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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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28 Equivalent” (TEDE), as the sum of the effective dose equivalent (for external exposures) and the CEDE (for
29 internal exposures).

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

31 10 CFR 20.1007, “Communications,”

32 10 CFR 20.1202, “Compliance with Requirements for Summation of External and Internal Doses,”

33 10 CFR 20.1204, “Determination of Internal Exposure,”

34 10 CFR 20.1206, “Planned Special Exposures,”

35 10 CFR 20.1207, “Occupational Dose Limits for Minors,”

36 10 CFR 20.1208, “Dose Equivalent to an Embryo/Fetus,”

37 10 CFR 20.1501 “Subpart F – Surveys and Monitoring” (General)

38 10 CR 20.1703, “Use of Individual Respiratory Protection Equipment,”

39 10 CFR 20.2106, “Records of Individual Monitoring Results,”

40 10 CFR 20.2206, “Reports of Individual Monitoring,” and

41 10 CFR Part 19, “Notices, Instructions, and Reports to Workers: Inspection and Investigations,”
42 (Ref. 3).

43 **Related Guidance**

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

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

- 56 • Regulatory Guide 8.35, Revision 1, "Planned Special Exposures" (Ref. 6), provides guidance
57 on conducting planned special exposures,
- 58 • Regulatory Guide 8.36, "Radiation Dose to the Embryo/Fetus" (Ref. 7), provides methods of
59 calculating doses to the embryo/fetus, and
- 60 • Regulatory Guide 8.40, "Methods for Measuring Effective Dose Equivalent from External
61 Exposure," (Ref. 8), provides details on acceptable methods of determining the effective
62 dose equivalent (from external exposure).

63 **Purpose of Regulatory Guides**

64 The NRC issues regulatory guides to describe to the public methods that the staff considers
65 acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the
66 staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants.
67 Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and
68 solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis
69 for the findings required for the issuance or continuance of a permit or license by the Commission.

70 **Paperwork Reduction Act**

71 This regulatory guide discusses information collection requirements covered by 10 CFR Part 20 and
72 10 CFR Part 50 that the Office of Management and Budget (OMB) approved under OMB control numbers
73 3150-0014 and 3150-0011 respectively. The NRC may neither conduct nor sponsor, and a person is not
74 required to respond to, an information collection request or requirement unless the requesting document
75 displays a currently valid OMB control number.

76 **B. DISCUSSION**

77 **Reason for Revision**

78 On December 4, 2007, the NRC revised the definition of the TEDE in 10 CFR Parts 20 and 50.
79 Under the revised rule, the TEDE means the sum of the effective dose equivalent for external exposures
80 (hereafter referred to as the EDEX) and the committed effective dose equivalent for internal exposures
81 (hereafter referred to as the CEDE). This revision of RG 8.34 provides updated regulatory guidance on
82 monitoring criteria and methods of calculating occupational dose based on the revised definition of the
83 TEDE. This regulatory guide also provides updated guidance on acceptable methods of:

- 84 • Determining the need for monitoring and demonstrating compliance,
- 85 • Monitoring alpha intakes and determining internal dose,
- 86 • Placement of dosimetry and resolving differences between passive and electronic
87 dosimeters,
- 88 • Assessing intakes and committed dose equivalent from wounds, and

- Additional calculational methods of determining internal doses.

90 **Background**

91 On December 4, 2007, the NRC revised the definition of the total effective dose equivalent (TEDE)
92 in 10 CFR Part 20, “Standards for Protection against Radiation,” Section 20.1003, “Definitions” and 10 CFR
93 50, Section 50.2, “Definitions,” (72 FR 68043 (Ref. 9)). The revision subsequently affected the methods of
94 monitoring and calculating occupational radiation doses and demonstrating compliance with the occupational
95 dose limits. Previously, the definition of the TEDE was the sum of the deep dose equivalent (DDE) to
96 account for external exposure and the committed effective dose equivalent (CEDE) to account for internal
97 exposure. Under the revised rule, 10 CFR Part 20, Section 20.1003, “Definitions,” the TEDE was redefined
98 by replacing the DDE with the EDEX.

99 Old definition: $TEDE = DDE + CEDE$

100 New definition: $TEDE = EDEX + CEDE$

101 In uniform radiation fields, the EDEX is normally determined by measuring the DDE and, therefore,
102 the revised TEDE definition has little impact on monitoring methods. However, for exposures in
103 non-uniform radiation fields, the revised TEDE definition provides greater monitoring flexibility and
104 accuracy for licensees in monitoring worker exposures. Under non-uniform conditions, the previous TEDE
105 definition tended to provide dose assessments that were excessively conservative.

