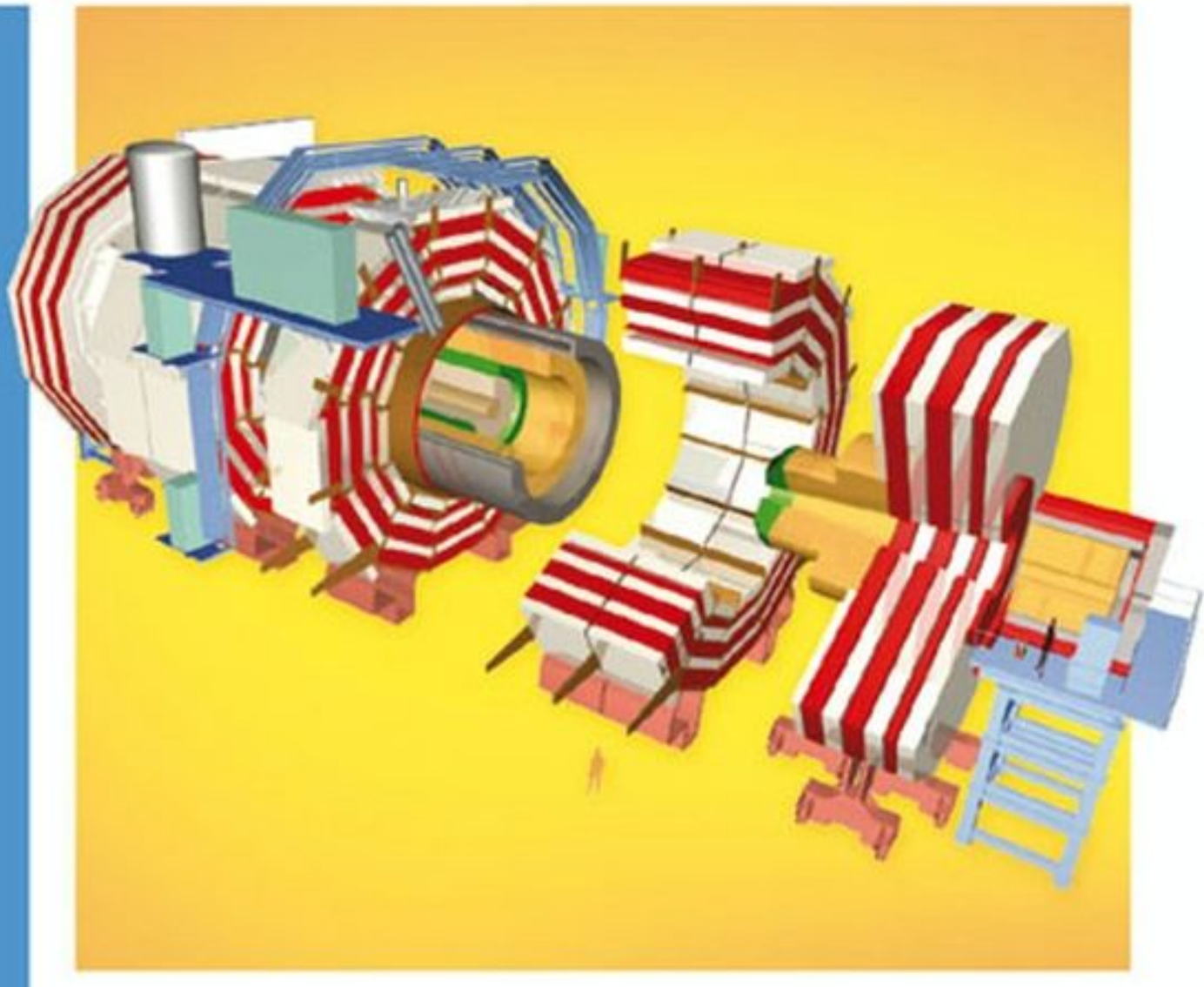
Joseph John Bevelacqua

WILEY-VCH

Contemporary Health Physics

Problems and Solutions

Second, Updated and Enlarged Edition



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Contemporary Health Physics

Problems and Solutions

Second, Updated and Enlarged Edition



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Cover

3-D cutaway view of the compact muon solenoid (CMS) at the Large Hadron Collider at CERN (CMS) detector

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This book is dedicated to my wife, Terry. Her love and understanding have been of great assistance to the completion of this text.

Preface to the Second Edition

The second edition of *Contemporary Health Physics: Problems and Solutions* has several new features. There is a new chapter on nonionizing radiation and four new appendixes. The new appendixes provide a compilation of key health physics relationships, a discussion of production equations and their applications, a mathematical review, and a set of radionuclides of health physics significance and their associated data. The text also incorporates a more extensive use of SI units. New problems are cast within an SI format, and a number of first edition problems are converted to SI units.

Additional discussion has been added to Chapters 2–7 to reflect new health physics recommendations, new reports, and emerging technologies. In the Medical Health Physics chapter, discussion has been added regarding neutron and heavy-ion therapy and the use of alpha-emitting radiopharmaceuticals. Recent recommendations of the National Council on Radiation Protection and Measurements (NCRP) regarding shielding in imaging and therapy facilities, and management of radionuclide therapy patients are incorporated into the second edition.

A section on research reactors and an expanded presentation of fusion energy have been added to the University Health Physics chapter. Discussion of open and closed fuel cycles and the Tokai Mura criticality were added to the Fuel Cycle chapter. The Power Reactor chapter has been enhanced by including a description of Generation I, II, III, and IV reactors, major instrumentation systems, updated radioactive waste processing approaches, and new NCRP hot-particle recommendations.

The Environmental Health Physics chapter has been expanded to address the isotopes and pathways associated with reprocessing options in open and closed fuel cycles and the intentional dispersal of radioactive material. A presentation of muon colliders and associated neutrino doses, synchrotron light sources, cascade reactions, the Large Hadron Collider, and ALARA aspects of shielding high-energy accelerators have been added to the Accelerator Health Physics chapter.

The problem and solution set was expanded from 375 to over 500 entries including nonionizing radiation questions. This expansion further develops the text material and provides additional practical application examples. These problems also attempt to capture the evolving nature of the Part II American Board

of Health Physics Certification Examination. Detailed solutions are provided for all problems.

In addition to new NCRP and International Commission on Radiological Protection (ICRP) Reports, a discussion of updated internal dosimetry models, including the human alimentary tract and the human respiratory tract models, have been incorporated into Appendix IV. Appendix IV has also been revised to include a discussion of the 2007 Recommendations of the ICRP. Appendix V has been expanded to incorporate a discussion of the BEIR VII Report that evaluates health risks from exposure to low levels of ionizing radiation.

Since one of the purposes of this text is to maintain the technical focus for students preparing for the American Board of Health Physics Certification Examinations, the majority of the problems were derived from questions that appeared on previous examinations. As a prior panel member, vice chair, and chair of the Part II Examination Panel, I would like to thank my panel and all others whose exam questions have been utilized in formulating questions for this textbook.

The author is also fortunate to have worked with colleagues, students, mentors, and teachers who have shared their wisdom and knowledge, provided encouragement or otherwise influenced the content of this text. The following individuals are acknowledged for their assistance during the author's career: Dick Amato, John Auxier, Lee Booth, Ed Carr, Paul Dirac, Bill Halliday, Tom Hess, Gordon Lodde, Bob Nelson, John Philpott, Lew Pitchford, John Poston, John Rawlings, Don Robson, Bob Rogan, Mike Slobodien, Jim Tarpinian, Jim Turner, and George Vargo. The continuing encouragement of my wife Terry is gratefully acknowledged.

I would also like to thank the staff of Wiley-VCH with whom I have enjoyed working, particularly Anja Tschörtner, Ulrike Werner, Hans-Jochen Schmitt and Dr. Alexander Grossmann. The advice and encouragement of George Telecki of John Wiley and Sons, Inc. is also acknowledged.

Richland, Washington USA
June 15, 2008

Joseph John Bevelacqua, PhD, CHP
President,
Bevelacqua Resources

Preface to the First Edition

This book contains over 375 problems in health physics and discusses their practical applications. It assumes that the reader is familiar with the science of radiation protection and is either an active participant in that field or interested in learning more about the health physics profession. In particular, this text is particularly useful to individuals preparing for the American Board of Health Physics Certification Examination.

The first part of this book provides an overview of the scientific basis for the field of health physics. The reader is provided with a comprehensive set of references supplemented by appendices that outline selected concepts required to fully appreciate the specialized Part II material. Over 130 problems and their solutions are provided to permit the reader to demonstrate a sound knowledge of health physics fundamentals. The problems are set within scenarios that are intended to enhance the reader's existing knowledge by demonstrating the basic principles in complex situations requiring a sound knowledge of both theoretical health physics principles and good judgment.

Part II provides the reader with examples of the concepts and calculations frequently encountered in the various fields of health physics. Chapter titles are selected to loosely conform to the various subfields of the health physics profession – that is, medical, university, fuel cycle, power reactor, environmental, and accelerator health physics. The problems are intended to illustrate general concepts within the framework of specific areas such as medical or power reactor health physics.

