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MONITORING CRITERIA AND METHODS TO CALCULATE OCCUPATIONAL RADIATION DOSES

A. INTRODUCTION

Purpose

This guide provides methods acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for monitoring the occupational radiation dose to individuals and for calculating occupational radiation doses. The regulatory guide (RG) applies to both reactor and materials licensees under both NRC and Agreement State licenses.

Applicable Rules and Regulations

The regulations established by the NRC in Title 10, "Energy," of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation" (Ref. 1), Section 20.1101, "Radiation Protection Programs," establish requirements for licensees (a) to keep individuals' exposures to radiation below the specified regulatory radiation dose limits and (b) to keep such radiation doses "as low as is reasonably achievable" (ALARA). To demonstrate compliance with the dose limits, licensees must perform surveys and, when appropriate, monitor individuals' radiation exposure and calculate the doses resulting from the exposure.

Also, 10 CFR 20.1201, "Occupational Dose Limits for Adults," establishes radiation dose limits for occupationally exposed individuals. These limits apply to the sum of the dose received from external exposure and the dose from internally deposited radioactive material. Conditions that require individual monitoring of external and internal occupational doses are specified in 10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose." Monitoring the intake of radioactive material and assessing the committed effective dose equivalent (CEDE) (for internal exposures) is required by 10 CFR 20.1502(b). The calculations that licensees are required to perform in order to comply with these regulations were affected by the 2007 revisions of 10 CFR 20.1003 and 10 CFR 50.2 (Ref. 2), both titled

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"Definitions." This revision redefined the "total effective dose equivalent" (TEDE) as the sum of the effective dose equivalent (for external exposures) and the CEDE (for internal exposures).

The following regulatory requirements are also discussed in this guide:

- 10 CFR 20.1007, "Communications"
- 10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations"(Ref. 3)
- 10 CFR 20.1202, "Compliance with Requirements for Summation of External and Internal Doses"
- 10 CFR 20.1204, "Determination of Internal Exposure"
- 10 CFR 20.1206, "Planned Special Exposures"
- 10 CFR 20.1207, "Occupational Dose Limits for Minors"
- 10 CFR 20.1208, "Dose Equivalent to an Embryo/Fetus"
- 10 CFR 20.1501, "General," in 10 CFR 20 Subpart F, "Surveys and Monitoring"
- 10 CFR 20.1703, "Use of Individual Respiratory Protection Equipment"
- 10 CFR 20.2106, "Records of Individual Monitoring Results"
- 10 CFR 20.2206, "Reports of Individual Monitoring"

Related Guidance

The NRC has developed guidance related to calculating occupational doses for monitored individuals and has provided criteria regarding which individuals should be monitored for radiation exposure. Such guidance includes the following:

- RG 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data" (Ref. 4)
- RG 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program" (Ref. 5)
- RG 8.11, "Applications of Bioassay for Uranium" (Ref. 6)
- RG 8.25, Revision 1, "Air Sampling in the Workplace" (Ref. 7)
- RG 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure" (Ref. 8)

- RG 8.35, Revision 1, "Planned Special Exposures" (Ref. 9)
- RG 8.36, "Radiation Dose to the Embryo/Fetus" (Ref. 10)
- RG 8.40, "Methods for Measuring Effective Dose Equivalent from External Exposure" (Ref. 11)

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. RGs are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs 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.

Paperwork Reduction Act

This RG discusses information-collection requirements covered by 10 CFR Part 20 and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," that the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0014 and 3150-0011 respectively. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information-collection request or requirement unless the requesting document displays a currently valid OMB control number.

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B. DISCUSSION

Reason for Revision

This revision of RG 8.34 provides updated regulatory guidance on monitoring criteria and methods of calculating occupational dose based on the revised definition of the TEDE. This RG also provides updated guidance on acceptable methods of:

- Determining the need for monitoring and demonstrating compliance with occupational dose limits.
- Monitoring alpha intakes and determining internal dose from alpha-emitting radionuclides.
- Assessing deep-dose equivalent (DDE) when the measurements of the primary monitoring device (dosimeter) are inconsistent with other radiological measurements (e.g., surveys or electronic dosimeters).
- Assessing intakes and committed dose equivalent (CDE) from wounds.
- Examples of calculational methods to assess intakes and internal doses.

Background

On December 4, 2007, the NRC revised the definition of the TEDE in 10 CFR 20.1003 and 10 CFR 50.2 (as published in the *Federal Register* at 72 FR 68043 (Ref. 12)). The revision subsequently affected the methods of monitoring and calculating occupational radiation doses and demonstrating compliance with the occupational dose limits. Previously, the definition of the TEDE was the sum of the DDE (to account for external exposure) and the CEDE (to account for internal exposure). Under the revised rule 10 CFR 20.1003, the TEDE was redefined by replacing the DDE with the effective dose equivalent-external (EDEX).

Old definition: TEDE = DDE + CEDE

New definition: TEDE = EDEX + CEDE

Regulations in 10 CFR 20.1201(c) require that, when external exposure is determined by measurement with an external personal monitoring device, the DDE for the part of the body receiving the highest exposure be used in place of the effective dose equivalent (i.e., the EDEX) unless the EDEX is determined by a dosimetry method approved by the NRC (see RG 8.40). In uniform radiation fields, the EDEX is normally determined by measuring the DDE and, therefore, the revised TEDE definition has little impact on monitoring methods. However, for exposures in non-uniform radiation fields, the revised TEDE definition provides greater monitoring flexibility and accuracy for licensees in monitoring worker exposures. Under non-uniform conditions, the previous TEDE definition tended to provide dose assessments that were excessively conservative.

Occupational dose limits are applicable during routine operations, planned special exposures, and during emergencies. Doses received during declared nuclear emergencies (including international emergencies) must be included in the determination of annual occupational dose. However, the potential for exceeding a dose limit during a declared emergency should not prevent a licensee from taking necessary actions to protect health and safety.

Occupational Dose Limits for Adults, Minors, and Embryos/Fetuses

For adults, occupational dose limits (except for planned special exposures) are established in 10 CFR 20.1201(a) as follows:

- For protection against stochastic effects, the annual TEDE limit is 5 rem (50 millisieverts (mSv)).
- For protection against nonstochastic effects, the annual total organ dose equivalent (TODE) limit is 50 rem (500 mSv).
- For protection of the lens of the eye, the annual lens dose equivalent (LDE) limit is 15 rem (150 mSv).
- For protection of the skin of the whole body or of the skin of any extremity, the annual shallow-dose equivalent (SDE) limit is 50 rem (500 mSv).

For minors, occupational dose limits are established in 10 CFR 20.1207, "Occupational Dose Limits for Minors," as annual limit at 10 percent of the adult dose limits.

For the embryo/fetus of a declared pregnant woman, a dose equivalent limit during the entire pregnancy is established in 10 CFR 20.1208, "Dose Equivalent to an Embryo/Fetus," as 0.5 rem (5 mSv).

Planned Special Exposures (PSEs)

PSEs are subject to the conditions specified in 10 CFR 20.1206, "Planned Special Exposures" (e.g., exceptional circumstances, specific authorizations, and informing and instructing the worker). RG 8.35, "Planned Special Exposures," provides guidance on conducting PSEs. For dose-accounting purposes, dose received during a PSE is in addition to and accounted for separately from the dose that is limited by 10 CFR 20.1201.