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

111 **Occupational Dose Limits for Adults, Minors, and Embryo/Fetus**

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

- 114 • For protection against stochastic effects, the annual TEDE limit of is 5 rem (50
115 millisieverts (mSv)).
- 116 • For protection of adults against nonstochastic effects, the annual total organ dose
117 equivalent (TODE) limit is 50 rem (500 mSv).
- 118 • For protection of the lens of the eye, the annual lens dose equivalent (LDE) limit is
119 15 rem (150 mSv).
- 120 • For protection of the skin of the whole body or to the skin of any extremity, the annual
121 shallow-dose equivalent (SDE) limit of 50 rem (500 mSv).

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

124 For protection of an embryo/fetus of a declared pregnant woman, 10 CFR 20.1208, “Dose
125 Equivalent to an Embryo/Fetus,” establishes a dose equivalent limit of 0.5 rem (5 mSv).

126 **Planned Special Exposures (PSEs)**

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

132 **Surveys¹**

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

137 When a licensee assigns or permits the use of respiratory protection equipment to limit the intake of
138 radioactive material, 10 CFR 20.1703(c)(2) requires surveys and bioassays, as necessary, to evaluate actual
139 intakes. Indications of an intake could include facial contaminations, nasal contamination, malfunctioning
140 respiratory protection equipment, loss of engineering controls creating an airborne radioactivity area, and
141 work in unknown or unplanned airborne radioactivity areas.

142 During operations, licensees should perform airborne radioactivity surveys as required in
143 10 CFR 20.1502 to characterize the radiological hazards that may be present and, as appropriate, use
144 engineering and respiratory protection equipment to reduce intakes. When it is not practical to use process or
145 engineering controls to reduce the concentrations of airborne radioactivity to values below those that define
146 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.

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

150 **Monitoring At Levels Sufficient to Demonstrate Compliance**

151 10 CFR 20.1502 requires monitoring at levels sufficient to demonstrate compliance with the
152 occupational dose limits; therefore monitoring methods should be reasonably accurate. Radiological surveys
153 and exposure times should be used as needed to account for dose not measured by a dosimetry system (e.g.,
154 due to dosimetry system sensitivity, or dosimeter placement, or dosimeter capability (e.g., not capable of
155 measuring minor amounts of dose from neutrons or low energy photons).

156 Licensees may voluntarily issue individual monitoring devices or use calculation methodologies
157 for reasons other than for required personnel monitoring under the requirements in 10 CFR 20.1502 (e.g., to
158 provide for worker knowledge or concern). The results of monitoring when voluntarily provided, but not
159 required by 10 CFR 20.1502, are not subject to the dose recording or reporting requirements in 10 CFR
160 Part 20, Subpart L, "Records" or Subpart M, "Reporting." However, licensees may voluntarily provide these
161 reports to the exposed individual(s) and to the NRC.

162 **Use of Effective DACs**

163 10 CFR 20.1204(e) provides for methods of determining internal exposure when the identity and
164 concentration in a mixture of radionuclides are present. The identity and concentration of radionuclides may
165 be determined by surveys requiring the specific radionuclides and their relative mix. Once the relative mix is
166 known, licensees may make use of this knowledge, and apply scaling factors applicable to the mixture for use
167 in calculating DACs and tracking DAC-hours as specified in 20.1204(e). This is commonly referred as
168 effective DACs, and is applicable for beta/gamma activity, alpha activity, and hard-to-detect radionuclides.

169 The use of effective DAC values may be needed in operational radiological protection programs to
170 establish airborne radioactivity postings, alarm set points for continuous air monitors, determining the need
171 for respiratory protection, estimating internal dose, or determining when bioassay may be needed.

172 **Alpha Monitoring at Nuclear Power Plants**

173 For reactor facilities that have experienced significant fuel defects, alpha contamination may be a
174 radiological hazard requiring specific evaluation. Alpha contamination (when present) requires specific
175 evaluation because the alpha DACs are generally orders of magnitude more restrictive than DACs for other
176 beta-emitting and gamma-emitting isotopes.

177 Each facility should characterize and periodically update its alpha source term, based on historical
178 and current survey data and alpha spectroscopy measurements. Alpha source term characterization should not
179 be based solely on the samples of dry activated waste collected for 10 CFR 61 waste classification purposes.
180 Loose contamination surveys may not be sufficient to identify fixed alpha contamination that may be present
181 and a hazard during abrasive work (e.g., grinding, cutting or welding). A site-specific characterization should
182 determine the extent of the alpha hazard within specific areas of the plant (such as contained within localized
183 areas within the primary reactor coolant boundary or having spread to generally contaminated areas).

184 The extent of the radiological characterization that is needed depends on the relative significance of
185 the alpha source term compared to other radiological contaminants. A site-specific alpha source term may be
186 used to identify radionuclides and determined their relative concentrations in a mixture, such as to comply
187 with the requirements of 10 CFR 20.1204(f). Once the relative concentrations are known, an effective
188 Derived Air Concentration (DAC) may be determined and used in radiological protection and dose
189 assessment.

