monitoring of external and internal occupational dose,” requires monitoring of exposures at levels sufficient to demonstrate compliance with the occupational dose limits.
- 10 CFR 20.1701, “Use of process or other engineering controls,” requires the use of process or engineering controls to control the concentration of radioactive material in air.
- 10 CFR 20.1702, “Use of other controls,” requires that when process and engineering controls are not practical to control airborne radioactivity, additional means be used to control airborne radioactive material.

- 10 CFR 20.1703, “Use of individual respiratory protection equipment,” requires that respiratory protection equipment be tested and certified by the National Institute for Occupational Safety and Health.
- 10 CFR 20.1704, “Further restrictions on the use of respiratory protection equipment,” provides that the NRC may impose additional restrictions to ensure that respiratory protection programs are adequate to include limiting the use of respiratory protection equipment instead of the use of process or engineering controls.
- 10 CFR 20.1705, “Application for use of higher assigned protection factors,” requires licensees to obtain authorization from the NRC to use higher respiratory protection factors in excess of those specified in 10 CFR Part 20, Appendix A, “Assigned Protection Factors for Respirators.”
- 10 CFR 20.2103, “Records of surveys,” requires that records of surveys and calibrations be maintained and retained for 3 years.
- 10 CFR 20.2104, “Determination of prior occupational dose,” requires a determination of the dose in the current monitoring year for all persons for whom monitoring is required under 10 CFR 20.1502.
- 10 CFR 20.2106, “Records of individual monitoring results,” requires that dose records be maintained for all individuals for whom monitoring is required under 10 CFR 20.1502, including doses received during planned special exposures, accidents, and emergency conditions.
- 10 CFR 20.2206, “Reports of individual monitoring,” requires certain categories of licensees to submit an annual report of required monitoring results by April 30 of each year to the NRC’s Radiation Exposure Information and Reporting System (REIRS) available at <http://www.reirs.com>.
- 10 CFR 20.2301, “Applications for exemptions,” provides a method for licensees to obtain exemptions from the 10 CFR Part 20 regulations.

### **Related Guidance**

- RG 8.7, “Instructions for Recording and Reporting Occupational Radiation Dose Data” (Ref. 3), describes methods and procedures acceptable for the preparation, retention, and reporting of occupational radiation dose.
- RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program” (Ref. 4), describes methods acceptable for estimating the intake of radionuclides using bioassay measurements.
- RG 8.15, “Acceptable Programs for Respiratory Protection” (Ref. 5), describes an acceptable respiratory protection program and guidance on performing evaluations to optimize the use of respirators to keep dose ALARA.
- RG 8.22, “Bioassay at Uranium Mills” (Ref. 6), provides guidance on acceptable bioassay programs at uranium mills during uranium recovery operations and at uranium conversion facilities.

- RG 8.25, “Air Sampling in the Workplace” (Ref. 7), provides guidance on performing air sampling (not applicable to licensees under 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”).
- RG 8.26, “Applications of Bioassay for Fission and Activation Products” (Ref. 8), provides guidance on performing bioassay measurements.
- RG 8.30, “Health Physics Surveys in Uranium Recovery Facilities” (Ref. 9), provides guidance on acceptable health physics survey methods at uranium recovery facilities.
- RG 8.36, “Radiation Dose to the Embryo/Fetus” (Ref. 10), provides guidance on calculating dose to the embryo/fetus.
- RG 8.38, “Control of Access to High and Very High Radiation Areas in Nuclear Power Plants” (Ref. 11), provides guidance on the controls for access to high and very high radiation areas at nuclear power plants.
- RG 8.40, “Methods for Measuring Effective Dose Equivalent from External Exposure” (Ref. 12), provides guidance for determining the effective dose equivalent (for external exposure) (EDEX).

### **Purpose of Regulatory Guides**

The NRC issues RGs to describe to the public methods that are acceptable to the staff for implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

### **Paperwork Reduction Act**

This RG provides guidance for implementing the mandatory information collections in NRC Form 4, NRC Form 5, and 10 CFR Parts 19 and 20 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0005, 3150-0006, 3150-0044 and 3150-0014 respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to [Infocollects.Resource@nrc.gov](mailto:Infocollects.Resource@nrc.gov), and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0005, 3150-0006, 3150-0044 and 3150-0014), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC 20503; e-mail: [oir\\_submission@omb.eop.gov](mailto:oir_submission@omb.eop.gov).

### **Public Protection Notification**

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

## B. DISCUSSION

### Reason for Revision

RG 8.34, Rev. 1 is revised to provide the following guidance:

- to revise the definition of the total effective dose equivalent (TEDE) as the sum of the effective dose equivalent (for external exposure) (EDEX) and the committed effective dose equivalent (CEDE),
- to provide guidance on performing prospective dose evaluations to determine the need for required monitoring to meet the occupational dose monitoring requirements of 10 CFR 20.1502,
- to provide guidance on monitoring of unplanned, unintended doses when monitoring was not performed,
- to provide guidance on monitoring dose from hot particles or contamination on or near the skin,
- to define the term “dosimetry processing” and explain when there are requirements for processing by an accredited National Voluntary Laboratory Accreditation Program (NVLAP) processor,
- to provide guidance on assessing dose from intakes of radioactive material by wound injuries, and
- to provide guidance on calculating soluble uranium intakes.

### Background

In the 1991 revision of 10 CFR Part 20, the regulation at 10 CFR 20.1201 established revised radiation dose limits for occupationally exposed adults. The 1991 revision also defined the total effective dose equivalent (TEDE) as equal to the sum of the deep-dose equivalent (DDE) and the committed effective dose equivalent (CEDE).

In the 2007 revision of 10 CFR Part 20, the TEDE was redefined as the sum of the effective dose equivalent (for external exposures) (EDEX) and the CEDE. In 10 CFR 20.1201, the annual dose limits for adults are the more limiting of 5 rem (50 mSv) TEDE or 50 rem (500 mSv) total organ dose equivalent (TODE) to any single organ or tissue (other than the lens of the eye). In addition, the annual dose limits are an LDE of 15 rem (150 mSv) and an SDE of 50 rem (500 mSv) to the skin of the whole body or the skin of any extremity.

The TODE limit of 50 rem (500 millisieverts (mSv)) specified in 10 CFR 20.1201(a)(1)(ii) applies to the sum of the DDE and the committed dose equivalent (CDE) to any individual organ or tissue. When monitoring is required for both external and internal doses, the regulations in 10 CFR 20.1202 require the summing of external and internal doses to demonstrate compliance with the dose limits of 10 CFR 20.1201. In addition to the annual dose limits, 10 CFR 20.1201 limits the soluble uranium intake by an individual to 10 milligrams (mg) in a week in consideration of chemical toxicity.

The occupational dose limits for minors in 10 CFR 20.1207 are 10 percent of the dose limit for adults. In 10 CFR 20.1208, a dose limit is established for the embryo/fetus of 0.5 rem (5 mSv) during the entire pregnancy for a declared pregnant woman.

The 10 CFR 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, “Occupational Dose Record for a Monitoring Period,” or equivalent (see RG 8.7) and submitted to the NRC’s “Radiation Exposure Information and Reporting System” (REIRS) Project Manager or via the REIRS web site by April 30 of each following year pursuant to 10 CFR 20.2206(c). The REIRS serves as the central repository for all NRC radiation exposure monitoring records that are recorded and reported to the NRC.

### **Consideration of International Standards**

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports, reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA Safety Requirements and Safety Guides pursuant to the Commission’s “International Policy Statement,” published in the *Federal Register* on July 10, 2014 (Ref. 13), and Management Directive 6.6, “Regulatory Guides” dated May 2, 2016 (Ref. 14).

The following reports in the IAEA Safety Standards Series incorporate similar design and preoperational testing guidelines and are related with the basic safety principles considered in developing this RG:

- IAEA Safety Reports Series No. 114 (1996), “Direct Methods for Measuring Radionuclides in the Body” (Ref. 15), provides information on the establishment and operation of facilities for the in vivo measurement of body activity by direct methods (whole-body counting),
- IAEA Safety Reports Series No. 18 (2000), “Indirect Methods for Assessing Intakes of Radionuclides” (Ref. 16), provides technical advice on the collection and analysis of biological and physical samples (in vitro samples) used to estimate intakes of radionuclides, and
- IAEA Safety Reports Series No. 37 (2004), “Methods for Assessing Occupational Radiation Doses Due to Intakes of Radionuclides” (Ref. 17), contains practical advice on the interpretation of monitoring results and the assessment of committed effective doses to workers, using the standard models of the International Commission on Radiological Protection (ICRP).