Surveys[1]

Surveys (i.e., evaluations of the radiological conditions and potential hazards) should be conducted as necessary in support of radiological monitoring and calculation of occupational dose. Instruments and equipment used in performing surveys must be calibrated periodically for the type of radiation measured in accordance with 10 CFR 20.1501(c).

When a licensee assigns or permits the use of respiratory protection equipment to limit the intake of radioactive material, 10 CFR 20.1703(c)(2) requires surveys and bioassays, as necessary, to evaluate actual intakes. Indications of an intake could include facial contamination, nasal

1 "Survey" means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation or concentrations or quantities of radioactive material present.

contamination, malfunctioning respiratory protection equipment, loss of engineering controls creating an airborne radioactivity area, and work in unknown or unplanned airborne radioactivity areas.

During operations, licensees should perform airborne radioactivity surveys as required in 10 CFR 20.1502 to characterize the radiological hazards that may be present and, as appropriate, use engineering and respiratory protection equipment to reduce intakes. When it is not practical to use process or engineering controls to reduce the concentrations of airborne radioactivity to values below those that define an airborne radioactivity area, licensees are required under 10 CFR 20.1702(a), to be consistent with keeping the TEDE ALARA, to increase monitoring (e.g., perform air sampling and track Derived Air Concentration (DAC)-hours and bioassay measurements) and to limit intakes by using access controls, limiting exposure times, or having individuals use respiratory protection equipment.

Monitoring at Levels Sufficient To Demonstrate Compliance

Regulations in 10 CFR 20.1502 require monitoring at levels sufficient to demonstrate compliance with the occupational dose limits; therefore, monitoring methods should be reasonably accurate. In addition, licensees may voluntarily issue individual monitoring devices or use calculational methodologies for reasons other than for required personnel monitoring under the requirements in 10 CFR 20.1502 (e.g., to inform individuals of exposure conditions, or to alleviate safety concerns). The results of monitoring that is voluntarily provided but not required by 10 CFR 20.1502 are not subject to the dose recording or reporting requirements in 10 CFR Part 20, Subpart L, "Records," or Subpart M, "Reporting." However, licensees may voluntarily provide these reports to the exposed individual(s) and to the NRC.

Use of Effective DACs

The regulation at 10 CFR 20.1204(e) provides a method for determining internal exposure when the identity and concentration of each radionuclide in a mixture is known. The identities and concentrations of radionuclides may be determined based on representative radiological surveys identifying the specific radionuclides and quantifying their relative mix. Once the relative mix is known, licensees may apply scaling factors applicable to the mixture for use in calculating DACs and tracking DAC-hours as specified in 20.1204(e). This is commonly referred to as "effective DACs" and is applicable to beta/gamma activity, alpha activity, and hard-to-detect radionuclides.

The use of effective DAC values may be needed in operational radiological protection programs to establish airborne radioactivity postings, determining alarm set points for continuous air monitors, determining the need for respiratory protection, estimating internal dose, or determining when bioassay measurements may be needed. When using effective DACs, licensees may disregard those radionuclides in the mixture (based on prior representative surveys) having a concentration less than 10% of the radionuclide's DAC, given that the sum of disregarded radionuclides does not exceed 30% (see 10 CFR 20.1204(g)).

Alpha Monitoring at Nuclear Power Plants

For reactor facilities that have experienced significant fuel defects, alpha contamination may be a radiological hazard requiring specific evaluation. Alpha contamination (when present)

requires specific evaluation because the DAC values for alpha emitting isotopes are generally orders of magnitude more restrictive than DACs for beta-emitting and gamma-emitting isotopes.

Each facility should characterize and update its alpha source term as needed based on the facility's operational history. Alpha source-term characterization should not be based solely on the samples of dry activated waste collected for waste-classification purposes under 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." Loose contamination surveys may not be sufficient to identify fixed alpha contamination that may pose a hazard during abrasive work (e.g., grinding, cutting, or welding). The characterization should determine the extent of the alpha hazard within the facility such as within localized areas.

The extent of the radiological characterization that is needed depends on the relative significance of the alpha source term compared to other radiological contaminants. The characterization may be used to determine the specific alpha radionuclides and to determine their relative concentrations in a mixture. Once the relative concentrations are known, an effective DAC may be determined and used in radiological protection and dose assessment (in lieu of using the most restrictive DAC of any radionuclide in the mixture as required by 10 CFR 20.1204(f)).

The principal transuranic nuclides producing alpha radiological hazards include the isotopes of curium, plutonium, and americium. For historical fuel failures (e.g., ten years have passed since significant fuel failure), the shorter-lived curium-242 will have largely decayed, leaving the longer-lived alpha radionuclides with more restrictive DACs and annual limits on intake (ALI) as the most prevalent hazard. However, investigations of more recent fuel failures are likely to identify curium-242 as the most abundant alpha-emitting nuclide, which has less restrictive DAC and ALI values. Therefore, effective DAC values must be updated as needed to account for the time-dependent (decayed) mix of alpha radionuclides. In addition, consideration should be given to transuranic isotopes which decay by other than alpha emission (e.g., Pu-241).

The extent of radiological protection measures against alpha radionuclides may be determined based on:

- knowledge of the specific alpha radionuclide mix
- knowledge of the solubility/insolubility of the radionuclides
- conservative assumptions about the most restrictive radionuclide in the mixture
- determination of site-specific effective-DAC alpha values

Discrete Radioactive-Particle Monitoring and SDE

A discrete radioactive particle (DRP) is a small (usually microscopic) and highly radioactive particle emitting either only beta or both beta and gamma radiation and having relatively high specific activity. DRPs are primarily an external exposure hazard to the skin, as measured by the SDE.

In 2002, the NRC amended its regulations related to the shallow-dose equivalent/skin-dose limit in 10 CFR Part 20 (at 67 FR 16298 (Ref. 13); see also Regulatory Issue Summary 2002-10, "Revision of the Skin Dose Limit in 10 CFR Part 20" (Ref. 14)). The amended regulations changed the definition and method of calculating SDEs by specifying that the assigned SDE must be the dose averaged over the contiguous 10 cm^2 of skin receiving the highest exposure.

Harmonization with International Standards

The NRC has a goal of harmonizing its guidance (to the extent that this is practical) with international standards. The International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) have issued a significant number of standards, guidance and technical documents, and recommendations addressing good practices in most aspects of radiation protection. The NRC encourages licensees to consult the international documents noted throughout this guide and implement the applicable good practices they contain that are consistent with NRC regulations.