In addition to illustrating the fundamental concepts of health physics, the collection includes a large number of detailed problems that are often encountered by the radiation protection professional. Some of these problems involve considerable effort, whereas others are more simplistic and can be solved from traditional lectures in health physics. In addition, there are problems which address topics not usually covered in existing texts. These problems are not presented as isolated bits of health physics knowledge, but are introduced within a scenario that stimulates an integrated professional approach to the problem. Professional judgment and sound health physics principles are emphasized.

The third part of this book provides the solutions to the problems presented in the first and second parts. Many of these are worked in considerable detail to further illustrate and emphasize the concepts introduced in Parts I and II.

The present collection of problems is largely based upon the American Board of Health Physics Comprehensive Examination. The author was privileged to serve for four years as a member, Vice-Chairman, and Chairman of the ABHP Comprehensive Panel of Examiners. The experience gained in the development of this examination and the weaknesses of candidates attempting this examination have affected the content of this work.

The author is deeply indebted to the members of the examination panels and the ABHP Board for their professional interaction which greatly expanded the author's own health physics knowledge. The opinions and interpretations reflected in this work are the author's and do not necessarily reflect those of his current or previous employers.

Wisconsin Electric Power Company *Joseph John Bevelacqua*

A Note on Units

In the United States many regulations, most reporting requirements, and a large portion of practicing health physicists utilize traditional units (Ci, R, rad, rem, etc.). The use of traditional units is currently in conflict with much of the international community and scientific publications which have adopted the SI system (Bq, C kg^{-1}, Gy, Sv, etc.).

The Second Edition of *Contemporary Health Physics* utilizes both traditional and SI units. Traditional units are selected because they are what the practicing health physicist will most frequently encounter in daily assignments and they can be easily related to their SI counterparts. Traditional units are also utilized to ensure that communications between the health physicist and the health physics technician are clearly understood.

The Second Edition attempts to strike a balanced view of units. All new problems have been recast in terms of SI units. This acknowledges the inevitable adoption of the international set of units. Although many of the existing problems are in traditional units, a number of First Edition problems have been converted to the SI system.

The conflict of units will remain until the United States adopts the SI system in its regulations. This should be done over a period of years in order to ensure that all health physicists are thoroughly familiar and comfortable with the SI units.

For those that feel more comfortable with the traditional system, the following conversion factors are provided:

SI Unit	Traditional Unit
Bq	2.70×10^{-11} Ci
Gy	100 rad
C kg^{-1} of air	3881 R
Sv	100 rem

As the reader can quickly note, the choice of units is more a matter of familiarity rather than scientific rigor. By using these simple factors, the reader should begin to feel more comfortable with either set of units.

Contents

Preface to the Second Edition *VII*

Preface to the First Edition *IX*

A Note on Units *XI*

Part I Basic Concepts: Theory and Problems *1*

1 Introduction *3*
1.1 Scenarios *4*
Scenario 1.1 *4*
Scenario 1.2 *4*
Scenario 1.3 *5*
Scenario 1.4 *5*
Scenario 1.5 *7*
Scenario 1.6 *8*
Scenario 1.7 *9*
Scenario 1.8 *9*
Scenario 1.9 *10*
Scenario 1.10 *10*
Scenario 1.11 *11*
Scenario 1.12 *12*
Scenario 1.13 *13*
Scenario 1.14 *14*
Scenario 1.15 *15*
Scenario 1.16 *15*
Scenario 1.17 *17*
Scenario 1.18 *17*
Scenario 1.19 *18*
Scenario 1.20 *19*
Scenario 1.21 *20*
Scenario 1.22 *21*

Scenario 1.23 *22*
Scenario 1.24 *23*
Scenario 1.25 *24*
Scenario 1.26 *25*
Scenario 1.27 *26*
Scenario 1.28 *28*
Scenario 1.29 *28*
Scenario 1.30 *29*
Scenario 1.31 *31*
Scenario 1.32 *33*
Scenario 1.33 *34*
Scenario 1.34 *35*
Scenario 1.35 *36*
Scenario 1.36 *37*
Scenario 1.37 *38*
Scenario 1.38 *39*
References *40*

Part II Specialized Areas: Theory and Problems *45*

2 Medical Health Physics *47*
2.1 Historical Perspective *47*
2.2 Medical Accelerator Physics *48*
2.3 Diagnostic Nuclear Medicine *51*
2.3.1 X-rays *51*
2.3.2 Nuclear Medicine *52*
2.3.3 Computed Tomography *52*
2.3.4 Tracer Studies and Radioisotope Administration *53*
2.4 Therapeutic Nuclear Medicine *53*
2.4.1 Radionuclide Administration *53*
2.4.2 External Beam Therapy *55*
2.4.3 Brachytherapy *57*
2.5 Facility Design *58*
2.6 Shielding Design *59*
2.7 X-ray Shielding *60*
2.8 NCRP-49 *61*
2.8.1 Primary Barrier *61*
2.8.2 Secondary Barrier *62*
2.8.3 Leakage Radiation *63*
2.9 NCRP-147 *64*
2.9.1 Unshielded Air Kerma *64*
2.9.2 Shielding Calculations *65*
2.10 NCRP-151 *66*
2.10.1 Primary Barrier *67*

2.10.2 Secondary Barriers *68*
2.10.3 Scattering *68*
2.10.4 Leakage *68*
2.11 Management of Radionuclide Therapy Patients *69*
2.12 Ventilation Considerations *70*
2.13 Scenarios *71*
Scenario 2.1 *71*
Scenario 2.2 *72*
Scenario 2.3 *73*
Scenario 2.4 *75*
Scenario 2.5 *76*
Scenario 2.6 *76*
Scenario 2.7 *77*
Scenario 2.8 *78*
Scenario 2.9 *79*
Scenario 2.10 *82*
References *82*

3 University Health Physics *87*
3.1 Research Utilizing Radionuclides *87*
3.1.1 H-3 *88*
3.1.2 C-14 *88*
3.1.3 P-32 *88*
3.1.4 Co-60 *89*
3.1.5 I-125/I-131 *89*
3.1.6 Cf-252 *90*
3.2 Engineering Considerations *90*
3.2.1 Engineering Controls *90*
3.3 Sample Counting *91*
3.4 Intake of Radionuclides *92*
3.5 Other Research Activities *93*
3.6 Agricultural/Environmental Research *93*
3.7 Research Reactors *94*
3.7.1 Operational Characteristics *94*
3.7.2 Reactor Systems and Associated Radionuclide Production *96*
3.7.3 Reactor Effluents *96*
3.7.4 Gaseous Effluents *96*
3.7.5 Liquid Effluents *97*
3.8 Particle Accelerators *97*
3.9 Materials Research Via X-ray Diffraction Techniques *97*
3.10 Fusion Energy Research *98*
3.11 Overview of an Initial Fusion Power Facility *100*
3.11.1 General Radiological Characteristics *101*
3.11.2 ALARA-Confinement Methods and Fusion Process Types *102*
3.12 Scenarios *103*

Scenario 3.1 *103*
Scenario 3.2 *105*
Scenario 3.3 *106*
Scenario 3.4 *108*
Scenario 3.5 *110*
Scenario 3.6 *111*
Scenario 3.7 *112*
Scenario 3.8 *113*
Scenario 3.9 *113*
Scenario 3.10 *114*
Scenario 3.11 *115*
References *116*

4 Fuel Cycle Health Physics *119*
4.1 Radiation in Fuel Cycle Facilities *119*
4.1.1 Occupational Exposure *120*
4.2 Nuclear Fuel Cycle *121*
4.3 Uranium Fuel Cycle *121*
4.3.1 Open and Closed Fuel Cycles *123*
4.3.2 Uranium Ore and Chemical Processing *124*
4.3.3 Enrichment *125*
4.3.4 Gaseous Diffusion *125*
4.3.5 Gas Centrifuge *127*
4.3.6 Laser Isotope Separation *128*
4.3.6.1 MLIS 130
4.3.6.2 AVLIS 130
4.3.7 Spent Power Reactor Fuel *131*
4.4 Thorium Fuel Cycle *131*
4.5 Radioactive Waste *132*
4.5.1 High-Level Wastes *133*
4.6 Criticality *133*
4.6.1 Tokaimura Criticality *134*
4.6.2 Critical Mass *135*
4.6.3 Geometry or Shape *136*
4.6.4 Enrichment of the Fissile Isotope *136*
4.6.5 Moderation and Reflection *136*
4.6.6 Neutron Absorbers or Poison Material *137*
4.6.7 Consequences of a Criticality Event *138*
4.7 Dispersion of Radioactive Gas from a Continuous Source *138*
4.8 Dispersion of Radioactive Particulates from a Continuous Source *140*
4.9 Fuel Cycle Facilities *140*
4.10 Detection of Fuel Cycle Facility Activity *142*
4.11 Scenarios *143*
Scenario 4.1 *143*
Scenario 4.2 *144*

Scenario 4.3 *147*
Scenario 4.4 *149*
Scenario 4.5 *150*
Scenario 4.6 *151*
Scenario 4.7 *151*
Scenario 4.8 *153*
Scenario 4.9 *154*
References *155*

