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NRC Training
RG 1.21 and RG 8.34 Revisions

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RG 1.21 and RG 8.34 Revisions

Session 1

RG 1.21, Rev. 3, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste"

Session 2

RG 8.34, Rev. 1, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses"

Session 1 – RG 1.21, Rev. 3

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U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE RG 1.21, REVISION 3

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MEASURING, EVALUATING, AND REPORTING RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS AND SOLID WASTE

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for the following uses:

(1) measuring, evaluating, and reporting licensed (plant-related) radioactivity in effluents and solid radioactive waste shipments from nuclear power plants and spent fuel storage facilities, and

(2) assessing and reporting the public dose to demonstrate compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation"

RG 1.21 ~ Measuring Effluents

- Significant changes to RG 1.21
  - when to update long-term, annual average χ/Q and D/Q values
  - environmental monitoring for iodine-131 in drinking water
  - ODCM - making changes to effluent and environmental programs
  - incorporates Regulatory Issue Summary 2008-03, “Return/Reuse of Previously Discharged Radioactive Effluents”
  - calibration of accident-range radiation monitors

RG 1.21
χ/Q and D/Q values

- Long-term annual-average χ/Q and D/Q should be based on 5 or more years of meteorological data
- χ/Q and D/Q values should be reevaluated periodically (e.g., every 3–5 years).
- If χ/Q and D/Q values are substantially nonconservative (e.g., higher by 20–30 percent or more), revise χ/Q and D/Q values used in dose assessment

RG 1.21 Environmental monitoring I -131 in drinking water

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- Some licensees believed they were required to perform unnecessary and expensive I-131 analyses in drinking water
- Licensees have brought concerns to NRC's attention at REEW
- HQ investigated the basis for NUREG-1301/1302 guidance
- Some licensees have been misinterpreting NUREG-1301 guidance on the need for sampling for I-131 in drinking water
- HQ determined that this guidance was applicable to setting up an environmental monitoring program
- Licensees should perform an evaluation of the likelihood of doses exceeding 1 mrem/yr (max organ and max age group) per NUREG-1301/1302

NUREG-1301/1302
I -131 in drinking water

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

			TYPE AND FREQUENCY OF ANALYSIS
3.	Waterborne (Cont	Composite sample over 2-week period[6] when I-131 analysis is performed; monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year[8]. Composite for gross beta and gamma isotopic analyses[4]
	c. Drinking		

(8) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

NUREG-1301/1302 - Environmental monitoring for I -131 in drinking water

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1) (2)

LOWER LIMIT OF DETECTION (LLD)(3)

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)
I-131	1**	0.07

** If no drinking water pathway exists, a value of 15 pCi/l may be used.

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NUREG-1301/1302 - Environmental monitoring for I -131 in drinking water

- In order to get to 1 pCi/L LLD, labs need to do a resin column separation of iodine from water
- A resin column separation is an expensive analysis
- In order to get to 15 pCi/L LLD, labs can do a simple gamma scan

RG 1.21
I-131 in Drinking Water

NUREG-1301 and NUREG-1302 provide guidance on implementing the environmental monitoring program for I-131 analysis on each composite of drinking water. The sampling and analysis of a drinking water composite sample should be performed when the projected annual thyroid dose from I-131 in drinking water is greater than 1 mrem.

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RG 1.21 Environmental monitoring I -131 in drinking water

- Licensees should perform a prospective dose evaluation to determine if the likely dose from I-131 in drinking water is > than 1 mrem/yr

- If likely dose is > 1 mrem/yr, perform I-131 sampling & analysis using an LLD of 1 pCi/L
  (resin column separation and gamma scan)

- If likely dose is < 1 mrem/yr, perform I-131 sampling and analysis using an LLD of 15 pCi/L
  (perform a gamma scan of drinking water)

RG 1.21 - Revising ODCMs

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- ODCMs need to be kept current
- Plant changes affect ODCMs:
  - Plant operating status (operating or decommissioning)
  - Failed fuel or change to decommissioning status – noble gas and iodine have been eliminated
  - Installation of new or removal of out-of-service radwaste processing equipment

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10 CFR 50.59 Changes, Tests, and Experiments – Exception (4) below

(c)(1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to Sec. 50.90 only if:

(i) A change to the technical specifications incorporated in the license is not required, and

(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.

(4) The provisions in this section do not apply to changes to the facility or procedures when the applicable regulations establish more specific criteria for accomplishing such changes.

Tech Specs specify how ODCMs are to be revised

- A specific change mechanism is specified in Technical Specs on how to make changes to the ODCM

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

Tech Specs specify how ODCMs are to be revised (Cont'd)

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Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,

b. Shall become effective after the approval of the plant manager, and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the

10 CFR 50.34a – Carbon-14 (C-14)

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§ 50.34a Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors.

(a) An application for a construction permit shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences. In the case of an

and in relation to the use of atomic energy in the public interest. The guides set out in appendix I to this part provide numerical guidance on design objectives for light-water-cooled nuclear power reactors to meet the requirements that radioactive material in effluents released to unrestricted areas be kept as low as is reasonably achievable. These numerical guides for design objectives and limiting conditions for operation are not to be construed as radiation protection standards.

(c) Each application for an operating license shall include:

(1) A description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, under paragraph (a) of this section; and

- There is no waste processing equipment for removal of C-14 from gaseous effluents

10 CFR 50, Appendix I – C-14 is not included as an organ dose criteria

Sec. II. *Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50 or part 52 of this chapter.* The guides on design objectives set forth in this section may be used by an applicant for a construction permit as guidance in meeting the requirements of § 50.34a(a), or

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

C. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

Tech Spec requirements

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5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM,

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

EPA 40 CFR 190.10(a) (excerpt below)

(a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its

RG 1.21
C-14 Dose Assessments

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- C-14 is likely a principal radionuclide in operating reactor effluents
- C-14 is likely the most significant organ dose radionuclide
- Appendix I limits organ dose to radionuclides in iodine or particulate form (not C-14)
- 10 CFR 50, Appendix I ~ Design Objectives

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

- Note: Tech specs add tritium in the organ dose limit

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and

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C-14 Source Term

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- C-14 source term
  - Scaling factors can be based on power generation, or
  - NUREG-0016 GALE computer codes, or
  - NCRP-81's C-14 report, or
  - EPRI Report No. 1021106
  - PWRs release ~ 6 curies, (~ 4.6 mrem max dose)
  - BWRs release ~ 9 curies, (~4.7 mrem max dose)

RG 1.21
C-14 Dose Assessments

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- C-14 dose is primarily from eating vegetables from local gardens
- Local gardens are identified in the land use census
  - Licensees have problems with communities growing up around nuclear plants
  - Licensees have problems locating/moving air samplers
- Vegetables are grown year-round in the southern USA, but less than ~ 6 months in the northern USA
- Vegetables absorb CO_2 during photosynthesis (daylight hours)

RG 1.21
Compliance with EPA dose limits

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- EPA dose limits required by 10 CFR 20.1301(e) are 25 mrem/yr whole body and any organ (except thyroid)
- C-14 is dominant source of organ dose (bone)
  - maximum organ doses in 2020 (incl. C-14) were:
    - BWR = 4.7 mrem
    - PWR = 4.6 mrem
    - Assume dose from direct radiation is < 10 mrem/yr
  - Total organ dose = effluent dose plus direct radiation
    - Total dose = 4.7 mrem + <10 mrem = <14.70 mrem
    - < 14.70 mrem is less than 25.00 mrem EPA 40 CFR 190

RIS-2008-03 Return/Reuse of Previously Discharged Radioactive Effluents

- Regulatory Issue Summary (RIS) 2008-03 (ML072120368)
- RIS 2008-03 states radioactive material properly released in gaseous or liquid effluent is not considered licensed material (if less than exempt concentrations)
- Note: RIS 2008-03 does not apply to solid materials
- The unlicensed material can be used and returned to the environment without being considered a new radioactive material effluent release
- Licensees are responsible for evaluating any new on-site or off-site exposure pathways created by returned/recycled material that exceed 10% of total dose (per RG 1.109 guidance)

RG 1.21 Reporting Abnormal Discharges (leaks and spills)

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- RG 1.21, Section 9.5 Supplemental Information
  - Discusses guidance on abnormal releases from plant equipment into onsite groundwater
  - Reporting thresholds include:
    - Voluntary reports under NEI 07-07
    - Abnormal discharges to the unrestricted area
  - Information should include:
    - Date, duration, volume, etc.
    - Doses to public

RG 1.21 Groundwater reference to RG 4.25

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OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE

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Revision 0

Technical Lead
Thomas Nicholson

REGULATORY GUIDE 4.25

(Draft was issued as DG-4025, dated December 2015)

