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**DRAFT REGULATORY
GUIDE**

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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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Electronic copies of this regulatory guide, previous versions of this guide, and other recently issued guides are available through the NRC’s public Web site under Regulatory Guides document collection of the NRC Library at <http://www.nrc.gov/reading-rm/doc-collection/>. The regulatory guide is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>, under ADAMS Accession No. **MLXXXXXXXXX**. The regulatory analysis may be found in ADAMS under Accession No. **MLXXXXXXXXX** and the staff responses to the public comments on DG-8031 may be found under ADAMS Accession No. **MLXXXXXXXXX**.

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

33 The following regulatory requirements are also discussed in this guide:

- 34 • 10 CFR 20.1007, “Communications”
35
- 36 • 10 CFR Part 19, “Notices, Instructions, and Reports to Workers: Inspection and
37 Investigations”(Ref. 3)
38
- 39 • 10 CFR 20.1202, “Compliance with Requirements for Summation of External and
40 Internal Doses”
41
- 42 • 10 CFR 20.1204, “Determination of Internal Exposure”
43
- 44 • 10 CFR 20.1206, “Planned Special Exposures”
45
- 46 • 10 CFR 20.1207, “Occupational Dose Limits for Minors”
47
- 48 • 10 CFR 20.1208, “Dose Equivalent to an Embryo/Fetus”
49
- 50 • 10 CFR 20.1501, “General,” in 10 CFR 20 Subpart F, “Surveys and Monitoring”
51
- 52 • 10 CFR 20.1703, “Use of Individual Respiratory Protection Equipment”
53
- 54 • 10 CFR 20.2106, “Records of Individual Monitoring Results”
55
- 56 • 10 CFR 20.2206, “Reports of Individual Monitoring”

57 **Related Guidance**

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

- 61 • RG 8.7, “Instructions for Recording and Reporting Occupational Radiation
62 Exposure Data” (Ref. 4)
63
- 64 • RG 8.9, Revision 1, “Acceptable Concepts, Models, Equations, and Assumptions
65 for a Bioassay Program” (Ref. 5)
66
- 67 • RG 8.11, “Applications of Bioassay for Uranium” (Ref. 6)
68
- 69 • RG 8.25, Revision 1, “Air Sampling in the Workplace” (Ref. 7)
70
- 71 • RG 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure”
72 (Ref. 8)
73

- 74 • RG 8.35, Revision 1, "Planned Special Exposures" (Ref. 9)
75
76 • RG 8.36, "Radiation Dose to the Embryo/Fetus" (Ref. 10)
77
78 • RG 8.40, "Methods for Measuring Effective Dose Equivalent from External
79 Exposure" (Ref. 11)
80

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82 **Purpose of Regulatory Guides**

83 The NRC issues RGs to describe to the public methods that the staff considers acceptable
84 for use in implementing specific parts of the agency's regulations, to explain techniques that the
85 staff uses in evaluating specific problems or postulated accidents, and to provide guidance to
86 applicants. RGs are not substitutes for regulations and compliance with them is not required.
87 Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they
88 provide a basis for the findings required for the issuance or continuance of a permit or license by the
89 Commission.

90 **Paperwork Reduction Act**

91 This RG discusses information-collection requirements covered by 10 CFR Part 20 and
92 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," that the Office of
93 Management and Budget (OMB) approved under OMB control numbers 3150-0014
94 and 3150-0011 respectively. The NRC may neither conduct nor sponsor, and a person is not
95 required to respond to, an information-collection request or requirement unless the requesting
96 document displays a currently valid OMB control number.

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B. DISCUSSION

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138 Reason for Revision

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

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

150 Background

151 On December 4, 2007, the NRC revised the definition of the TEDE in 10 CFR 20.1003 and
152 10 CFR 50.2 (as published in the *Federal Register* at 72 FR 68043 (Ref. 12)). The revision
153 subsequently affected the methods of monitoring and calculating occupational radiation doses and
154 demonstrating compliance with the occupational dose limits. Previously, the definition of the
155 TEDE was the sum of the DDE (to account for external exposure) and the CEDE (to account for
156 internal exposure). Under the revised rule 10 CFR 20.1003, the TEDE was redefined by replacing
157 the DDE with the effective dose equivalent-external (EDEX).