5 Power Reactor Health Physics *157*
5.1 Overview *157*
5.2 Generation I, II, III, and IV Reactors *158*
5.3 Health Physics Hazards *160*
5.3.1 Buildup of Filter or Demineralizer Activity *161*
5.3.2 Activation of Reactor Components *162*
5.3.3 Cladding Failures *163*
5.3.4 Reactor Coolant System Leakage *164*
5.3.5 Hot-Particle Skin Dose *165*
5.4 NCRP-130 Hot Particle Recommendations *169*
5.5 Health Physics Program Elements *170*
5.5.1 ALARA *170*
5.5.2 Effluents *171*
5.5.3 Radioactive Waste *172*
5.5.4 Decontamination *172*
5.6 Outages *176*
5.7 Major Radiation Instrumentation Systems *176*
5.7.1 Primary System Monitors *176*
5.7.2 Secondary System Monitors *177*
5.8 Radiological Considerations During Reactor Accidents *178*
5.9 Mitigation of Accident Consequences *180*
5.10 Scenarios *181*
Scenario 5.1 *181*
Scenario 5.2 *183*
Scenario 5.3 *185*
Scenario 5.4 *188*
Scenario 5.5 *190*
Scenario 5.6 *192*
Scenario 5.7 *194*
Scenario 5.8 *195*
Scenario 5.9 *196*
References *197*

6 Environmental Health Physics *201*
6.1 Naturally Occurring Radioactive Material *201*
6.2 Radon *203*

6.2.1 Buildup of Radon from Inleakage *206*
6.2.2 Evolution of Radon from the Household Water Supply *206*
6.2.3 Radon Risk Assessments *207*
6.3 Environmental Monitoring Programs *207*
6.4 Environmental Releases *208*
6.5 Accumulation of Activity in Ponds and Surfaces *208*
6.6 Pathways Associated with Open and Closed Fuel Cycles *210*
6.6.1 High-Level Waste *210*
6.6.2 Open Fuel Cycles *210*
6.6.3 Closed Fuel Cycles *211*
6.7 Regulatory Guidance for Effluent Pathways *211*
6.8 Doses from Liquid Effluent Pathways *212*
6.8.1 Potable Water *212*
6.8.2 Aquatic Foods *213*
6.8.3 Shoreline Deposits *214*
6.8.4 Irrigated Foods *214*
6.8.4.1 Irrigated Foods (Tritium) 214
6.8.4.2 Irrigated Foods (Radionuclides Other than Tritium) 215
6.9 Doses from Gaseous Effluent Pathways *216*
6.9.1 Annual Gamma Air Dose from Noble Gas Releases from Free-Standing Stacks Higher than 80 Meters *216*
6.9.2 Annual Air Dose from All Noble Gas Releases *217*
6.9.2.1 Annual Gamma Air Dose from All Noble Gas Releases 217
6.9.2.2 Annual Beta Air Dose from All Noble Gas Releases 217
6.9.2.3 Annual Total Body Dose Equivalent from Noble Gas Releases from Free-Standing Stacks More than 80 Meters High 218
6.9.2.4 Annual Skin Dose from Noble Gas Releases from Free-Standing Stacks Higher than 80 Meters 218
6.10 Annual Doses from All Other Noble Gas Releases *219*
6.10.1 Annual Total Body Dose Equivalent from All Other Noble Gas Releases *219*
6.10.2 Annual Skin Dose from All Other Noble Gas Releases *219*
6.11 Doses from Radioiodines and Other Radionuclides Released to the Atmosphere *220*
6.11.1 Annual Organ Dose from External Irradiation from Radionuclides Deposited Onto the Ground Surface *220*
6.11.2 Annual Organ Dose from Inhalation of Radionuclides in Air *220*
6.11.3 Annual Organ Dose from Ingestion of Atmospherically Released Radionuclides in Food *221*
6.12 Pathway Selection *222*
6.13 Model Parameters *222*
6.14 Intentional Dispersal of Radioactive Materials *222*
6.15 Protection of the Environment *225*
6.16 Scenarios *226*
Scenario 6.1 *226*

Scenario 6.2 *227*
Scenario 6.3 *228*
Scenario 6.4 *229*
Scenario 6.5 *230*
Scenario 6.6 *230*
Scenario 6.7 *231*
Scenario 6.8 *232*
Scenario 6.9 *233*
References 234

7 Accelerator Health Physics *237*
7.1 High-Energy Interactions *237*
7.2 Radiation Types *239*
7.3 Proton Accelerators *239*
7.4 Electron Accelerators *240*
7.4.1 Bremsstrahlung *240*
7.4.2 Synchrotron Radiation *242*
7.4.3 Electromagnetic Cascade *243*
7.5 Light Sources *243*
7.6 Heavy-Ion Accelerators *244*
7.7 Large Hadron Collider *245*
7.7.1 Antiprotons *245*
7.7.2 Proton Reactions *245*
7.7.3 Neutrons *246*
7.7.4 Muons *246*
7.7.5 Hadronic (Nuclear) Cascade *246*
7.8 Muon Colliders *248*
7.8.1 Bounding Neutrino Effective Dose – Linear Muon Collider *248*
7.8.2 Bounding Neutrino Effective Dose – Circular Muon Collider *249*
7.8.3 ALARA Impacts of Muon Colliders *250*
7.9 Radiation Types of Concern *250*
7.9.1 Residual Radioactivity *251*
7.9.2 Activation of Water *251*
7.9.3 Activation of the Soil *252*
7.9.4 Activation of Air *252*
7.9.5 Buildup of Radioactive and Toxic Gases in an Irradiation Cell *253*
7.9.6 Other Radiation Sources *253*
7.10 Shielding *255*
7.11 Accelerator Beam Containment *257*
7.12 Dose Equivalent Rate from the Accelerator Target *258*
7.13 Beam Current *258*
7.14 Pulsed Radiation Fields *259*
7.15 Scenarios *259*
Scenario 7.1 *259*
Scenario 7.2 *261*

Scenario 7.3 *263*
Scenario 7.4 *264*
Scenario 7.5 *265*
Scenario 7.6 *266*
Scenario 7.7 *266*
Scenario 7.8 *267*
Scenario 7.9 *269*
References *271*

8 Nonionizing Radiation Health Physics *273*
8.1 Sources of Radiofrequency and Microwave Radiation *273*
8.2 Characteristics of Electromagnetic Waves *274*
8.3 Antennas *275*
8.3.1 Stationary Antennas *275*
8.3.2 Rotating Antennas *277*
8.4 Attenuation by Biological Systems *278*
8.5 Biological Effects *279*
8.6 Protection Standards *279*
8.7 Measurement of Electromagnetic Fields *280*
8.8 Laser Radiation *281*
8.8.1 Radiometric and Photometric Terms and Units *281*
8.8.2 Properties of the Laser *282*
8.9 Biological Effects from Laser Radiation *282*
8.9.1 Eye *283*
8.9.2 Skin *284*
8.10 Laser Standards *284*
8.10.1 Intrabeam Exposures *285*
8.10.2 Nominal Ocular Hazard Distance *286*
8.10.3 Diffuse Reflections *286*
8.10.4 Nominal Hazard Zone *287*
8.10.5 Skin Exposures *287*
8.11 Free Electron Lasers *288*
8.12 Federal Regulations and Laser Standards *289*
8.12.1 Laser Classes *289*
8.12.2 Laser Safety Calculations *291*
8.12.2.1 Limiting Aperture 292
8.12.2.2 Exposure Time/Maximum Permissible Exposure 292
8.13 Controlling Laser Radiation *293*
8.14 Personnel Protective Equipment *294*
8.15 Spectral Effectiveness of Ultraviolet Radiation *294*
8.16 Scenarios *295*
Scenario 8.1 *295*
Scenario 8.2 *296*
Scenario 8.3 *297*
Scenario 8.4 *298*

Scenario 8.5 *300*
Scenario 8.6 *302*
Scenario 8.7 *302*
Scenario 8.8 *303*
Scenario 8.9 *304*
References *306*

Part III Answers and Solutions *309*

Solutions for Chapter 1 *311*
Scenario 1.1 *311*
Scenario 1.2 *312*
Scenario 1.3 *313*
Scenario 1.4 *313*
Scenario 1.5 *314*
Scenario 1.6 *316*
Scenario 1.7 *317*
Scenario 1.8 *318*
Scenario 1.9 *319*
Scenario 1.10 *321*
Scenario 1.11 *322*
Scenario 1.12 *325*
Scenario 1.13 *326*
Scenario 1.14 *327*
Scenario 1.15 *328*
Scenario 1.16 *329*
Scenario 1.17 *329*
Scenario 1.18 *331*
Scenario 1.19 *334*
Scenario 1.20 *336*
Scenario 1.21 *338*
Scenario 1.22 *339*
Scenario 1.23 *341*
Scenario 1.24 *343*
Scenario 1.25 *344*
Scenario 1.26 *345*
Scenario 1.27 *348*
Scenario 1.28 *352*
Scenario 1.29 *355*
Scenario 1.30 *356*
Scenario 1.31 *357*
Scenario 1.32 *358*
Scenario 1.33 *359*
Scenario 1.34 *360*