Such documents include the following:

- ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection" (Ref. 15)
- ICRP Publication 30, (7-volume set including supplements), "Limits for Intakes of Radionuclides by Workers" (Ref. 16)
- ICRP Publication 54, "Individual Monitoring for Intakes of Radionuclides by Workers" (Ref. 17)
- ICRP Publication 60, "1990 Recommendations of the International Commission on Radiological Protection" (Ref. 18)
- ICRP Publication 68, "Dose Coefficients for Intakes of Radionuclides for Workers" (Ref. 19)
- ICRP Publication 78, "Individual Monitoring for Internal Exposure of Workers" (Ref. 20)
- ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection" (Ref. 21)

Documents Discussed in Staff Regulatory Guidance

Although this RG uses information, in part, from one or more reports developed by external organizations and other third-party guidance documents, the RG does not endorse these references other than as specified in this RG. These reports and third-party guidance documents may contain references to other reports or third-party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference in NRC regulations as a requirement, licensees and applicants must comply with that requirement in the regulation.

If the secondary reference has been endorsed in an RG as an acceptable approach for meeting an NRC requirement, the reference constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference in NRC regulations nor endorsed in an RG, the secondary reference is neither a legally binding requirement nor a "generic" NRC approval as an acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if it is appropriately justified and consistent with current regulatory practice, in ways consistent with applicable NRC requirements such as those in 10 CFR Part 20.

C. STAFF REGULATORY GUIDANCE

1. Monitoring Criteria

Regulations in 10 CFR 20.1502 require individual monitoring of external and internal occupational dose at levels sufficient[2] to demonstrate compliance with the occupational dose limits. As a minimum, licensees must monitor occupational exposure to radiation from licensed and unlicensed radiation sources[3] under the control of the licensee.

For external occupational exposure, licensees are required to supply and require the use of individual monitoring devices if the external occupational dose:

- for adults, is likely to exceed 10 percent of the occupational dose limits in 10 CFR 20.1201(a);
- for minors, in one year, is likely to exceed a deep-dose equivalent of 0.1 rem (1 mSv), a lens dose equivalent of 0.15 rem (1.5 mSv), or a shallow-dose equivalent to the skin of the whole body or to the skin of the extremities of 0.5 rem (5 mSv); or
- for declared pregnant women, during their entire pregnancy, is likely to exceed a deep-dose equivalent of 0.1 rem (1 mSv), and
- for individuals entering a high or very high radiation area.

For internal occupational exposure, licensees are required to monitor the intake of radioactive material and assess the CEDE by 10 CFR 20.1502(b) if the intake is likely to exceed:

- 10 percent of the applicable annual limit on intake (ALI) for adults;
- 0.1 rem (1 mSv) for minors in one year; or
- 0.1 rem (1 mSv) for declared pregnant women during the entire pregnancy.

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2 Monitoring performed to assess the magnitude of an inadvertent or unplanned exposure (from external radiation or from intakes of radionuclides) is required monitoring per 10 CFR 20.1502 (i.e., required to demonstrate compliance with the dose limits in Part 20) and are subject to the recording requirements in 20.2106(a) and the reporting requirements 20.2206(b).

3 Unlicensed sources are radiation sources not licensed by the NRC or Agreement States; such as products or sources covered by exemptions from licensing requirements (e.g., 10 CFR 30.14, "Exempt Concentrations"; 10 CFR 30.15, "Certain Items Containing Byproduct Material"; 10 CFR 30.18, "Exempt Quantities"; 10 CFR 30.19, "Self-Luminous Products Containing Tritium, Krypton-85, or Promethium-147"; 10 CFR 30.20, "Gas and Aerosol Detectors Containing Byproduct Material"; 10 CFR 30.22, "Certain Industrial Devices"; or 10 CFR 40.13, "Unimportant Quantities of Source Material"), naturally occurring radioactive materials that are not covered by the Atomic Energy Act, radioactive materials possessed by or nuclear facilities operated by another Federal entity such as the U.S. Department of Defense or the U.S. Department of Energy, and machines that produce radiation (such as x-ray radiography machines and x-ray machines used by security staff).

2. Occupational Dose

The definition of occupational dose in 10 CFR 20.1003 includes dose received during the course of employment in which assigned duties involve exposure to radiation or radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or of another person. The definition of occupational dose was changed in 1995 (at 60 FR 36038) (Ref. 22) so that occupational dose applies to workers whose assigned duties involve exposure to radiation, irrespective of their location inside or outside a restricted area. Note: A member of the public does not become an occupationally exposed individual simply as a result of entering a restricted area.

Individuals who receive occupational exposure and are likely to receive more than 100 mrem must be instructed in accordance with 10 CFR 19.12, "Instruction to Workers." See RG 8.29 for further information.

3. Prospective Assessments of the Need for Occupational Dose Monitoring

Licensees must identify those individuals receiving occupational dose, either individually or as a group or category of individuals. Individuals pre-designated by the licensee as receiving occupational dose are subject to the occupational dose limits; otherwise, individuals must be considered as members of the public subject to public dose limits in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

Once occupationally exposed individuals are identified, licensees should perform a prospective assessment to determine whether those individuals are "likely to exceed" the minimum exposure levels specified in 10 CFR 20.1502 (i.e., to determine the need for monitoring of the occupational dose). The potential for unlikely exposures and accident conditions need not be considered because these events, by definition, are unlikely. However, as discussed at 60 FR 36039, the term "likely to receive" includes "normal situations as well as abnormal situations involving exposure to radiation which can reasonably be expected to occur during the life of the facility." Therefore, licensees should consider normal operations and anticipated operational occurrences (e.g., unplanned onsite events, such as sudden increases in external radiation levels, or localized areas of high airborne radioactivity) but would not need to consider design-basis accidents

The prospective assessment determines the type of monitoring required (e.g., external-dose or internal-dose monitoring). In performing a prospective assessment, an evaluation should be performed based on planned work activities and likely exposure conditions. In the prospective assessment, licensees may take credit for the use of engineering controls (e.g., containment, decontamination, ventilation, and filtration). However, if licensees are using respiratory protection equipment to limit the intake of radioactive material, licensees must establishing a respiratory protection program and perform air sampling, surveys, and bioassays to evaluate intakes and estimate dose in accordance with the 10 CFR 20.1703. Prospective assessments should be revised when there are substantial changes to the radiological conditions of personnel exposure (e.g., changes in work activities, airborne concentrations, beta energy spectra, or use of radiation-producing equipment emitting new or different types of energies).

The requirements for monitoring in 10 CFR 20.1502 refer to exposures that might occur at each licensee individually. Doses that have already been received while in the employ of another licensee, or that might be received in the future while in the employ of another licensee or unlicensed entity, are excluded from consideration in a licensee's determination of the need to

monitor an individual. The need for monitoring should be based on the anticipated exposure to licensed or unlicensed sources under the control of a single licensee.

4. Determination of External Doses

a. Determination of the TEDE

Under 10 CFR 20.1202, if a licensee is required to monitor both external dose and internal dose, the licensee must demonstrate compliance with the dose limits by summing external and internal doses (i.e., TEDE = EDEX + CEDE). However, if the licensee is required to monitor only external doses under 10 CFR 20.1502(a) or only internal doses under 10 CFR 20.1502(b), summation is not required to demonstrate compliance with the occupational dose limits. For example, if the internal dose is not monitored, the CEDE can be assumed to be equal to zero and the TEDE is equal to the EDEX. Similarly, if the external dose is not monitored, the EDEX can be assumed to be equal to zero and the TEDE is equal to the CEDE.

b. Determination of the EDEX

The EDEX is determined using one or more combinations of the following methods in accordance with 10 CFR 20.1201(c). These methods are described in RG 8.40 as follows:

1. Measuring the DDE at the most highly exposed part of the whole body with an external personal monitoring device, as required by 10 CFR 20.1201(c), when an NRC method for determining EDEX is not used.

