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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 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, Part 20, of the *Code of Federal Regulations* (10 CFR Part 20), "Standards for Protection against Radiation," (Ref. 1), Section 20.1101, "Radiation Protection Programs," establishes requirements for licensees to limit radiation exposures to individuals within the specified regulatory radiation dose limits and are "as low as is reasonably achievable" (ALARA). To demonstrate compliance with the dose limits, licensees must perform surveys and, when appropriate, monitor the radiation exposure and calculate the resultant doses.

Also, Section 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 licensees are required to perform in order to comply with these regulations were affected by the 2007 revision of 10 CFR Part 20, Section 20.1003, "Definitions" and 10 CFR 50, Section 50.2, "Definitions," (Ref. 2). This revision redefined the "Total Effective Dose

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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 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 "Subpart F – Surveys and Monitoring" (General)

10 CR 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," and

10 CFR Part 19, "Notices, Instructions, and Reports to Workers: Inspection and Investigations," (Ref. 3).

Related Guidance

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

- Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," U.S. Nuclear Regulatory Commission, Washington, DC.
- Regulatory Guide 8.9, Revision 1, "Interpretation of Bioassay Measurements" (Ref. 4), provides methods of determining intakes from bioassay results,
- Regulatory Guide 8.11, "Applications of Bioassay for Uranium"
- Regulatory Guide 8.25, Revision 1, "Air Sampling in the Workplace" (Ref. 5), provides methods of determining intakes from air sampling measurements,
- Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure"

- Regulatory Guide 8.35, Revision 1, "Planned Special Exposures" (Ref. 6), provides guidance on conducting planned special exposures,

- Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus" (Ref. 7), provides methods of calculating doses to the embryo/fetus, and

- Regulatory Guide 8.40, "Methods for Measuring Effective Dose Equivalent from External Exposure," (Ref. 8), provides details on acceptable methods of determining the effective dose equivalent (from external exposure).

Purpose of Regulatory Guides

The NRC issues regulatory guides 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. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This regulatory guide discusses information collection requirements covered by 10 CFR Part 20 and 10 CFR Part 50 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.

B. DISCUSSION

Reason for Revision

On December 4, 2007, the NRC revised the definition of the TEDE in 10 CFR Parts 20 and 50. Under the revised rule, the TEDE means the sum of the effective dose equivalent for external exposures (hereafter referred to as the EDEX) and the committed effective dose equivalent for internal exposures (hereafter referred to as the CEDE). 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 regulatory guide also provides updated guidance on acceptable methods of:

- Determining the need for monitoring and demonstrating compliance,

- Monitoring alpha intakes and determining internal dose,

- Placement of dosimetry and resolving differences between passive and electronic dosimeters,

- Assessing intakes and committed dose equivalent from wounds, and

- Additional calculational methods of determining internal doses.

Background

On December 4, 2007, the NRC revised the definition of the total effective dose equivalent (TEDE) in 10 CFR Part 20, "Standards for Protection against Radiation," Section 20.1003, "Definitions" and 10 CFR 50, Section 50.2, "Definitions," (72 FR 68043 (Ref. 9)). 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 deep dose equivalent (DDE) to account for external exposure and the committed effective dose equivalent (CEDE) to account for internal exposure. Under the revised rule, 10 CFR Part 20, Section 20.1003, "Definitions," the TEDE was redefined by replacing the DDE with the EDEX.

Old definition: TEDE = DDE + CEDE

New definition: TEDE = EDEX + CEDE

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 Embryo/Fetus

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 of is 5 rem (50 millisieverts (mSv)).
- For protection of adults 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 to the skin of any extremity, the annual shallow-dose equivalent (SDE) limit of 50 rem (500 mSv).

For minors, 10 CFR 20.1207, "Occupational Dose Limits for Minors," establishes an annual limit at 10 percent of the adult limits.

For protection of an embryo/fetus of a declared pregnant woman, 10 CFR 20.1208, "Dose Equivalent to an Embryo/Fetus," establishes a dose equivalent limit of 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). Regulatory Guide 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 contaminations, nasal 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), consistent with maintaining

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.

the TEDE ALARA, to increase monitoring (e.g., perform air sampling and Derived Air Concentrations (DAC)-hour tracking and bioassay) and to limit intakes by the use of access controls, limiting exposures times, or use respiratory protection equipment.