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

193 The principal transuranic nuclides producing alpha radiological hazards include the isotopes of
194 curium, plutonium, and americium. For historical fuel failures (e.g., ten years since significant fuel failure),
195 the shorter-lived curium-242 has largely decayed leaving the longer-lived alpha radionuclides with more
196 restrictive DACS and ALIs as the most prevalent hazard. However, more recent fuel failures are likely to
197 identify curium-242 as the most abundant alpha emitting nuclide, which has less restrictive DAC and ALI
198 values. Therefore, the use of effective DAC values must account for the time dependent mix of alpha
199 radionuclides.

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

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

205 **Discrete Radioactive Particle Monitoring and SDE**

206 A discrete radioactive particle (DRP) is a radioactive particle that is a small, usually microscopic,
207 highly radioactive beta or beta-gamma emitting particles having relatively high specific activity. DRPs are
208 primarily an external exposure hazard to the skin, as measured by the SDE.

209 In 2002, the NRC amended its regulations related to the shallow dose equivalent/skin dose limit in
210 10 CFR Part 20 (67 FR 16298, (Ref. 12) (see also Regulatory Issue Summary 2002-10, “Revision of the Skin
211 Dose Limit in 10 CFR Part 20,” (Ref. 23). The amended regulations changed the definition and method of
212 calculating shallow-dose equivalents (SDE) by specifying that the assigned SDE must be the dose averaged
213 over the contiguous 10 cm² of skin receiving the highest exposure, rather than 1 cm² as previously
214 recommended by the NCRP (NCRP Report No. 106, Limit for Exposure to Hot Particles on the Skin” (1980).

215 **Harmonization with International Standards**

216 The NRC has a goal of harmonizing its guidance with international standards, to the extent
217 practical. The International Commission on Radiological Protection (ICRP) and the International Atomic
218 Energy Agency (IAEA) have issued a significant number of standards, guidance and technical documents,
219 and recommendations addressing good practices in most aspects of radiation protection. The NRC

220 encourages licensees to consult these international documents noted throughout this guide and implement the
221 good practices, where applicable that are consistent with NRC regulations. These documents are:

- 222 • ICRP Publication 26, “Recommendations of the International Commission on
223 Radiological Protection,” (Ref. 14),
- 224 • ICRP Publication 30, (7-volume set including supplements), “Limits for Intakes of
225 Radionuclides by Workers,” (Ref. 15),

226 **Documents Discussed in Staff Regulatory Guidance**

227 Although this regulatory guide utilizes information, in part, from one or more reports developed by
228 external organizations and other third party guidance documents, the regulatory guide does not endorse these
229 references other than as specified in this regulatory guide. These reports and third party guidance documents
230 may contain references to other reports or third party guidance documents (“secondary references”). If a
231 secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then
232 licensees and applicants must comply with that requirement in the regulation.

233 If the secondary reference has been endorsed in a regulatory guide as an acceptable approach for
234 meeting an NRC requirement, then the reference constitutes a method acceptable to the NRC staff for meeting
235 that regulatory requirement as described in the specific regulatory guide. If the secondary reference has
236 neither been incorporated by reference into NRC regulations nor endorsed in a regulatory guide, then the
237 secondary reference is neither a legally-binding requirement nor a “generic” NRC approval as an acceptable
238 approach for meeting an NRC requirement. However, licensees and applicants may consider and use the
239 information in the secondary reference, if appropriately justified and consistent with current regulatory
240 practice, consistent with applicable NRC requirements such as 10 CFR Part 20.

241 **C. STAFF REGULATORY GUIDANCE**

242 **1. Monitoring Criteria**

243 10 CFR 20.1502, “Conditions Requiring Individual Monitoring of External and Internal
244 Occupational Dose,” requires individual monitoring of external and internal occupational dose under the
245 radiological conditions specified below. Monitoring of external radiation exposure (i.e., the EDEX) is
246 required by 10 CFR 20.1502(a) for any individual entering a high or very high radiation area from licensed

247 and unlicensed² sources under the control of the licensee. Monitoring is also required for any individual if the
248 external occupational dose is likely to exceed:

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

255 Monitoring the intake of radioactive material and assessing the CEDE is required by
256 10 CFR 20.1502(b) if the intake is likely to exceed:

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

260 2. Occupational Dose

261 The definition of occupational dose, in 10 CFR 20.1003, “Definitions,” includes dose received
262 during the course of employment in which assigned duties involve exposure to radiation or radioactive
263 material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other
264 person. The definition of occupational dose was changed in 1995 (60 FR 36038) (Ref. 19) such that
265 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.

266 their location inside or outside a restricted area. Note: A member of the public does not become an
267 occupationally exposed individual as a result of just entering a restricted area.