The ICRP functions under the auspices of the International Congress of Radiology and is the primary international body providing recommendations for protection against ionizing radiation. The ICRP is an independent, nongovernmental organization created by the 1928 International Congress of Radiology to advance for the public benefit the science of radiological protection.

The ICRP and the International Organization for Standardization (ISO) provide recommendations and guidance on protection against the risks associated with ionizing radiation, from artificial sources widely used in medicine, from general industry, from nuclear enterprises, and from naturally occurring sources. The following publications are relevant to the NRC’s mission to license and regulate the Nation’s civilian use of radioactive materials to protect public health and safety, promote the common defense and security, and protect the environment:

- ICRP Publication 2 (1959), “Report of Committee II on Permissible Dose for Internal Radiation” (Ref. 18), provided the basis for the historical (c. 1960–1991) NRC dose limits for blood-forming organs, gonads, and lenses of the eyes that were based on 3 rem during any consecutive weeks (e.g., per quarter) and 5(N-18) rem annual limits where N is the age in years. Recommendations

were also provided for skin and thyroid gland. ICRP-2 provided maximum permissible concentrations and maximum permissible body burden values, which were previously used prior to 1991 in 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,”

- ICRP Publication 26 (1977), “Recommendations of the International Commission on Radiological Protection” (Ref. 19), provides the current basis for the NRC dose limits. ICRP-26 introduced a distinction between “stochastic” effects and “nonstochastic effects” and the concepts of “detriment,” “dose equivalent,” “dose equivalent commitment,” and “committed dose equivalent.” ICRP-26 provides radiation-induced risk factors and the annual dose limitations system of 5 rem (50 mSv) for limiting stochastic effects with tissue weighting factors that are specified in the current 10 CFR Part 20. ICRP-26 also recommended use of an annual dose limitation system for nonstochastic effects of 50 rem (500 mSv),
- ICRP Publication 30 (1982), “Limits for Intakes of Radionuclides by Workers” (Ref. 20), provides a model of the gastrointestinal tract and replaced the concepts of maximum permissible concentration and maximum permissible body burden with the concepts of “Annual Limits on Intake” (ALI) and “Derived Air Concentrations” (DACs). These ALI and DAC values are in current use in 10 CFR Part 20, Appendix B,
- International Organization for Standardization (ISO), 20553:2006, “Radiation Protection—Monitoring of Workers Occupationally Exposed to a Risk of Internal Contamination with Radioactive Material—First Edition” (Ref. 21), provides guidance on the design of programs to monitor workers exposed to the risk of internal contamination,
- ISO 27048:2011, “Radiation Protection—Dose Assessment for the Monitoring of Workers for Internal Radiation Exposure—First Edition” (Ref. 22), provides procedures and assumptions for interpretation of internal contamination monitoring data and quantification of exposures, and
- ISO 20031:2020, “Radiological Protection—Monitoring and Dosimetry for Internal Exposures Due to Wound Contamination with Radionuclides” (Ref. 23), provides monitoring programs to assess contamination via a wound and methods of performing dose assessment.

## C. STAFF REGULATORY GUIDANCE

### 1 Monitoring Criteria

Table 1 below summarizes the minimum individual monitoring requirements in 10 CFR Part 20. The monitoring requirements apply to both routine and nonroutine exposure scenarios. The monitoring requirements apply separately to each external dose type (i.e., DDE, LDE, SDE to the extremities, and SDE to the skin of the whole body). For external dose monitoring, 10 CFR 20.1502(a) requires the licensee to supply and require the use of individual monitoring devices.

#### 1.1 Monitoring External Radiation Exposure

- For an occupationally exposed adult, external monitoring is required:
  - if the occupational dose is likely to exceed 10 percent of the adult's annual dose limits.
  - for any individual entering a high or very high radiation area (in accordance with 10 CFR 20.1502(a)(4)).
  - for assessment of an unplanned, unintended dose when a monitoring device was not provided, and it is necessary to demonstrate compliance with the occupational dose limits in accordance with 10 CFR 20.1502 (see Section 2.5).
- For an occupationally exposed minor, monitoring is required if exposures are likely to be in excess of 0.1 rem (1 mSv) DDE, in excess of 0.15 rem (1.5 mSv) LDE, or in excess of 0.5 rem (5 mSv) SDE in accordance with 10 CFR 20.1502(a)(2).
- For a DPW, monitoring is required if the DDE is likely to be in excess of 0.1 rem (1 mSv) during the entire pregnancy (in accordance with 10 CFR 20.1502(a)(3)).

#### 1.2 Monitoring Intakes of Radioactive Material

- For an occupationally exposed adult, monitoring the intake and assessment of the CEDE is required:
  - if the intake is likely to exceed 10 percent of any applicable ALI in 1 year, or
  - for assessment of unplanned, unintended dose when it is necessary to demonstrate compliance with the occupational dose limits in accordance with 10 CFR 20.1502 (see Section 2.5).
- For an occupationally exposed minor, monitoring the intake and assessment of the CEDE is required if the CEDE is likely to exceed 0.1 rem (1 mSv) in 1 year.
- For a DPW, monitoring the intake and assessment of the CEDE is required if the CEDE is likely to exceed 0.1 rem (1 mSv) during the entire pregnancy.

Note: Individual monitoring devices are not required for monitoring the intake of radioactive material.



## Table 1. Summary of 10 CFR Part 20 Monitoring Requirements

### External exposure monitoring 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 (5 mSv) TEDE (e.g., 0.4 rem EDEX plus 0.1 rem CEDE)
- 5 rem (50 mSv) TODE
- 1.5 rem (15 mSv) LDE
- 5 rem (50 mSv) SDE to the skin
- 5 rem (50 mSv) SDE to any extremity

For DPW who are likely to receive a dose during the entire gestation period in excess of 0.1 rem (1.0 mSv) DDE.

For investigational radiological surveys following unplanned, unintended exposure events when necessary to demonstrate compliance with the occupational dose limits.

For minors who are likely to receive an annual dose in excess of any of the following (each evaluated separately):

- 0.1 rem (1.0 mSv) DDE
- 0.15 rem (1.5 mSv) LDE
- 0.5 rem (5 mSv) SDE to the skin of the whole body
- 0.5 rem (5 mSv) SDE to any extremity

For all individuals entering a high or a very high radiation area.

Note: The definition of a “high radiation area” requires that the area be accessible to individuals. Accessibility is determined by whether an individual can reasonably occupy the area with a major portion of their whole body, where the term “whole body” is as defined in 10 CFR 20.1003.

### Internal exposure monitoring is required:

For adults likely to receive in 1 year an intake in excess of 10 percent of the applicable ALIs for ingestion and inhalation.

For minors likely to receive in 1 year a CEDE in excess of 0.1 rem (1.0 mSv).

For DPW likely to receive during the entire gestation period a CEDE in excess of 0.1 rem (1.0 mSv).

For investigational radiological surveys following unplanned, unintended exposure events when necessary to demonstrate compliance with the occupational dose limits.

## **2 Determining the Need for Monitoring**

### **2.1 Establishing Categories of Workers for Consideration of the Need for Monitoring**

Licensees should<sup>1</sup> evaluate potential exposure scenarios to determine if annual doses to individuals are likely to exceed monitoring criteria (i.e., by performing a prospective dose evaluation – see Section 2.2 below). 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, routine operational guidelines may be established for categories of workers who will be monitored. For example, criteria or procedures may be established for monitoring based on anticipated area access or work functions.

### **2.2 Prospective Evaluation of Doses Likely to Exceed Monitoring Criteria**

The evaluation should include exposure scenarios from both licensed and unlicensed sources (e.g., x-ray producing machines) under the licensee’s control as described in 10 CFR 20.1502. Unless there is a documented prospective dose evaluation that individual monitoring was not required, the fact that monitoring was provided is considered de facto evidence that the licensee had previously determined the monitoring was required by 10 CFR 20.1502. A retrospective determination that the monitoring results did not exceed the 10 CFR 20.1502 monitoring criteria does not reverse the prior monitoring decision.