158 Old definition: $TEDE = DDE + CEDE$

159 New definition: $TEDE = EDEX + CEDE$

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

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

174

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

176 For adults, occupational dose limits (except for planned special exposures) are established in
177 10 CFR 20.1201(a) as follows:

- 178 • For protection against stochastic effects, the annual TEDE limit is 5 rem
179 (50 millisieverts (mSv)).
- 180 • For protection against nonstochastic effects, the annual total organ dose equivalent
181 (TODE) limit is 50 rem (500 mSv).
- 182 • For protection of the lens of the eye, the annual lens dose equivalent (LDE) limit is
183 15 rem (150 mSv).
- 184 • For protection of the skin of the whole body or of the skin of any extremity, the
185 annual shallow-dose equivalent (SDE) limit is 50 rem (500 mSv).

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

188 For the embryo/fetus of a declared pregnant woman, a dose equivalent limit during the entire
189 pregnancy is established in 10 CFR 20.1208, “Dose Equivalent to an Embryo/Fetus,” as 0.5 rem
190 (5 mSv).

191 **Planned Special Exposures (PSEs)**

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

197 **Surveys¹**

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

202 When a licensee assigns or permits the use of respiratory protection equipment to limit the
203 intake of radioactive material, 10 CFR 20.1703(c)(2) requires surveys and bioassays, as necessary,
204 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.

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

208 During operations, licensees should perform airborne radioactivity surveys as required in
209 10 CFR 20.1502 to characterize the radiological hazards that may be present and, as appropriate,
210 use engineering and respiratory protection equipment to reduce intakes. When it is not practical to
211 use process or engineering controls to reduce the concentrations of airborne radioactivity to values
212 below those that define an airborne radioactivity area, licensees are required under
213 10 CFR 20.1702(a), to be consistent with keeping the TEDE ALARA, to increase monitoring
214 (e.g., perform air sampling and track Derived Air Concentration (DAC)-hours and bioassay
215 measurements) and to limit intakes by using access controls, limiting exposure times, or having
216 individuals use respiratory protection equipment.

217 **Monitoring at Levels Sufficient To Demonstrate Compliance**

218 Regulations in 10 CFR 20.1502 require monitoring at levels sufficient to demonstrate
219 compliance with the occupational dose limits; therefore, monitoring methods should be reasonably
220 accurate. In addition, licensees may voluntarily issue individual monitoring devices or use
221 calculational methodologies for reasons other than for required personnel monitoring under the
222 requirements in 10 CFR 20.1502 (e.g., to inform individuals of exposure conditions, or to alleviate
223 safety concerns). The results of monitoring that is voluntarily provided but not required by
224 10 CFR 20.1502 are not subject to the dose recording or reporting requirements in 10 CFR Part 20,
225 Subpart L, "Records," or Subpart M, "Reporting." However, licensees may voluntarily provide these
226 reports to the exposed individual(s) and to the NRC.

227 **Use of Effective DACs**

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

235 The use of effective DAC values may be needed in operational radiological protection
236 programs to establish airborne radioactivity postings, determining alarm set points for continuous air
237 monitors, determining the need for respiratory protection, estimating internal dose, or determining
238 when bioassay measurements may be needed. When using effective DACs, licensees may disregard
239 those radionuclides in the mixture (based on prior representative surveys) having a concentration less
240 than 10% of the radionuclide's DAC, given that the sum of disregarded radionuclides does not exceed
241 30% (see 10 CFR 20.1204(g)).