Scenario 1.35 *364*
Scenario 1.36 *366*
Scenario 1.37 *369*
Scenario 1.38 *372*

Solutions for Chapter 2 *375*
Scenario 2.1 *375*
Scenario 2.2 *378*
Scenario 2.3 *381*
Scenario 2.4 *384*
Scenario 2.5 *386*
Scenario 2.6 *388*
Scenario 2.7 *389*
Scenario 2.8 *391*
Scenario 2.9 *395*
Scenario 2.10 *398*

Solutions for Chapter 3 *403*
Scenario 3.1 *403*
Scenario 3.2 *406*
Scenario 3.3 *408*
Scenario 3.4 *410*
Scenario 3.5 *412*
Scenario 3.6 *413*
Scenario 3.7 *417*
Scenario 3.8 *419*
Scenario 3.9 *421*
Scenario 3.10 *423*
Scenario 3.11 *425*

Solutions for Chapter 4 *433*
Scenario 4.1 *433*
Scenario 4.2 *435*
Scenario 4.3 *438*
Question 4.12 *439*
Scenario 4.4 *440*
Scenario 4.5 *442*
Scenario 4.6 *444*
Scenario 4.7 *445*
Scenario 4.8 *449*
Scenario 4.9 *452*

Solutions for Chapter 5 *455*
Scenario 5.1 *455*
Scenario 5.2 *457*

Scenario 5.3 *459*
Scenario 5.4 *460*
Scenario 5.5 *462*
Scenario 5.6 *466*
Scenario 5.7 *468*
Scenario 5.8 *470*
Scenario 5.9 *472*

Solutions for Chapter 6 *475*
Scenario 6.1 *475*
Scenario 6.2 *477*
Scenario 6.3 *480*
Scenario 6.4 *483*
Scenario 6.5 *485*
Scenario 6.6 *486*
Scenario 6.7 *488*
Scenario 6.8 *489*
Scenario 6.9 *491*

Solutions for Chapter 7 *499*
Scenario 7.1 *499*
Scenario 7.2 *501*
Scenario 7.3 *503*
Scenario 7.5 *504*
Scenario 7.6 *506*
Scenario 7.7 *507*
Scenario 7.8 *510*
Scenario 7.9 *514*

Solutions for Chapter 8 *521*
Scenario 8.1 *521*
Scenario 8.2 *526*
Scenario 8.3 *530*
Scenario 8.4 *533*
Scenario 8.5 *537*
Scenario 8.6 *545*
Scenario 8.7 *547*
Scenario 8.8 *549*
Scenario 8.9 *554*

Part IV Appendices *559*

Appendix I Serial Decay Relationships *561*
References *563*

Appendix II Basic Source Geometries and Attenuation Relationships *565*
References 571

Appendix III Neutron-Induced Gamma Radiation Sources *573*
References 576

Appendix IV Selected Topics in Internal Dosimetry *577*
References 624

Appendix V Radiation Risk and Risk Models *627*
References 638

Appendix VI Key Health Physics Relationships *639*
References 652

Appendix VII Production Equations in Health Physics *653*
References 659

Appendix VIII Mathematical Review *661*
References 671

Appendix IX Selected Data on Radionuclides of Health Physics Interest *673*
References 678

Subject Index *679*

Part I
Basic Concepts: Theory and Problems

1
Introduction

Health physics or radiation protection is the science dealing with the protection of radiation workers and the general public from the harmful effects of radiation. Health physicists work in a variety of environments, including medical facilities, facilities utilizing nonionizing radiation, universities, accelerator complexes, power reactors, and fuel cycle facilities. The health physicist is responsible for the radiological safety aspects of facility equipment and services. Radiological assessments of plant equipment, facility modifications, design changes, employee exposures, or the assessment of radiological effluents are key functions of a health physicist.

The fundamental tools of the health physicist include the fields of mechanics, electricity and magnetism, energy transfer, quantum mechanics. Atomic and nuclear structure, radioactive transformations, and the interaction of radiation with matter are the cornerstones of health physics knowledge. Application of these fundamental tools permits the health physicist to measure, quantify, and control radiation exposures to affected groups.

Introductory health physics texts typically cover these topics in several hundred pages. Because the scope of this text builds upon these fundamental concepts, we will not repeat them herein. The reader is referred to the texts listed as references to this chapter for a discussion of health physics fundamentals. We will, however, provide several appendices that illustrate selected fundamental concepts. Also included is an extensive set of scenarios, including over 160 worked examples, that illustrate the fundamental concepts and permit the reader to assess his or her knowledge of these concepts. Because the fundamentals are needed to fully understand the remaining chapters in this text, a review of the scenarios in this chapter is recommended.

1.1 Scenarios

Scenario 1.1

One of your neighbors, while digging up his back yard to build a pool, has discovered some old planks. Another neighbor, who has been investigating the possibility of the existence of a Viking settlement in the area, believes that the planks may be significant. He wishes to conduct an archeological expedition prior to any further construction. You offer to carbon date the wood to help settle the argument.

1.1 Carbon dating is possible because:
   a. The specific activity of carbon-14 in living organisms has changed over time, and one can identify the era of time the organism lived based on its current specific activity.
   b. Carbon-14 is in secular equilibrium with its daughter.
   c. The specific activity of carbon-14 in living organisms is relatively constant through time, but decays after the death of the organism.
   d. The specific activity of carbon-14 in wood increases over time due to shrinkage of the wood.

1.2 Calculate the approximate age of the wood given the following:

C-14 $T_{1/2}$ = 5715 years

Specific activity for C-14 in a nearby living tree = 1.67×10^{-1} Bq/g

Specific activity for C-14 in the old wooden plank = 1.50×10^{-1} Bq/g

Scenario 1.2

A nearby hospital has received a shipment of a Mo-99 generator. The shipment contained 1000 mCi of Mo-99 when manufactured. It arrived at the hospital 48 h after its production. The decay scheme is illustrated in Figure 1.1.

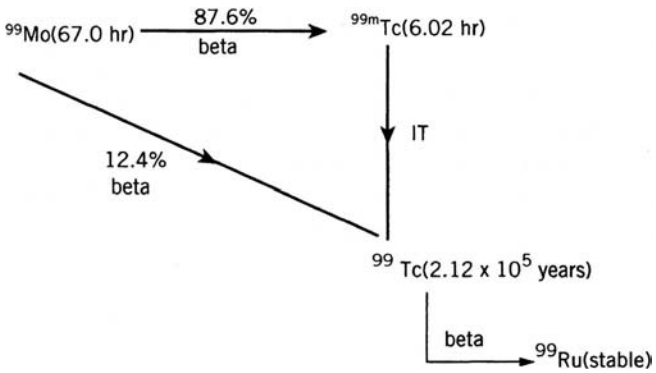


Figure 1.1 Decay scheme for Mo-99.

1.3 If the generator is milked exactly upon arrival at the hospital, how much Tc-99m will be obtained? Assume that 95% of the available Tc-99m is eluted.

1.4 If the generator is milked 24 hr after the initial milking, how much Tc-99m will be obtained?

Scenario 1.3

Consider a parent radioisotope A ($T_{1/2}$ = 10 hr) that decays to a daughter radioisotope B ($T_{1/2}$ = 1 hr).

1.5 Which of the following statements is true concerning these radioisotopes?
   a. Because $\lambda_A > \lambda_B$, the parent and daughter will eventually reach the condition of transient equilibrium.
   b. Because $\lambda_A >> \lambda_B$, the parent and daughter will eventually reach the condition of secular equilibrium.
   c. Because $\lambda_A = \lambda_B$, no state of equilibrium can ever exist between the parent and daughter.
   d. Because $\lambda_B > \lambda_A$, the parent and daughter will eventually reach the condition of transient equilibrium.
   e. Because $\lambda_B >> \lambda_A$, the parent and daughter will eventually reach the condition of secular equilibrium.

1.6 Assuming that the activity of the daughter is zero at time zero, at what time (t) will the daughter reach its maximum activity?

Scenario 1.4

The plant manager at your facility has requested that you review the following questions and provide the best solution. These questions will be used to assess the qualification of health physics candidates for entry-level positions in your facility's radiological controls department.

1.7 Tissue dose from thermal neutrons arises principally as a result of:
   a. (n, γ) reactions with hydrogen.
   b. (n, γ) reactions with hydrogen and (n, p) reactions with nitrogen.
   c. (n, p) reactions with carbon.
   d. (n, α) reactions with carbon.
   e. (n, α) reactions with carbon and (n, γ) reactions with hydrogen.

1.8 Tissue dose from fast neutrons (0.1 to 14 MeV) is due principally to:
   a. Resonance scattering with nuclei.
   b. Inelastic scattering with nuclei.
   c. Coulomb scattering with nuclei.
   d. Nuclear capture and spallation.
   e. Elastic scattering with nuclei.

1.9 The most probable process for energy deposition by a 1-MeV photon in tissue is:
   a. Photoelectric absorption.
   b. Pair production.
   c. Compton scattering.
   d. Photonuclear absorption.
   e. Bremsstrahlung.

1.10 The principal mechanism of dose deposition by a 5-MeV alpha particle that stops in tissue is:
   a. Inelastic scattering by atomic electrons.
   b. Elastic scattering by atomic electrons.
   c. Elastic scattering by atomic nuclei.
   d. Inelastic scattering by atomic nuclei.
   e. Nuclear spallation.