268 Individuals who receive occupational exposure and are likely to receive more than 100 mrem must
269 be instructed in accordance with 10 CFR 19.12. See Regulatory Guide 8.29, “Instruction Concerning Risks
270 from Occupational Radiation Exposure” for further information.

271 3. Prospective Assessments of the Need for Occupational Dose Monitoring

272 Licensees must identify those individuals receiving occupational dose, either individually or
273 as a group or category of individuals. Individuals pre-designated by the licensee as receiving occupational
274 dose are subject to the occupational dose limits; otherwise, individuals must be considered as members of the
275 public subject to public dose limits in 10 CFR 20.1301, “Dose limits for individual members of the public.”

276 Once occupationally exposed individuals are identified, licensees should perform a
277 prospective assessment to determine if those individuals are “likely to exceed” the minimum exposure levels
278 specified in 10 CFR 20.1502 (i.e., determine the need for monitoring of the occupational dose). As discussed
279 in 60 FR 36039 (1995) (Ref. 22), the term “likely to receive” includes “normal situations as well as abnormal
280 situations involving exposure to radiation which can reasonably be expected to occur during the life of the
281 facility.” Reactor licensees should consider normal operations and anticipated operational occurrences (e.g.,
282 unplanned onsite events, such as sudden increases in external radiation levels, or localized high airborne
283 radioactivity areas) but would not need to consider design basis accidents.

284 The prospective assessment determines the type of monitoring required (e.g., external dose
285 or internal dose monitoring). In performing a prospective assessment, an evaluation should be performed
286 based on planned work activities and likely exposure conditions. Prospective assessments should be revised
287 when there are substantial changes to the radiological conditions of personnel exposure (e.g., changes in work
288 activities, airborne concentrations, beta energy spectrums, or use of new or different types or energies of
289 radiation producing equipment.)

290 The requirements for monitoring in 10 CFR 20.1502 refers to exposures that may occur at
291 each licensee individually. Doses that have already been received under another licensee, or may be
292 received in the future from employment by another licensee or unlicensed entity, are excluded from
293 consideration in a licensee’s determination of the need to monitor an individual. The need for monitoring
294 should be based on the anticipated exposure to licensed or unlicensed sources under the control of a single
295 licensee.

296 4. Determination of External Doses

297 a. Determination of the TEDE

298 Under 10 CFR 20.1202, if a licensee is required to monitor both external dose and internal
299 dose, the licensee must demonstrate compliance with the dose limits by summing external and internal doses
300 (i.e., $TEDE = EDEX + CEDE$). However, if the licensee is only required to monitor external doses under
301 10 CFR 20.1502(a), or only internal doses under 10 CFR 20.1502(b), then summation is not required to
302 demonstrate compliance with the occupational dose limits. For example, if the internal dose is not

303 monitored, the CEDE can be assumed equal to zero, and the TEDE is equal to the EDEX. Similarly, if the
304 external dose is not monitored, the EDEX can be assumed equal to zero, and the TEDE is equal to the CEDE.

305 **b. Determination of the EDEX**

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

- 308 • Measuring the DDE at the highest exposed part of the whole body with an external
309 personal monitoring device, as required by 10 CFR 20.1201(c), when an NRC method
310 for determining EDEX is not used.
- 311 • Measuring external exposure with one or more external personal monitoring devices and
312 determining EDEX using an NRC approved method such as those provided in
313 Regulatory Guide 8.40, or as specifically approved by the NRC.
- 314 • Calculating the EDEX based on survey data obtained under 10 CFR 20.1501 or other
315 radiological data, such as known source activity, dose rates, and exposure times using
316 scientifically sound technical methods. This may be required under unique exposure
317 situations (e.g., partial body exposed to radiation streaming of narrow beam geometries)
318 or when the individual monitoring device was not in the region of the highest whole body
319 exposure (per 10 CFR 20.1201(c)), or the results of the individual monitoring are not
320 available (i.e., damaged or lost device).

321 Note: Within the same monitoring period, a licensee may use a combination of methods
322 above; e.g., a licensee may routinely determine EDEX for the majority of a monitoring period using
323 method (1) above, and then use the methods (2) and/or (3) for special exposure situations at other
324 times. The results of the different dosimetry methods must be combined to determine the EDEX for
325 the entire monitoring period.

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

327 The DDE is typically measured by the use of an individual monitoring device(s) and is
328 determined at a tissue depth of 1 centimeter (cm) (1,000 mg/cm²). The DDE can also be calculated if
329 the appropriate parameters (i.e., radiation source strength, exposure geometry, full or partial
330 shielding) are known.

331 A single DDE located at the highest exposed part of the whole body is a conservative (and for
332 uniform exposures, a reasonably accurate) estimate of the EDEX from external sources. However,
333 there are several other NRC approved methods for determining EDEX provided in RG 8.40 that use
334 external monitoring devices measuring DDE at specific locations on the whole body. See the RG
335 8.40 for the use and limitations of each method.