A prospective dose evaluation of the likelihood of doses exceeding the monitoring criteria should be based on the potential occupational dose to the individual for the year (or the gestation period in the case of a DPW). A review of anticipated work activities, prior monitoring results, workplace monitoring (e.g., area badge monitoring) and occupancy factors should be considered, as appropriate, in evaluating potential exposures and the likelihood of exceeding monitoring requirements. 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 individually. 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.

For prospective dose evaluations of external exposures, historical surveys of dose rates and estimates of occupancy times may be used to estimate expected external doses. Evaluations of historical dosimetric data may also be considered in projecting likely doses, assuming anticipated work activities are expected to be similar to prior work activities. Credit for beta shielding provided by protective clothing (e.g., rubber gloves) may also be taken in the prospective dose evaluation.

Similarly, for prospective dose evaluations of internal exposures, historical contamination surveys, historical airborne radioactivity measurements, anticipated work activities, expected duration of exposures and predictions of future airborne radionuclide concentrations may be used to estimate likely radionuclide intakes. The planned use of and credit for respiratory protective equipment may be considered in the evaluations of internal exposures, provided use of the equipment complies with the respiratory protection requirements of 10 CFR 20.1703.

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<sup>1</sup> As used in the body of these standards, the term “should” denote a recommendation and the term “may” denotes permission (neither a requirement nor a recommendation).

“Likely exposures” include exposures received during normal situations as well as abnormal situations which can reasonably be expected to occur during the life of the facility, during both normal operations and anticipated operational occurrences, as discussed in Volume 60 of the *Federal Register*, page 36039 (1995).<sup>2</sup> Examples of anticipated operational occurrences include small, unplanned events involving spills of reactor coolant, sudden increases in external radiation levels (loss of shielding), and a loss of control of radioactive materials leading to a localized high airborne radioactivity area. However, licensees under 10 CFR Part 50 and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. 24), do not need to consider design-basis accidents analyzed in the facility’s Final Safety Analysis Report. Precautionary monitoring for unlikely exposures and for potential accident conditions are not required because these events, by definition, are not likely.

### **2.3 Prospective Evaluation of Doses Not Likely to Exceed Monitoring Criteria**

Potential exposure scenarios involving small doses (e.g., work activities in low dose rate areas or brief exposure durations, low-levels of anticipated extremity doses or low-levels of airborne radioactivity) may be evaluated in the prospective dose evaluation and determined as “not likely” to result in doses exceeding monitoring criteria. These anticipated exposure scenarios involving small exposures below the monitoring criteria are not subject to monitoring requirements.

### **2.4 Change in Exposure Conditions**

If the radiation exposure conditions change during the year, the need to provide individual monitoring should be reevaluated. If the licensee determines that, under a new job assignment, a worker’s dose is likely to exceed 10 percent of the annual dose limit, then 10 CFR 20.1502 requires that monitoring be provided. The licensee should estimate, record, and report the prior dose received before monitoring began. These estimates can be based on a combination of work location radiation survey results, monitoring results of other individuals in similar work situations, or other methods to produce a “best estimate” of the actual dose received when monitoring was not provided.

If reevaluation of a monitored individual’s anticipated annual occupational dose indicates that the dose is likely to be below 10 percent of the annual dose limits, monitoring may be terminated. However, the doses measured while required monitoring was provided must be recorded pursuant to 10 CFR 20.2106 because the monitoring was initially provided to meet 10 CFR 20.1502.

### **2.5 Monitoring after an Unplanned, Unintended Exposure Event When Necessary to Demonstrate Compliance with Occupational Dose Limits**

After an unplanned, unintended exposure event, a follow-up investigational survey performed for purposes of demonstrating compliance with occupational dose limits becomes required monitoring pursuant to 10 CFR 20.1502. Therefore, these dose assessment results must be recorded pursuant to 10 CFR 20.2106 and reported pursuant to 10 CFR 19.13 and 10 CFR 20.2206.<sup>3</sup> For example, assume an unplanned, unintended exposure event occurred which may have exceeded occupational dose limits. The follow-up investigational survey is considered required monitoring, and dose assessment results must be recorded in accordance with 10 CFR 20.2106 and reported in accordance with 10 CFR 19.13 and 10 CFR 20.2206.

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<sup>2</sup> Volume 60 of the *Federal Register*, page 36039 (1995): The term “likely to receive” in 10 CFR 20.1502 includes normal situations as well as abnormal situations involving exposure to radiation that can reasonably be expected to occur during the life of the facility, during both normal operations and anticipated operational occurrences (e.g., unplanned exposure events conditions, such as sudden increases in external radiation levels, or localized areas of elevated concentrations of airborne radioactivity) but do not need to consider design-basis accidents.

<sup>3</sup> Memoranda to NRC’s Regional Offices from the Office of Nuclear Reactor Regulation, regarding recording and reporting occupational dose (ADAMS Accession Nos. ML15187A388 and ML16137A098).

However, follow-up investigational surveys are not considered required monitoring when performed to validate the assumptions in the prospective dose evaluation. For example, assume the prospective dose evaluation determined that small unplanned, unintended extremity exposures may occur that are not likely to exceed external occupational dose limits, and/or that minor facial contamination and/or intakes may occur that are not likely to exceed internal occupational dose limits. If an unplanned, unintended exposure event occurred, whereby a worker is identified as having an extremity exposure that is not likely to have exceeded occupational dose limits, then a follow-up dose evaluation is not considered required monitoring. Another example is when a facial contamination occurs with a potential intake. An investigational follow-up whole body count is performed, and a small intake is determined. These scenarios do not involve required monitoring because the prospective dose evaluation determined that these types of unplanned, intended exposures may occur and were not likely to exceed occupational dose limits. Although these scenarios do not involve required monitoring, NRC recommends voluntary recording and reporting of dose assessment results to the worker and to the NRC in accordance with 10 CFR 19.13, 10 CFR 20.2106 and 10 CFR 20.2206..

## **2.6 Voluntary Monitoring**

Voluntary monitoring beyond that required in 10 CFR 20.1502 may be performed. The results of voluntary monitoring obtained when monitoring was not required by 10 CFR 20.1502 are not subject to dose recording requirements. However, to keep the occupationally exposed individual better informed, a licensee should record and report the results of this monitoring or establish criteria for when to record and report voluntary monitoring (see RG 8.7).

## **2.7 Detection Sensitivity**

The monitoring criteria in 10 CFR 20.1502 do not establish required levels of detection sensitivity (e.g., the lower limit of detection). For example, it may not be feasible to confirm intakes of  $\leq 10$  percent of the ALI, particularly for bioassay measurements of some alpha-emitting radionuclides. Therefore, monitoring thresholds should not be considered requirements for the sensitivity of a particular measurement.

## **3 Determination of External Doses**

According to the definitions in 10 CFR 20.1003, the DDE to the whole body is considered to be at a tissue depth of 1 centimeter (cm) (1,000 milligrams per square cm ( $\text{mg}/\text{cm}^2$ ), the SDE to the skin or extremities is to be determined at 0.007 cm ( $7 \text{ mg}/\text{cm}^2$ ), and the LDE is to be determined at 0.3 cm ( $300 \text{ mg}/\text{cm}^2$ ). In evaluating the SDE and LDE, it is acceptable to take credit for the shielding provided by gloves and protective lenses, respectively.

### **3.1 Placement of Individual Monitoring Devices for Measuring DDE**

External dose (DDE) is typically determined using individual monitoring devices, such as passive dosimetry (e.g., thermoluminescence or optically stimulated dosimetry) and/or electronic dosimetry. The assigned DDE must be for the part of the body receiving the highest exposure (10 CFR 20.1201(c)). When the whole body is exposed uniformly, the individual monitoring device is typically worn on the front of the upper torso. When dosimetry is not capable of measurement, external dose from very low energy photons (e.g., Xe-133) may be determined by calculation.

If the radiation field 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 moved near that part of the whole body. For

example, if nonuniform exposure to the head of an individual results in substantially higher dose than the dose to the trunk of the body, a monitoring device should be located on or close to the head.

If post-exposure 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. Significant discrepancies between dosimetry results (e.g., passive and electronic dosimetry) or between dosimetry results and radiation surveys should be investigated, and the actual maximum dose should be assigned as the dose of record.