1.11 The principal mechanism of dose deposition by a 100-keV beta particle that stops in tissue is:
   a. Elastic scattering by atomic electrons.
   b. Elastic scattering by atomic nuclei.
   c. Inelastic scattering by atomic nuclei.
   d. Inelastic scattering by atomic electrons.
   e. Bremsstrahlung.

1.12 The average number of ion pairs produced by 100-keV beta particle that stops in air is approximately:
   a. 300
   b. 30
   c. 30 000
   d. 3000
   e. 300 000.

1.13 The average number of ion pairs produced by a 100-keV beta particle that stops in a germanium semiconductor is:
   a. 30 000
   b. 30
   c. 300
   d. 3000
   e. 300 000.

1.14 A nuclide that undergoes orbital electron capture:
   a. Emits an electron, a neutrino, and the characteristic X-rays of the daughter.
   b. Emits a neutrino and the characteristic X-rays of the daughter.
   c. Also decays by positron emission.
   d. Also emits internal conversion electrons.
   e. Makes an isomeric transition.

1.15 The specific gamma-ray emission rate for Cs-137 in units of R hr^{-1} Ci^{-1} m^2 is approximately:
   a. 1.3

b. 0.12
c. 0.33
d. 0.05
e. 0.77.

1.16 An example of an organ or tissue for which the Annual Limit on Intake (ALI) is determined by the limit for nonstochastic effects is the:
a. Red bone marrow.
b. Gonads.
c. Lung.
d. Breast.
e. Thyroid.

Scenario 1.5

The radioisotope I-126 (atomic number 53) can decay into stable Te-126 (atomic number 52) by orbital electron capture (EC) or by positron emission. It can, alternatively, decay by negative beta emission into stable Xe-126 (atomic number 54). The fractions of the transformations that take place via these modes are: EC 55%, positron emission 1%, and beta decay 44%. An I-126 source also emits gamma photons of energy 386 keV and 667 keV as well as characteristic X-rays of Te. The energy equivalents (Δ) of the mass excesses of the atoms involved in these transformations are (Δ = atomic mass – atomic mass number):

Atom	Δ (MeV)
Te-126	–90.05
I-126	–87.90
Xe-126	–89.15

The energy equivalent of the electron rest mass is 0.511 MeV, and the binding energy of the K-shell electron in I-126 is 32 keV.

For the following questions, choose the best answer.

1.17 The energy release (*Q*-value) by the decay of I-126 via capture of a K-shell electron, going directly to the ground state of Te-126, is:
a. 0.03 MeV.
b. 1.13 MeV.
c. 2.12 MeV.
d. 2.15 MeV.
e. 2.18 MeV.

1.18 The energy released (*Q*-value) by the decay of I-126 via position emission to the ground state of Te-126 is:
a. 0.51 MeV.
b. 1.02 MeV.

c. 1.13 MeV.
d. 1.64 MeV.
e. 2.15 MeV.

1.19 The energy released in the decay of I-126 to the ground state of Xe-126 by beta emission is:
a. 0.20 MeV.
b. 0.23 MeV.
c. 0.90 MeV.
d. 0.74 MeV.
e. 1.25 MeV.

1.20 Of the following kinds of radiation emitted from I-126, which is the single least significant potential contributor to internal dose?
a. Annihilation photons.
b. Bremsstrahlung.
c. Internal-conversion electrons.
d. Auger electrons.
e. Antineutrino.
How would your answer change if external dose contributions were under consideration?

1.21 Why are the 32-keV Te X-rays present with an I-126 source?
a. The nucleus of Te-126 has excess energy after the EC event. This excess energy is released by Te-126 as X-rays.
b. Stable Te-126 has excess energy after the positron emission. This excess energy is released by Te-126 as X-rays.
c. Electrons rearranging between the L and M shells produce X-rays.
d. Te X-rays are released when the EC event creates a vacancy in the inner shells, and electrons from outer shells fill the vacancy.
e. Te X-rays are equivalent to the bremsstrahlung radiation emitted by I-126.

Scenario 1.6

The nuclide Sr-90 (atomic number 38) decays by beta emission into Y-90 (atomic number 39), which then decays by beta emission into Zr-90 (atomic number 40), with the half-lives noted below:

$$\text{Sr-90} \xrightarrow[27.7\text{ years}]{} \text{Y-90} \xrightarrow[64.2\text{ hr}]{} \text{Zr-90}$$

1.22 What is the mean, or average, lifetime of a Y-90 atom?
a. 31.1 hr.
b. 44.5 hr.
c. 77.04 hr.
d. 92.6 hr.
e. 128.4 hr.

1.23 What is the specific activity of Y-90 in SI units?
   a. 5.42×10^5 Bq/kg
   b. 7.22×10^{16} Bq/kg
   c. 2.01×10^{19} Bq/kg
   d. 7.22×10^{19} Bq/kg
   e. 6.49×10^{21} Bq/kg.

1.24 Starting with a pure Sr-90 sample at time $t = 0$, a researcher finds that the Y-90 activity is 3.4 MBq at $t = 72.0$ hours. What was the activity of the Sr-90 at $t = 0$?
   a. 1.84 MBq
   b. 3.40 MBq
   c. 4.37 MBq
   d. 6.29 MBq
   e. 7.39 MBq.

Scenario 1.7

You have been asked to assist in the technical evaluation of an ionization chamber and environmental sampling results. Your boss has requested answers to the following questions. Assume the density of air at STP = 1.293×10^{-6} kg cm^{-3}.

1.25 A free air ionization chamber shows a flow of electrical charge of 1×10^{-9} A. The chamber has a sensitive volume of 4 cm^3. The reading is taken at 10 °C and 755 mm Hg. Find the exposure rate in R/s based on STP conditions.

1.26 You are asked to provide immediate, on-site measurement results for a series of environmental samples that are being collected every 100 min. It has been requested that each sample count be preceded by a background count. From past experience, you estimate that the net sample and background counting rates should be approximately 2400 and 300 cpm, respectively. Assuming that each sample must be analyzed before the next one is received, how long would you count the sample to minimize the standard deviation estimate for the sample's net activity?

1.27 A water sample that was counted for 10 min yielded 600 counts. A 40-min background count yielded a background rate of 56 cpm. At a 95% confidence level (one-tail test), determine whether or not there was any net activity in the sample.

Scenario 1.8

You are responsible for operating the counting room at a nuclear facility. You need to minimize the counting time required for air samples because of the heavy workload and a need to streamline operations in the count room. The bulk of your air sample workload is counting I-131. The following parameters are applicable to your operation:

Counting efficiency = 20%

Background count time = sample count time

Background count rate = 50 cpm

Sampling flow rate = 5 liters/min

Sample collection time = 10 min

Iodine collection efficiency = 70%

DAC for iodine = 900 Bq/m^3

1.28 Calculate the minimum sample and background counting time required to ensure an LLD at the 95% confidence level less than or equal to 0.10 DAC for I-131.

1.29 List methods that could be used in the field or in the counting room to reduce the time required to process I-131 samples. Explain how each method reduces processing time.

Scenario 1.9

As a health physicist at a nuclear facility, you are asked to develop a program to characterize the radioactive particulate emissions through the facility's main ventilation stack. The following questions relate to various aspects of this assignment.

1.30 In designing the sampling system, you have determined that the stack internal diameter is 0.5 m and the volumetric flow through the stack is 20 m^3 min^{-1}. You want to use a vacuum source which will provide a constant volumetric flow of 200 liters/min through your sampling train. Assuming laminar flow, what should the internal diameter of the sampling nozzle be to ensure isokinetic sampling conditions?

1.31 To ensure that your sample is representative of laminar flow conditions (nonturbulent, constant velocity) within the stack, discuss factors that you should consider relative to the location of your sampling nozzle within the stack.

1.32 You have decided to use filtration techniques to capture your sample and are evaluating three types of media (cellulose, glass-fiber, and membrane filters). List advantages and disadvantages of each.

Scenario 1.10

You are responsible for a high-volume air sampler located downwind from a Department of Energy (DOE) facility following a suspected release of Pu-239. The air sampler has a calibrated volumetric flow rate of 55 SCFM, and the filter has an alpha self-absorption factor and filter collection efficiency of 0.4 and 0.8, respectively. The air sampler is operated at this flow rate for 1 hr, and the filter surface is measured with a gross alpha probe detector having an active detection

area of 60 cm^2 and a background count of 20 counts in 100 min. The detector efficiency for alpha is 0.3 cpm dpm^{-1}, and the active filter area is 500 cm^2. Assume that the filter face velocity is uniform.

Data

Half-Life for Pu-239 = 24 100 years

Alpha yield = 100%

LLD (95%) = 4.66 σ_b (where σ_b is the standard deviation of the background)

1.33 The initial filter-face alpha count immediately after the 1 hr sampling period was 2000 for a 10-min counting interval. Forty-eight hours later, the same filter is measured again with the same detector, and the count was 220 in 100 min. Explain why the count rate is lower 48 hr later.

1.34 What is Pu-239 airborne activity (in dpm/m^3) and the standard deviation for this measured quantity?

1.35 What is the lower limit of detection (LLD) at the 95% confidence level for this air sampling and detection system (in dpm/m^3) for the same sampling conditions?