ASSESSMENT OF ABNORMAL RADIONUCLIDE DISCHARGES IN GROUND WATER TO THE UNRESTRICTED AREA AT NUCLEAR POWER PLANT SITES

RG 4.25
Groundwater Discharges (GW)

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- RG 4.25 – guidance on calculating discharges from "on-site" groundwater to "off-site" groundwater

- RG 4.25 is not a calculation of releases into "on-site" groundwater

RG 4.25 - On-site Discharges into On-site Ponds

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- Some licensees dispose of liquid effluents to on-site ponds
- Releases are reported in ARERR as though releases were to the "unrestricted area" (10 CFR 50.36a)
- Most on-site ponds leak into on-site groundwater
- There is an important footnote in RG 4.25 that provides an exclusion for reporting leakage from on-site ponds into GW
  - Leakage from bottom of lake or pond to groundwater does not need to be reported (again)
  - However, the potential dose (through a new groundwater pathway) must be assessed **if** the off-site dose from leakage from onsite ponds is greater than 10% of all pathways combined (see RG 1.109)

List of Leaks and Spills (L&S)

- NRC publishes a list of L&S

  http://www.nrc.gov/reactors/operating/ops-experience/tritium/sites-grndwtr-contam.html

- 55 currently operating sites (2020 data)
- 38 sites historically have had L&S of H-3 ≥ 20,000 pCi/L reported
- 7 sites currently have residual radioactive ground water with H-3 ≥ 20,000 pCi/L

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Remediation of Leaks and Spills

- SRM-SECY-13-108, Remediation of Residual Radioactivity During Operations
- Evaluate feasibility of prompt remediation
- NRC Commission determined that prompt remediation is not a requirement

RG 4.13 Environmental Dosimetry & Direct Radiation Dose Assessment

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- RG 4.13, Rev. 2 was revised in June 2019
- NRC endorsed ANSI/HPS N13.37, ~ Environmental Dosimetry
- RG 4.13 provides an NRC-approved method of determining facility-related dose (FRD) from direct radiation
- RG 4.13 methods can be used in the demonstration of compliance with 10 CFR 20.1302 surveys requirements and EPA 40 CFR 190's dose limit of 25 mrem/yr

RG 4.13 - Data Analysis Method for Direct Radiation Using Environmental Dosimetry

- At each location, using historical data, determine the baseline background dose rate and its standard deviation (**σ**)
- Then, perform a 2-step quarterly data analysis process:
  - At each location, determine if there is there a detectable increase **greater than 3σ above the baseline dose rate?** (a yes/no question)
  - If **> 3σ,** determine the facility-related dose (FRD)
    - Subtract current quarterly reading from baseline background dose rate
    - Note: Do not subtract the **3σ value**
- Environmental dosimetry systems can measure FRD dose at: ~ 5 mrem/qtr, and ~ 10 mrem/yr

Decomm Planning Rule (DPR)

76 FR (2012) pp. 35512 – see NRC website at https://www.nrc.gov/reading-rm/doc-collections/fedreg/notices/

- The DPR was first revised in 2007 for applicants to minimize contamination
- The DPR was revised in 2012 for plant operations to minimize contamination [10 CFR 20.1406(c)]
  - Licensees shall minimize residual radioactivity (contamination), including subsurface (ground water)

10 CFR 20.1501
Radiological Surveys and Monitoring

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- § 20.1501 was revised (during the DPR - 2012) to require surveys of the "**subsurface**" (i.e., soil and ground water)
- NEI groundwater guidance documents
  - NEI 07-07 (~ Ground Water Protection Initiative (GPI))
  - NEI 08-08 (~ FSAR template for minimizing contamination), and
  - NEI 09-14 (~ Underground Pipes) are used as guidance for the ground water monitoring program defining:
    - how to "minimize" (prevent) leaks into ground water
    - how to "survey" subsurface (ground water)

Decommissioning Programs

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- Groundwater monitoring may need to be increased in support of license termination
- Licensees must maintain and update 10 CFR 50.75(g) record keeping files to include leaks and spills
- Decommissioning-related RGs
  - RG 4.22, Decommissioning Planning During Operations
  - RG 1.185, Standard Format and Content for Post-Shutdown Decommissioning Activities Report
  - NUREG-1757, Rev. 1 (2006) and draft Rev. 2 (2020) “Consolidated Decommissioning Guidance”

RG 1.21 Accident-Range Gaseous Effluent Monitoring

- RG 1.21 summarizes previously issued NRC requirements and guidance in:
  - NUREG-0660 ~ TMI Action Plan
  - NUREG-0737, “Clarification of TMI Action Plan Requirements” (ML051400209)
  - HPPOS-001, “Guidance on Calibration and Surveillance to meet Item II.F.1, Additional Accident-Monitoring Instrumentation”

Accident-Range Gaseous Effluent Monitoring

- NUREG-0737, TMI Action Plan Requirements (ML051400209)
- Item II.F.1 is “Additional Accident-Monitoring Instrumentation, requiring:
  - Noble gas effluent monitoring (Item II.F.1-1)
  - Iodine and particulate sampling and analysis (Item II.F.1-2)
  - Containment high range radiation monitoring (II.F.1-3)
- Specifications for radiation monitoring equipment are in Tables II.F.1-1, II.F.1-2, and II.F.1-3

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Accident-Range Radiation Monitors

- Three different instrument criteria to discuss:
  - Instrument design criteria
  - Instrument calibration criteria
  - Instrument measurement criteria

Three different criteria:

- Design criteria:
  - RG 1.97 establishes a factor of 2 for "design criteria," is not a "calibration criteria"
- Calibration criteria:
  - NUREG-0737 – **sufficient to perform intended function**
  - ANSI N320-1979 and IEEE-497 – generally ± 40% – ± 50%
- Measurement criteria:
  - Effluent monitors should be able to measure fresh noble gas mixtures (0 – 10 days) within overall system accuracy factor of 2
  - CHRMs should be able to measure within factor of 2

RG 1.97 footnotes
Design Criteria for Effluent Monitors

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- BWRs, RG 1.97, Rev. 2, Table 1, footnote 9
  - 0 – 10-day gas mixtures, overall system accuracy within a factor of 2
- BWRs, RG 1.97, Rev. 3, Table 2, footnote 9
  - 0 – 10-day gas mixtures, overall system accuracy within a factor of 2
- PWRs, RG 1.97, Rev. 2, Table 2, footnote 8
  - 0 – 10-day gas mixtures, overall system accuracy within a factor of 2
- PWRs, RG 1.97, Rev. 3, Table 3, footnote 9
  - 0 – 10-day gas mixtures, overall system accuracy within a factor of 2

RG 1.97, Accident Instrumentation Design Criteria for CHRMs

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- BWRs, RG 1.97, Rev. 2, Table 1, footnote 7
  - 60 keV – 100 keV within factor of 2
  - 100 keV – 3 MeV within +/-20%
- BWRs, RG 1.97, Rev. 3, Table 2, footnote 7
  - 60 keV – 100 keV within factor of 2
  - 100 keV – 3 MeV within +/-20%
- PWRs, RG 1.97, Rev. 2, Table 2, footnote 7
  - 60 keV – 100 keV within factor of 2
  - 100 keV – 3 MeV within +/-20%
- PWRs, RG 1.97, Rev. 3, Table 2, footnote 8
  - 60 keV – 3 MeV within factor of 2

Calibration of Accident-Range Radiation Monitoring Equipment

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- Health Physics Position (HPPOS-001) is a summary of the NRC guidance on meeting NUREG-0737, Item II.F.1 calibration and surveillance criteria is in https://www.nrc.gov/about-nrc/radiation/protects-you/hppos/hppos001.html

Proposed Guidance for Calibration and Surveillance Requirements to Meet Item II.F.1 of NUREG-0737

HPPOS-001 PDR-9111210074

- Actual guidance is in a letter from NRR to NRC Regional Administrators (ML103420044)

See the memorandum from D. G. Eisenhut to Regional Administrators

NUREG-0737 Item II.F.1-1
Noble Gas Effluent Monitoring

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- GM detector, scintillator or CdTe(Cl) detector output is in cpm or mR/hr
- Manufacturer provides energy response characterization from low to high gamma energy (~81 keV to 3 MeV)
- Manufacturer provides instrument response factor (efficiency factors) for Xe-133 (and Kr-85 for scintillators and CdTe(Cl) detectors)
- $- \frac{uCi/cc}{mR/hr}$ *or* $\frac{\frac{uCi}{cc}}{cpm}(Xe-133 \text{ or } Kr-85)$
- Licensees perform periodic calibration checks with a solid source