242

243 **Alpha Monitoring at Nuclear Power Plants**

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

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

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

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

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

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

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

276 **Discrete Radioactive-Particle Monitoring and SDE**

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

281 In 2002, the NRC amended its regulations related to the shallow-dose equivalent/skin-dose
282 limit in 10 CFR Part 20 (at 67 FR 16298 (Ref. 13); see also Regulatory Issue Summary 2002-10,
283 "Revision of the Skin Dose Limit in 10 CFR Part 20" (Ref. 14)). The amended regulations
284 changed the definition and method of calculating SDEs by specifying that the assigned SDE must
285 be the dose averaged over the contiguous 10 cm² of skin receiving the highest exposure.

286 **Harmonization with International Standards**

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

294 Such documents include the following:

- 295 • ICRP Publication 26, “Recommendations of the International Commission on
296 Radiological Protection” (Ref. 15)
- 297 • ICRP Publication 30, (7-volume set including supplements), “Limits for Intakes of
298 Radionuclides by Workers” (Ref. 16)
- 299 • ICRP Publication 54, “Individual Monitoring for Intakes of Radionuclides by Workers”
300 (Ref. 17)
- 301 • ICRP Publication 60, “1990 Recommendations of the International Commission on
302 Radiological Protection” (Ref. 18)
- 303 • ICRP Publication 68, “Dose Coefficients for Intakes of Radionuclides for Workers”
304 (Ref. 19)
- 305 • ICRP Publication 78, “Individual Monitoring for Internal Exposure of Workers” (Ref. 20)
- 306 • ICRP Publication 103, “The 2007 Recommendations of the International Commission on
307 Radiological Protection” (Ref. 21)

308 **Documents Discussed in Staff Regulatory Guidance**

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

315 If the secondary reference has been endorsed in an RG as an acceptable approach for
316 meeting an NRC requirement, the reference constitutes a method acceptable to the NRC staff for
317 meeting that regulatory requirement as described in the specific RG. If the secondary reference has
318 neither been incorporated by reference in NRC regulations nor endorsed in an RG, the secondary
319 reference is neither a legally binding requirement nor a “generic” NRC approval as an acceptable
320 approach for meeting an NRC requirement. However, licensees and applicants may consider and
321 use the information in the secondary reference, if it is appropriately justified and consistent with
322 current regulatory practice, in ways consistent with applicable NRC requirements such as those in
323 10 CFR Part 20.

324

C. STAFF REGULATORY GUIDANCE

1. Monitoring Criteria

Regulations in 10 CFR 20.1502 require individual monitoring of external and internal occupational dose at levels sufficient² to demonstrate compliance with the occupational dose limits. As a minimum, licensees must monitor occupational exposure to radiation from licensed and unlicensed radiation sources³ 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.

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).

351 **2. Occupational Dose**

352 The definition of occupational dose in 10 CFR 20.1003 includes dose received during the
353 course of employment in which assigned duties involve exposure to radiation or radioactive
354 material from licensed and unlicensed sources of radiation, whether in the possession of the
355 licensee or of another person. The definition of occupational dose was changed in 1995 (at
356 60 FR 36038) (Ref. 22) so that occupational dose applies to workers whose assigned duties involve
357 exposure to radiation, irrespective of their location inside or outside a restricted area. Note:
358 A member of the public does not become an occupationally exposed individual simply as a result of
359 entering a restricted area.

360 Individuals who receive occupational exposure and are likely to receive more than
361 100 mrem must be instructed in accordance with 10 CFR 19.12, "Instruction to Workers." See
362 RG 8.29 for further information.

363 **3. Prospective Assessments of the Need for Occupational Dose Monitoring**

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

369 Once occupationally exposed individuals are identified, licensees should perform a
370 prospective assessment to determine whether those individuals are "likely to exceed" the minimum
371 exposure levels specified in 10 CFR 20.1502 (i.e., to determine the need for monitoring of the
372 occupational dose). The potential for unlikely exposures and accident conditions need not be
373 considered because these events, by definition, are unlikely. However, as discussed at
374 60 FR 36039, the term "likely to receive" includes "normal situations as well as abnormal
375 situations involving exposure to radiation which can reasonably be expected to occur during the life
376 of the facility." Therefore, licensees should consider normal operations and anticipated operational
377 occurrences (e.g., unplanned onsite events, such as sudden increases in external radiation levels, or
378 localized areas of high airborne radioactivity) but would not need to consider design-basis
379 accidents