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

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

347 **d. Determining the LDE**

348 The LDE is defined at a tissue depth of 0.3 cm (300 mg/cm²). If the LDE is being monitored
349 with a dosimeter, then that dosimeter should be calibrated to measure the dose at a tissue depth of
350 7 mg/cm². Alternately, the LDE may be conservatively determined based on SDE measurements. In
351 many exposure situations, shield glasses can be worn to prevent exposures to the lens of the eye from
352 low energy (or low penetrating) radiations, eliminating the need for monitoring the LDE.

353 **e. Determination of the SDE**

354 The SDE is defined as the external exposure of the skin of the whole body or extremities,
355 which can result from external skin contamination such as from radioactive solids, discrete
356 radioactive particles (hot particles), or liquids on the surface of the skin or on protective clothing.
357 The SDE is defined only for external exposure at a tissue depth of 0.007 cm (7 mg/cm²), and is the
358 dose averaged over the contiguous 10 cm² of skin receiving the highest exposure. If the SDE is being
359 measured with a dosimeter, then that dosimeter should be calibrated to measure the dose at a tissue
360 depth of 7 mg/cm². The latest version of NUREG/CR-6918, "VARSKIN: A Computer Code for
361 Assessing Skin Dose from Skin Contamination" can be used to assess SDE.

362 The SDE for exposure to submersion class radionuclides containing low energy betas are not
363 readily measureable by direct survey techniques or dosimetry methods, and hence may need to be
364 calculated based on air sample analyses and DAC-hr tracking. This submersion exposure
365 information may be needed for informing workers of radiological exposure conditions (e.g., SDE
366 rates used for pre-job briefings), and also to account for the SDE that may not be adequately
367 measured by dosimeters because of the dosimeter lack of response to low energy beta spectrums.

368 **5. Determination of Intakes**

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

373 The assessment of intake should include the readily-detected radionuclides as well as the
374 hard-to-detect radionuclides (not directly measured). The activity of hard-to-detect radionuclides may be
375 based on scaling factors correlated to the amount of readily-detected radionuclides. See Regulatory Guide

376 8.25, “Air Sampling in the Workplace” and Regulatory Guide 8.9, “Acceptable Concepts, Models, Equations,
377 and Assumptions for a Bioassay Program” for further guidance on determining uptakes and intakes.

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

382 If respiratory protection is used to limit the intake of radioactive materials, 10 CFR
383 20.1703(c)(4)(i) requires internal monitoring be implemented as part of the respiratory protection program.
384 When respiratory protection is provided, the intake is adjusted by dividing the ambient air concentration by
385 the appropriate Assigned Protection Factor (APF) listed in 10 CFR 20, Appendix A. If the ambient air
386 concentration is determined by performing breathing zone air sampling inside the respiratory protective
387 device (such as with a lapel air sampler inside a loose fitting supplied air hood or suit), then no APF
388 adjustment is made to the ambient air concentration as measured by the breathing zone air sample.

389 **a. Determining the Intake based on Air Sampling**

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

392
$$I = C_{\text{Air sample}} (\mu\text{Ci/ml}) * \text{breathing rate (ml/minutes)} * \text{exposure time (minutes);}$$

393 where the breathing rate of "Reference Man" under light working
394 conditions is 2E+4 ml/minute.

395 The intake of radionuclides can also be estimated by “DAC-Hour” tracking in which the
396 ambient airborne concentration (expressed as a fraction of the DAC) is multiplied by exposure time
397 (expressed in hours).

398 **b. Determining the Intake based on Bioassay Measurements**

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

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

411 Note: The amount of the “intake” may be assessed using newer, updated biokinetic models
412 (e.g., ICRP Publications 60, “1990 Recommendations of the International Commission on
413 Radiological Protection,” and ICRP Publication 103, “The 2007 Recommendations of the
414 International Commission on Radiological Protection”). However, the CEDE must be calculated
415 using the existing 10 CFR 20.1003 organ weighting factors (unless the use of other weighting factors
416 have been specifically approved by the NRC). In other words, the use of more recent tissue or organ
417 dose weighting factors is not acceptable (since the regulations in 10 CFR Part 20 list the specific
418 organ dose weighting factors that must be used).

419 **c. Determining Alpha Intakes**

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

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

432 **6. Determination of Internal Dose**

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

434 The dose quantity for protection against stochastic effects of internal dose is the CEDE; i.e., a
435 50-year committed effective dose equivalent from intakes occurring during the monitoring period. There are
436 three fundamental methods described below for calculating the CEDE:

437 • Using dose coefficients³ from the U.S. Environmental Protection Agency’s Federal
438 Guidance Report No. 11 (FGR-11) (Ref. 25).

439 • Using ALI methods.

440 • Using DAC-hour methods.