Monitoring At Levels Sufficient to Demonstrate Compliance

10 CFR 20.1502 requires monitoring at levels sufficient to demonstrate compliance with the occupational dose limits; therefore monitoring methods should be reasonably accurate. Radiological surveys and exposure times should be used as needed to account for dose not measured by a dosimetry system (e.g., due to dosimetry system sensitivity, or dosimeter placement, or dosimeter capability (e.g., not capable of measuring minor amounts of dose from neutrons or low energy photons).

Licensees may voluntarily issue individual monitoring devices or use calculation methodologies for reasons other than for required personnel monitoring under the requirements in 10 CFR 20.1502 (e.g., to provide for worker knowledge or concern). The results of monitoring when 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

10 CFR 20.1204(e) provides for methods of determining internal exposure when the identity and concentration in a mixture of radionuclides are present. The identity and concentration of radionuclides may be determined by surveys requiring the specific radionuclides and their relative mix. Once the relative mix is known, licensees may make use of this knowledge, and 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 as effective DACs, and is applicable for 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, alarm set points for continuous air monitors, determining the need for respiratory protection, estimating internal dose, or determining when bioassay may be needed.

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 alpha DACs are generally orders of magnitude more restrictive than DACs for other beta-emitting and gamma-emitting isotopes.

Each facility should characterize and periodically update its alpha source term, based on historical and current survey data and alpha spectroscopy measurements. Alpha source term characterization should not be based solely on the samples of dry activated waste collected for 10 CFR 61 waste classification purposes. Loose contamination surveys may not be sufficient to identify fixed alpha contamination that may be present and a hazard during abrasive work (e.g., grinding, cutting or welding). A site-specific characterization should determine the extent of the alpha hazard within specific areas of the plant (such as contained within localized areas within the primary reactor coolant boundary or having spread to generally contaminated 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. A site-specific alpha source term may be used to identify radionuclides and determined their relative concentrations in a mixture, such as to comply with the requirements of 10 CFR 20.1204(f). Once the relative concentrations are known, an effective Derived Air Concentration (DAC) may be determined and used in radiological protection and dose assessment.

Note: Methods and criteria that are acceptable for identifying and controlling alpha hazards are described in the EPRI guidelines, "EPRI Alpha Monitoring and Control Guidelines for Operating Nuclear Power Stations, Revision 2, August, 2013 (ML14083A535) (Ref. 11).

The principal transuranic nuclides producing alpha radiological hazards include the isotopes of curium, plutonium, and americium. For historical fuel failures (e.g., ten years since significant fuel failure), the shorter-lived curium-242 has largely decayed leaving the longer-lived alpha radionuclides with more restrictive DACS and ALIs as the most prevalent hazard. However, 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, the use of effective DAC values must account for the time dependent mix of alpha radionuclides.

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

- Knowledge of the specific alpha radionuclide mix;
- Conservatively assuming the most restrictive radionuclide in the mixture; or
- Determining site specific, effective-DAC alpha values.

Discrete Radioactive Particle Monitoring and SDE

A discrete radioactive particle (DRP) is a radioactive particle that is a small, usually microscopic, highly radioactive beta or beta-gamma emitting particles 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 (67 FR 16298, (Ref. 12) (see also Regulatory Issue Summary 2002-10, "Revision of the Skin Dose Limit in 10 CFR Part 20," (Ref. 23). The amended regulations changed the definition and method of calculating shallow-dose equivalents (SDE) by specifying that the assigned SDE must be the dose averaged over the contiguous 10 cm^2 of skin receiving the highest exposure, rather than 1 cm^2 as previously recommended by the NCRP (NCRP Report No. 106, Limit for Exposure to Hot Particles on the Skin" (1980).

Harmonization with International Standards

The NRC has a goal of harmonizing its guidance with international standards, to the extent practical. 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 these international documents noted throughout this guide and implement the good practices, where applicable that are consistent with NRC regulations. These documents are:

- ICRP Publication 26, "Recommendations of the International Commission on Radiological Protection," (Ref. 14),
- ICRP Publication 30, (7-volume set including supplements), "Limits for Intakes of Radionuclides by Workers," (Ref. 15),

Documents Discussed in Staff Regulatory Guidance

Although this regulatory guide utilizes information, in part, from one or more reports developed by external organizations and other third party guidance documents, the regulatory guide does not endorse these references other than as specified in this regulatory guide. 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 into NRC regulations as a requirement, then licensees and applicants must comply with that requirement in the regulation.