NUREG-0737 Item II.F.1-2 Iodine and Particulate Monitoring

- Real-time monitoring is not practical
- Licensees must develop procedures for collection and analysis of samples
- Iodine releases can be calculated based on partitioning (scaling) factors to noble gas

NUREG-0737 Item II.F.1-3
Containment High Range Monitor (CHRMs)

- High Range measurement is up to 10 million R/hr
- Output used in estimating containment conditions and assessing Core Damage
- Manufacturer provides the instrument response factor; e.g., ion chamber is ~ 1E-11 $\frac{amps}{R/hr}$
- Licensees perform a periodic solid source calibration check in the 1 – 10 R/hr range
- Perform electronic calibration above 10 R/hr

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Instrument Calibration Process

Effluent Monitors
Initial Vendor Calibrations

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- Vendors perform initial calibrations:
  - perform dose-rate linearity check to high dose rates
  - determine the detector's energy response characteristics
  - determine the efficiency factor (instrument response factor) (cpm per µCi/cc or mR per hour per µCi/cc) for a standard gas (Xe-133 or Kr-85)
  - build a field calibrator with a Cs-137 source for licensee's use for in-plant calibration checks
  - determine and provide a Transfer Factor (µCi/cc) / (cpm) or (µCi/cc) / (mR/hr)

In-plant calibration checks

- Instrument and Control (I&C) technicians do a one-point radiological calibration check in the first scale/decade
- I&C do electrical calibration checks for higher scales/decades
- HP normally only provides radiological support (RWP, pre-job briefings, and job coverage)
- “Instrument response factors” (efficiency factors) are normally NOT adjusted during calibration checks

Instrument Response Factors (Efficiency Factors)

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- Noble gas effluent monitoring instruments are GM detectors, plastic scintillators, and CdTe(Cl) solid-state detectors, typically ~2 mm x 2 mm x 5 mm size
- Each solid-state detector has its own counting efficiency
- GM and ion chambers are typically calibrated to Xe-133; i.e., to low energy, 81 keV photons with low yield (~36%)
- Plastic scintillators and solid-state detectors are calibrated to Xe-133 (gamma) and Kr-85 (beta)

Instrument Response Factors (Efficiency Factors)

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- Detector output is a count rate or a dose rate
- Output is converted to a Xe-133 concentration, µCi/cc
- Concentration (µCi/sec) times flow rate (cubic feet per sec)
- µCi/cc x flow rate = release rate (µCi/sec) of Xe-133

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Potential Errors in Use of Efficiency Factors (instrument response factors)

- Licensees may be using wrong calibration geometry
- Licensees may be using wrong efficiency factors; i.e., incorrectly:
  - assume 1 efficiency factor fits all detectors
  - replace detectors and do not update efficiency factors (particularly General Atomics CdTe(Cl) solid state detectors
  - apply Xe-133 efficiency factor to the radionuclide mix

Radionuclide Mix

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- Gaseous effluent is not just Xe-133
- Gaseous effluent is a mix of noble gases, and is very time dependent
- Generally, short-lived noble gas nuclides have higher energy gammas than long-lived nuclides
- Efficiency factors are 10 – 30 times than Xe-133 for higher energy gammas
- A time-dependent efficiency factor (instrument response factor) is needed

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Accident Source Term: 13 Noble Gases

There are ~ 60 different gamma energies and gamma yields from 13 noble gas nuclides to consider

6 Kryptons

- 1. Kr-83m
- 2. Kr-85m
- 3. Kr-85
- 4. Kr-87
- 5. Kr-88
- 6. Kr-89

7 Xenons

7. Xe-131m
8. Xe-133m
9. Xe-133
10. Xe-135m
11. Xe-135
12. Xe-137
13. Xe-138

~ 60 Gamma Energies

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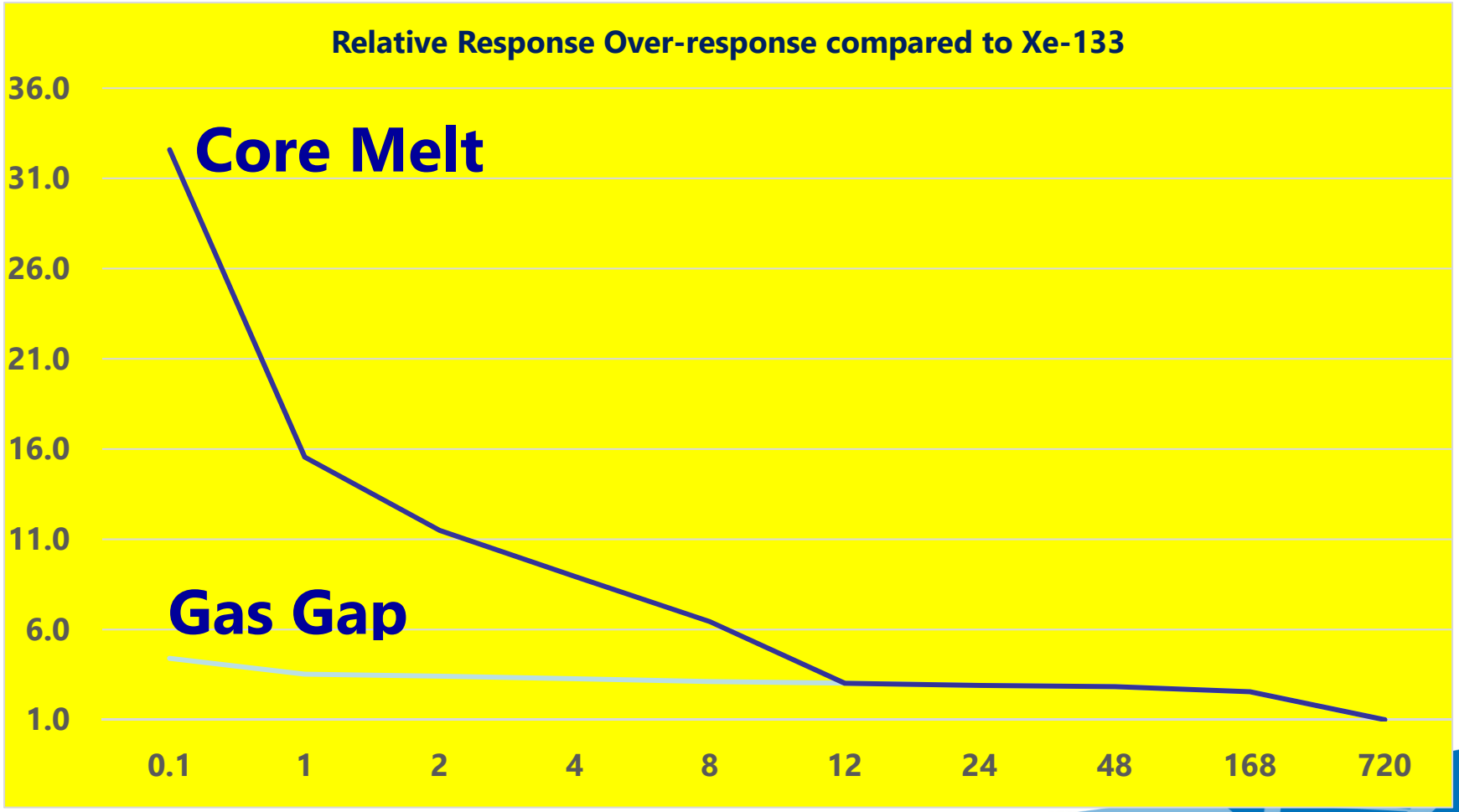
Nuclide	Half Life (hours)	gamma energy keV
Kr-83m	1.9	9
Kr-85m	4.5	150
	4.5	305
Kr-85	94,000	514
Kr-87	1.3	403
	1.3	674
	1.3	845
	1.3	1,175
	1.3	1,740
	1.3	2,010

Nuclide	Half Life (hours)	gamma energy keV
Kr-88	2.8	166
	2.8	196
	2.8	362
	2.8	835
	2.8	986
	2.8	1,000
	2.8	1,140
	2.8	1,180
	2.8	1,530

Nuclide	Half Life (hours)	gamma energy keV
Kr-89	0.053	197
	0.053	221
	0.053	345
	0.053	369
	0.053	411
	0.053	498
	0.053	586
	0.053	696
	0.053	738
	0.053	776
	0.053	836
	0.053	867
	0.053	904
	0.053	1,108
	0.053	1,117
	0.053	1,274
	0.053	1,324
	0.053	1,473
	0.053	1,501
	0.053	1,532
	0.053	1,694
	0.053	1,903

Nuclide	Half Life (hours)	gamma energy keV
	0.053	1,903
Xe-131m	288	164
Xe-133m	55	223
Xe-133	127	81
Xe-135m	0.25	527
Xe-135	9.2	250
	9.2	608
Xe-137	0.065	455
	0.065	1,491
Xe-138	0.3	153
	0.3	242
	0.3	258
	0.3	396
	0.3	401
	0.3	434
	0.3	1,114
	0.3	1,768
	0.3	1,851
	0.3	2,005
	0.3	2,016