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

442 Note: When performing CEDE calculations using the ALI and DAC-hour methods, the ALI
443 and DAC values provided in Appendix B to 10 CFR Part 20 must be used, unless the licensee has
444 obtained prior NRC approval in accordance with 10 CFR 20.1204(c)(2) to adjust the ALI or DAC
445 values.

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

447 The CDE is the 50-year committed dose equivalent from intake of radioactive material.
448 Methods and examples of calculating the CDE are described in Appendix A. The special case of
449 calculating the CDE from wound intakes is discussed in Section C.7.d below.

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

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

453
$$\text{TODE} = \text{DDE} + \text{CDE}$$

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

456 If only internal monitoring is being performed, the TODE is equal to the CDE to the highest
457 exposed organ (since the DDE was not monitored and is assumed equal to zero). Further details on
458 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).

459 If both internal and external monitoring are being performed, the licensee must demonstrate
460 that both the 5 rem TEDE and the 50 rem TODE limits are met. One method of demonstrating
461 compliance with the TODE limit is by summing the DDE and the CDE to the highest exposed organ.
462 Another acceptable method of demonstrating that the TODE limit is met is by maintaining the DDE
463 to less than 5 rem, and the CEDE to less than 1 rem⁴, then the TODE cannot exceed the 50 rem TODE
464 limit. In this case, the CDE does not need to be determined since compliance was demonstrated by
465 calculation. If the CEDE does exceed 1 rem, the CDE must be determined in order to demonstrate
466 compliance with the dose limits.

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

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

470 10 CFR 20.1201 also specifies two annual dose limits:

- 471 • TODE limits (Section 20.1201(a)(1)(ii)) - the sum of the DDE and the CDE to any
472 individual organ or tissue other than the lens of the eye) being equal to 50 rem (0.5 Sv)),
473 and
- 474 • SDE limits (Section 20.1201(a)(2)(ii)) – the SDE of 50 rem (0.5 Sv) to the skin of the
475 whole body or skin of any extremity.

476 However, because the SDE is defined only for external exposure, the SDE limit is not
477 applicable (to dose from wound intakes). Therefore, the TODE dose limit becomes the only
478 applicable limit; i.e., a CDE limit of 50 rem to any individual organ (e.g., skin). Note: In most skin
479 exposure situations, the dose is from external exposure (and therefore the dose to the skin organ is
480 commonly equal to the SDE). However, when the dose to the skin (organ dose) is from a wound, the
481 CDE (organ) dose limit applies (not the SDE).

482 In making the TODE dose calculation (to the skin organ) under 20.1201(a)(1)(ii), the DDE
483 component is zero, since for intakes by wounds, the DDE is zero (since DDE is an external
484 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.

485 highest exposed, contiguous 10 cm² area at a depth of 0.007 cm (in a manner similar to SDE
486 calculations).

487 In summary, the CDE to the skin is the appropriate quantity to be calculated (50-year
488 integrated dose (until the source is removed), at a depth of 0.007 centimeters below the surface of the
489 skin, and averaged over the highest exposed 10 cm² of the basal layer of the skin. In order to do this
490 calculation, the location (depth) of the source must be determined as an input parameter, and the most
491 recent version of Varskin computer code may be used in performing calculations.

492 For wound intakes with systemic uptakes, an evaluation must be performed of the CEDE and
493 TEDE. Additional information on assessing intakes through wounds is available ICRP-54 (Ref. 26),
494 ICRP-78 (Ref. 27), NCRP-87 (Ref. 28), and technical articles by Toohey (Ref. 29) and Ishique (Ref.
495 30).

496 Note: With respect to tissue dose, there is no regulatory limit for small volume, localized
497 tissue dose. However, licensees should estimate the committed dose to underlying tissues (e.g., 1
498 cm³ of flesh) at the wound site for purposes of determining the potential for tissue function
499 impairment and whether medical intervention is warranted (e.g., surgical removal). The guidance in
500 NCRP Report No. 156, "Development of a Biokinetic Model for Radionuclide-Contaminated
501 Wounds and Procedures for Their Assessment, Dosimetry, and Treatment" is acceptable for this
502 evaluation (Ref. 31).

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

504 A number of methods are acceptable for calculating the CDE and CEDE from the intake of
505 radioactive materials. Some of these methods are described below. However, calculations of the
506 CEDE must be based on the 10 CFR Part 20 organ weighting factors and specified tissues. The more
507 recent ICRP Publication 68 dose coefficients cannot be used, (unless their use has been specifically
508 approved by the NRC). This is because the ICRP 68 and ICRP 103 tissues and weighting factors are
509 different from those in 10 CFR Part 20.