### **3.2 Use of More Than One Dosimeter for Measuring DDE**

An acceptable alternative approach for highly nonuniform radiation fields is to use more than one dosimeter (i.e., multi-badging) to separately track doses to different parts of the whole body. At the end of the year, doses from each location would be summed. The DDE to be recorded would be that of the dosimeter location receiving the highest dose.

### **3.3 Determination of Effective Dose Equivalent (for External Exposure) (EDEX)**

When the external exposure is determined by measurement with an external personal monitoring device, the DDE must be used in place of the EDEX, unless the EDEX is determined by a dosimetry method approved by the NRC pursuant to 10 CFR 20.1201(c). RG 8.40 describes currently approved dosimetry methods for determining the EDEX; however, licensees may request NRC approval of additional dosimetry methods on a case-by-case basis. For external exposure whole-body dose determinations, licensees may also apply to the NRC to use other weighting factors (see 10 CFR 20.1003, Footnote 2).

### **3.4 Determination of Shallow-Dose Equivalent (SDE)**

When the SDE is being monitored, and the amount of SDE is expected to differ substantially from the DDE, the SDE should be monitored separately. This may occur in specialized exposure situations such as in accelerator operations or medical applications or in other situations as needed.

If the licensee determines in the prospective dose evaluation that SDE is required to be monitored, it may be appropriate to use an SDE dosimeter for some, but not all, radiation exposures. When exposure is uniform, the SDE measured by a torso dosimeter would be representative of the SDE to the extremities or skin, and a separate SDE monitoring dosimeter would not be required. However, during periods of highly nonuniform exposure to the extremities, an extremity dosimeter should be provided. If protective gloves are used, it is acceptable to place the extremity dosimeter under the gloves.

For hot particles or contamination on or near the skin, SDE may be calculated using methods described in NUREG/CR-6918, Revision 4, "VARSKIN+ 1.0, A Computer Code for Skin Contamination and Dosimetry Assessments" issued in 2021 (Ref. 25) or more updated versions.

### **3.5 Determination of Embryo/Fetus Dose**

RG 8.36 contains guidance on calculating doses to the embryo/fetus.

### **3.6 Use of Federal Guidance Reports Nos. 12 and 15 to Determine DDE and EDEX**

Federal Guidance Report (FGR) No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," issued September 1993 (Ref. 26), and FGR No. 15, "External Exposure to Radionuclides in Air,

Water and Soil” issued August 2019 (Ref. 27), are primarily intended for dose assessments for dose assessments from environmental contamination. FGR 12 uses the same organ or tissue weighting factors as specified in 10 CFR 20.1003, which are based on the Federal Register Notice issued by the Environmental Protection Agency (EPA), “Radiation Protection Guidance to Federal Agencies for Occupational Exposure; Approval of Environmental Protection Agency Recommendations,” issued September 1993 (Ref. 28). When the source term and geometry of radionuclides in air submersion, water immersion, contaminated ground surface, or soil contamination at various depths are known, the dose rate coefficients of FGR No. 12 may be used to calculate the organ dose,<sup>4</sup> the EDEX, and the SDE.

FGR 12 and 15 should normally not be used to calculate external occupational dose because the environmental exposure geometry (air submersion, water immersion, or ground contamination is not representative of in-facility exposure geometry. Additionally, the FGR 15 dose coefficients are based on organ or tissue weighting factors based on ICRP Publication 103, “The 2007 Recommendations of the International Commission on Radiological Protection” (Ref. 29), which are different than the 10 CFR 20.1003 organ or tissue weighting factors.

However, upon application by a licensee, on a case-by-case basis, the NRC staff is authorized to grant exemptions (e.g., from 10 CFR Part 20 organ or tissue weighting factors) in accordance with 10 CFR 20.2301. If an exemption request is approved to use alternate dosimetry methods with different organ or tissue weighting factors than given in 10 CFR 20, the EDEX may be calculated using those approved methods.

### **3.7 Dosimeter Processing**

Personnel dosimetry that requires processing must be processed and evaluated by a dosimetry processor holding accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP) pursuant to 10 CFR 20.1501(c). The NRC interprets processing to mean a process, separate from and independent of the design of the dosimeter, that is required to extract dose information from the dosimeter after exposure to radiation. For example, film dosimeters, thermoluminescent dosimeters, and optically stimulated luminescence dosimeters, require processing by qualified technicians using separate equipment to obtain data to compute the dose measurement.

Conversely, dose measurements obtained from electronic dosimeters or digital output personnel dosimeters (e.g., direct ion storage dosimeters) do not require processing since the data is extracted directly from the dosimeter (i.e., through a process independent of dosimeter processing). Therefore, since processing is not required for this type of dosimeter, there is no requirement for processors to have NVLAP accreditation.

## **4 Determination of Internal Dose**

### **4.1 Assessing Intakes**

RG 8.9 provides guidance on determining intakes from bioassay results.

RG 8.25 (not applicable to 10 CFR Part 50 licensees) provides guidance on determining intakes from air sampling measurements.

RG 8.26 provides guidance on when bioassay programs are needed for those individuals subject to internal radiation exposure monitoring requirements.

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<sup>4</sup> The CDE is not applicable because the CDE is defined as resulting from an intake.

## 4.2 Calculation of Committed Effective Dose Equivalent (CEDE) from Inhalation

The internal dose component needed for evaluating the TEDE is the CEDE. The inhalation CEDE is the 50-year effective dose equivalent that results when radioactive material is inhaled, and uptake occurs. The contributions from all occupational intakes for these modes of intake are added over the yearly time period for which the inhalation CEDE is being evaluated.

Most noble gases (except radon) are listed in 10 CFR Part 20, Appendix B, as “submersion” class and do not have inhalation ALI values. For radionuclides listed in the submersion class, the internal dose is negligible compared to the external dose and may be excluded when determining internal dose.

Acceptable methods for calculating CEDE from inhaled radioactive materials are described below. The calculation of CEDE must be based on the organ or tissue weighting factors specified in 10 CFR 20.1003, unless licensees have been granted an exemption from the regulations pursuant to 10 CFR 20.2301.

### 4.2.1 Use of Federal Guidance Report No. 11 to Calculate CEDE from Inhalation

FGR No. 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” issued September 1988 (Ref. 30), uses the same organ or tissue weighting factors as specified in 10 CFR 20.1003, which are based on the EPA radiation protection guidance to Federal agencies for occupational exposure. When the intake by inhalation has been determined, FGR No. 11, Table 2.1 may be used to calculate the CEDE.

### 4.2.2 Use of Stochastic Inhalation ALIs to Calculate CEDE from Inhalation

For individual radionuclides, 10 CFR Part 20, Appendix B, Table 1, Column 2, provides the inhalation ALI values. If only one ALI is listed in Table 1, Column 2, it is the stochastic ALI. If two ALIs are listed, the non-stochastic ALI appears first because it is the more limiting ALI with the most highly exposed organ shown below and the stochastic ALI shown in parenthesis. The stochastic ALIs correspond to an inhalation CEDE of 5 rem (50 mSv) and the non-stochastic ALIs correspond to an inhalation CDE of 50 rem (500 mSv) to an individual organ or tissue.

The CEDE for each radionuclide may be calculated, using the estimated radionuclide intake as shown in Equation 1:

$$H_{i,E} = \frac{5 \cdot I_i}{ALI_{i,stoc}} \quad \text{Equation 1}$$

where:

- $H_{i,E}$  = CEDE from radionuclide  $i$  (rem),  
(Note: Subscript “E” is nomenclature for effective dose.)
- $I_i$  = intake of radionuclide  $i$  by inhalation during the calendar year ( $\mu\text{Ci}$ ),
- $ALI_{i,stoc}$  = stochastic inhalation ALI of radionuclide  $i$  ( $\mu\text{Ci}$ ), and
- 5 = CEDE from intake of 1 inhalation  $ALI_{i,stoc}$  (rem)

If intakes of more than one radionuclide occur, the total CEDE will be the sum of the CEDE for all radionuclides. Note that the ALIs 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 the agency’s prior approval in accordance with 10 CFR 20.1204(c).