2. Measuring external exposure with one or more external personal monitoring devices and determining EDEX using an NRC-approved method (such as those provided in RG 8.40 or as specifically approved elsewhere by the NRC).

3. Calculating the EDEX based on survey data obtained under 10 CFR 20.1501 or on other radiological data (such as known source activity, dose rates, and exposure times) using scientifically sound technical methods. This might be required (a) under unique exposure situations (e.g., if an individual's body were partially exposed to radiation streaming in a narrow beam geometry), (b) when the individual's monitoring device was not in the region of the highest whole-body exposure (in accordance with 10 CFR 20.1201(c)), or (c) when the results of the individual monitoring are not available (i.e., the monitoring device is damaged or lost).

Note: Within the same monitoring period, a licensee may use a combination of the methods above: A licensee may routinely determine EDEX for the majority of a monitoring period using method 1 above, and then use method 2 or 3 for special exposure situations at other times. The results of the different dosimetry methods must be combined to determine the EDEX for the entire monitoring period.

c. Determination of the Deep-Dose Equivalent (DDE)

The DDE (external exposure of the whole body) is typically measured with a passive primary monitoring device that assesses the dose at a tissue depth of 1 centimeter (cm) (a mass thickness of 1,000 mg/cm^2). The DDE can also be calculated if the appropriate parameters are known (i.e., the radiation source strength, the exposure geometry, and whether full or partial shielding was in place).

An individual monitoring device located at the most highly exposed part of the whole body measuring the DDE is a conservative and (for uniform exposures) a reasonably accurate estimate of the EDEX. However, if the radiation dose is highly non uniform, 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. There are several other NRC-approved methods for determining EDEX provided in RG 8.40.

In many exposure situations, a required monitoring device (e.g., a passive dosimeter) may be voluntarily supplemented with an additional, active dosimeter (e.g., an electronic dosimeter used for work control and daily dose accounting purposes). Due to the differences in dosimeter design and detection technology, and the relative measurement errors associated with each type of dosimeter, there can be valid differences in readings of these two dosimeters for the same exposure, even if the dosimeters are co-located on the monitored individual. Within a reasonable, licensee pre-determined accuracy criteria (depending on dosimeter designs), small differences between measurements can be disregarded and either dosimetry value used as the measured dose (since both results are considered valid and equal within measurement error). However, a significantly higher reading on the voluntary dosimeter may indicate that the required dosimeter was not appropriately placed to measure the highest exposed part of the whole body. Licensees should investigate those cases where a significant discrepancy exists between dosimeters. If the differences cannot be resolved, an assessment must be performed to determine the DDE, LDE, and SDE for the highest exposed part of the whole body, as provided for in 10 CFR 20.1201(c).

d. Determining the LDE

If the LDE is being monitored with a dosimeter, that dosimeter should be calibrated to measure the dose at a tissue depth of 0.3 centimeter (cm) (a mass thickness 300 mg/cm^2). Alternatively, the LDE may be conservatively determined based on SDE measurements at 7 mg/cm^2. In many exposure situations, safety glasses can be worn to minimize exposures to the lens of the eye from low-energy (or poorly penetrating) radiations, potentially eliminating the need for monitoring the LDE.

e. Determination of the SDE

The SDE is defined only for external exposure at a tissue depth of 0.007 cm (a mass thickness of 7 mg/cm^2), and is the dose averaged over the contiguous 10 cm^2 of skin receiving the highest exposure. If the SDE is being measured with a dosimeter, that dosimeter should be calibrated to measure the dose at a tissue depth of 7 mg/cm^2. For skin contamination, the computer code described in NUREG/CR-6918, "VARSKIN: A Computer Code for Skin Contamination Dosimetry" (Ref. 23) may be used to assess the SDE. The SDE may also be determined from analytical calculational methods based on survey data when dosimetry methods are not representative of the actual exposure conditions.

The SDE for exposure to submersion-class radionuclides containing low-energy betas is not readily measurable by direct survey techniques or dosimetry methods and hence may need to be calculated based on air-sample analyses and DAC-hr tracking. This submersion exposure information may be needed for informing workers of radiological exposure conditions (e.g., informing workers of the SDE rates during pre-job briefings) and also to account in dose records for the SDE that might not be adequately measured by dosimeters (e.g., because of the dosimeter's lack of response to a low-energy beta spectrum).

5. Determination of Intakes

For those licensees monitoring internal dose in accordance with 10 CFR 20.1204, a determination must be made of the intake that can occur through inhalation, ingestion, absorption through the skin, or absorption through wounds. The amount of the intake may be assessed from suitable and timely measurements of airborne radionuclides or may be based on bioassay measurements.

The assessment of intake should include not only the readily detected radionuclides but also the hard-to-detect radionuclides if their dose contribution is significant. The activity of hard-to-detect radionuclides may be based on scaling factors that correspond to the amount of readily detected radionuclides. See RG 8.25, "Air Sampling in the Workplace," and Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," for further guidance on determining uptakes and intakes.

Unless respiratory protection is used, the concentration of radionuclides in the intake (i.e., the breathing-zone concentration) is assumed to be equal to the ambient concentration. Therefore, when selecting the air-sample location, one should consider engineered features such as containment, airflow, and filtration to ensure that the air sample is representative of the air breathed.

If respiratory protection is used to limit the intake of radioactive materials, 10 CFR 20.1703(c)(4)(i) requires internal monitoring to be implemented as part of the respiratory protection program. When respiratory protection is provided, the intake is adjusted by dividing the ambient air concentration by the appropriate Assigned Protection Factor (APF) listed in Appendix A, "Assigned Protection Factors for Respirators," to 10 CFR 20. If the ambient air concentration is determined by performing breathing-zone air sampling inside the respiratory protective device (such as with a lapel air sampler inside a loose-fitting supplied air hood or suit), no APF adjustment is made to the ambient air concentration as measured in the breathing-zone air sample.

a. Determining the Intake Based on Air Sampling

Intake (I) based on air-sampling results can be assessed by multiplying the airborne concentration (C) by the breathing rate and the exposure time:

$I = C_{Air\ sample}$ (µCi/ml) * breathing rate (ml/minutes) * exposure time (minutes), where the breathing rate of a "Reference Man" under light working conditions is 2E+4 ml/minute (20 liters/minute).

The intake of radionuclides can also be estimated by "DAC-hour" tracking in which the ambient airborne concentration (expressed as a fraction of the DAC) is multiplied by exposure time (expressed in hours).