If the secondary reference has been endorsed in a regulatory guide as an acceptable approach for meeting an NRC requirement, then the reference constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific regulatory guide. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a regulatory guide, then 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 appropriately justified and consistent with current regulatory practice, consistent with applicable NRC requirements such as 10 CFR Part 20.

C. STAFF REGULATORY GUIDANCE

1. Monitoring Criteria

10 CFR 20.1502, "Conditions Requiring Individual Monitoring of External and Internal Occupational Dose," requires individual monitoring of external and internal occupational dose under the radiological conditions specified below. Monitoring of external radiation exposure (i.e., the EDEX) is required by 10 CFR 20.1502(a) for any individual entering a high or very high radiation area from licensed

and unlicensed[2] sources under the control of the licensee. Monitoring is also required for any individual if the external occupational dose is likely to exceed:

- For adults, 10 percent of the occupational dose limits in 10 CFR 20.1201(a).
- For minors in one year, a deep-dose equivalent of 0.1 rem (1 mSv), a lens dose equivalent of 0.15 rem (1.5 mSv), and a shallow-dose equivalent to the skin of the whole body or to the skin of the extremities of 0.5 rem (5 mSv).
- For declared pregnant women during the entire pregnancy, a deep-dose equivalent of 0.1 rem (1 mSv).

Monitoring the intake of radioactive material and assessing the CEDE is required by 10 CFR 20.1502(b) if the intake is likely to exceed:

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

2. Occupational Dose

The definition of occupational dose, in 10 CFR 20.1003, "Definitions," 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 other person. The definition of occupational dose was changed in 1995 (60 FR 36038) (Ref. 19) such that occupational dose applies to workers whose assigned duties involve exposure to radiation, irrespective of

2 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;" 10 CFR 40.13, "Unimportant Quantities of Source Material"), naturally occurring radioactive materials that are not covered by the Atomic Energy Act, radioactive materials or nuclear facilities operated by another Federal entity such as the U.S. Department of Defense or the U.S. Department of Energy; as well as machines that produce radiation, such as x-ray radiography machines and x-ray machines used by security staff.

their location inside or outside a restricted area. Note: A member of the public does not become an occupationally exposed individual as a result of just 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. See Regulatory Guide 8.29, "Instruction Concerning Risks from Occupational Radiation Exposure" 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 if those individuals are "likely to exceed" the minimum exposure levels specified in 10 CFR 20.1502 (i.e., determine the need for monitoring of the occupational dose). As discussed in 60 FR 36039 (1995) (Ref. 22), 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." Reactor licensees should consider normal operations and anticipated operational occurrences (e.g., unplanned onsite events, such as sudden increases in external radiation levels, or localized high airborne radioactivity areas) 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. 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 spectrums, or use of new or different types or energies of radiation producing equipment.)

The requirements for monitoring in 10 CFR 20.1502 refers to exposures that may occur at each licensee individually. Doses that have already been received under another licensee, or may be received in the future from employment by 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 only required to monitor external doses under 10 CFR 20.1502(a), or only internal doses under 10 CFR 20.1502(b), then 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 equal to zero, and the TEDE is equal to the EDEX. Similarly, if the external dose is not monitored, the EDEX can be assumed 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:

- Measuring the DDE at the highest 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.

- Measuring external exposure with one or more external personal monitoring devices and determining EDEX using an NRC approved method such as those provided in Regulatory Guide 8.40, or as specifically approved by the NRC.

- Calculating the EDEX based on survey data obtained under 10 CFR 20.1501 or other radiological data, such as known source activity, dose rates, and exposure times using scientifically sound technical methods. This may be required under unique exposure situations (e.g., partial body exposed to radiation streaming of narrow beam geometries) or when the individual monitoring device was not in the region of the highest whole body exposure (per 10 CFR 20.1201(c)), or the results of the individual monitoring are not available (i.e., damaged or lost device).