510 7. Use of Individual or Material-Specific Information

511 The regulation at 10 CFR 20.1204(c) states that "when specific information on the physical
512 and biochemical properties of the radionuclides taken into the body or the behavior of the material in
513 an individual is known, the licensee may...use that information to calculate the committed effective
514 dose equivalent...." Prior NRC approval is not required, but detailed records must be kept to
515 demonstrate the acceptability of the dose assessment.

516 The characteristics most amenable to such individual or site-specific consideration are the
517 activity median aerodynamic diameter (AMAD) of the inhaled aerosol and the solubility of the
518 material in the lungs and in the GI tract. The use of specific information on the physical and
519 biochemical properties to calculate the CEDE requires the licensee to do considerably more work and
520 to have greater technical expertise than the other methods, and therefore, this method may not be
521 useful for small, infrequent intakes. Conversely, the use of specific information of the physical and
522 biochemical properties of radionuclides taken into the body may be appropriate in the case of

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

525 **8. Uranium Intake Limitation**

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

532 **D. IMPLEMENTATION**

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

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

540

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543

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628

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 $5 \cdot 10^{-2}$	$4.05 \cdot 10^{-9}$	$4.16 \cdot 10^{-9}$	$3.57 \cdot 10^{-8}$	$4.25 \cdot 10^{-9}$	$3.54 \cdot 10^{-9}$	$3.72 \cdot 10^{-9}$	$7.65 \cdot 10^{-9}$	$8.94 \cdot 10^{-9}$
	Y $5 \cdot 10^{-2}$	$4.76 \cdot 10^{-9}$	$1.84 \cdot 10^{-8}$	$3.45 \cdot 10^{-7}$	$1.72 \cdot 10^{-8}$	$1.35 \cdot 10^{-8}$	$1.62 \cdot 10^{-8}$	$3.60 \cdot 10^{-8}$	$5.91 \cdot 10^{-8}$

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.

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

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

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 μ Ci) of Co-60, Class Y aerosol. Calculate the CDE to the lung and the CEDE.

657 From Table 2.1 of FGR-11 (see excerpt below), the DCs for Class Y, Co-60 radionuclide are
 658 3.45E-7 Sv/Bq for the CDE and 5.91E-8 Sv/Bq for the CEDE.

659

660 Excerpt from Federal Guidance Report No. 11

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 ⁻³	3.45 10 ⁻⁷	1.72 10 ⁻³	1.35 10 ⁻³	1.62 10 ⁻³	3.60 10 ⁻³	5.91 10 ⁻³

661

662 The DCs in FGR-11 are tabulated in Sv/Bq and may be converted to mrem/μCi by
 663 multiplying by 3.7x10⁹.

664

665 $DC_{lung} = (3.45E-7 \text{ Sv/Bq}) * (3.7E+9) = 1,277 \text{ mrem}/\mu\text{Ci}$

666 $DC_{effective} = (5.91E-8 \text{ Sv/Bq}) * (3.7E+9) = 219 \text{ mrem}/\mu\text{Ci}$

667

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

669

670 $CDE_{lung} = (1,277 \text{ mrem}/\mu\text{Ci}) * (0.36 \mu\text{Ci}) = 460 \text{ mrem}$

671 $CEDE = (219 \text{ mrem}/\mu\text{Ci}) * (0.36 \mu\text{Ci}) = 79 \text{ mrem}$

672

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

674 The ALI values are listed in Table 1 of Appendix B to 10 CFR Part 20. Column 1 lists the
 675 values for oral ingestion, and Column 2 lists the values for inhalation. The stochastic ALI values can
 676 be used in the calculation of the CEDE, based on the fraction of the allowable annual intake and the 5
 677 rem (50 mSv) CEDE dose limit. When the ALI is defined by the stochastic limit, this value alone is
 678 given in the table.

679

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	-	-

680

681

682 Since the stochastic ALI corresponds to a 5 rem (50 mSv) CEDE dose limit, the CEDE may
 683 be calculated based on the ratio of the intake to the stochastic ALI, multiplied by 5 rem (50 mSv).

684
 685
$$\text{CEDE} = (I/\text{ALI}) * 5 \text{ rem}$$

 686

687 **Example 2:** Calculate the CEDE based on bioassay measurements using the stochastic ALIs.

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

690
 691 From Appendix B above, Table 1, Column 2, the ALI for Class Y Co-60 is:

692
 693
$$\text{ALI (stochastic)} = 30 \mu\text{Ci}$$

 694
$$\text{CEDE} = (I/\text{ALI}) * 5 \text{ rem}$$

 695
$$\text{CEDE} = (0.36 \mu\text{Ci}/30 \mu\text{Ci}) * 5 \text{ rem} = 0.06 \text{ rem} = 60 \text{ mrem}$$

 696

697 Note: Considering the precision of a 1 significant figure for the ALI values, this 60 mrem
 698 value compares favorably to the calculated CEDE value of 79 mrem determined in Example
 699 A above using the FGR-11 method. Either calculational method and/or result is acceptable in
 700 demonstrating compliance.
 701

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

703 The nonstochastic ALI values can be used in the calculation of the CDE, based on the
 704 fraction of the allowable annual intake and the 50 rem (500 mSv) CDE dose limit. When the ALI is
 705 defined by the nonstochastic limit, this value is listed first in the table with its corresponding organ
 706 (see excerpt below), and the corresponding stochastic ALIs are given in parenthesis below (e.g.,
 707 9E+1 μCi for ingestion and 2E+2 μCi for inhalation in excerpt below).