If the intake assessment is based on measurements from a lapel air sampler, the intake may be assessed by multiplying the activity on the lapel air sampler by the breathing rate divided by the lapel air sampler's flow rate as follows:

$I = A_{Air\ sample}$ (µCi) * breathing rate/air sampler flow rate (ml/min), where the breathing rate of a "Reference Man" under light working conditions is 2E+4 ml/minute (20 liters/minute).

b. Determining the Intake Based on Bioassay Measurements

The intake can be determined based on initial bioassay measurements of uptakes and on follow-up bioassay measurements to determine the retention/elimination rates (which can also assist in the evaluation of the mode of intake (inhalation or ingestion)). Time and motion conditions may support assessments of intake as well. Guidance on methods of estimating intake based on bioassay measurements of uptake is provided in NUREG/CR-4884, "Interpretation of Bioassay Measurements" (Ref. 24).

Any intake from wounds is generally assessed based on bioassay measurements using a combination of whole body *in vivo* bioassay and handheld instrumentation. The bioassay measurements should determine the location and depth of the injected source so that CDE dose calculations may be made to the most highly exposed 10 cm^2 area of the skin at a depth of 0.007 cm (see Section 6.d below).

Note: The amount of the "intake" may be assessed using newer, updated biokinetic models (e.g., those described in ICRP Publication 60, "1990 Recommendations of the International Commission on Radiological Protection," and ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection"). However, the CEDE must be calculated using the existing 10 CFR 20.1003 organ weighting factors (unless the use of other weighting factors has been specifically approved by the NRC).

c. Determining Intakes of Alpha Emitters

Alpha intakes may be assessed based on gross surface area and/or airborne surveys of the alpha-emitting isotopes present in the work area at the time of exposure. Scaling factors based on beta/gamma activity may be determined and used to assess the identity and relative concentration of alpha isotopes.

Internal doses may also be assessed based on whole-body count data and scaling factors when nominal (e.g., less than 500 mrem CEDE) alpha doses occur. However, when an alpha intake resulting in alpha doses exceeding a nominal quantity is considered likely, excreta sampling or lung counting may be needed to assess intakes and assign dose. When excreta sampling is to be initiated, sampling should begin as soon as possible following detection of the exposure and should continue for a 24-hour period or until at least one sample is collected (following the first void for urine). ANSI N13.39-2001 (R2011), "Design of Internal Dosimetry Programs" (Ref. 25), provides additional guidance on excreta sampling.

6. Determination of Internal Doses

a. Calculation of the Committed Effective Dose Equivalent (CEDE)

The dose quantity for protection against stochastic effects of internal dose is the CEDE; i.e., a 50-year committed effective dose equivalent from intakes occurring during the monitoring period. There are three fundamental methods described below for calculating the CEDE:

- Using dose coefficients[4] from the U.S. Environmental Protection Agency's Federal Guidance Report No. 11 (FGR-11) (Ref. 26).

- Using ALI methods.

- Using DAC-hour methods.

For details about and examples of calculating the CEDE, see Appendix A.

Note: When performing CEDE calculations using the ALI and DAC-hour methods, the ALI and DAC values provided in Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20 must be used unless the licensee has obtained prior NRC approval in accordance with 10 CFR 20.1204(c)(2) to adjust the ALI or DAC values.

b. Calculation of the Committed Dose Equivalent (CDE)

The CDE is the 50-year committed dose equivalent from the intake of radioactive material. For methods and examples of calculating the CDE, see Appendix A. The special case of calculating the CDE from wound intakes is discussed in Section 6.d below.

c. Calculation of the Total Organ Dose Equivalent (TODE)

The dose limit for protection against nonstochastic effects is expressed in terms of the TODE; i.e., the sum of the DDE and the CDE.

$$TODE = DDE + CDE$$

The TODE is determined by adding the DDE (measured at the most highly exposed part of the whole body) to the CDE.

If only internal monitoring is being performed, the TODE is equal to the CDE to the most highly exposed organ (given that the DDE was not monitored and is assumed to be equal to zero). Further details on acceptable methods of calculating the CDE are described in Appendix A.

If both internal and external monitoring are being performed, the licensee must demonstrate that both the 5-rem TEDE and the 50-rem TODE limits are met. One method of demonstrating compliance with the TODE limit is by summing the DDE and the CDE to the most highly exposed organ. Another acceptable method of demonstrating that the TODE limit is met is by keeping the maximum DDE below 5 rem and the CEDE below 1 rem[5]; if this is done, the TODE cannot exceed its 50-rem limit. In this case, the CDE does not need to be determined because

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[4] Note: Federal Guidance Report No. 11 (FGR-11) uses the terminology "dose conversion factors." However, more recent ICRP documents use the terminology "dose coefficients." This regulatory guide is adopting the newer terminology "dose coefficients" (this change in terminology is acceptable because the terminology is not incorporated in the regulations).

[5] The value of 1 rem is based on the most limiting tissue-weighting factor (i.e., the weighting factor for the thyroid tissue is 0.03; therefore, 1 rem divided by thyroid weighting factor of 0.03 results in a CDE of 33.3 rem. A CDE value of 33.3 rem, when added to an assumed 5-rem DDE value, is less than the CDE limit of 50 rem.

compliance was demonstrated by calculation. If the CEDE does exceed 1 rem, the CDE must be determined in order to demonstrate compliance with the dose limits.

d. Doses from Intakes through Wounds

In accordance with 10 CFR 20.1202(d), the licensee shall evaluate and, to the extent practical, account for intakes through wounds.

Regulations in10 CFR 20.1201 also specify two annual dose limits:

- TODE limits (10 CFR 20.1201(a)(1)(ii))—the sum of the DDE and the CDE to any individual organ or tissue other than the lens of the eye being equal to 50 rem (0.5 Sv)—and

- SDE limits (10 CFR 20.1201(a)(2)(ii))—the SDE to the skin of the whole body or skin of any extremity being equal to 50 rem (0.5 Sv).

However, because the SDE is defined only for external exposure, the SDE quantity and its dose limit are not applicable to dose from wound intakes. Therefore, the TODE dose limit becomes the only applicable limit; i.e., a CDE limit of 50 rem to any individual organ, including the skin. Note that in most skin-exposure situations, the skin dose is from external exposure (and therefore the dose to the skin is normally equal to the SDE). However, when the dose to the skin is from a wound, the CDE dose limit applies (not the SDE).

In making the TODE dose calculation (to the skin organ) under 20.1201(a)(1)(ii), the DDE component is zero (because DDE is specifically defined as an external whole-body exposure). As a result, the CDE is determined for the basal layer of the skin at a depth of 0.007 cm below skin surface for the most highly exposed, contiguous 10-cm^2 area.

In summary, the CDE to the skin is the appropriate quantity to be calculated as the integrated dose from the time of injection to the time the source is removed or by the 50-year integration period for committed dose. The CDE is to be determined at a depth of 0.007 centimeters below the surface of the skin, averaged over the most highly exposed 10 cm^2 of the basal layer of the skin. In order to do this calculation, the location (depth) of the source and distance to the basal layer must be determined as an input parameter. The VARSKIN computer code may be used in performing the CDE skin-dose calculations.

Bioassay measurements should be performed to determine whether there is a systemic uptake from the injected radioactive material. For wound intakes with systemic uptakes, an evaluation must be performed of the CEDE and TEDE. Additional information on assessing intakes through wounds is available in ICRP-54, ICRP-78, NCRP-87 (Ref. 27), and technical articles by Toohey (Ref. 28) and Ishigure (Ref. 29).