Note: Within the same monitoring period, a licensee may use a combination of methods above; e.g., a licensee may routinely determine EDEX for the majority of a monitoring period using method (1) above, and then use the methods (2) and/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 is typically measured by the use of an individual monitoring device(s) and is determined at a tissue depth of 1 centimeter (cm) (1,000 mg/cm^2). The DDE can also be calculated if the appropriate parameters (i.e., radiation source strength, exposure geometry, full or partial shielding) are known.

A single DDE located at the highest exposed part of the whole body is a conservative (and for uniform exposures, a reasonably accurate) estimate of the EDEX from external sources. However, there are several other NRC approved methods for determining EDEX provided in RG 8.40 that use external monitoring devices measuring DDE at specific locations on the whole body. See the RG 8.40 for the use and limitations of each method.

In many exposure situations, the passive dosimeter used for the single DDE measurement may be supplemented with an active dosimeter (e.g., electronic dosimeter) for work control or dose accounting purposes (i.e., an active dosimeter provides real time indication of the accrued dose and

possibly the dose rate). Due to the differences in dosimeter design and detection technology (and the relative measurement errors associated with each) there can be differences in reading of these two dosimeters for the same exposure, even if the dosimeters are co-located on the monitored individual. Within a reasonable pre-established accuracy criteria (depending of the dosimeter designs), the small differences can be disregarded and either dosimeter value used as the measured dose (i.e., the readings are considered the same value). However, if dosimeter readings are outside the established accuracy criteria, and unresolved, then the highest reading must be recorded as the dose received during the exposure period per the requirement in 20.1201(c).

d. Determining the LDE

The LDE is defined at a tissue depth of 0.3 cm (300 mg/cm^2). If the LDE is being monitored with a dosimeter, then that dosimeter should be calibrated to measure the dose at a tissue depth of 7 mg/cm^2. Alternately, the LDE may be conservatively determined based on SDE measurements. In many exposure situations, shield glasses can be worn to prevent exposures to the lens of the eye from low energy (or low penetrating) radiations, eliminating the need for monitoring the LDE.

e. Determination of the SDE

The SDE is defined as the external exposure of the skin of the whole body or extremities, which can result from external skin contamination such as from radioactive solids, discrete radioactive particles (hot particles), or liquids on the surface of the skin or on protective clothing. The SDE is defined only for external exposure at a tissue depth of 0.007 cm (7 mg/cm2), 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, then that dosimeter should be calibrated to measure the dose at a tissue depth of 7 mg/cm^2. The latest version of NUREG/CR-6918, "VARSKIN: A Computer Code for Assessing Skin Dose from Skin Contamination" can be used to assess SDE.

The SDE for exposure to submersion class radionuclides containing low energy betas are not readily measureable 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., SDE rates used for pre-job briefings), and also to account for the SDE that may not be adequately measured by dosimeters because of the dosimeter lack of response to low energy beta spectrums.

5. Determination of Intakes

For those licensees determining internal dose per 10 CFR 20.1204, a determination must be made of the intake that can occur through inhalation, ingestion, absorption through the skin, or 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 the readily-detected radionuclides as well as the hard-to-detect radionuclides (not directly measured). The activity of hard-to-detect radionuclides may be based on scaling factors correlated to the amount of readily-detected radionuclides. See Regulatory Guide

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, selecting the air sample location 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 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 10 CFR 20, Appendix A. 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), then no APF adjustment is made to the ambient air concentration as measured by the breathing zone air sample.

a. Determining the Intake based on Air Sampling

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

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

b. Determining the Intake based on Bioassay Measurements

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

The intake(s) from wounds is generally assessed based on bioassay measurements using a combination of whole body in vivo bioassay and hand-held instrumentation. The bioassay measurements should determine the location of the injected source, such that CDE dose calculations may be made to the highest exposed 10 cm2 area of the skin at a depth of 0.007 cm (see section 7.d below).

Note: The amount of the "intake" may be assessed using newer, updated biokinetic models (e.g., ICRP Publications 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 have been specifically approved by the NRC). In other words, the use of more recent tissue or organ dose weighting factors is not acceptable (since the regulations in 10 CFR Part 20 list the specific organ dose weighting factors that must be used).

c. Determining Alpha Intakes

Alpha intakes may be assessed based on radiological surveys and on a site-specific alpha source term. After the relative concentrations of alpha emitting isotopes are determined (e.g., by alpha spectroscopy), scaling factors for alpha to beta/gamma activity may be used to determine the alpha activity. Scaling factors based on surface area contamination or air samples should be representative of work area at the time of exposure.