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

391
392 The requirements for monitoring in 10 CFR 20.1502 refer to exposures that might occur at
393 each licensee individually. Doses that have already been received while in the employ of another
394 licensee, or that might be received in the future while in the employ of another licensee or
395 unlicensed entity, are excluded from consideration in a licensee's determination of the need to

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

398 **4. Determination of External Doses**

399 **a. Determination of the TEDE**

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

408 **b. Determination of the EDEX**

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

- 411 1. Measuring the DDE at the most highly exposed part of the whole body with an external
412 personal monitoring device, as required by 10 CFR 20.1201(c), when an NRC method for
413 determining EDEX is not used.
- 414 2. Measuring external exposure with one or more external personal monitoring devices and
415 determining EDEX using an NRC-approved method (such as those provided in RG 8.40 or
416 as specifically approved elsewhere by the NRC).
- 417 3. Calculating the EDEX based on survey data obtained under 10 CFR 20.1501 or on other
418 radiological data (such as known source activity, dose rates, and exposure times) using
419 scientifically sound technical methods. This might be required (a) under unique exposure
420 situations (e.g., if an individual's body were partially exposed to radiation streaming in a
421 narrow beam geometry), (b) when the individual's monitoring device was not in the region
422 of the highest whole-body exposure (in accordance with 10 CFR 20.1201(c)), or (c) when
423 the results of the individual monitoring are not available (i.e., the monitoring device is
424 damaged or lost).

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

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

431 The DDE (external exposure of the whole body) is typically measured with a passive
432 primary monitoring device that assesses the dose at a tissue depth of 1 centimeter (cm) (a mass
433 thickness of 1,000 mg/cm²). The DDE can also be calculated if the appropriate parameters are
434 known (i.e., the radiation source strength, the exposure geometry, and whether full or partial
435 shielding was in place).

436 An individual monitoring device located at the most highly exposed part of the whole body
437 measuring the DDE is a conservative and (for uniform exposures) a reasonably accurate estimate of
438 the EDEX. However, if the radiation dose is highly non uniform, causing a specific part of the
439 whole body (head, trunk, arms above the elbow, or legs above the knees) to receive a substantially
440 higher dose than the rest of the whole body, the individual monitoring device should be placed near
441 that part of the whole body expected to receive the highest dose. There are several other
442 NRC-approved methods for determining EDEX provided in RG 8.40.

443

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

458

d. Determining the LDE

459

460 If the LDE is being monitored with a dosimeter, that dosimeter should be calibrated to
461 measure the dose at a tissue depth of 0.3 centimeter (cm) (a mass thickness 300 mg/cm²).
462 Alternatively, the LDE may be conservatively determined based on SDE measurements at
463 7 mg/cm². In many exposure situations, safety glasses can be worn to minimize exposures to the
464 lens of the eye from low-energy (or poorly penetrating) radiations, potentially eliminating the need
465 for monitoring the LDE.

465

e. Determination of the SDE

466

467 The SDE is defined only for external exposure at a tissue depth of 0.007 cm (a mass
468 thickness of 7 mg/cm²), and is the dose averaged over the contiguous 10 cm² of skin receiving the
469 highest exposure. If the SDE is being measured with a dosimeter, that dosimeter should be
470 calibrated to measure the dose at a tissue depth of 7 mg/cm². For skin contamination, the computer
471 code described in NUREG/CR-6918, "VARSKIN: A Computer Code for Skin Contamination
472 Dosimetry" (Ref. 23) may be used to assess the SDE. The SDE may also be determined from
473 analytical calculational methods based on survey data when dosimetry methods are not
474 representative of the actual exposure conditions.