Note: With respect to tissue dose, there is no regulatory limit for small-volume localized tissue dose. However, licensees should estimate the committed dose to small volumes of underlying tissues (e.g., 1 cm3) at the wound site for purposes of determining the potential for tissue impairment and whether medical intervention is warranted (e.g., surgical removal). The guidance in National Council on Radiation Protection & Measurements (NCRP) Report No. 156, "Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry, and Treatment" (Ref. 30), is acceptable for this evaluation.

e. Calculating the CDE and CEDE for Inhalation, Submersion and Absorption

A number of methods are acceptable for calculating the CDE and CEDE from the intake of radioactive materials. Some of these methods are described below. However, calculations of the CEDE must be based on organ weighting factors and tissues specified in 10 CFR Part 20. The dose coefficients based on ICRP Publication 60 cannot be used unless specifically approved by the NRC, because ICRP 60 and ICRP 103 tissues and weighting factors are different from those in 10 CFR Part 20.

7. Use of Individual or Material-Specific Information

The regulation in 10 CFR 20.1204(c) states 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 [...]." Prior NRC approval is not required, but detailed records must be kept to demonstrate the acceptability of the dose assessment.

The characteristics most amenable to such individual or site-specific consideration are the activity median aerodynamic diameter (AMAD) of the inhaled aerosol and the solubility (or insolubility) of the material in the lungs and in the gastrointestinal (GI) tract (particularly for alpha intakes). The use of specific information on the physical and biochemical properties to calculate the CEDE requires the licensee to do considerably more work and to have greater technical expertise than the other methods, so this method might not be useful for small infrequent intakes. Conversely, the use of specific information on the physical and biochemical properties of radionuclides taken into the body might be appropriate in the cases of accidental large exposures if more accurate information would lead to a better estimate of the actual dose.

8. Limitation on Uranium Intake

In accordance with 10 CFR 20.1201(e), in addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to 10 mg in a week, in consideration of its chemical toxicity. RG 8.11, "Applications of Bioassay for Uranium," describes methods acceptable for the design of bioassay programs for protection against intake of uranium, conditions under which bioassay is necessary, minimum quantifiable values for direct and indirect bioassay measurements, protection guidelines, and objectives.

9. 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 in clear and legible records containing all the information required by NRC Form 5. Regulatory Guide 8.7 provides further guidance for recording and reporting occupational radiation dose data.

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 round-off. 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 round-offs.

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."

D. IMPLEMENTATION

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

Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative complies with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged**.**

REFERENCES[6]

1. *U.S. Code of Federal Regulations* (CFR), "Standards for Protection against Radiation," Part 20, Chapter I, Title 10, "Energy."

2. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

3. 10 CFR 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," Part 19, Chapter I, Title 10, "Energy."

4. U.S. Nuclear Regulatory Commission (NRC), "Instructions for Recording and Reporting Occupational Radiation Exposure Data," RG 8.7, Revision 2, November 2005, Agencywide Documents Access and Management System (ADAMS) Accession No. ML052970092.

5. NRC, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," RG 8.9, Revision 1, July 1993, ADAMS Accession No. ML003739554.

6. NRC, "Applications of Bioassay for Uranium," RG 8.11, June 1974, ADAMS Accession No. ML003739450.

7. NRC, "Air Sampling in the Workplace," RG 8.25, Revision 1, June 1992, ADAMS Accession No. ML003736916.

8. NRC, "Instruction Concerning Risks from Occupational Radiation Exposure," RG 8.29, Revision 1, February 1996, ADAMS Accession No. ML003739438.

9. NRC, "Planned Special Exposures," RG 8.35, Revision 1, August 2010, ADAMS Accession No. ML101370008.

10. NRC, "Radiation Dose to the Embryo/Fetus," RG 8.36, July 1992, ADAMS Accession No. ML003739548.

11. NRC, "Methods for Measuring Effective Dose Equivalent from External Exposure," RG 8.40, July 2010, ADAMS Accession No. ML100610534.

12. NRC, "Occupational Dose Records, Labeling Containers, and the Total Effective Dose Equivalent," *Federal Register*, Vol. 72, No. 232, December 4, 2007, pp. 68043-68059 (72 FR 68043).[7]

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[6] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

[7] Printed copies of *Federal Register* notices are available for a fee from the U.S. Government Printing Office, 732 N. Capitol Street NW, Washington, DC 20401, telephone (866) 521-1800, or they may be downloaded for free from the Government Printing Office Web site, http://www.gpo.gov/fdsys/.

13. NRC, "Revision of the Skin Dose Limit," *Federal Register*, Vol. 67, No. 66, April 5, 2002, pp. 16298-16301 (67 FR 16298).

14. NRC, "Revision of the Skin Dose Limit in 10 CFR Part 20," Regulatory Issue Summary 2002-2010, July 9 2002, ADAMS Accession No. ML021860332.

15. International Commission on Radiological Protection (ICRP), "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, Oxford, UK: Pergamon Press, 1977.

16. ICRP, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30 (7-volume set including supplements), Oxford, UK: Pergamon Press, 1982.

17. ICRP, "Individual Monitoring for Intakes of Radionuclides by Workers," ICRP Publication 54, Oxford, UK: Pergamon Press, 1989, specifically Sections 4.2 and 4.3.

18. ICRP, "1990 Recommendations of the International Commission on Radiological Protection," ICRP Publication 60, Oxford, UK: Pergamon Press, 1990.

19. ICRP, "Dose Coefficients for Intakes of Radionuclides for Workers," ICRP Publication 68, Oxford, UK: Pergamon Press, 1994.

20. ICRP, "Individual Monitoring for Internal Exposure of Workers," ICRP Publication 78, Oxford, UK: Pergamon Press, 1997, specifically Section 4.2.

21. ICRP, "The 2007 Recommendations of the International Commission on Radiological Protection," ICRP Publication 103, Oxford, UK: Pergamon Press, 2007.

22. NRC, "Radiation Protection Requirements: Amended Definitions and Criteria," *Federal Register*, Vol. 60, No. 134, July 13, 1995, pp. 36038-36043 (60 FR 36038).

23. NRC, "VARSKIN 5: A Computer Code for Skin Contamination Dosimetry," NUREG/CR-6918, Rev. 2, July 2014, Accession No. ML14204A361.

24. NRC, "Interpretation of Bioassay Measurements," NUREG/CR-4884, June 1990, ADAMS Accession No. ML11285A018.

25. American National Standards Institute (ANSI), "Design of Internal Dosimetry Programs," ANSI N13.39-2001 (R2011), Washington, DC, 2011.

26. Eckerman, K.F., 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," Federal Guidance Report No. 11 (EPA 520/1-8-020), U.S. Environmental Protection Agency, Washington, DC, 1988.

27. National Council on Radiation Protection & Measurements (NCRP), "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition," NCRP Report No. 87, Bethesda, MD, March 1987, specifically subsections 5.3.1, 5.3.2, and 5.4.6.

28. Toohey, R.E., et al., "Dose Coefficients for Intakes of Radionuclides via Contaminated Wounds," *Health Physics* 100(5):508-14, May 2011; a much larger and revised version (Ver. 2, August 2014) is available from the Oak Ridge Institute for Science and Education

at http://orise.orau.gov/reacts/resources/retention-intake-publication.aspx (accessed October 10, 2014).