GM Detector Instrument Response Factors (based on calibration to Xe-133)



Plant staff responsibilities

- Plant staff:
  - Some plants may not have rad-engineering expertise
  - I&C may do calibration checks without knowledge of radiological response characteristics
- Plant staff should know:
  - what equipment is installed
  - which department is in charge
  - how equipment works (vendor manuals and calibrations)
  - how calibration checks are performed
  - the basis for efficiency factors and detector specific factors
  - how monitor output interfaces with dose assessment codes

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Accident-Range Gaseous Effluent Monitoring Calibration and Time-Dependent Instrument Response Factors

ADAMS ML18171A035

Steve Garry, CHP

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Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

Radiological Effluents and Environmental Workshop
June 27, 2018
New Orleans, LA

Iodine and Particulate (I&P) Monitoring

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- NUREG-0737, TMI Action Plan Requirements, Item II.F.1-2 (ML051400209)
- Real-time iodine and particulate monitoring is not required
- However, licensees should have procedures for sample collection and analysis of hot iodine and particulate samples
- Real-time dose assessment can be performed using scaling factors to noble gas

Containment High Range Monitors (CHRM)

- CHRM measurements are used in Emergency Action Levels (EALs) and for core damage assessment
- Core damage assessment methods are in NUREG-1940, Radiological Assessment System for Consequence Analysis (RASCAL) section 1.2.8 and NUREG-1940, Supplement 1, Section 2.6
- Licensee staff perform a one-point radiological calibration check below 10 R/hr
- Licensee staff perform an electronic calibration check for each decade above 10 R/hr

NRC staff training
CHRMs

- NRC developed, provided, and recorded training on CHRMs calibration in 2021

- Training slides are publicly available at ML21327A271

Session 2 - RG 8.34, Rev. 1

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U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 8.34, REVISION 1

Issue Date: July 2022
Technical Lead: Steven Garry

MONITORING CRITERIA AND METHODS TO CALCULATE OCCUPATIONAL RADIATION DOSES

A. INTRODUCTION

Purpose

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

Applicability

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

RG 8.34
Calculating Occupational Dose

- Reasons for revision:
  - to revise the definition of the total effective dose equivalent (TEDE) as the sum of the EDEX and the committed effective dose equivalent (CEDE)
  - to provide guidance on performing prospective dose evaluations to determine the need for required monitoring to meet the occupational dose monitoring requirements of 10 CFR 20.1502
  - to provide guidance on monitoring of unplanned, unintended doses when monitoring was not performed

RG 8.34 (Cont'd)
Calculating Occupational Dose

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- Reasons for Revision (continued):
  - to provide guidance on monitoring dose from hot particles or contamination on or near the skin
  - to define the term "dosimetry processing" and explain when there are requirements for processing by an accredited National Voluntary Laboratory Accreditation Program (NVLAP) processor
  - to provide guidance on assessing dose from intakes of radioactive material by wound injuries
  - to provide guidance on calculating soluble uranium intakes

Background
DDE vs EDEX

- In 1991, 10 CFR 20 defined TEDE as equal to DDE plus CEDE
- In 2007, 10 CFR 20 was revised to define TEDE as equal to EDE (for external dose) plus the CEDE
- The term “EDEX” was coined as an acronym for EDE (for external exposure)
- 10 CFR 20.1201 occupational dose requires NRC approval for methods of calculating EDEX (other than using DDE)

10 CFR 20.1201(c)
Occupational dose limits for adults

- When the external exposure is determined by measurement with an external personal monitoring device
  - the assigned deep dose equivalent must be for the part of the body receiving the highest exposure
  - the deep dose equivalent must be used in place of the EDEX, unless the EDEX is determined by a dosimetry method approved by the NRC

RG 8.38 ~ HRAs and LHRAs

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U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
May 2006

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.38

(*Draft was issued as DG-8028, dated September 2005*)

CONTROL OF ACCESS TO HIGH AND VERY HIGH RADIATION AREAS IN NUCLEAR POWER PLANTS

A. INTRODUCTION

RG 8.38
HRAs and LHRAs

- The definition of a “high radiation area” requires that the area be “accessible” to individuals
- Accessibility is determined by whether an individual can reasonably occupy the area with a major portion of their whole body
- An area into which an individual can only insert an extremity, or a portion of an extremity (e.g., a finger) is not “accessible to individuals”
- However, the upper arm, the head, the eye, and the male gonads are considered to be major portions of the whole body

RG 8.40 ~ EDEX

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U.S. NUCLEAR REGULATORY COMMISSION July 2010

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.40

(Draft was issued as DG-8039, dated August 2009)
(New Regulatory Guide)

METHODS FOR MEASURING EFFECTIVE DOSE EQUIVALENT FROM EXTERNAL EXPOSURE

A. INTRODUCTION

This regulatory guide describes dosimetry methods that the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for determining the effective dose equivalent (EDE) for external (EDEX) radiation exposures. These methods provide a conservative estimate of the EDEX and may be used to calculate the total effective dose equivalent (TEDE) in demonstrating compliance with TEDE-based NRC regulatory requirements.

RG 8.40 ~ EDEX

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- 10 CFR 20 requires the use of the EDEX in determining TEDE
- If the DDE is used as the EDEX, the EDEX can be overly conservative
- Several methods of determining EDEX are acceptable to the NRC staff based on dose measurements on the surface of the whole body
- Each of these methods generally involves the measurement of the DDE at one or more locations on the whole body.
- The EDEX is then determined by applying a weighting factor to each dosimeter result

RG 8.40 - Compartment Factors

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- Method 1 - Using Compartment Factors

$$EDEX = \sum W_C \, DDE_C.$$

Table 1. Compartment Factors

AREA OF THE BODY/COMPARTMENT	COMPARTMENT FACTOR (W_C)
Head and neck	0.10
Thorax, above the diaphragm	0.38
Abdomen, including the pelvis	0.50
Upper right arm	0.005
Upper left arm	0.005
Right thigh	0.005
Left thigh	0.005

RG 8.40 - Compartment Factors

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- Method 1 - Using Compartment Factors
  - Each compartment is monitored separately
  - The (TLD/OSL) DDE dose measured for each compartment is then multiplied (weighted) by its “compartment factor” (see previous slide)
  - The resulting weighted doses are then summed to determine the EDEX for the whole body

RG 8.40 – EPRI Method

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- Method 2 - Use Two Dosimeters – front and back
  - a dosimeter worn on the front of the body (abdomen or thorax) is combined with a reading of a second dosimeter worn on the back of the body
  - EDEX = 3/4 High dosimeter + 1/4 Low dosimeter

RG 8.40 – Other Methods

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- Method 3 – Medical scenarios wearing an apron during medical X-ray procedures
  - Agreement States can develop their own EDEX calculational method, or
  - One monitoring device worn on the neck outside the lead apron with calculational procedures
  - Other specific methods as described in RG 8.40

RG 8.34 (Cont'd)
Prospective Dose Evaluations

- 10 CFR 20.1502- licensees need to determine if the occupational dose is "likely to exceed" 10% of the dose limits
- One method of determining "likelihood" is to do a prospective dose evaluation
- What is a prospective dose evaluation?
- Some licensees are unsure "how" to do a prospective dose evaluation

RG 8.34 (Cont'd)
Prospective Dose Evaluations

- **Section C.2 Determining the Need for Monitoring**
- **2.1 Establishing Categories of Workers for Consideration of the Need for Monitoring**

- Licensees should[1] evaluate potential exposure scenarios to determine whether annual doses to individuals are likely to exceed monitoring criteria (i.e., by performing a prospective dose evaluation

- [1] The term "should" denotes a recommendation and the term "may" denotes permission (neither a requirement nor a recommendation)

RG 8.34
Section C.2.2 Likely Exposures

- Likely exposures[1] include exposures during normal and anticipated operational occurrences
- Likely exposures do not include design-basis accidents
- Precautionary monitoring for unlikely exposures is not required
- [1] As discussed in 60 FR 36039

RG 8.34
Section C.2.3 - Prospective Evaluation of Doses Not Likely to Exceed Monitoring Criteria

- Potential exposure scenarios involving small doses:
  - may be evaluated in the prospective dose evaluation
  - may be determined as “not likely” to result in doses exceeding monitoring criteria
  - therefore, are not subject to monitoring requirements

Examples of "not likely to exceed monitoring criteria"