474

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

481 **5. Determination of Intakes**

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

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

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

498 If respiratory protection is used to limit the intake of radioactive materials,
499 10 CFR 20.1703(c)(4)(i) requires internal monitoring to be implemented as part of the respiratory
500 protection program. When respiratory protection is provided, the intake is adjusted by dividing the
501 ambient air concentration by the appropriate Assigned Protection Factor (APF) listed in
502 Appendix A, "Assigned Protection Factors for Respirators," to 10 CFR 20. If the ambient air
503 concentration is determined by performing breathing-zone air sampling inside the respiratory
504 protective device (such as with a lapel air sampler inside a loose-fitting supplied air hood or suit),
505 no APF adjustment is made to the ambient air concentration as measured in the breathing-zone air
506 sample.

507 **a. Determining the Intake Based on Air Sampling**

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

510 $I = C_{\text{Air sample}} (\mu\text{Ci/ml}) * \text{breathing rate (ml/minutes)} * \text{exposure time (minutes)}$, where the
511 breathing rate of a "Reference Man" under light working conditions is 2E+4 ml/minute
512 (20 liters/minute).

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

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

519 $I = A_{\text{Air sample}} (\mu\text{Ci}) * \text{breathing rate/air sampler flow rate (ml/min)}$, where the breathing rate
520 of a "Reference Man" under light working conditions is 2E+4 ml/minute
521 (20 liters/minute).

522 **b. Determining the Intake Based on Bioassay Measurements**

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

529 Any intake from wounds is generally assessed based on bioassay measurements using a
530 combination of whole body *in vivo* bioassay and handheld instrumentation. The bioassay
531 measurements should determine the location and depth of the injected source so that CDE dose
532 calculations may be made to the most highly exposed 10 cm² area of the skin at a depth of 0.007 cm
533 (see Section 6.d below).

534 Note: The amount of the “intake” may be assessed using newer, updated biokinetic models
535 (e.g., those described in ICRP Publication 60, “1990 Recommendations of the International
536 Commission on Radiological Protection,” and ICRP Publication 103, “The
537 2007 Recommendations of the International Commission on Radiological Protection”). However,
538 the CEDE must be calculated using the existing 10 CFR 20.1003 organ weighting factors (unless
539 the use of other weighting factors has been specifically approved by the NRC).

540 **c. Determining Intakes of Alpha Emitters**

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

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

553 **6. Determination of Internal Doses**

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

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

558 • Using dose coefficients⁴ from the U.S. Environmental Protection Agency’s Federal
559 Guidance Report No. 11 (FGR-11) (Ref. 26).

560 • Using ALI methods.

561 • Using DAC-hour methods.

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

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

569 **b. Calculation of the Committed Dose Equivalent (CDE)**

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

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

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

576
$$\text{TODE} = \text{DDE} + \text{CDE}$$

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

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

582 If both internal and external monitoring are being performed, the licensee must
583 demonstrate that both the 5-rem TEDE and the 50-rem TODE limits are met. One method of
584 demonstrating compliance with the TODE limit is by summing the DDE and the CDE to the most
585 highly exposed organ. Another acceptable method of demonstrating that the TODE limit is met is
586 by keeping the maximum DDE below 5 rem and the CEDE below 1 rem⁵; if this is done, the TODE
587 cannot exceed its 50-rem limit. In this case, the CDE does not need to be determined because

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.

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

590 **d. Doses from Intakes through Wounds**

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

593 Regulations in 10 CFR 20.1201 also specify two annual dose limits:

- 594 • TODE limits (10 CFR 20.1201(a)(1)(ii))—the sum of the DDE and the CDE to any
595 individual organ or tissue other than the lens of the eye being equal to 50 rem
596 (0.5 Sv)—and
- 597 • SDE limits (10 CFR 20.1201(a)(2)(ii))—the SDE to the skin of the whole body or skin of
598 any extremity being equal to 50 rem (0.5 Sv).