29. Ishigure, N., "Implementation of the NCRP Wound Model for Interpretation of Bioassay Data for Intake of Radionuclides Through Contaminated Wounds," *Journal of Radiation Research* 50(3):267-76, May 2009.

30. NCRP, "Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry, and Treatment," NCRP Report No. 156, Bethesda, MD, 2007.

31. Oak Ridge National Laboratory, ORNL/TM-13188, "Recommended ALIs and DACs for 10 CFR 20: A Consistent Numerical Set" (1996), ADAMS Accession No. ML14322A420.

Appendix A
Methods of Calculating Internal Dose

1. Calculations of the CDE and the CEDE Based on Bioassay Measurements Using Federal Guidance Report No. 11 (FGR-11)

This method is based on using tabulated dose coefficients to calculate the dose. FGR-11 provides tables of dose coefficients (DCs) (FGR-11 uses the terminology "dose conversion factors") for intakes by inhalation and by ingestion (see excerpt below for inhalation of cobalt-60 (Co-60)). FGR-11 provides two types of DCs:

(1) DCs for the CDE to an organ or tissue per unit of activity (DC_{organ}) (e.g., the heading "Lung" below) and

(2) DCs for the CEDE per unit of activity ($DC_{effective}$) (as shown in the far right column of the tables under the heading "Effective").

Excerpt from Federal Guidance Report No. 11:

Table 2.1, Inhalation, Cont'd

Committed Dose Equivalent per Unit intake (Sv/Bq)

Nuclide	Class/f_1	Gonad	Breast	Lung	R Marrow	B Surface	Thyroid	Remainder	Effective
Co-60	W 5 10^{-2}	4.05 10^{-9}	4.16 10^{-9}	3.57 10^{-8}	4.25 10^{-9}	3.54 10^{-9}	3.72 10^{-9}	7.65 10^{-9}	8.94 10^{-9}
	Y 5 10^{-2}	4.76 10^{-9}	1.84 10^{-8}	3.45 10^{-7}	1.72 10^{-8}	1.35 10^{-8}	1.62 10^{-8}	3.60 10^{-8}	5.91 10^{-8}

If site-specific information is known about the type of compound and its clearance class, the appropriate clearance class can be selected. If not, the class is normally selected based on the most conservative class; in Example 1, the DC for the lung is selected from clearance Class Y, which has a value of 3.45E-7). Multiplying the DCs by the intake (I) for that radionuclide yields the CDE and CEDE for that radionuclide.

$$\text{CDE (rem)} = DC_{organ}\ (\text{rem}/\mu\text{Ci [rem per millicurie]}) * I\ (\mu\text{Ci})$$
$$\text{CEDE (rem)} = DC_{effective}\ (\text{rem}/\mu\text{Ci}) * I\ (\mu\text{Ci})$$

Example 1: Calculations of the CDE and the CEDE for Co-60, based on bioassay measurements using the DCs from FGR-11. Note: The DCs in FGR-11 are tabulated in Sieverts per Becquerel (Sv/Bq) and may be converted to millirem per microcurie (mrem/μCi) by multiplying by 3.7E+9.

An intake by inhalation was estimated by a whole body count to be 360 nanocuries (nCi) (0.36 µCi) of Co-60 as a Class Y aerosol. Calculate the CDE to the lung and the CEDE.

From Table 2.1 of FGR-11 (see excerpt below), the DCs for the Class Y Co-60 radionuclide are 3.45E-7 Sv/Bq for the CDE and 5.91E-8 Sv/Bq for the CEDE.

Table 2.1, Inhalation, Cont'd

Committed Dose Equivalent per Unit intake (Sv/Bq)

Nuclide	Class/f_1	Gonad	Breast	Lung	R Marrow	B Surface	Thyroid	Remainder	Effective
Co-60	W 5 10^{-2}	4.05 10^{-9}	4.16 10^{-9}	3.57 10^{-8}	4.25 10^{-9}	3.54 10^{-9}	3.72 10^{-9}	7.65 10^{-9}	8.94 10^{-9}
	Y 5 10^{-2}	4.76 10^{-9}	1.84 10^{-8}	3.45 10^{-7}	1.72 10^{-8}	1.35 10^{-8}	1.62 10^{-8}	3.60 10^{-8}	5.91 10^{-8}

DC_{lung} = (3.45E-7 Sv/Bq) * (3.7E+9) = 1277 mrem/µCi
$DC_{effective}$ = (5.91E-8 Sv/Bq) * (3.7E+9) = 219 mrem/µCi

The doses are calculated by multiplying these DCs by the intake of 0.36 µCi:

CDE_{lung} = (1277 mrem/µCi) * (0.36 µCi) = 460 mrem
CEDE = (219 mrem/µCi) * (0.36 µCi) = 79 mrem

2. Calculation of the CEDE based on Bioassay Measurements using Stochastic ALIs

The ALI values are listed in Table 1 of 10 CFR 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations for Release to Sewerage." Column 1 lists the values for oral ingestion and Column 2 lists the values for inhalation. The stochastic ALI values can be used in the calculation of the CEDE, which is based on the fraction of the allowable annual intake and the 5-rem (50-millisievert (mSv)) CEDE dose limit. When the ALI is defined by the stochastic limit, this value alone is given in the table.

Atomic No.	Radionuclide	Class	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion ALI (µCi)	Inhalation ALI (µCi)	Inhalation DAC (µCi/ml)	Air (µCi/ml)	Water (µCi/ml)	Monthly Average Concentration (µCi/ml)
27	Cobalt-60	W, see ^{55}Co	5E+2	2E+2	7E-8	2E-10	3E-6	3E-5
		Y, see ^{55}Co	2E+2	3E+1	1E-8	5E-11	-	-

Because the stochastic ALI corresponds to a 5-rem (50-mSv) CEDE dose limit, the CEDE may be calculated based on the ratio of the intake to the stochastic ALI multiplied by 5 rem (50 mSv):

$$CEDE = (I/ALI) * 5 \text{ rem}$$

Example 2: Calculate the CEDE based on bioassay measurements using the stochastic ALI.

The intake by inhalation for a worker was estimated by bioassay to be 360 nCi (0.36 µCi) of Co-60 as a Class Y aerosol. Calculate the CEDE.

From Appendix B above, Table 1, Column 2, the ALI for Class Y Co-60 is:

ALI (stochastic) = 30 µCi
CEDE = (I/ALI) * 5 rem
CEDE = (0.36 µCi/30 µCi) * 5 rem = 0.06 rem = 60 mrem

Note: Doses calculated based on FGR-11 methods are generally more precise than doses calculated based on ALI values, because ALI values are given to only one significant figure. Additionally, the precision of the ALI values is limited by the calculational technique used in ICRP-30 (Section 4.7) whereby target organs that are not significantly irradiated were excluded (<10% rule), as well as dose from source organs contributing less than 1% were also excluded. For further information, see Oak Ridge National Laboratory, ORNL/TM-13188, "Recommended ALIs and DACs for 10 CFR 20: A Consistent Numerical Set" (Ref. 31).