Internal doses may be determined based on whole body count data and scaling factors when nominal alpha doses occur, such as less than 500 mrem CEDE. However, if an alpha intake exceeding a nominal level is considered likely, excreta sampling or lung counting may be needed to determine intakes and assign dose. When excreta sampling is to be initiated, sampling should begin as soon as possible following detection of the exposure, and continue for a 24 hour period or until at least one sample is collected (following the first void for urine). ANSI N13.39 (2011), "Design of Internal Dosimetry Programs" provides additional guidance on excreta sampling.

6. Determination of Internal Dose

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[3] from the U.S. Environmental Protection Agency's Federal Guidance Report No. 11 (FGR-11) (Ref. 25).
- Using ALI methods.
- Using DAC-hour methods.

Details and examples on calculating the CEDE are described in Appendix A.

Note: When performing CEDE calculations using the ALI and DAC-hour methods, the ALI and DAC values provided in Appendix B 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 intake of radioactive material. Methods and examples of calculating the CDE are described in Appendix A. The special case of calculating the CDE from wound intakes is discussed in Section C.7.d below.

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

The dose limit for protection against the 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 highest 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 highest exposed organ (since the DDE was not monitored and is assumed equal to zero). Further details on acceptable methods of calculating the CDE are described in Appendix A.

[3] 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 since the terminology is not incorporated into the regulations).

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 highest exposed organ. Another acceptable method of demonstrating that the TODE limit is met is by maintaining the DDE to less than 5 rem, and the CEDE to less than 1 rem[4], then the TODE cannot exceed the 50 rem TODE limit. In this case, the CDE does not need to be determined since 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.

10 CFR 20.1201 also specifies two annual dose limits:

- TODE limits (Section 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 (Section 20.1201(a)(2)(ii)) – the SDE of 50 rem (0.5 Sv) to the skin of the whole body or skin of any extremity.

However, because the SDE is defined only for external exposure, the SDE limit is 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 (e.g., skin). Note: In most skin exposure situations, the dose is from external exposure (and therefore the dose to the skin organ is commonly equal to the SDE). However, when the dose to the skin (organ dose) is from a wound, the CDE (organ) 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, since for intakes by wounds, the DDE is zero (since DDE is an external whole-body exposure). As a result, the calculated dose is only the CDE to the skin calculated to the

[4] 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.

highest exposed, contiguous 10 cm^2 area at a depth of 0.007 cm (in a manner similar to SDE calculations).

In summary, the CDE to the skin is the appropriate quantity to be calculated (50-year integrated dose (until the source is removed), at a depth of 0.007 centimeters below the surface of the skin, and averaged over the highest exposed 10 cm^2 of the basal layer of the skin. In order to do this calculation, the location (depth) of the source must be determined as an input parameter, and the most recent version of Varskin computer code may be used in performing calculations.

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 ICRP-54 (Ref. 26), ICRP-78 (Ref. 27), NCRP-87 (Ref. 28), and technical articles by Toohey (Ref. 29) and Ishique (Ref. 30).

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 underlying tissues (e.g., 1 cm^3 of flesh) at the wound site for purposes of determining the potential for tissue function impairment and whether medical intervention is warranted (e.g., surgical removal). The guidance in NCRP Report No. 156, "Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry, and Treatment" is acceptable for this evaluation (Ref. 31).

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 the 10 CFR Part 20 organ weighting factors and specified tissues. The more recent ICRP Publication 68 dose coefficients cannot be used, (unless their use has been specifically approved by the NRC). This is because the ICRP 68 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 at 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 of the material in the lungs and in the GI tract. 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, and therefore, this method may not be useful for small, infrequent intakes. Conversely, the use of specific information of the physical and biochemical properties of radionuclides taken into the body may be appropriate in the case of

accidental large exposures if more accurate information would lead to a better estimate of the actual dose.

8. Uranium Intake Limitation

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 chemical toxicity. Regulatory Guide 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.

D. IMPLEMENTATION

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

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**.**

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4. Regulatory Guide 8.9, Revision 1, "Interpretation of Bioassay Measurements" U.S. Nuclear Regulatory Commission, Washington, DC 20555.