Scenario 1.11

This scenario deals with the working-level unit.

With the passage of the Radon Control Act of 1988, the Environmental Protection Agency (EPA) is now instructed by the Congress to assess public risks of radon exposure in public buildings (including schools) throughout the nation. Regarding the measurement, detection, and health physics of radon-222 and its daughter products, answer the following questions:

1.36 Historically, an operational definition of the working-level exposure unit (WL) for radon-222 daughters has been 100 pCi/liter of each short-lived daughter product in secular equilibrium. Using this definition and the data provided derive the total alpha energy per liter of air (MeV/liter) associated with a concentration of one working level. Radon and its short-lived daughters include:

Nuclide	Alpha Energy (MeV)	Half-life
Radon-222	5.49	3.82 days
Polonium-218	6.00	3.05 min
Lead-214	0	26.8 min
Bismuth-214	0	19.7 min
Polonium-214	7.68	1×10^{-6} min

1.37 Using the data provided, calculate the concentration of radon-222 gas in air determined from a single-count, filter collection method for radon daughters. Assume a 50% equilibrium between radon-222 and its daughters. Neglect special considerations for radioactive growth and decay during sampling and counting. The following data are provided:

Sample collection period = 5 min

Counting time = 1 min

Total alpha counts = 230

Counting efficiency = 0.3

Pump flow rate = 10 liters min^{-1}

Conversion factor = 150 dpm alpha $liter^{-1}$ WL^{-1}

1.38 List common methods for the detection and measurement of radon and/or its daughters for use in assessing public exposure in building structures.

Scenario 1.12

A common type of portable beta–gamma survey instrument uses an air ionization chamber vented to atmospheric pressure. The cylindrical detector is 3 in. high and 3 in. in diameter with a 7-mg/cm^2 beta window and a 400-mg/cm^2 beta shield. The side walls are 600 mg/cm^2. Answer the following questions with respect to the instrument's response versus the 'true' dose rates specifically associated with the following conditions.

1.39 Briefly describe a potential source of error associated with measuring gamma and beta dose rates while moving in and out of a noble gas environment.

1.40 List and briefly explain two harsh environmental conditions which could have an adverse effect on the accuracy of the instrument response while in the area.

1.41 Briefly describe the most significant source of error associated with measuring true beta and gamma surface dose rates from contact measurements of small sources.

1.42 Briefly explain a source of error associated with measuring beta dose rates from large-area sources, with each source comprised of a different radionuclide.

1.43 Briefly describe a source of error associated with measuring beta dose rates from high-energy beta sources using open minus closed window readings.

Scenario 1.13

ANSI N13.11-1983, 'American National Standard for Dosimetry – Personal Dosimetry Performance Criteria for Testing', is used as a basis for testing the performance of suppliers of dosimetry services. This standard provides criteria for testing personnel dosimetry performance for any type of dosimeter whose reading is used to provide a lifetime cumulative personal radiation record. The test procedure in this standard evaluates the absorbed dose and dose equivalent at two irradiation depths (0.007 cm and 1.0 cm). The radiation sources used for the performance tests are Cs-137, Sr-90/Y-90, heavy water moderated Cf-252, and an X-ray machine. The X-ray machine is used to generate several photon beams with average energies between 20 keV and 70 keV. Choose the single answer which is most correct.

1.44 The provisions of this standard apply:
   a. to neither pocket dosimeters nor extremity dosimeters.
   b. to pocket dosimeters but not to extremity dosimeters.
   c. only to beta and gamma radiation.
   d. to extremity dosimeters but not to pocket dosimeters.
   e. to film badges but not to thermoluminescent dosimeters (TLDs).

1.45 Because of the particular irradiation depths chosen for the tests, a dosimetry system which is calibrated with the standard tests may be reporting doses which are different than the actual dose received. For which of the following tissues (red bone marrow, skin, gonads, lens of the eye, or whole body) is this difference most significant?

1.46 Because of the particular radiation sources specified, the standard least adequately tests for radiations emitted by:
   a. C-14, power reactor leakage neutrons.
   b. P-32, Cf-252.
   c. Y-90/Sr-90, Am–Be source.
   d. Co-60, Ni-65.
   e. Uranium slab, Cf-252.

1.47 A dosimeter of a processor who has passed the test category for:
   a. beta radiation, is appropriate for measuring low-energy photons.
   b. beta radiation, is not appropriate for measuring beta radiation from all sources.
   c. low-energy photons, can be used to pass the performance test for beta radiation.
   d. high-energy photons and the category for low-energy photons, can be assumed to pass the test for mixtures of high-energy and low-energy photons.
   e. neutrons, is appropriate for measuring neutron radiation from any sources.

1.48 This standard:
   a. forms the basis for the National Voluntary Laboratory Accreditation Program for dosimetry processors.
   b. provides guidance for individual variability from reference man.
   c. provides guidance for summing the internal and external dose.
   d. is applicable to the entire range of gamma energies.
   e. is not required to be implemented by 10 CFR 20.

Scenario 1.14

For each of the situations below (1.49 to 1.53), select the personnel dosimeter which is most suitable for the purpose of establishing primary dose records. In each case substantiate your choice of dosimeter. Limit your choice of dosimeter to the following:

1. A common film badge with 300 mg/cm^2 plastic filtration over all areas except for the 14-mg/m^2 mylar window.
2. A TLD albedo containing both Li-6 and Li-7 elements.
3. A TLD albedo containing both Li-6 and B-11 elements.
4. A calcium sulfate, manganese-activated TLD element in a tissue equivalent holder.
5. A proton recoil film badge.
6. A four-element TLD with lithium borate phosphors, 300-mg/cm^2 plastic filtration over two elements, aluminum over the third element, and lead over the fourth element.
7. A four-element TLD with lithium borate phosphors, a thin mylar filter over one element, plastic filters over two elements, and an aluminum filter over the fourth element.
8. A natural LiF TLD element.
9. A calcium sulfate, dysprosium-activated TLD element in a tissue equivalent holder.
10. A two-element TLD with lithium borate phosphors and 300-mg/cm^2 plastic filters.

1.49 An accelerator facility using tritiated targets with 14-MeV deuteron beams.
1.50 A mixed neutron and gamma field where gamma dose predominates.
1.51 A radiographer using a 320-kVp X-ray machine.
1.52 A field of high-energy, 6-MeV photons.
1.53 A field of mixed beta (average energy of 200 keV) and gamma (average energy of 800 keV) radiation.

Scenario 1.15

You supervise an in-house TLD system for occupationally exposed workers governed by US Nuclear Regulatory Commission regulations. The TLD badge consists of two LiF chips of 235-mg/cm^2 thickness. Chip 1 is covered by 7 mg/cm^2 of plastic, and Chip 2 is shielded by 850 mg/cm^2 of lead and 150 mg/cm^2 of plastic. The TLD system is calibrated by exposing badges to known quantities of beta and gamma radiations and plotting the TL reader output versus mrem dose equivalent. Both the gamma and beta calibration curves are linear and pass through the origin (0, 0) on the graph. The gamma calibration curve indicates that 6000 TL units equals 500 mrem of gamma dose equivalent, and the beta curve yields 750 TL units per 1000 mrem of beta dose equivalent.

The following data are provided:

1. The control dosimeter reads 120 TL units on both Chips 1 and 2. (Both Chips have the same gamma sensitivity.)
2. Chip 1 = 12 270 TL units.
   Chip 2 = 11 520 TL units.
3. The beta calibration curve for other tissue depths includes the following:

Tissue Depth (mg/cm^2)	Percentage of Dose Equivalent at 7 mg/cm^2
7	100
100	50
300	25
500	10
1000	1

4. The gamma dose equivalent remains constant at tissue depths from 7 to 1000 mg/cm^2.

1.54 Calculate the skin and whole-body dose equivalents for the exposed TLDs noted above in item 2.
1.55 Calculate the dose to the lens of the eye.
1.56 Explain if any dose limits were exceeded. Justify your answer by stating the limits and identifying the source of the limits that you applied.

Scenario 1.16

A facility is in the process of setting up a neutron dosimetry program. You have been asked to consult on this matter. The following dosimeters are under consideration: TLD, recoil track-etch, neutron track type A (NTA) film, and bubble detectors. A final option is to use stay time calculations based on survey results from a 'rem-ball' that has been calibrated using D_2O-moderated Cf-252.

1.57 Which one of the following statements is incorrect?
   a. There is no neutron dosimetry system in use today that is adequate (±50% of the true dose equivalent) for all situations where neutron dosimetry is required.
   b. The neutron quality factor between 0 and 20 MeV is relatively constant at a value of about 10.
   c. Neutron energies can span nine decades in some monitoring situations.
   d. Neutron monitoring is usually performed in a mixed field of neutron and gamma radiation.
   e. Stay-time calculations, though often used, may be unreliable due to variations of neutron dose rates and energies in a given neutron radiation area.

1.58 In a field of mixed neutron and gamma radiation, the gamma dose measured on a phantom is:
   a. greater than the gamma dose measured in air due to the $H(n, \gamma) D$ reaction in the phantom.
   b. less than the dose measured in air due to the moderation of neutrons in the phantom.
   c. the same as the measured dose in air because phantoms do not influence gamma irradiation.
   d. less than the dose measured in air because some incident gamma rays are absorbed in the phantom.
   e. not a quantity of interest in a dosimetry program.