For Co-60, a 60-mrem value based on an ALI calculation compares to a calculated CEDE value of 79 mrem using the FGR-11 method as determined in Example 1 above. For other radionuclides such as Co-58, the differences might be larger. However, either calculational method and/or result is acceptable in demonstrating compliance with regulatory limits.

3. Calculation of the CDE Based on Bioassay Measurements Using Nonstochastic ALI

The 10 CFR 20 Appendix B , Table 1, Column 2, nonstochastic ALI values can be used in the calculation of the CDE, based on the fraction of the allowable annual intake and the 50-rem (500-mSv) CDE dose limit. When the ALI is defined by the nonstochastic limit, this value is listed first in the table with its corresponding organ (see excerpt below), and the corresponding stochastic ALI are given in parentheses (e.g., 9E+1 µCi (90 µCi) for ingestion and 2E+2 µCi (200 µCi) for inhalation in the excerpt below).

Iodine-131

Atomic No.	Radionuclide	Class	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion ALI (µCi)	Inhalation				
				ALI (µCi)	DAC (µCi/ml)	Air (µCi/ml)	Water (µCi/ml)	Monthly Average Concentration (µCi/ml)
53	Iodine-131	D, all compounds	3E+1 Thyroid	5E+1 Thyroid	2E-8	-	-	-
			(9E+1)	(2E+2)	-	2E-10	1E-6	1E-5

Because the nonstochastic ALI corresponds to a 50-rem (500-mSv) CDE dose limit, the CDE may be calculated based on the ratio of the intake to the nonstochastic ALI multiplied by 50 rem (500 mSv):

$$CDE = (I/ALI) * 50 \text{ rem}$$

Note: For a mixture of radionuclides, the "sum of the fractions" technique as described in 10 CFR 20.1202(b) must be used.

Example 3: Calculate the CDE based on bioassay measurements using the nonstochastic ALIs.

The intake by inhalation for a worker was estimated by bioassay to be 131 nCi (0.131 µCi) of iodine-131 (I-131) as a Class D aerosol. Calculate the CDE to the thyroid.

From Appendix B above, Table 1, Column 2, the ALI for Class D I-131 is:

ALI (nonstochastic) = 5E+1 µCi = 50 µCi
CDE = (0.131 µCi/50 µCi) * 50 rem = 0.131 rem = 131 mrem

4. Calculation of the CDE Based on Air Sampling and Nonstochastic DAC-Hours (DAC-hr)

For nonstochastic radionuclides, an exposure to an airborne concentration of 1 DAC for 2000 hours results in a 50-rem CDE, or 50,000 mrem/2000 hours, or a 25-mrem CDE per DAC-hr.

$$CDE = (25 \text{ mrem per DAC-hr}) * \text{number of DAC-hr}$$

where the number of DAC-hr = (air concentration / DAC value) * exposure time.

Example 4: Calculate the CDE based on air sampling and nonstochastic DAC-hr.

Calculate the CDE to the thyroid for a 30-minute exposure based on an air-sample result of 2.1E-7 µCi/ml from I-131.

The nonstochastic DAC for I-131 is listed in Appendix B (see the excerpt below) as 2E-8 µCi/ml.

Iodine-131

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (µCi)	Table 1 Occupational Values Col. 2 Inhalation ALI (µCi)	Table 1 Occupational Values Col. 3 Inhalation DAC (µCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (µCi/ml)	Table 2 Effluent Concentrations Col. 2 Water (µCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (µCi/ml)
53	Iodine-131	D, all compounds	3E+1 Thyroid	5E+1 Thyroid	2E-8	-	-	-
			(9E+1)	(2E+2)	-	2E-10	1E-6	1E-5

CDE = 25 mrem/DAC-hr * (2.1E-7 µCi/ml / 2E-8 µCi/ml) number of DACs * (0.5 hr) = 131 mrem

5. Calculations of the CEDE Based on Air Sampling and Stochastic DAC-hr

For stochastic radionuclides (e.g., Co-60), an exposure to an airborne concentration of 1 DAC results in a 5000-mrem CEDE in 2000 hours of exposure time (5000 mrem/2000 hours) or a 2.5-mrem CEDE per stochastic DAC-hr.

$$\text{CEDE} = 2.5 \text{ mrem/DAC-hr} * \text{number of DAC-hr}$$

where the number of DAC-hr = (air concentration / DAC value) * exposure time.

Example 5: Calculate the CEDE based on air sampling and stochastic DAC-hr.

Calculate the CEDE for a 30-minute exposure based on an air sample result of 2.1E-7 µCi/ml from Co-60.

From Appendix B below, the stochastic DAC for Co-60 in a clearance Class Y compound is 1E-8 µCi/ml.

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (µCi)	Table 1 Occupational Values Col. 2 Inhalation ALI (µCi)	Table 1 Occupational Values Col. 3 Inhalation DAC (µCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (µCi/ml)	Table 2 Effluent Concentrations Col. 2 Water (µCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (µCi/ml)
27	Cobalt-60	W, see ^{55}Co	5E+2	2E+2	7E-8	2E-10	3E-6	3E-5
		Y, see ^{55}Co	2E+2	3E+1	1E-8	5E-11	-	-

CEDE = (2.5 mrem/DAC-hr) * [(2.1E-7 µCi/ml) / (1E-8 µCi/ml)] number of DACs * (0.5 hr) = 26 mrem

6. Calculation of the CEDE Based on Air Sampling and Calculated Stochastic DAC-hr

CEDE = 2.5 mrem/DAC-hr * number of DAC-hr

Number of DAC-hr = air concentration / calculated DAC value * exposure time

Note: Appendix B to 10 CFR Part 20 does not list the stochastic DAC values (as shown in the empty circled cell below) for radionuclides with intakes that have nonstochastic limits. However, the stochastic DAC values may be calculated based on the stochastic ALI values. These stochastic ALI values are listed (in parentheses) below the limiting nonstochastic organ (see circled value of 2E+2 µCi in the table below).

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
Atomic No.	Radionuclide	Class	Oral Ingestion ALI (µCi)	Inhalation ALI (µCi)	Inhalation DAC (µCi/ml)	Air (µCi/ml)	Water (µCi/ml)	Monthly Average Concentration (µCi/ml)
53	Iodine-131	D, all compounds	3E+1 Thyroid	5E+1 Thyroid	2E-8	-	-	-
			(9E+1)	(2E+2)	-	2E-10	1E-6	1E-5

Example 6: Calculate the CEDE based on air sampling and calculated stochastic DAC-hr.

Calculate the CEDE for a 30-minute exposure based on an air-sample result of 2.1E-7 µCi/ml from I-131.

The stochastic DAC value is first calculated by dividing the stochastic ALI by the breathing rate of 2.4E+9 ml/yr.

The calculated stochastic DAC for I-131 = (2E+2 µCi) / (2.4E+9 ml/yr) = 8E-8 µCi/ml or µCi/cc (because 1 ml = 1 cc).

CEDE = (2.5 mrem/hr/DAC-hr) * [(2.1E-7 µCi/ml) / (8E-8 µCi/ml)] DACs * (0.5 hr) = 3.3 mrem