### 4.2.3 Use of DACs to Calculate CEDE from Inhalation

CEDE may also be calculated from inhalation exposures expressed in terms of DAC-hours. If the DAC for a radionuclide is a stochastic DAC, the DAC value may be used directly. However, the stochastic DACs are not listed whenever the non-stochastic DAC is listed because it is a more limiting value. However, the stochastic DAC may be calculated based on the stochastic ALI (as listed in the parentheses) using Equation 2:

$$DAC_{i,stoc} = \frac{ALI_{i,stoc}}{2.4E9} \quad \text{Equation 2}$$

where:

$DAC_{i,stoc}$  = stochastic DAC for radionuclide  $i$  ( $\mu\text{Ci}/\text{milliliter (ml)}$ ),  
 $ALI_{i,stoc}$  = stochastic ALI for radionuclide  $i$  ( $\mu\text{Ci}$ ), and  
 $2.4E9$  = volume of air inhaled by a worker in a work year (ml),

and then the CEDE may be determined using Equation 3:

$$H_{i,E} = \frac{5 \cdot C_i \cdot t}{2000 \cdot DAC_{i,stoc}} \quad \text{Equation 3}$$

where:

$H_{i,E}$  = CEDE from radionuclide  $i$  (rem),  
 $C_i$  = the airborne concentration of radionuclide  $i$  to which the worker is exposed ( $\mu\text{Ci}/\text{ml}$ ),  
 $t$  = the duration of the exposure (hours),  
 $2,000$  = the number of hours in a work year, and  
 $5$  = CEDE from annual intake of 2,000 DAC-hours (rem).

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 in accordance with 10 CFR 20.1204(g). These radionuclides may be disregarded if the following conditions are met:

- the concentration of any radionuclide disregarded is less than 10 percent of its DAC,
- the sum of these percentages for all radionuclides disregarded in the mixture does not exceed 30 percent, and
- the total activity of the mixture is used in demonstrating compliance with the dose limits and monitoring requirements.

### 4.2.4 Use of ICRP Publication 30 to Calculate CEDE from Inhalation

The supplements to ICRP-30 list both the “committed dose equivalents” and the “weighted committed dose equivalent to target organs or tissues per intake of unit activity” for inhalation in Sv/Bq. The CEDE is then the sum of the weighted committed dose equivalents over all the organs or tissues.

### 4.2.5 Use of Individual or Material-Specific Information



NRC regulations in 10 CFR 20.1204(c) state the following:

“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 as stated in 10 CFR 20.2106(a)(4), and the correct tissue or organ factors given in 10 CFR 20.1003 must be used.

This approach requires considerably more work and greater technical expertise than the other approaches. Thus, the approach is unlikely to be attractive for small routine intakes. On the other hand, it might be valuable 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 that are not “significantly irradiated”<sup>5</sup> may be excluded from the calculation in accordance with 10 CFR 20.1202(b)(3).

### 4.3 Calculation of CEDE from Ingestion

Appendix B to 10 CFR Part 20 provides ALIs for occupational ingestion of radioactive material. For each chemical formula (e.g., halide, nitrate, oxide, hydroxide, and all other compounds), there is only one ingestion ALI given for each radionuclide compound.

If ingestion has occurred, the methods for determining the CEDE are similar to the methods used for estimating inhalation dose. Acceptable methods are described below. Noble gas radionuclides 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.

#### 4.3.1 Use of Federal Guidance Report No. 11 to Calculate CEDE from Ingestion

FGR No. 11 lists in Table 2.2 the CEDE per unit of intake by ingestion in units of Sv/Bq. These values may be used directly after converting the units from Sv/Bq to rem/ $\mu$ Ci by multiplying the Sv/Bq value by 3.7E6.

#### 4.3.2 Use of Stochastic ALIs to Calculate CEDE from Ingestion

If the intake of ingested radioactive material is known, the stochastic ingestion ALIs may be determined using Equation 4:

$$H_{i,E} = \frac{5 \cdot I_i}{ALI_{i,stoc}} \quad \text{Equation 4}$$

where:

$H_{i,E}$  = CEDE from radionuclide  $i$  (rem),

$I_i$  = intake of radionuclide  $i$  by ingestion during the calendar year ( $\mu$ Ci),

$ALI_{i,stoc}$  = value of the stochastic ingestion ALI for the CEDE from Column 1 of Table 1 in Appendix B to 10 CFR Part 20 ( $\mu$ Ci), and

<sup>5</sup> An organ or tissue is deemed to be significantly irradiated if, for that organ or tissue, the product of the weighting factor,  $w_T$ , and the CDE,  $H_{T,50}$ , per unit intake is greater than 10 percent of the maximum weighted value of  $H_{T,50}$ , (i.e.,  $w_T H_{T,50}$ ) per unit intake for any organ or tissue.

5 = CEDE from an annual intake of 1 ALI (rem).

#### 4.3.3 Use of ICRP Publication 30 to Calculate CEDE from Ingestion

The supplements to ICRP-30 list the “weighted committed dose equivalent to target organs or tissues per intake of unit activity” for oral intake (ingestion) in units of Sv/Bq. The CEDE is then the sum of the weighted committed dose equivalents over all the organs or tissues.

#### 4.3.4 Use of Individual or Material-Specific Information

NRC regulations in 10 CFR 20.1204(c) allow the CEDE 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 methods previously described for inhalation.

#### 4.4 Determination of CDE from Inhalation or Ingestion

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 CDE. The organ-specific CDE is calculated for an individual organ. Tissue weighting factors are not used.

CDE need be calculated only if the CEDE exceeds 1 rem (10 mSv) or if an overexposure has occurred, because if the CEDE is less than 1 rem (10 mSv) and no overexposure has occurred, the 50 rem (500 mSv) nonstochastic organ limit cannot be exceeded.

Acceptable methods to calculate the organ-specific CDE are described below.

##### 4.4.1 Use of Federal Guidance Report No. 11 to Calculate CDE from Inhalation or Ingestion

The organ-specific dose per unit intake conversion factors presented in FGR No. 11, Table 2.1 (for inhalation), and FGR No. 11, Table 2.2 (for ingestion), provide data for calculating individual organ doses using Equation 5:

$$H_{i,T} = I_i \cdot DCF_{i,T} \cdot 3.7E6 \quad \text{Equation 5}$$

where:

- $H_{i,T}$  = CDE to the tissue (T) or organ from radionuclide i (rem),
- $I_i$  = intake of radionuclide i ( $\mu\text{Ci}$ ),
- $DCF_{i,T}$  = dose conversion factor for radionuclide i for organ (T) from Table 2.1 (Inhalation) or Table 2.2 (Ingestion) in FGR No. 11 (Sv/Bq), and
- $3.7E6$  = conversion factor to convert from Sv/Bq to rem/ $\mu\text{Ci}$ .

##### 4.4.2 Use of Nonstochastic ALIs to Calculate CDE from Inhalation or Ingestion

The CDE component of a radionuclide with a nonstochastic ALI can be calculated for those radioactive materials for which the nonstochastic ALI is limiting. The nonstochastic ALIs are those in which the organ is identified under the Inhalation ALI or the Oral Ingestion ALI in 10 CFR 20, Appendix B. The CDE component of nonstochastic radionuclides may be determined using Equation 6:

$$H_{i,T} = \frac{50 \cdot I_i}{ALI_{i,T}} \quad \text{Equation 6}$$

where:

- $H_{i,T}$  = CDE from radionuclide  $i$  to tissue (T) or organ (rem),
- $I_i$  = intake of radionuclide  $i$  during the calendar year ( $\mu\text{Ci}$ ),
- $ALI_{i,T}$  = value of the nonstochastic ALI for radionuclide  $i$  (based on the organ-specific CDE) ( $\mu\text{Ci}$ ), and
- 50 = CDE to maximum-exposed organ (rem).

The CDE for all radionuclides combined is then the sum of the CDE from nonstochastic radionuclides and the CDE from stochastic radionuclides.

#### 4.4.3 Use of DACs to Calculate CDE from Inhalation

If a radionuclide has an ALI based on a nonstochastic dose limit to an organ, the corresponding DAC may be used to calculate the CDE component from the nonstochastic radionuclide using Equation 7:

$$H_{i,T} = \frac{50 \cdot C_i \cdot t}{2000 \cdot DAC_i} \quad \text{Equation 7}$$

where:

- $H_{i,T}$  = CDE from radionuclide  $i$  to tissue (T) or organ (rem),
- $C_i$  = concentration of the radionuclide  $i$  ( $\mu\text{Ci/ml}$ ),
- $DAC_i$  = nonstochastic DAC for radionuclide  $i$  ( $\mu\text{Ci/ml}$ ),
- $t$  = duration of the exposure (hours),
- 2,000 = number of hours in the work year, and
- 50 = CDE to maximum-exposed organ from annual intake of 2,000 DAC-hours (rem).