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

605 In making the TODE dose calculation (to the skin organ) under 20.1201(a)(1)(ii), the DDE
606 component is zero (because DDE is specifically defined as an external whole-body exposure). As a
607 result, the CDE is determined for the basal layer of the skin at a depth of 0.007 cm below skin
608 surface for the most highly exposed, contiguous 10-cm² area.

609 In summary, the CDE to the skin is the appropriate quantity to be calculated as the
610 integrated dose from the time of injection to the time the source is removed or by the 50-year
611 integration period for committed dose. The CDE is to be determined at a depth of
612 0.007 centimeters below the surface of the skin, averaged over the most highly exposed 10 cm² of
613 the basal layer of the skin. In order to do this calculation, the location (depth) of the source and
614 distance to the basal layer must be determined as an input parameter. The VARSKIN computer
615 code may be used in performing the CDE skin-dose calculations.

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

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

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

629 A number of methods are acceptable for calculating the CDE and CEDE from the intake of
630 radioactive materials. Some of these methods are described below. However, calculations of the
631 CEDE must be based on organ weighting factors and tissues specified in 10 CFR Part 20. The dose
632 coefficients based on ICRP Publication 60 cannot be used unless specifically approved by the
633 NRC, because ICRP 60 and ICRP 103 tissues and weighting factors are different from those in
634 10 CFR Part 20.

635 **7. Use of Individual or Material-Specific Information**

636 The regulation in 10 CFR 20.1204(c) states that “when specific information on the
637 physical and biochemical properties of the radionuclides taken into the body or the behavior of the
638 material in an individual is known, the licensee may [...] use that information to calculate the
639 committed effective dose equivalent [...]” Prior NRC approval is not required, but detailed records
640 must be kept to demonstrate the acceptability of the dose assessment.

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

650 **8. Limitation on Uranium Intake**

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

657 **9. Recording Of Individual Monitoring Results**

658 The requirements for recording individual monitoring results are contained in 10 CFR 20.2106,
659 which requires that the recording be done on NRC Form 5, or in clear and legible records
660 containing all the information required by NRC Form 5. Regulatory Guide 8.7 provides further
661 guidance for recording and reporting occupational radiation dose data.

662 Licensees should avoid entering doses on NRC Form 5 with more significant figures than justified
663 by the precision of the basic measured values. In general, it is appropriate to enter dose values with
664 two significant figures on NRC Form 5 using the standard rules for round-off. Thus, a
665 computer-generated calculated dose of "1.726931 rems" should be entered on NRC Form 5 as "1.7
666 rems." However, licensees should generally carry at least three significant figures in calculations to
667 avoid loss of accuracy due to multiple round-offs.

668 In addition, licensees should not enter doses smaller than 0.001 rem on NRC Form 5 because
669 smaller values are insignificant relative to the dose limits. Therefore, a calculated committed
670
671

672 effective dose equivalent of "0.006192 rem" should be entered as "0.006 rem," and a value of
673 "0.000291 rem" should be entered as "0 rem."

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D. IMPLEMENTATION

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

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

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Appendix A Methods of Calculating Internal Dose

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

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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:

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

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(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

Nuclide	Class/ f_1	Committed Dose Equivalent per Unit Intake (Sv/Bq)							Effective
		Gonad	Breast	Lung	R Marrow	B Surface	Thyroid	Remainder	
Co-60	W $5 \cdot 10^{-2}$	$4.05 \cdot 10^{-9}$	$4.16 \cdot 10^{-9}$	$3.57 \cdot 10^{-8}$	$4.25 \cdot 10^{-9}$	$3.54 \cdot 10^{-9}$	$3.72 \cdot 10^{-9}$	$7.65 \cdot 10^{-9}$	$8.94 \cdot 10^{-9}$
	Y $5 \cdot 10^{-2}$	$4.76 \cdot 10^{-9}$	$1.84 \cdot 10^{-8}$	$3.45 \cdot 10^{-7}$	$1.72 \cdot 10^{-8}$	$1.35 \cdot 10^{-8}$	$1.62 \cdot 10^{-8}$	$3.60 \cdot 10^{-8}$	$5.91 \cdot 10^{-8}$

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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.