1.59 If no corrections are made to the dosimeter response for neutron energy, TLD albedo dosimeters calibrated with a bare Cf-252 source will:
   a. give accurate indications (±50%) of neutron dose equivalent in soft (thermal or epithermal) spectra.
   b. underestimate the neutron dose equivalent by as much as a factor of 2 in soft (thermal or epithermal) spectra.
   c. overestimate the neutron dose equivalent regardless of the incident spectrum.
   d. underestimate the neutron dose equivalent regardless of the incident spectrum.
   e. overestimate the neutron dose equivalent in soft (thermal or epithermal) spectra.

1.60 Which one of the following statements is true regarding neutron bubble detectors?
   a. They are insensitive to intermediate-energy neutrons.
   b. They are accurate within ±30% in neutron dose rates of over 1000 rad/hr.
   c. They are affected by temperature.
   d. They cannot measure the total integrated dose.
   e. They are not yet commercially available.

1.61 Which of the following choices would most accurately measure the neutron dose equivalent for commercial power reactor containment entries?
   a. A TLD albedo dosimetry system calibrated to D_2O-moderated Cf-252.
   b. A TLD albedo dosimetry system calibrated to AmBe.
   c. A proton-recoil dosimetry system calibrated to D_2O-moderated Cf-252.
   d. A proton-recoil dosimetry system calibrated to AmBe.
   e. An NTA film dosimetry system calibrated to D_2O-moderated Cf-252.

Scenario 1.17

A large community hospital wishes you to set up a personnel monitoring program. The following organization information is provided:

Department A: The nuclear medicine department is a well-equipped department using technetium-99m for all its studies. The Tc-99m is milked from a generator, and the radiopharmaceuticals are prepared within the nuclear medicine department. The department has sealed sources of cobalt-57, cesium-137, and barium-133 for calibrating the dose calibrator.

Department B: The X-ray department is an active group using fluoroscopic procedures, general diagnostic X-ray procedures, and some special procedures.

Department C: The radiation therapy department is an active group using a Co-60 teletherapy device and a 4.0-MeV linear accelerator, but no brachytherapy.

Department D: The research department is a fairly active department using only hydrogen-3 and carbon-14.

1.62 What departments will require personnel monitoring for photons?
1.63 What department will require personnel monitoring for neutrons?
1.64 What departments will benefit from both a personnel monitor at the belt (under leaded apron) and one at the collar?
1.65 What department would need ring badges?
1.66 In what department would the assessment of skin dose be important?
1.67 What department might require bioassay?
1.68 List positive characteristics of film dosimeters for personnel monitoring.
1.69 List negative characteristics of film dosimeters for personnel monitoring.
1.70 List positive characteristics of TLDs for personnel monitoring.
1.71 List negative characteristics of TLDs for personnel monitoring.

Scenario 1.18

This scenario involves the properties of gas-filled detectors.

Data

Air density = 1.29 kg m^{-3} at STP

1 torr = 1 mm Hg at 0 °C

1.72 Consider two cylindrical gas ionization chambers, A and B. The chamber of detector A has the dimensions 0.5 cm in diameter and 5 cm in height. Detector B has the dimensions 1.0 cm in diameter and 5 cm in height. both detectors have the same chamber wall material and thickness, fill gas, and chamber pressure. If detector A shows an output current of 1.0×10^{-10} A when placed in an isotropic gamma field, what theoretical response should be given by detector B when placed in the same field? Neglect detector end effects.
   a. 2.5×10^{-11} A
   b. 4.0×10^{-10} A
   c. 2.0×10^{-10} A
   d. 5.0×10^{-11} A
   e. 1.0×10^{-10} A.

1.73 The gas fill pressure in detector A is 7600 torr, and the detector sensitivity is 1.2×10^{-10} A-hr/R. What would the detector sensitivity be if the gas fill pressure was increased to 11 400 torr?

1.74 Assuming a chamber pressure of 7600 torr, a chamber volume of 100 cm^3, and a temperature of 20 °C, calculate the dose equivalent rate in mSv/hr for a tissue equivalent wall ion chamber if the saturated ion current is 9.0×10^{-14} A. For this question, assume 100 mR = 100 mrem = 1 mSv.

1.75 An ambient-pressure air ion chamber is calibrated at 7000-feet altitude in New Mexico at 20 °C, 591.6-torr air pressure, to read correctly under those conditions. What dose equivalent rate will it indicate in a 1 mSv/hr field at sea level in the Marshall Islands at 36 °C, 760.0-torr air pressure?
   a. 0.74 mSv/hr
   b. 0.82 mSv/hr
   c. 1.00 mSv/hr
   d. 1.22 mSv/hr
   e. 1.36 mSv/hr.

Scenario 1.19

You are the station health physicist at a nuclear power station. The Chemistry Manager has asked you to review a purchase requisition for an N-16 calibration source. The source generates N-16 via an (α, p) reaction involving 160 mCi of curium-244 and carbon-13. The source gamma emission strength is 2.2×10^6 gammas/s, and the neutron emission strength is 2.0×10^5 neutrons/s. Assume a gamma energy of 6.1 MeV and an average neutron energy of 2.5 MeV. The following information is provided for your evaluation:

Physical Quantity (6.1 MeV)	Water	Air	Muscle	Lead
Density (g/cm^3)	1.00	0.001293	1.0400	11.35
Mass-energy absorption coefficient (cm^2/g)	0.0180	0.0163	0.0178	—
Mass-attentuation coefficient (cm^2/g)	0.0277	0.0252	0.0274	0.0435

Point Source Dose Buildup Factors in Lead (*ux*).

Energy = 6.0 MeV	1	2	4	7	10	15	20
	1.18	1.40	1.97	3.34	5.69	13.8	32.7

The neutron flux to dose equivalent (k) at 2.5 MeV is $k = 20$ n/cm^2-s = 2.5 mrem/hr.

1.76 Calculate the total gamma dose equivalent rate at 1 ft. Assume a 100% emission rate from the principal gamma peak.
1.77 Calculate the total neutron dose equivalent rate at 1 ft.
1.78 Lead and polyethylene are available to shield the source. How would you arrange these materials to yield the lowest overall dose rate?
   a. Polyethylene followed by lead.
   b. Lead only.
   c. Polyethylene only.
   d. Lead followed by polyethylene.
   e. No shielding is necessary because the 12-in. air gap will sufficiently scatter/attenuate the neutrons.
1.79 What is the shielding requirement (cm of lead) to reduce the gamma dose rate at 1 ft by a factor of 5?

Scenario 1.20

As the HP supervisor at a reactor decommissioning project, the project engineer has asked you to assist in the evaluation of methods to reduce radiation levels emanating from a neutron-activated concrete shield to meet release limits for unrestricted use. The preferred method requires you to predict the depth to which a slab of neutron-activated concrete must be excavated to allow free release. Other methods she has asked to be evaluated include delayed decommissioning and the addition of shielding.

Assume that the neutron relaxation length in concrete is 15 cm. The current exposure rate is 20 μR/hr 1 m from the slab. The only applicable release limit is 5 μR/hr 1 m from the surface. The concrete source term, based upon a single concrete core sample 1 in. deep, is as follows:

Activation Data for Concrete Source Term.

Nuclide	Radiations			Specific Activity (pCi/g)	Gamma Constant (R/Ci-hr @ 1 m)
	Decay Mode	Energy (MeV)	$T_{1/2}$		
H-3	Beta	0.0186	12.3 years	1000	—
C-14	Beta	0.156	5715 years	500	—
Mn-54	Gamma	0.835	312 days	2500	0.47
Co-60	Gamma Gamma Beta	1.332 1.173 0.314	5.27 years	2500	1.32

1.80 For each of the following three methods for meeting the release limit, list two advantages and two disadvantages:
1. Time to allow decay.
2. Immediate removal.
3. Add shielding.

1.81 Based upon the data provided, estimate the depth of the excavation required to allow free release.

1.82 Assuming that no concrete removal occurs, predict the time necessary to allow the principal radionuclide of interest, Co-60, to decay to the release limit. Assume that the 20 μR/hr exposure rate is due solely to the Co-60.

1.83 How much more concrete shielding would be needed to reduce the exposure rate at 1 m to the release limit? Neglect the geometry considerations. The mass attenuation coefficient is 0.06 cm^2/g, and the density of concrete is 2.5 g/cm^3. Buildup is assumed to be a constant factor of 2.

Scenario 1.21

You are involved in an assessment of the results of an activation experiment that produced Na-24. The buildup and decay of this source and the resultant dose rates require your attention. Answer the following questions regarding the shielding and activation of the Na-24 source.

1.84 What is the flux in particles per square centimeter per second which will produce 4.0×10^7 Bq of Na-24 at saturation in an aluminum target of 1-cm^2 cross-section and 1-g weight? Assume that the production cross-section is 20 mb.

Data

Atomic weight of aluminum = 27

Avogadro's number = 6.02×10^{23}

1 barn = 1.0×10^{-24} cm^2

1.85 Immediately after an irradiation time of 30 hr, what would be the amount of Na-24 present? Assume no initial activity. The half-life of Na-24 is 15 hr.