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, each organ dose may be calculated individually or one of the other methods described in this RG may be used.

The CDE for all radionuclides combined is then the sum of the CDE from nonstochastic radionuclides and the CDE from stochastic radionuclides.

#### 4.4.4 Use of ICRP Publication 30 to Calculate CDE

The supplements to ICRP-30 list the “committed dose equivalent to target organs or tissues per intake of unit activity” (CDE) in units of Sv/Bq to significantly exposed organs for both oral (ingestion) and inhalation.

#### 4.4.5 Use of Individual or Material-Specific Information

NRC regulations in 10 CFR 20.1204(c) state that the CEDE 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 CDE may also be calculated based on specific information on the physical and biochemical properties of radionuclides taken into the body.

In general, if specific information is used to calculate the CEDE, it should also be used to calculate the organ-specific CDE so that both dose calculations have the same basis.

### 4.5 Dose from Intakes Through Wounds

Intakes through wounds must be evaluated and, to the extent practical, accounted for pursuant to 10 CFR 20.1202(d). Wound dose assessments may be performed as appropriate using the Wound Dosimetry Model in the VARSKIN+ 1.0 computer code described in NUREG/CR-6918, Revision 4, “VARSKIN+ 1.0, A Computer Code for Skin Contamination and Dosimetry Assessments,” or more updated versions of NUREG/CR-6918.

Analytical methods and parameters to perform wound dose assessments may be used as described in National Council on Radiation Protection (NCRP) Report No. 156, “Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry, and Treatment” issued in 2015 (Ref. 31), if the metabolic modeling and dosimetry methods are consistent with NRC regulations.

NCRP-156, Chapter 5, “Exposure Assessment and Dosimetry of Radionuclide-Contaminated Wounds,” provides dosimetric methods for quantification of the source term, skin dosimetry methods, and local dose rates in a contaminated wound based on a 10 millimeter diameter sphere (about 0.5 cm<sup>3</sup>) surrounding a point source, with an NCRP recommendation for use of a 1 cm<sup>3</sup> default value in the absence of sufficient information to determine the exact volume of the irradiated tissue. Dose estimates in spheres of various sizes may also be based on calculations by Jeffrey A. Siegel and M.G. Stabin (Ref. 32).

NCRP-156 also provides an extensive list of references to technical literature on wound models and dosimetry. The ISO provides additional information that may be useful in performing dose assessments in its standard ISO 20031:2020.

For whole-body dose assessments, calculations of effective dose and organ-equivalent doses may be based on R.E. Toohey, et al. (Ref. 33 and Ref. 34) for soluble intakes via contaminated wounds. Note that the 10 CFR Part 20 ALIs do not apply to intakes through wounds since the ALIs are based on inhalation and ingestion models.

#### **4.5.1 Dose Limits for Wound Exposures**

The following dose limits apply to wound exposures:

- TEDE - The annual TEDE dose limit is 5 rem (50 mSv).

Note 1: For wound exposures, the EDEX component of TEDE is zero since the EDEX is an external exposure. For assessment of the CEDE component of TEDE, the mobility (solubility) of a radioactive wound intake should be determined by bioassay or other appropriate calculational methods. The CEDE quantity (if present) may be determined based on the systemic uptake.

- TODE - The annual TODE limit is 50 rem (500 mSv) to any individual organ or tissue (e.g., skin or muscle tissue) as stated in 10 CFR 20.1201(a)(1)(ii).

Note 2: The DDE is defined as an external whole-body exposure, therefore, the DDE component of the TODE from wound exposures is zero. Thus, the TODE is determined based on the CDE component to organs or tissues.

Note 3: There is a difference between the “CDE for skin dose and the “SDE.” The SDE is not applicable to skin dose from wound exposures because the SDE defined as an external exposure of the skin. The CDE for “skin dose” is averaged over the mass of the skin organ, whereas the SDE is dose from external exposures to the highest exposed 10-cm<sup>2</sup> area of the skin.

In summary, the dose limits for internal wound injuries are 5 rem (50 mSv) CEDE and 50 rem (500 mSv) CDE (averaged over the mass of the organ, including the skin organ).

#### **4.5.2 Calculating Skin Dose from Wound Exposures**

For purposes of medical evaluation, if the skin is exposed from the radioactive material imbedded in or below the skin surface, dose assessments should be made to the basal layer of the skin (the most sensitive portion of the skin). The NCRP-156 report dosimetry models are based on a surface area of 1-cm<sup>2</sup>. The basal layer of the skin varies between a depth of 7 mg/cm<sup>2</sup> (70 microns (μm) and 500 mg/cm<sup>2</sup> (5,000 μm)) below the skin surface. The 1999 NCRP Report No. 130, “Biological Effects and Exposure Limits for ‘Hot Particles’ (Ref. 35), provides data on the estimated thickness of the epidermal skin for different skin sites.

NCRP-156, Section 5.2.1 describes skin (shallow) dosimetry models. Note: Table 5.1 in NCRP-156 provides shallow-dose coefficients for selected radionuclides on the external skin surface to a 1 cm<sup>2</sup> area. Similarly, ISO 20031:2020, Annex E, also provides external dose coefficients for a 1 cm<sup>2</sup> area for some common radionuclides that may be used for correlation to skin dose assessments from subcutaneous contamination.

#### **4.5.3 Calculating the CDE to Localized Tissue from Wound Exposures**

Muscle tissue is likely to be exposed from a radioactive wound injury. NCRP-156, Section 5.2.2. describes dosimetry methods for penetrating wounds, such as injected radioactive material in the geometry of a line source. For medical assessment purposes, consistent with NCRP-156 recommendations an initial conservative assessment of the dose to a reference volume of 1 cm<sup>3</sup> of soft tissue should be performed to determine the likelihood of deterministic effects at the wound site.

Note: The dose to a small sphere of tissue surrounding a contaminated wound injury is primarily from beta or alpha radiations, with minimal gamma dose because of the range of a gamma photon and its resulting low mass-energy absorption coefficient.

NCRP-156, Table 5.2, provides local dose rates from a wound contaminated with common radionuclides within a 10 millimeter diameter sphere (about 0.5 cm<sup>3</sup>) of tissue surrounding a point source. Medical professionals should evaluate the impact of the localized dose to these tissues to determine the potential for tissue function impairment and to determine whether medical intervention (e.g., surgical removal) is warranted. The decision on medical intervention is ultimately the choice of the wounded individual.

#### **4.5.4 Calculating the CDE to Organs from Wound Exposures**

If appropriate (e.g., when a radioactive source is located adjacent to, or uptake has occurred into, an organ as the result of a systemic uptake), the organ dose should be assessed for those organs defined in 10 CFR 20.1003. Consistent with the recommendations of ICRP Publications 26 and 30, to determine compliance with the dose limits in 10 CFR Part 20, the CDE should be assessed as the dose equivalent received during the 50-year period following the intake. As defined in ICRP-26, the term “dose equivalent” refers to the mean dose equivalent averaged over the entire organ or tissue.

### **4.6 Dose from Absorption of Radionuclides Through the Skin**

Dose from tritium absorption through the skin is included in the DAC value in Appendix B to 10 CFR Part 20. The intake by skin absorption of other airborne radioactive materials usually does not need to be considered because the intake through skin absorption will be negligible compared to the intake from inhalation. Intakes through the skin should be considered when liquid solutions containing dissolved radioactive material come into contact with the skin.

## **5 Other Requirements**

### **5.1 Monitoring Periods**

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. As 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.

### **5.2 Summation of External and Internal Doses (TEDE)**

In 10 CFR 20.1202, the NRC requires summation of external and internal doses when both external monitoring and internal monitoring of an individual are needed to meet 10 CFR 20.1502(a) and (b). The requirement for summation of external and internal doses applies to the occupationally exposed adult and minor and to the embryo/fetus of a DPW. The internal and external doses must be based on the organ or tissue weighting factors as defined in 10 CFR 20.1003, "Definitions" unless otherwise authorized.