- If the prospective dose evaluation determined that:
  - small, unplanned, unintended extremity exposures may occur but are not likely to exceed external monitoring thresholds, or
  - that minor facial contamination or intakes may occur that are not likely to exceed internal monitoring thresholds, then
- a follow up dose evaluation is not considered required monitoring

RG 8.34, Section C.2.5
Voluntary Monitoring and Reporting

- Voluntary monitoring is monitoring beyond 10 CFR 20.1502 requirements

- The results of voluntary monitoring obtained when 10 CFR 20.1502 did not require monitoring are not subject to dose recording and reporting requirements

Section 2.6
Change in Exposure Conditions

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- If the radiation exposure conditions change during the year:
  - the need to provide individual monitoring should be reevaluated
  - if a new job assignment, a worker's dose is likely to exceed 10 percent of the annual dose limit, then the licensee should provide monitoring
  - prior dose should be estimated, recorded, and reported

Section C.2.7 Detection Sensitivity

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- The monitoring criteria in 10 CFR 20.1502 are not required levels of detection sensitivity (e.g., the lower limit of detection)
- For example, it may not be feasible to confirm intakes of 10 percent of the ALI
- This is true particularly for bioassay measurements of some alpha-emitting radionuclides

Section C.3
Determination of External Doses

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- The DDE to the whole body is considered to be at a tissue depth of 1 centimeter (cm) (1,000 milligrams per square cm (mg/cm^2)
- The SDE to the skin or extremities is to be determined at 0.007 cm (7 mg/cm^2)
- The LDE is to be determined at 0.3 cm (300 mg/cm^2)
- In evaluating the SDE and LDE, it is acceptable to take credit for the shielding provided by gloves and protective lenses

RG 8.34 Section C.3.1
Placement of dosimeters

- Licensees can use passive (TLD/OSL) or electronic dosimetry
- If a portion of the whole body if receiving substantially more dose than rest of the whole body, then move dosimetry to highest exposed portion of the whole (substantial is a judgment call)
- External dose not measured by dosimetry (e.g., from low energy gammas; e.g., Xe-133 at 81 keV) or radiation beams may be calculated per 10 CFR 20.1201(c)
- If dosimetry was not moved and substantially higher dose was received to a part of the body, the DDE dose should be estimated by calculation

RG 8.34
Section C.3.2 ~ Multi-badging

- Use of more than one dosimeter
- Used for job-specific tasks to track dose to different parts of the body
- Determine which dosimeter had the highest reading
- Add that job-specific DDE to the whole-body DDE measured by a torso dosimeter (with subtraction)

RG 8.34
Section C.3.3 ~ Determining EDEX

- When external exposure is measured with a dosimeter
- Use DDE in place of EDEX, unless…
  - EDEX is calculated by a method approved by NRC
  - RG 8.40 provides approved methods
  - Licensees may apply for use of other weighting factors, see 10 CFR 20.1003, footnote 2 below

[2] For the purpose of weighting the external whole body dose (for adding it to the internal dose), a single weighting factor, w_T=1.0, has been specified. The use of other weighting factors for external exposure will be approved on a case-by-case basis until such time as specific guidance is issued.

RG 8.34
Section C.3.4 ~ Determining SDE

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- When exposure is uniform, the SDE measured by a torso dosimeter is expected to be representative of the SDE
- If SDE is expected to differ substantially from SDE measured by the torso dosimeter, the SDE should be monitored separately
- “Substantial” is a judgment based on circumstances
- If so, licensees should use an SDE dosimeter for some, but not all radiation exposures.
- Extremity dosimeters may be worn under gloves
- Note: Credit may be taken for protective equipment (gloves and safety glasses)

RG 8.34
Section C.3.4 ~ Determining SDE

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- For hot particles, SDE may be calculated using VARSKIN+1.0
- VARSKIN+ 1.0 has added a wound model, eye dose model, and new alpha and neutron skin dose model
- The RAMP website has additional information on VARSKIN+ 1.0 at https://ramp.nrc-gateway.gov/
- The HQ ARCB (HP) SharePoint site has a presentation on the chronology and overview of VARSKIN+ 1.0 at: NRR Radiation Protection - Varskin + - All Documents (sharepoint.com)
- NUREG/CR-6918 has a full description of VARSKIN+ 1.0

RG 8.34
Section C.3.5 ~ Embryo/Fetus

- RG 8.36 provides outdated guidance on calculating dose to the Embryo/Fetus
- RG 8.36 relies mainly on the guidance provided by ICRP 30 (1992). However, this guidance is outdated
- More up-to-date models are available at:
  - ICRP Publication 73, “Radiological Protection and Safety in Medicine” (paragraphs 76 and 77), and
  - ICRP Publication 75, “General Principles for the Radiation Protection of Workers” (paragraph 124)
  - ICRP Publication 88 (2002), “Doses to the Embryo and Fetus from Intakes of Radionuclides by the Mother”

RG 8.34
Section C.3.6, EPA Federal Guidance Reports (FGR) 12 and 15

- FGR-12 and FGR-15 are methods of dose assessment for external geometry from environmental contamination (airborne, soils, ground surface, water immersion)
- The exposure geometry for environmental contamination is typically not representative of nuclear plant exposures
- FGR-12 and FGR-15 methods should normally not be used in occupational dose assessments

Section C.3.6, EPA Federal Guidance Reports (FGR) 12 and 15

- FGR-12 (1993) ~ Doses from External Environmental Contamination – EDEX, organ doses (not CDE), skin dose
  - Note: By definition, CDE is from an internal source (not an external source) so FGR-12 calls it “organ dose” in lieu of “committed” dose
- FGR-15 (2018) ~ EDEX from Environmental Contamination – organ doses from external radiation (not CDE from intakes), skin dose, and EDE from external radiation
  - 6 age groups and 1,252 radionuclides
  - Tissue weighting factors from ICRP-103 (not 10 CFR 20)

RG 8.34, Section C.3.7 Dosimetry Processing

- 10 CFR 20.1501(d) - All personnel dosimeters (except for direct and indirect reading pocket ionization chambers …) that require processing must be processed and evaluated by an NVLAP accredited dosimetry processor
- OGC and NMSS interpreted processing to mean a method, separate from and independent of the design of the dosimeter, that is required to extract dose information from the dosimeter
- Alarming dosimeters and direct ion storage dosimeters do not require processing, so NVLAP accreditation is not required

RG 8.34 Section C.4 – Internal Dose
Section C.4.1 – Assessing Intakes

- Associated Regulatory Guides:
  - RG 8.9 - determining intakes from bioassay results
  - RG 8.22 - conducting bioassay programs at uranium mills
  - RG 8.25 (not applicable to 10 CFR Part 50 licensees) - guidance on determining intakes from air sampling measurements
  - RG 8.26 provides guidance on when bioassay programs are needed for those individuals subject to internal radiation exposure monitoring requirements
  - RG 8.30 acceptable survey methods at uranium recovery facilities

Section C.4
Assessing Intakes

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Intakes can be determined two ways:

1. Air sampling (not a reliable method)

2. Whole body counting (WBC) (best method)
   - WBC determines “uptakes”
   - “Uptakes” need to be converted to “Intakes”
   - NUREG/CR-4884 does that (see next slide)