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788

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

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

789

790

791

792

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/ μ Ci) by multiplying by $3.7E+9$.

793

794

795 An intake by inhalation was estimated by a whole body count to be 360 nanocuries (nCi)
796 (0.36 μCi) of Co-60 as a Class Y aerosol. Calculate the CDE to the lung and the CEDE.

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

799

Nuclide	Class/f ₁	Committed Dose Equivalent per Unit Intake (Sv/Bq)							
		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 ⁻⁵	3.45 10 ⁻⁷	1.72 10 ⁻⁵	1.35 10 ⁻⁵	1.62 10 ⁻⁵	3.60 10 ⁻⁵	5.91 10 ⁻⁵

800

801 $DC_{lung} = (3.45E-7 \text{ Sv/Bq}) * (3.7E+9) = 1277 \text{ mrem}/\mu\text{Ci}$
 802 $DC_{effective} = (5.91E-8 \text{ Sv/Bq}) * (3.7E+9) = 219 \text{ mrem}/\mu\text{Ci}$

803

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

805

806 $CDE_{lung} = (1277 \text{ mrem}/\mu\text{Ci}) * (0.36 \mu\text{Ci}) = 460 \text{ mrem}$
 807 $CEDE = (219 \text{ mrem}/\mu\text{Ci}) * (0.36 \mu\text{Ci}) = 79 \text{ mrem}$

808

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

811 The ALI values are listed in Table 1 of 10 CFR 20, Appendix B, “Annual Limits on Intake
812 (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure;
813 Effluent Concentrations for Release to Sewerage.” Column 1 lists the values for oral ingestion and
814 Column 2 lists the values for inhalation. The stochastic ALI values can be used in the calculation
815 of the CEDE, which is based on the fraction of the allowable annual intake and the 5-rem
816 (50-millisievert (mSv)) CEDE dose limit. When the ALI is defined by the stochastic limit, this
817 value alone is given in the table.

818

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	Monthly Average Concentration (μCi/ml)
			Oral Ingestion ALI (μCi)	Inhalation		Air (μCi/ml)	Water (μCi/ml)	
ALI (μCi)	DAC (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	-	-

819

820

821 Because the stochastic ALI corresponds to a 5-rem (50-mSv) CEDE dose limit, the CEDE
822 may be calculated based on the ratio of the intake to the stochastic ALI multiplied by 5 rem
823 (50 mSv):

824

$$825 \text{ CEDE} = (I/\text{ALI}) * 5 \text{ rem}$$

826

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

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

830

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

832

$$833 \text{ ALI (stochastic)} = 30 \mu\text{Ci}$$

$$834 \text{ CEDE} = (I/\text{ALI}) * 5 \text{ rem}$$

$$835 \text{ CEDE} = (0.36 \mu\text{Ci}/30 \mu\text{Ci}) * 5 \text{ rem} = 0.06 \text{ rem} = 60 \text{ mrem}$$

836

837 Note: Doses calculated based on FGR-11 methods are generally more precise than doses
838 calculated based on ALI values, because ALI values are given to only one significant figure.
839 Additionally, the precision of the ALI values is limited by the calculational technique used in
840 ICRP-30 (Section 4.7) whereby target organs that are not significantly irradiated were excluded
841 (<10% rule), as well as dose from source organs contributing less than 1% were also excluded. For
842 further information, see Oak Ridge National Laboratory, ORNL/TM-13188, "Recommended ALIs
843 and DACs for 10 CFR 20: A Consistent Numerical Set" (Ref. 31).