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Appendix A

1. Calculations of the CDE and the CEDE for Any Radionuclide, based on Bioassay Measurements using the Dose Coefficients from Federal Guidance Report No. 11

This method is based on using tabulated dose coefficients to calculate the dose. The 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 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 A, the DC for the lung is selected from clearance Class Y having a value of 3.45E-7). Multiplying the DCs by the intake (I) for that radionuclide calculates the CDE and CEDE for that radionuclide.

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

Example 1: Calculations of the CDE and the CEDE for Co60, based on bioassay measurements using the DCs from FGR-11.

An intake by inhalation was estimated by a whole body count to be 360 nCi (0.36 µCi) of Co-60, 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 Class Y, Co-60 radionuclide are 3.45E-7 Sv/Bq for the CDE and 5.91E-8 Sv/Bq for the CEDE.

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^{-5}	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}

The DCs in FGR-11 are tabulated in Sv/Bq and may be converted to mrem/µCi by multiplying by 3.7×10^9.

DC_{lung} = (3.45E-7 Sv/Bq) * (3.7E+9) = 1,277 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} = (1,277 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 Appendix B to 10 CFR Part 20. 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, based on the fraction of the allowable annual intake and the 5 rem (50 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	-	-

Since 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 ALIs.

The intake by inhalation for a worker was estimated by bioassay to be 360 nCi (0.36 µCi) of Co-60, 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: Considering the precision of a 1 significant figure for the ALI values, this 60 mrem value compares favorably to the calculated CEDE value of 79 mrem determined in Example A above using the FGR-11 method. Either calculational method and/or result is acceptable in demonstrating compliance.

3. Calculation of the CDE based on Bioassay Measurements Using Nonstochastic ALIs

The 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 ALIs are given in parenthesis below (e.g., 9E+1 µCi for ingestion and 2E+2 µCi for inhalation in 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)	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

Since 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 I-131, 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-hrs

For nonstochastic radionuclides, an exposure to an airborne concentration of 1 DAC results for a 2000 hour exposure time results in 50 rem CDE; or 50,000 mrem/2,000 hours, or 25 mrem CDE per DAC-hour.

CDE = [25 mrem per DAC-hr] * number of DAC-hrs

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

Example 4: Calculation of the CDE based on air sampling and nonstochastic DAC-hrs.

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

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

Iodine-131								
			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	Inhalation				
Atomic No.	Radionuclide	Class	ALI (µCi)	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

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

5. Calculations of the CEDE based on air sampling and stochastic DAC-hrs

For stochastic radionuclides (e.g., Co-60), an exposure to an airborne concentration of 1 DAC results in 5,000 mrem CEDE in 2,000 hours of exposure time; or 5,000 mrem/2,000 hours, or 2.5 mrem CEDE per stochastic DAC-hr.

CEDE = [(2.5 mrem)/DAC-hr] * No. of DAC-hrs

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

Example 5: Calculation of the CEDE based on air sampling and stochastic DAC-hrs.

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

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

			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	Inhalation				
Atomic No.	Radionuclide	Class	ALI (µCi)	ALI (µCi)	DAC (µCi/ml)	Air (µCi/ml)	Water (µCi/ml)	Monthly Average Concentration (µCi/ml)
27	Cobalt-60	W, see 55Co	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)] No. of DACs* (0.5 hrs) = 26 mrem

6. Calculation of the CEDE based on air sampling and calculated stochastic DAC-hrs

CEDE = [2.5 mrem/DAC-hr] * No. of DAC-hrs

No. DAC-hrs = [air concentration / calculated DAC value] * [exposure time]

Note: Appendix B to 10 CFR Part 20 does not list the stochastic DAC values (see empty circled cell below) for radionuclides with intakes limited by the nonstochastic limits. However, the stochastic DAC values may be calculated based on the stochastic ALI values. These stochastic ALI values are listed (in parenthesis) 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: Calculation of the CEDE based on air sampling and calculated stochastic DAC-hrs.

Calculate the CEDE for a 30-minute exposure based on an air sample result of 2.1E-7 µCi/ml (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 (I-131) = (2E+2 µCi) / (2.4E+9 ml/yr) = 8E-8 µCi/ml or µCi/cc (since 1 ml = 1 cc)

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