The requirements for summation of external and internal doses specified in 10 CFR 20.1202(a) are not applicable to the SDE to the skin or extremities or to the LDE. Only external dose is considered in evaluating the SDE to the skin and the extremities and the LDE. If the licensee is required to monitor both external and internal doses, the TEDE is calculated by summing external and internal doses (i.e., the EDEX and the CEDE). Likewise, the TODE is calculated by summing the DDE and the CDE.

### **5.3 Required Units and Significant Decimal Places**

The required U.S. conventional units are the curie, rad, and rem, including multiples and subdivisions. They must be clearly indicated as units of all quantities on records pursuant to 10 CFR 20.2101(a). NRC regulations (10 CFR 20.2101(b)) permit the use of units from the International System of Units (SI) in parentheses following each of the U.S. conventional units. Dose assessments and intakes should be recorded and reported using three significant decimal places (e.g., 1.435 rem or 0.002  $\mu$ Ci) for consistency with operational detection thresholds and with units of rem and  $\mu$ Ci in NRC Form 4 and Form 5.

### **5.4 Recording and Reporting Occupational Doses**

RG 8.7 describes methods and procedures for the preparation, retention, and reporting of occupational radiation doses, including the use of NRC Form 4, "Cumulative Occupational Dose History," and NRC Form 5.

## **D. IMPLEMENTATION**

The NRC staff may use this RG as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this RG to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, “Backfitting,” 10 CFR 70.76, “Backfitting,” 10 CFR 72.62, “Backfitting,” or 10 CFR 76.76, “Backfitting,” and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in NRC Management Directive 8.4 (Ref. 36). If a licensee believes that the NRC is using this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

## REFERENCES<sup>6</sup>

1. *U.S. Code of Federal Regulations (CFR)*, “Standards for Protection Against Radiation,” Part 20, Chapter 1, Title 10, “Energy.”
2. CFR, “Notices, Instructions, and Reports to Workers: Inspection and Investigations,” Part 19, Chapter 1, Title 10, “Energy.”
3. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 8.7, “Instructions for Recording and Reporting Occupational Radiation Dose Data,” Washington, DC.
4. NRC, RG 8.9, “Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program,” Washington, DC.
5. NRC, RG 8.15, “Acceptable Programs for Respiratory Protection,” Washington, DC.
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8 ICRP documents may be purchased from the publishing organization at <http://www.icrp.org>.

9 Copies of International Organization for Standardization (ISO) documents may be obtained by writing to the International Organization for Standardization, 1, ch. de la Voie-Creuse, CP 56, CH-1211 Geneva 20, Switzerland, Telephone: +41 22 749 01 11, Fax: +41 22 749 09 47, by E-mail at [sales@iso.org](mailto:sales@iso.org), or on-line at the ISO Store Web site: <http://www.iso.org/iso/store.htm>.

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- ICRP Publication 60 (1991), “1990 Recommendations of the International Commission on Radiological Protection,” replaced the “effective dose equivalent” with the term “effective dose” and provides updated tissue weighting factors. The ICRP-60 weighting factors are not consistent with those specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, “Standards for Protection against Radiation,” and should not be used without specific NRC approval.
- ICRP Publication 66 (1993), “Human Respiratory Tract Model for Radiological Protection,” provides a revision of the ICRP Publication 30 lung model using increased knowledge of the anatomy and physiology of the respiratory tract and the deposition, clearance, and biological effects of inhaled radioactive particles.
- ICRP Publication 68 (1995), “Dose Coefficients for Intakes of Radionuclides by Workers” provides updated dose coefficients based on new models of the respiratory tract and metabolism of certain elements. The ICRP-68 biokinetic models describe the distribution, tissue retention, and excretion of individual elements and their radioisotopes following their absorption into blood following inhalation. These new dose coefficients are slightly lower (within a factor of two) than those published in ICRP-30 and should not be used without specific NRC approval.
- ICRP Publication 75 (1997), “General Principles for the Radiation Protection of Workers” provides guidance for the radiological protection of workers during normal and emergency situations.
- ICRP Publication 78 (1997), “Individual Monitoring for Internal Exposure of Workers” provides updated guidance on individual monitoring for intake of radionuclides by workers based on information provided by recently updated ICRP publications.
- ICRP Publication 88 (2002), “Doses to the Embryo and Fetus from Intakes of Radionuclides by the Mother” provides dose coefficients for the embryo and fetus after intake of radionuclides by the mother.
- ICRP Publication 103 (2007), “The 2007 Recommendations of the International Commission on Radiological Protection,” recommends an effective dose limit of 2 rem (20 millisieverts (mSv)) per year, averaged over defined 5-year periods (10 rem (100 mSv) in 5 years), with the further provision that the effective dose should not exceed 5 rem (50 mSv) in any single year. ICRP-103 provides the latest available scientific information on how the biology and physics of radiation exposure affect the calculation of effective dose. ICRP-103 updated the radiation weighting factors and expanded the tissue weighting factors for a wider range of organs.

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<sup>11</sup> ICRP publications can be obtained by subscribing to *Annals of the ICRP* or by purchase of individual issues from SAGE Publications Ltd., 1 Oliver’s Yard - 55 City Road - London EC1Y 1SP – UK or [reprints@sagepub.co.uk](mailto:reprints@sagepub.co.uk) +44 (0)20 7324 8500.

- ICRP Publication 119 (2011), “Compendium of Dose Coefficients based on ICRP Publication 60” provides a compilation of dose coefficients for intakes of radionuclides by workers and members of the public and external dose coefficients for use in occupational radiological protection.
- ICRP Publication 130 (2015), “Occupational Intakes of Radionuclides: Part 1” is the first in a series of reports replacing ICRP Publications 30 and 68 to provide revised dose coefficients for occupational intakes of radionuclides by inhalation and ingestion. The revised dose coefficients were calculated using the Human Alimentary Tract Model (Publication 100) and a revision of the Human Respiratory Tract Model (Publication 66).
- ICRP Publication 134 (2016), “Occupational Intakes of Radionuclides: Part 2” describes the assessment of internal occupational exposure to radionuclides, biokinetic and dosimetric models, methods of individual and workplace monitoring, and general aspects of retrospective dose assessment. Dosimetric data are provided for committed effective dose per unit intake (Sv per Bq intake) for inhalation and ingestion, tables of committed effective dose per content (Sv per Bq measurement) for inhalation, and graphs of retention and excretion data per Bq intake for inhalation of 14 individual elements: hydrogen, carbon, phosphorus, sulphur, calcium, iron, cobalt, zinc, strontium, yttrium, zirconium, niobium, molybdenum, and technetium.
- ICRP Publication 137 (2017), “Occupational Intakes of Radionuclides: Part 3” provides dosimetric data on committed effective dose per intake (Sv per Bq intake) for inhalation and ingestion, tables of committed effective dose per content (Sv per Bq measurement) for inhalation, and graphs of retention and excretion data per Bq intake for the following elements: ruthenium, antimony, tellurium, iodine, cesium, barium, iridium, lead, bismuth, polonium, radon, radium, thorium, and uranium.
- ICRP Publication 141 (2019), “Occupational Intakes of Radionuclides: Part 4” provides dosimetric data on the committed effective dose per intake (Sv per Bq intake) for inhalation and ingestion, tables of committed effective dose per content (Sv per Bq measurement) for inhalation, and graphs of retention and excretion data per Bq intake for inhalation for the following elements: lanthanum, cerium, praseodymium, neodymium, promethium, samarium, europium, gadolinium, terbium, dysprosium, holmium, erbium, thulium, ytterbium, lutetium, actinium, protactinium, neptunium, plutonium, americium, curium, berkelium, californium, einsteinium, and fermium.

## APPENDIX A

### EXAMPLE OF THE CALCULATION OF OCCUPATIONAL DOSES

This example below illustrates the calculation of dose information needed for NRC Form 5, “Occupational Exposure Record for a Monitoring Period.” In this example, it is assumed that the individual was exposed to external radiation and received an intake by inhalation of five airborne radionuclides. Note: On NRC Form 4, “Cumulative Occupational Dose History,” and Form 5, intakes are reported in units of microcuries ( $\mu\text{Ci}$ ) and dose equivalents are reported in units of rem.

#### Effective Dose Equivalent (for External Exposure) (EDEX)

Monitoring was provided for deep-dose equivalent (DDE) and effective dose equivalent (for external exposure) (EDEX) based on the prospective dose evaluation of likely annual occupational dose exceeding a dose in excess of 10 percent (0.5 rem) (50 millisieverts (mSv)) of the total effective dose equivalent (TEDE). The sum of the dosimeter readings for the year is assumed to be the DDE = 1.435 rem (14.35 mSv). The EDEX is assumed to be equal to the DDE since the licensee was not using an EDEX dosimetry method.