844

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

849

850 3. Calculation of the CDE Based on Bioassay Measurements Using 851 Nonstochastic ALI

852 The 10 CFR 20 Appendix B, Table 1, Column 2, nonstochastic ALI values can be used in
853 the calculation of the CDE, based on the fraction of the allowable annual intake and the 50-rem
854 (500-mSv) CDE dose limit. When the ALI is defined by the nonstochastic limit, this value is listed
855 first in the table with its corresponding organ (see excerpt below), and the corresponding stochastic
856 ALI are given in parentheses (e.g., $9\text{E}+1 \mu\text{Ci}$ (90 μCi) for ingestion and $2\text{E}+2 \mu\text{Ci}$ (200 μCi) for
857 inhalation in the excerpt below).

858

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	Monthly Average Concentration (µCi/ml)
			Oral Ingestion ALI (µCi)	Inhalation		Air (µCi/ml)	Water (µCi/ml)	
	ALI (µCi)	DAC (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

859

860

861 Because the nonstochastic ALI corresponds to a 50-rem (500-mSv) CDE dose limit, the
 862 CDE may be calculated based on the ratio of the intake to the nonstochastic ALI multiplied by
 863 50 rem (500 mSv):

864

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

865

866 Note: For a mixture of radionuclides, the “sum of the fractions” technique as described in
 867 10 CFR 20.1202(b) must be used.

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

869

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

872

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

874

$$ALI \text{ (nonstochastic)} = 5E+1 \text{ } \mu\text{Ci} = 50 \text{ } \mu\text{Ci}$$

$$CDE = (0.131 \text{ } \mu\text{Ci}/50 \text{ } \mu\text{Ci}) * 50 \text{ rem} = 0.131 \text{ rem} = 131 \text{ mrem}$$

877

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

880 For nonstochastic radionuclides, an exposure to an airborne concentration of 1 DAC for
 881 2000 hours results in a 50-rem CDE, or 50,000 mrem/2000 hours, or a 25-mrem CDE per DAC-hr.

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

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

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

885

886 Calculate the CDE to the thyroid for a 30-minute exposure based on an air-sample result of
 887 2.1E-7 µCi/ml from I-131.

888
889
890
891

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			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	Monthly Average Concentration (μCi/ml)
			Oral Ingestion ALI (μCi)	Inhalation		Air (μCi/ml)	Water (μCi/ml)	
				ALI (μCi)	DAC (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

892

893 CDE = 25 mrem/DAC-hr * (2.1E-7 μCi/ml / 2E-8 μCi/ml) number of DACs * (0.5 hr) =
894 131 mrem

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

896 For stochastic radionuclides (e.g., Co-60), an exposure to an airborne concentration of
897 1 DAC results in a 5000-mrem CEDE in 2000 hours of exposure time (5000 mrem/2000 hours) or
898 a 2.5-mrem CEDE per stochastic DAC-hr.

899
$$CEDE = 2.5 \text{ mrem/DAC-hr} * \text{number of DAC-hr}$$

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

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

902

903 Calculate the CEDE for a 30-minute exposure based on an air sample result of
904 2.1E-7 μCi/ml from Co-60.

905

906 From Appendix B below, the stochastic DAC for Co-60 in a clearance Class Y compound
907 is 1E-8 μCi/ml.
908

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	Monthly Average Concentration (μCi/ml)
			Oral Ingestion ALI (μCi)	Inhalation		Air (μCi/ml)	Water (μCi/ml)	
				ALI (μCi)	DAC (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	-	-

909

910 CEDE = (2.5 mrem/DAC-hr) * [(2.1E-7 μCi/ml) / (1E-8 μCi/ml)] number of DACs *
 911 (0.5 hr) = 26 mrem

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

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

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

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

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	Monthly Average Concentration (μCi/ml)
			Oral Ingestion ALI (μCi)	Inhalation		Air (μCi/ml)	Water (μCi/ml)	
				ALI (μCi)	DAC (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

921

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

923 Calculate the CEDE for a 30-minute exposure based on an air-sample result of
 924 2.1E-7 μCi/ml from I-131.

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

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

929 CEDE = (2.5 mrem/hr/DAC-hr) * [(2.1E-7 μCi/ml) / (8E-8 μCi/ml)] DACs * (0.5 hr)
 930 = 3.3 mrem