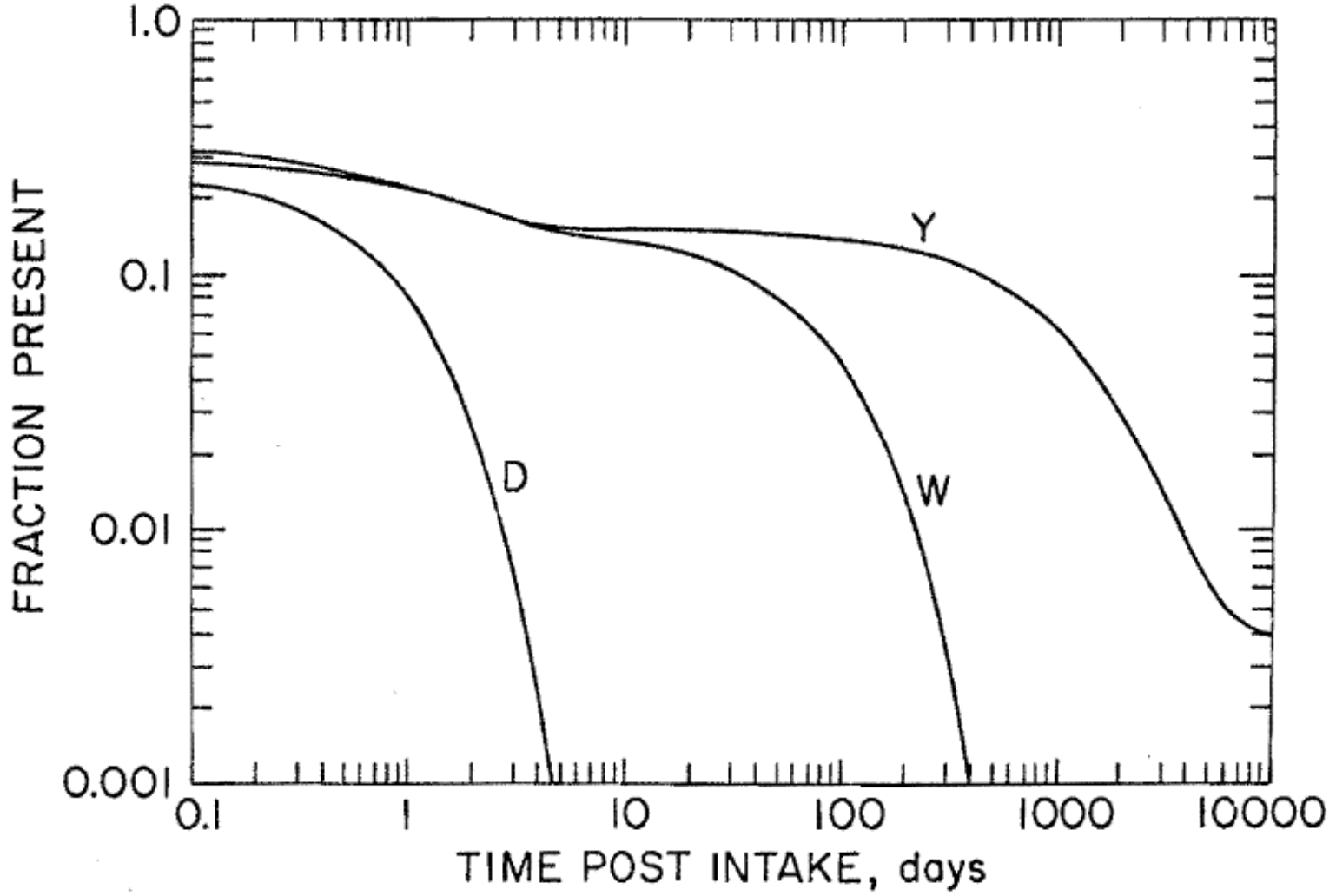
NUREG/CR-4884

U.S.NRC
United States Nuclear Regulatory Commission

NUREG/CR-4884
BNL-NUREG-52063
1988

Interpretation of Bioassay Measurements

NUREG/CR-4884
Retention of Elemental Cobalt



NUREG/CR-4884
Retention of Inhaled Co-60

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CLASS W AMAD = 1 MICRON HALFLIFE= 1.92E+03 DAYS COBALT 60

TIME AFTER SINGLE INTAKE	FRACTION OF INITIAL INTAKE IN:				
DAYS	SYSTEMIC ORGANS	LUNGS	NASAL PASSAGES	GI TRACT	TOTAL BODY
1.00E-01	6.72E-02	2.76E-01	2.35E-01	5.68E-02	6.35E-01
2.00E-01	6.42E-02	2.65E-01	1.97E-01	1.04E-01	6.30E-01
3.00E-01	6.21E-02	2.56E-01	1.66E-01	1.42E-01	6.26E-01
4.00E-01	6.04E-02	2.47E-01	1.39E-01	1.73E-01	6.20E-01
5.00E-01	5.89E-02	2.40E-01	1.17E-01	1.99E-01	6.14E-01
6.00E-01	5.76E-02	2.33E-01	9.86E-02	2.18E-01	6.07E-01
7.00E-01	5.63E-02	2.26E-01	8.29E-02	2.33E-01	5.99E-01
8.00E-01	5.51E-02	2.21E-01	6.97E-02	2.43E-01	5.89E-01
9.00E-01	5.39E-02	2.16E-01	5.86E-02	2.50E-01	5.78E-01
1.00E+00	5.28E-02	2.11E-01	4.93E-02	2.53E-01	5.66E-01

NUREG/CR-4884
Retention of Inhaled Co-60

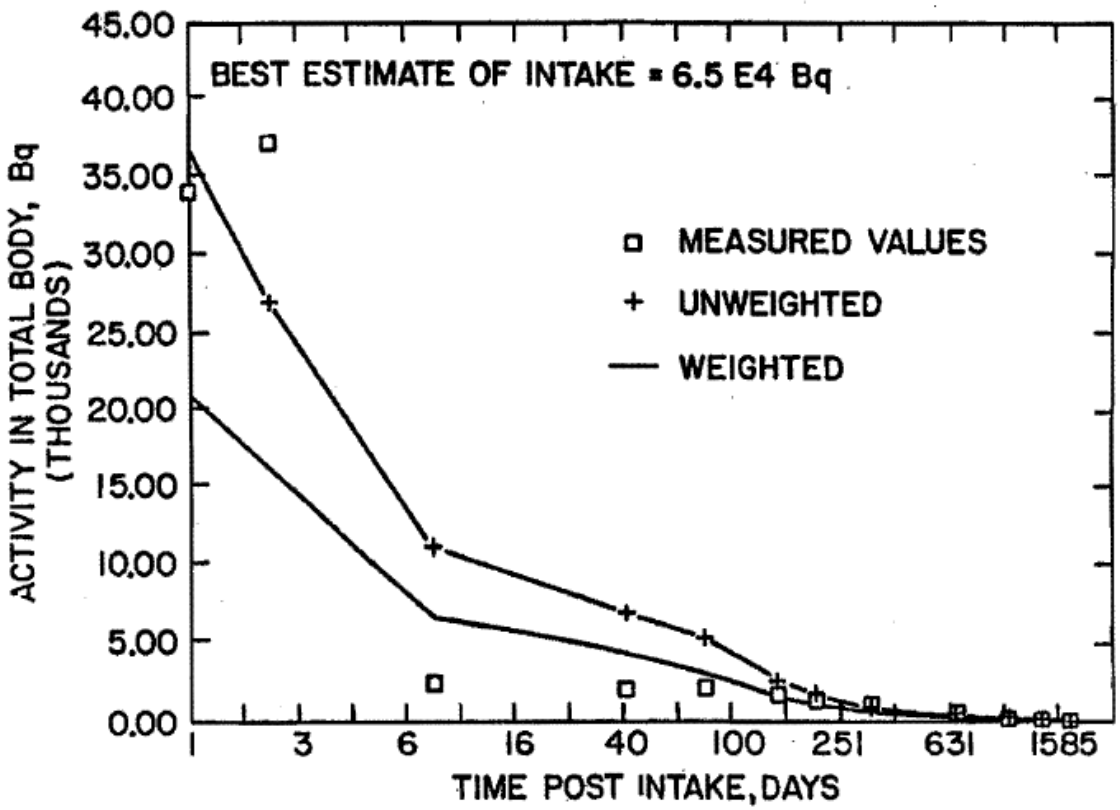

FIGURE A.7.16 Calculated and Measured Values of Total Body Activity Following Inhalation of Class W Co-60.

NUREG/CR-4884 – Co-60 fraction retained and excreted

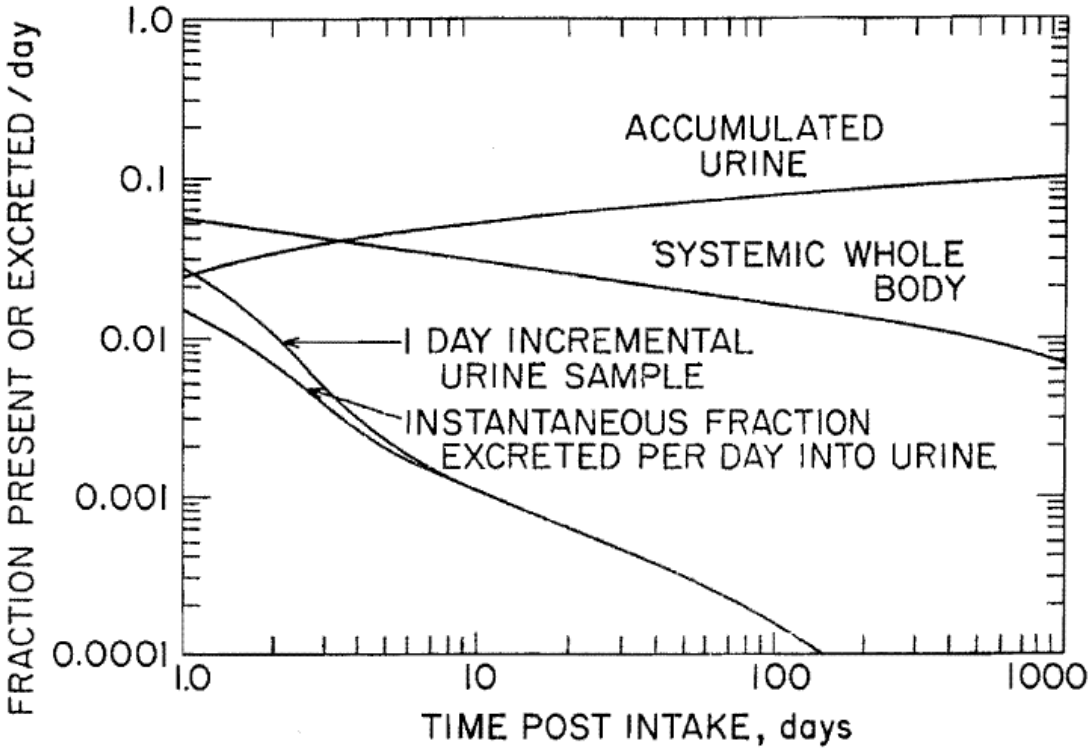

FIGURE 2.4 Systemic retention and urinary excretion post single inhalation intake of 1 micron AMAD aerosols of stable Class W cobalt for which $f_1 = 0.05$ and $f_u = 0.8$.