708
 709

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	

710
 711

712 Since the nonstochastic ALI corresponds to a 50 rem (500 mSv) CDE dose limit, the CDE
713 may be calculated based on the ratio of the intake to the nonstochastic ALI, multiplied by 50 rem (500
714 mSv).

$$715 \quad \text{CDE} = (I/\text{ALI}) * 50 \text{ rem}$$

716
717 Note: For a mixture of radionuclides, the sum of the fractions technique as described in 10 CFR
718 20.1202(b) must be used.

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

720
721 The intake by inhalation for a worker was estimated by bioassay to be 131 nCi (0.131 μCi) of
722 I-131, Class D aerosol. Calculate the CDE to the thyroid.

723
724 From Appendix B above, Table 1, Column 2, the ALI for Class D I-131 is:

$$725 \quad \text{ALI (nonstochastic)} = 5\text{E}+1 \mu\text{Ci} = 50 \mu\text{Ci}$$
$$726 \quad \text{CDE} = (0.131 \mu\text{Ci}/50 \mu\text{Ci}) * 50 \text{ rem} = 0.131 \text{ rem} = 131 \text{ mrem}$$

728

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

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

$$733 \quad \text{CDE} = [25 \text{ mrem per DAC-hr}] * \text{number of DAC-hrs}$$

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

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

736
737 Calculate the CDE to the thyroid for a 30-minute exposure based on an air sample result of
738 $2.1\text{E}-7 \mu\text{Ci/ml}$ (I-131).

739
740 The nonstochastic DAC for I-131 is listed in Appendix B (see excerpt below) as $2\text{E}-8$
741 $\mu\text{Ci/ml}$.

742

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

743

744

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

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

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

751
$$\text{CEDE} = [(2.5 \text{ mrem})/\text{DAC-hr}] * \text{No. of DAC-hrs}$$

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

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

754 Calculate the CEDE for a 30-minute exposure based on an air sample result of 2.1E-7 µCi/ml
755 (Co-60).
756

757 From Appendix B below, the stochastic DAC for Co-60 clearance Class Y compound is 1E-8
758 µCi/ml.
759
760

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	-	-

761

762 CEDE = [2.5 mrem/DAC-hr] * [(2.1E-7 µCi/ml) / (1E-8 µCi/ml)] No. of DACs* (0.5 hrs) = 26 mrem

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

764 $CEDE = [2.5 \text{ mrem/DAC-hr}] * \text{No. of DAC-hrs}$

765 $\text{No. DAC-hrs} = [\text{air concentration} / \text{calculated DAC value}] * [\text{exposure time}]$

766 Note: Appendix B to 10 CFR Part 20 does not list the stochastic DAC values (see empty
 767 circled cell below) for radionuclides with intakes limited by the nonstochastic limits.
 768 However, the stochastic DAC values may be calculated based on the stochastic ALI values.
 769 These stochastic ALI values are listed (in parenthesis) below the limiting nonstochastic organ
 770 (see circled 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 ($\mu\text{Ci/ml}$)
			Oral Ingestion ALI (μCi)	Inhalation		Air ($\mu\text{Ci/ml}$)	Water ($\mu\text{Ci/ml}$)	
				ALI (μCi)	DAC ($\mu\text{Ci/ml}$)			
53	Iodine-131	D, all compounds	3E+1 Thyroid	5E+1 Thyroid	2E-8	-	-	-
			(9E+1)	(2E+2)	-	2E-10	1E-6	1E-5

771

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

773 Calculate the CEDE for a 30-minute exposure based on an air sample result of 2.1E-7 $\mu\text{Ci/ml}$ (I-131).

774 The stochastic DAC value is first calculated by dividing the stochastic ALI by the breathing rate of
 775 2.4E+9 ml/yr.

776 The calculated stochastic DAC (I-131) = (2E+2 μCi) / (2.4E+9 ml/yr) = 8E-8 $\mu\text{Ci/ml}$ or $\mu\text{Ci/cc}$ (since
 777 1 ml = 1 cc)

778 $CEDE = [2.5 \text{ mrem/hr/DAC-hr}] * [(2.1E-7 \mu\text{Ci/ml}) / (8E-8 \mu\text{Ci/ml})] \text{ DACs} * (0.5 \text{ hrs}) = 3.3 \text{ mrem}$

779