#### Lens Dose Equivalent (LDE)

The lens dose equivalent (LDE) was monitored because the dose to the lens of the eye was likely to exceed 1.5 rem (15 mSv). The total annual dose measured at a depth of 0.3 centimeters (cm) by a dosimeter worn on the trunk was 1.720 rem (17.20 mSv).

#### Shallow-Dose Equivalent (SDE)

The shallow-dose equivalent (SDE) to the skin or extremities must be monitored if the SDE is likely to exceed 5 rem in the year, as stated in Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1502(a) and 10 CFR 20.1201(a). In this example, the licensee concluded at the start of the year that the SDE was not likely to exceed 5 rem, and therefore, monitoring of the SDE was not required by 10 CFR 20.1502, “Conditions Requiring Individual Monitoring of External and Internal Occupational Dose.” Nevertheless, the licensee performed voluntary SDE monitoring because the dosimeter supplier automatically provided an SDE reading on all badges. The annual monitored total of the SDE was 1.850 rem, confirming that monitoring of the SDE was not necessary.

#### Radionuclide Intakes

Based on air sampling data, worker exposure duration, whole-body counts, and respirator protection factors when applicable, the intakes (I) from inhalation were calculated (shown in Table A-1) using Equation A.1:

$$I_i = \frac{C_i \cdot BR \cdot t}{APF} \quad \text{Equation A.1}$$

where:

- $I_i$  = intake from radionuclide  $i$  ( $\mu\text{Ci}$ )
- $C_i$  = the concentration of radionuclide  $i$  ( $\mu\text{Ci}/\text{milliliter (ml)}$ ),
- BR = the worker’s breathing rate of 20,000 (ml/minute)
- $t$  = duration of the worker’s exposure (minutes), and
- APF = assigned respiratory protection factor (dimensionless).

**Table A-1. Radionuclide Intakes**

<b>RADIONUCLIDE<sup>12</sup></b>	<b>SOLUBILITY CLASS</b>	<b>INTAKE MODE</b>	<b>INTAKE (μCi)</b>
U-238	D (Day)	Inhalation	0.022
U-235	D	Inhalation	0.003
U-234	D	Inhalation	0.060
Cs-137	D	Inhalation	1.870
Ce-144	Y (Year)	Inhalation	2.070

**Soluble Uranium Intakes**

To demonstrate compliance with the 10 CFR 20.1201(e) limit of 10 mg per week intake of soluble uranium (U) compounds (i.e., solubility class D and W), the licensee may use air sampling data, worker exposure duration, and assigned respiratory protection factors (APF) using Equation A.2 below. For the purposes of 10 CFR 20.1201(e), Class D and W compounds are considered “soluble” uranium compounds and Class Y compounds are considered insoluble.

$$m_{i,U} = \frac{C_i \cdot BR \cdot 2.4}{APF \cdot SA_{i,U}} \quad \text{Equation A.2}$$

where

- $m_{i,U}$  = mass intake of uranium isotope i (mg),
- $SA_{i,U}$  = specific activity of uranium isotope i (Ci/g), and
- 2.4 = a unit conversion constant value based on a 40-hr work week.

An example calculation of the intakes of three uranium isotopes is shown in Table A-2 based on an APF = 100 and assumed weekly-average air concentrations. If bioassay monitoring indicates additional intake occurred by ingestion, the total of the inhalation and ingestion intakes should be compared to the weekly limit.

**Table A-2. Soluble Uranium Intakes**

<b>RADIONUCLIDE</b>	<b>AIR CONCENTRATION (μCi/mL)</b>	<b>SPECIFIC ACTIVITY (Ci/g)</b>	<b>INTAKE MASS (mg)</b>
U-238	8.5E-10	3.4x10 <sup>-7</sup>	1.200
U-235	1.2E-10	2.2x10 <sup>-6</sup>	0.026
U-234	2.3E-09	6.3x10 <sup>-3</sup>	0.00018
Total intake			1.226

<sup>12</sup> Radionuclide abbreviations: uranium (U), cesium (Cs), cerium (Ce).

### Committed Effective Dose Equivalent (CEDE)

The committed effective dose equivalent (CEDE) from each stochastic radionuclide is calculated by using Equation A.3:

$$H_{i,E} = \frac{5 \cdot I_i}{ALI_{i,stoc}} \quad \text{Equation A.3}$$

where:

- $H_{i,E}$  = CEDE from radionuclide  $i$  (rem),
- $I_i$  = intake from radionuclide  $i$  ( $\mu\text{Ci}$ ), and
- $ALI_{i,stoc}$  = stochastic annual limit of intake from radionuclide  $i$  ( $\mu\text{Ci}$ ).

Table A-2 shows the data used in calculating CEDE.

**Table A-3. Calculation of CEDE**

RADIONUCLIDE AND CLASS	INTAKE, $I_i$ ( $\mu\text{Ci}$ )	$ALI_{i,stoc}$ ( $\mu\text{Ci}$ )	CEDE (rem)
U-238 (D)	0.022	2	0.055
U-235 (D)	0.003	2	0.008
U-234 (D)	0.060	2	0.150
Cs-137 (D)	1.87	200	0.047
Ce-144 (Y)	2.07	10	1.035
Sum			1.295

### Total Effective Dose Equivalent (TEDE)

The TEDE is the sum of the EDEX and the CEDE from all radionuclides. In this case, the TEDE is equal to the sum of the EDEX ((1.435 rem (14.35 mSv)) (from the EDEX example above) plus the CEDE ((1.295 rem (12.95 mSv)) (from the example above), such that TEDE = 2.730 rem (27.30 mSv).

### Committed Dose Equivalent (CDE)

The CDE ( $H_T$ ) should be calculated because the CEDE exceeds 1 rem. The CDE dose conversion factors in Federal Guidance Report (FGR) No. 11 may be used. Table A-3 reproduces the CDE dose conversion factors from Table 2.1 of that report. The dose conversion factors for the “remainder” listed in FGR No. 11 is not listed here or used to calculate CDE because the “remainder” does not represent a dose to a particular individual organ.

**Table A-4. Dose Conversion Factors from Federal Guidance Report No. 11**

RADIONUCLIDE	Dose Per Unit Intake (Sv/Bq)					
	GONAD	BREAST	LUNG	RED MARROW	BONE 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

To calculate the CDE, multiply the intake by the organ dose conversion and a unit conversion factor to convert from sievert/becquerel (Sv/Bq) to rem/ $\mu$ Ci using Equation A.3:

$$H_{i,T} = I_i \cdot DCF_{i,T} \cdot 3.7E6 \quad \text{Equation A.4}$$

where:

- $H_{i,T}$  = 50-year committed dose to organ or tissue T from radionuclide i (rem),
- $I_i$  = the intake of radionuclide i ( $\mu$ Ci), and
- $DCF_{i,T}$  = the dose conversion factor from radionuclide i, for organ or tissue T (Sv/Bq).
- 3.7E6 = unit conversion factor from Sv/Bq to rem/ $\mu$ Ci

Table A-5 shows the results.

**Table A-5 Calculated CDE**

RADIONUCLIDE	INTAKE ( $\mu$ Ci)	CDE (rem)					
		GONAD	BREAST	LUNG	RED MARROW	BONE SURFACE	THYROID
U-238(D)	0.022	0.002	0.002	0.023	0.054	0.796	0.002
U-235(D)	0.003	0.000	0.000	0.003	0.007	0.112	0.000
U-234(D)	0.060	0.006	0.006	0.071	0.155	2.420	0.006
Cs-137(D)	1.870	0.061	0.054	0.061	0.057	0.055	0.055
Ce-144(Y)	2.070	0.002	0.003	6.060	0.022	0.036	0.002
Sum		0.071	0.065	6.218	0.295	3.419	0.065

**Total Organ Dose Equivalent (TODE)**

The total organ dose equivalent (TODE) to the most exposed organ is the sum of the DDE and the CDE to the organ with the largest dose. In this case, the DDE is 1.435 rem. The lung is the organ with the highest CDE (6.218 rem). The TODE = 1.435 rem DDE plus 6.218 rem CDE = 7.653 rem.