RG 8.34
Section C.4.2 ~ CEDE from Inhalation

- Dose from inhalation is the 50-year committed dose assigned to the year of intake

- Licensees must use the 10 CFR 20 organ or tissue weighting factors

Organ or Tissue	W_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder	[1]0.30
Whole Body	[2]1.00

RG 8.34
Section C.4.2.1 FGR-11 (1988)

EPA

Limiting Values of Radionuclide Intake And Air Concentration

and

Dose Conversion Factors For Inhalation, Submersion, And Ingestion

Comparison between 10 CFR 20 and FGR-11 for ALI and DACs

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10 CFR 20

Radionuclide	Class	Table 1 Occupational Values: Col. 1 Oral Ingestion ALI (μCi)	Table 1 Occupational Values: Col. 2 Inhalation ALI (μCi)	Table 1 Occupational Values: Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations: Col. 1 Air (μCi/ml)	Table 2 Effluent Concentrations: Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers: Monthly Average Concentration (μCi/ml)
Co-60	W 0.05		6	0.003		0.05	20
5.271 y	Y 0.05		1	$5\ 10^{-4}$		0.3	7

- FGR-11

Table 1.a, Cont'd.

Nuclide	Class/f_1	Inhalation ALI MBq	Inhalation DAC MBq/m³	Ingestion f_1	Ingestion ALI MBq
Co-60	W 0.05	6	0.003	0.05	20
5.271 y	Y 0.05	1	$5\ 10^{-4}$	0.3	7

FGR-11
Description of Lung Models

- 10 CFR 20, Appendix B is based in part on ICRP-30

- FGR-11 provides a simplified description of the ICRP-2 lung model and ICRP-30 lung models

FGR-11 (pg 13)
Description of ICRP-2 lung model

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- ICRP-2 lung model for inhalation
  - 25% is exhaled
  - 25% is deep into lungs
  - 50% is cleared from the throat into GI tract
  - Considers material as either soluble or non-soluble

FGR-11
ICRP-30 Lung Model

- A refined lung model
  - Considers particle size AMAD (activity median aerodynamic diameter)
  - AMAD is assumed to be 1 μm (diameter of a particle)
  - Activity transferred to the GI tract is calculated based on linear differential equations
  - Solubility (clearance from lung) is classified as days (D), weeks (W) or years (Y)
  - ICRP-23 describes transit times through GI tract

RG 8.34 Section C.4.2.1
Use of FGR 11 to calculate CEDE

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- FGR-11 (~inhalation and ingestion)
  - Uses the same organ weighting factors as 10 CFR 20
  - Table 2.1 provides inhalation dose coefficients
  - Table 2.2 provides ingestion dose coefficients
  - Dose coefficients are expressed in Sv/Bq
  - Convert to mrem/μCi by multiplying the listed Sv/Bq values by 3.7E9

FGR-11 - Inhalation
CDE and CEDE values for Co-60

- Table 2.1 (pg 124) – f_1 is the fraction transfer from GI tract to blood
  - Class W - f_1 is 0.05 (5% transfer to blood)
  - Class Y - f_1 is 0.05 (5% transfer to blood)

Table 2.1, Inhalation, Cont'd.

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

Comparison between 10 CFR 20 and FGR-11 for inhalation dose

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- 10 CFR 20
  - Appendix B does not provide dose factors
  - RG 1.109 uses ICRP-2 dose methods
- FGR-11

TABLE 2.1

Exposure-to-Dose Conversion Factors for Inhalation

Table 2.1, Inhalation, Cont'd.

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

RG 8.34 Section C.4.2.2
Use Stochastic ALIs to calculate CEDE

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- 10 CFR 20, Appendix B provides ALIs
- stochastic ALI (SALI)
- nonstochastic ALI (NALI)
- The more limiting ALI is listed first in App B
- CEDE = $$H_{i,E} = \frac{5 \cdot I_i}{ALI_{i,stoc}}$$

where

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

I_i = intake of radionuclide i by inhalation during the calendar year (µCi)

$ALI_{i,stoc}$ = stochastic inhalation ALI of radionuclide i (µCi)

5 = CEDE from intake of one inhalation $ALI_{i,stoc}$ (rem)

RG 8.45 Section C.4.2.3
Using DACs to calculate CEDE

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- Licensees perform air sampling to measure airborne concentrations and track exposure times (e.g., Rx Bldg entries)
- Calculate CEDE ($H_{i,E}$)

$$H_{i,E} = \frac{5 \cdot C_i \cdot t}{2000 \cdot DAC_{i,stoc}}$$ where

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

RG 8.34, Section C.4.2.4
Use of ICRP-30 to calculate CEDE

- ICRP 30, Part 1, "Supplement" lists the CDE and the weighted CDE (the CDE multiplied by its weighting factor)

WEIGHTED COMMITTED DOSE EQUIVALENT IN TARGET ORGANS OR TISSUES
PER INTAKE OF UNIT ACTIVITY (Sv/Bq) OF CO-60M

ORAL		INHALATION	
		CLASS W	CLASS Y
f_1=5.E-02	f_1=3.E-01	f_1=5.E-02	f_1=5.E-02
ST WALL	ST WALL	LUNGS	LUNGS
8.2E-13	8.2E-13	3.4E-13	5.0E-13
SI WALL	SI WALL		
1.2E-13	1.2E-13		

RG 8.34, Section C.4.2.5 ~ Exceptions

- 10 CFR 20.1204(c) allows the use of individual or material specific Information
  - Physical properties (e.g., known AMAD values)
  - Biochemical properties (chemical form)
  - NRC approval is NOT necessary
  - Caveats – must use:
    - 10 CFR 20 tissue or organ weighting factors
    - Records must be kept per 10 CFR 20.2106(a)(4)

RG 8.34, Section C.4.3
Calculating CEDE using 10 CFR 20, App B (Ingestion Doses)

- 10 CFR 20, Appendix B only provides 1 ingestion ALI value for each radionuclide, regardless of chemical form

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

RG 8.34, Section C.4.3.1
Use FGR-11, Table 2.2 Ingestion

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- Use of FGR-11, Table 2.2 (pg 157) - Ingestion
- There is no Class D, W, or Y for ingestion
- f_1 is the fraction transfer from GI tract to the blood - f_1 is 0.05 (5%) transferred
- First (f_1) value is for oxides, second (f_1) value is organics

Table 2.2. Exposure-to-Dose Conversion Factors for Ingestion

		Committed Dose Equivalent per Unit Intake (Sv/Bq)							
Nuclide	f_1	Gonad	Breast	Lung	R Marrow	B Surface	Thyroid	Remainder	Effective
Co-60	$5\ 10^{-2}$	$3.19\ 10^{-9}$	$1.10\ 10^{-9}$	$8.77\ 10^{-10}$	$1.32\ 10^{-9}$	$9.39\ 10^{-10}$	$7.88\ 10^{-10}$	$4.97\ 10^{-9}$	$2.77\ 10^{-9}$
	$3\ 10^{-1}$	$7.23\ 10^{-9}$	$5.08\ 10^{-9}$	$4.96\ 10^{-9}$	$5.49\ 10^{-9}$	$4.81\ 10^{-9}$	$4.68\ 10^{-9}$	$1.06\ 10^{-8}$	$7.28\ 10^{-9}$

RG 8.34, Section C.4.3.2
Stochastic ALIs for CEDE Ingestion

- Stochastic ALIs can be used to calculate

CEDE = $$H_{i,E} = \frac{5 \cdot I_i}{ALI_{i,stoc}}$$

where:

$H_{i,E}$	=	CEDE from radionuclide i (rem)
I_i	=	intake of radionuclide i by ingestion during the calendar year (μCi)
$ALI_{i,stoc}$	=	value of the stochastic ingestion ALI for the CEDE from column 1 of table 1 in Appendix B to 10 CFR Part 20 (μCi)
5	=	CEDE from an annual intake of 1 ALI (rem)

RG 8.34, Section C.4.3.3
Use ICRP-30 for CEDE - Ingestion

- ICRP-30, Part 1 supplement, lists the CDE and the "weighted" CDE per unit intake (Sv/Bq)
- CEDE is the sum of weighted CDE
- See next slide for Co-60 example

ICRP-30, Supplement to Part 1
Co-60 "weighted" CDE

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WEIGHTED COMMITTED DOSE EQUIVALENT IN TARGET ORGANS OR TISSUES
PER INTAKE OF UNIT ACTIVITY (Sv/Bq) OF CO-60

ORAL		INHALATION	
		CLASS W	CLASS Y
f_1=5.E-02	f_1=3.E-01	f_1=5.E-02	f_1=5.E-02
ULI WALL 3.4E-10	ULI WALL 5.7E-10	LIVER 5.5E-10	
LLI WALL 6.7E-10	LLI WALL 8.1E-10	REMAINDER 4.8E-10	
LIVER 1.4E-10	LIVER 7.7E-10		
REMAINDER 1.2E-10	REMAINDER 5.2E-10		

ICRP publications

- ICRP-2 (1959) ~ 5 (N-18) and 3 rem/quarter, MPCs, internal dose and use of critical organ concept
- ICRP-26 (1977) ~ Recommendations
  - Stochastic, non-stochastic, dose-equivalent, committed dose equivalent, ALIs, tissue weighting factors, planned special exposures, occupational dose of 5 rem/yr, public dose of 500 mrem/yr and ALARA concepts
- ICRP-30 ~ Limits for intakes by workers
  - New lung model (chapter 5), new GI Tract model (chapter 6), new bone dosimetry model, etc.
  - New metabolic data to use in the models to calculate DACs and ALIs

RG 8.34, Section C.4.3.4
Ingestion CEDE using Individual or Material Specific Information

- 10 CFR 20.1204(c) allows use of specific information on physical or biochemical properties
- Individual or material specific information in not commonly available for nuclear power plants
- Sometimes material specific information is used by fuel facilities

RG 8.34, Section C.4.4
CDE is calculated if CEDE > 1 rem

- CDE only needs to be calculated if the CEDE exceeds 1 rem or an overexposure occurred

- If there is no overexposure, the 50 rem CDE cannot be exceeded unless the CEDE exceeds 1 rem

RG 8.34, Section C.4.4.1
Use FGR-11 to calculate CDE

- FGR-11, Table 2.1 for inhalation
- FGR-11, Table 2.2 for ingestion
- CDE = $H_{i,T} = I_i \cdot DCF_{i,T} \cdot 3.7E6$
- CDE (rem) = µCi x Sv/Bq x 100 rem/Sv x 3.7E4 Bq/uCi

RG 8.34, Section C.4.4.2
Use of nonstochastic ALI to calculate CDE

- Appendix B lists the nonstochastic ALIs

Table 1
Occupational Values

Atomic No.	Radionuclide	Class	Col. 1 Oral Ingestion ALI (μCi)	Col. 2 Inhalation ALI (μCi)	Col. 3 Inhalation DAC (μCi/ml)
53	Iodine-131	D, all compounds	3E+1 Thyroid (9E+1)	5E+1 Thyroid (2E+2)	2E-8 -

- CDE = $H_{i,T} = \frac{50 \cdot I_i}{ALI_{i,T}}$

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

RG 8.34, Section C.4.4.3
Use of DACs to calculate CDE inhalation

- For nonstochastic radionuclides, the DAC is listed in 10 CFR 20, Appendix B

			ALIs uCi		DAC uCi/ml
53	Iodine-131	D, all compounds	3E+1 Thyroid (9E+1)	5E+1 Thyroid (2E+2)	2E-8 -

- CDE = $$H_{i,T} = \frac{50 \cdot C_i \cdot t}{2000 \cdot DAC_i}$$

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

RG 8.34, Section C.4.4.4
Use ICRP-30 to calculate CDE

- ICRP-30 Supplement to Part 1 lists the CDE per Sv/Bq for both ingestion and inhalation

COMMITTED DOSE EQUIVALENT IN TARGET ORGANS OR TISSUES
PER INTAKE OF UNIT ACTIVITY (Sv/Bq) OF CO-60

ORAL		INHALATION	
		CLASS W	CLASS Y
f_1=5.E-02	f_1=3.E-01	f_1=5.E-02	f_1=5.E-02
GONADS	GONADS	GONADS	LUNGS
3.2E-09	7.2E-09	4.0E-09	3.4E-07
		(35, 21, 44)	(0, 0,100)
BREAST	BREAST	BREAST	
1.1E-09	5.1E-09	4.2E-09	
		(19, 17, 64)	
R MARROW	R MARROW	R MARROW	
1.3E-09	5.5E-09	4.2E-09	
		(20, 17, 63)	
LUNGS	LUNGS	LUNGS	
8.7E-10	5.0E-09	3.6E-08	
		(2, 2, 96)	

RG 8.34, Section C.4.4.5
Use Individual or Material Specific Values for CDE

- 10 CFR 20.1204(c) allows use of individual or material specific values for CDE
- Physical or biochemical properties may be taken into account

RG 8.34, Section C.4.5 Wound Dose

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- Decontamination and radiological assessment should not interfere with medical treatment
- Radiological wound injuries are considered intakes under 10 CFR 20.1202(d)
- VARSKIN+ 1.0 has a wound model, neutron dosimetry and eye dosimetry models) – see NUREG/CR-6918, Rev 4
- Potential types of wound doses
  - Tissue dose (volumetric)
  - Skin dose
  - Whole body dose from uptake
- ALIs do not apply since wounds are not an ingestion or inhalation exposure

RG 8.34, Section C.4.5
NCRP-156 Wound Model

- NCRP-156 ~ Wound Dosimetry & Assessment
  - analytical methods and parameters may be used to assess dose if consistent with NRC regulations
  - Chapter 5 provides dosimetry methods for skin dose and 10 mm (0.5 cm^3) sphere
- NCRP-156 recommends a default volume of 1 cm^3 when exact volume is unknown
- NCRP-156 also provides an extensive list of references

RG 8.34, Section C.4.5.1
Dose limits for wound intakes

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- 10 CFR 20 dose limits were not designed for wound dose exposures:
  - TEDE = 5 rem
    - EDEX = 0 since external exposure
    - CEDE is approx. zero since unlikely to have soluble uptake from wounds
  - TODE = 50 rem
    - DDE component = zero since DDE is external exposure
    - CDE component needs to be assessed
    - CDE component likely to be minimal as dose is averaged over the volume of the exposed organ

RG 8.34, Section C.4.5.2
Calculating Skin Dose from wounds

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- Assess dose to basal layer of skin at 7 mg/cm^2 (70 µm) from embedded RAM
- NCRP-130 provides estimated skin thickness for different parts of the body (e.g, calluses)
- NCRP-156 describes external skin dosimetry models
- NCRP-156, table 5.1 provides external skin dose coefficients to 1 cm^2 surface area

RG 8.34, Section C.4.5.2
Calculating skin dose from wounds

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- Dose limits (Cont'd)
  - SDE = 0, since SDE is an external exposure
  - CDE to the skin is minimal since CDE is averaged over the volume of the skin
- Conclusion:
  - dose limits are not a major consideration in evaluation wound injuries
  - However, dose assessments should be performed for medical purposes

RG 8.34, Section C.4.5.3
Calculating CDE to local tissues

- Muscle tissue is most likely to be exposed
- Muscle “tissue” has no real dose limit (due to dose averaging over volume of the tissue)
- Dose assessment is primarily in support of medical diagnosis
- Dose is primarily from beta and alpha (not gamma)
- For comparison, NCRP-156, table 5.2 provides dose rates to a 10 mm (0.5 cm^3) tissue sphere around a point source

RG 8.34, Section C.4.5.4
Calculating CEDE from wounds

- Organ dose should be assessed if:
  - wound source is near an organ (e.g., thyroid or lung)
  - an uptake has occurred into an organ
  - assessed as the 50-year committed dose
  - dose is assessed over the entire organ or tissue per ICRP-26

RG 8.34, Section C.4.6
Dose from absorption through skin

- Tritium DAC values in 10 CFR 20, App B include absorption through the skin
- Absorption of other RAM through the skin is negligible compared to intake through inhalation
- Intakes through the skin should be considered when liquid solutions with RAM come into contact with the skin

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Questions & Discussion