1.86 What is the dose equivalent rate to a person standing 1 m from the unshielded Na-24 source in air? The following information should be considered in your answer:

Gamma 1 = 1.4 MeV @ 100%

Gamma 2 = 2.8 MeV @ 100%

Air density = 0.00129 g/cm^3

Energy absorption coefficient = 2.3×10^{-5} cm^{-1} @ 1.4 and 2.8 MeV

Assume 1 mrem = 1 mrad

1 MeV = 1.6×10^{-6} erg

Scenario 1.22

You are a consulting health physicist. A client plans to build a 10-MCi Co-60 irradiation facility to sterilize surgical equipment. You may assume that the activity is in the form of a point source. The following data may be useful:

Co-60 gamma constant = 13.2 R-cm^2 hr^{-1} mCi^{-1}

Table of Linear Attenuation Coefficients and Fluence Buildup Factors for Concrete.

Energy (MeV)	u (cm^{-1})	B			
		$ux = 7$	$ux = 10$	$ux = 15$	$ux = 20$
0.5	0.204	16.6	29.0	58.1	98.3
1.0	0.149	11.7	18.7	33.1	50.6
1.17	0.140	11.0	17.5	30.6	46.4
1.25	0.135	10.7	16.9	29.4	44.4
1.33	0.130	10.4	16.3	28.2	42.4
1.5	0.121	9.7	15.0	25.7	38.2

1.87 What would be the exposure rate at a distance of 3 m from the unshielded 10-MCi Co-60 source?

1.88 When exposed, the source will be in the center of a room having internal dimensions of 5 m × 5 m × 5 m with walls of 1-m-thick concrete. The room layout is illustrated in Figure 1.2. Based upon this information, calculate the maximum photon fluence rate in photons cm^{-2} s^{-1} at a point on the exterior surface of the shield wall. State any assumptions used. Ignore scatter off air or walls other than the wall between the source and the reference point.

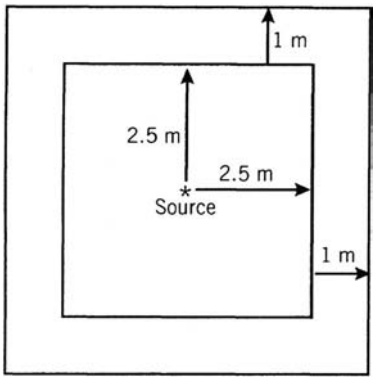


Figure 1.2 Proposed irradiation facility floor plan.

1.89 The buildup factor should only be used:
- a. for photons of energy below 3 MeV.
- b. for photons of energy above 0.5 MeV.
- c. in cases where the shield thickness exceeds 3 relaxation lengths.
- d. for situations involving 'broad-beam' or 'poor' geometry.
- e. for situations involving 'narrow-beam' or 'good' geometry.

Scenario 1.23

Consider both broad and narrow beams of 1-MeV photons, illustrated in Figure 1.3, that are normally incident on different thicknesses of uranium slabs.

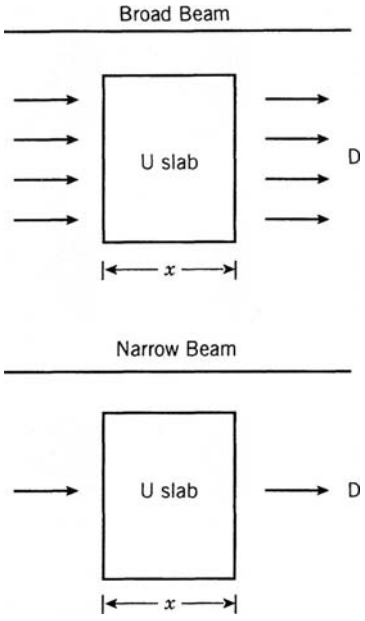


Figure 1.3 Geometry for broad and narrow photon beam scattering experiments. The detector position is indicated by the letter 'D'.

The measured radiation levels for three different thicknesses (x) are given below for both the broad-beam and narrow-beam situations. The following data are provided:

Density of uranium = 18.9 g/cm^3

Measured Radiation Levels for Various Thicknesses of Uranium.

Slab Thickness (cm)	Broad Beam (mR/hr)	Narrow Beam (mR/hr)
0.0	127.0	127.0
1.0	43.1	29.5
2.0	13.0	7.7
3.0	4.0	1.9

From these data, determine:

1.90 The linear attenuation coefficient of uranium for the narrow beam of 1-MeV photons.
1.91 The buildup factor for the broad beam with a slab thickness of 2.5 cm. Assume the linear attenuation coefficient is 2/cm.
1.92 What is the mass-attenuation coefficient of uranium for 1-MeV photons if the linear attenuation coefficient is 2/cm?

Scenario 1.24

The International Commission on Radiological Protection (ICRP) publishes various reports on basic radiation protection policy, practices, and research. Two such reports, Report Number 23 and Report Number 26, are of interest in this question.

1.93 ICRP Report Number 23 describes reference man as containing 140 g of potassium. The following data apply:

Mass of reference man = 70 000 g

0.012% of the potassium is K-40

K-40 decays by emitting a beta particle with a 90% probability

Maximum beta energy = 1.3 MeV

Half-life of K-40 = 1.2×10^9 years

Avogadro's number = 6.023×10^{23}

1.6021×10^{-6} erg/MeV

What is the average beta dose rate in rad per week to the whole body from K-40?

1.94 Which statement is most accurate?
   a. It is hard to identify K-40 in the presence of 10 nCi of Co-60.
   b. The quantity of K-40 does not vary by more than ±5% from individual to individual.
   c. K-40 has no regulatory significance in the whole-body counting program but serves as an important qualitative system check.
   d. K-40 should be omitted from the radionuclide library for the whole-body counting because it is of no regulatory interest.
   e. A multidetector counter will typically not identify K-40.

1.95 Strict adherence to ICRP Report Number 26 would allow:
   a. plutonium internal doses to be regulated using annual dose equivalent rather than committed dose equivalent.
   b. deletion of record-keeping for internal doses less than 50% of the allowable dose limit.
   c. consideration of internal and external dose limits separately.
   d. the worker to choose the type of respiratory protection device if use is required.
   e. use of air samples and stay-time calculations instead of respirator usage, if it is deemed to be ALARA.

1.96 The assumption of electronic equilibrium for a Co-60 source at 1-m distance is least likely to be correct at the:
   a. surface of the skin.
   b. center of a large muscle mass.
   c. bone–tissue interface.
   d. center of a large bone mass.
   e. internal surface of the lung.

Scenario 1.25

A worker at your facility received a diagnostic administration of I-131 as NaI for a thyroid function test. Your radiation protection program restricts workers to 0.1 times the ICRP-10 investigation level from certain work even if the exposure resulted from a medical procedure.

You are asked to estimate how long he must be placed on a restricted status. He is also concerned about the dose that he will receive from this diagnostic procedure. Use the following data to answer the questions for this scenario:

Administered activity = 1.0 μCi

Administrative limit (0.1 times the ICRP-10 investigation level) = 30 nCi

Biological half-life (thyroid) = 74 days

Biological half-life (whole body) = 0.4 days

Physical half-life = 8.08 days

$f_2 = 0.3$

Thyroid mass = 20 g

$S(T \leftarrow S)$ for the thyroid = $2.2 \cdot 10^{-2}$ rad/μCi-hr (MIRD-11)

1.97 Calculate the thyroid dose received from this procedure.

1.98 Based on the thyroid retention, how many days must pass until the worker can be released from restricted status?

1.99 Which one of the following statements is incorrect?

a. Because a thyroid abnormality is suspected, these calculations are only an estimate of the organ dose.
b. The most accurate method of assessing the actual dose is to obtain *in vivo* bioassay data and calculate an organ retention function.
c. The values of *S* take beta dose within the organ of interest into account, but do not consider beta doses between organs except for organs with walls and bone and bone marrow.
d. Because of the reciprocal dose theorem, the dose to testes from the thyroid is equal to the dose to the thyroid that would be produced if the same activity were in the testes.
e. The ICRP-10-derived investigation level for short-lived transportable radionuclides is based on one-quarter of the maximum permissible quarterly intake for short-lived transportable radionuclides.

Scenario 1.26

You are responsible for the health physics input to the design of a new laboratory which will be handling tritium. One room in the lab is 30 ft long, 20 ft wide, and 10 ft high and contains the primary tritium handling glove box. The glove box will contain a maximum of 10 Ci of tritium, all of which could be released into the room should an accident occur. You are concerned about the doses that could be received by an operator in the room and by an individual standing downwind at the site boundary, 1 mile away. The following data should be considered:

Breathing rate of both individuals = 3.5×10^{-4} m^3/s

Atmospheric diffusion factor at 1 mile = 1.0×10^{-4} s/m^3

Dose conversion factor for tritium (including absorption through the skin) = 158 rem/Ci inhaled

Time the operator remains in the room after the accident without respiratory protection = 30 min

Assume that the position of the operator in the room does not affect the dose received.

1.100 Determine the maximum dose equivalent that could be delivered to the operator.

1.101 Calculate the maximum dose equivalent that could be delivered to the person at the site boundary. Assume that all tritium is released to the environment in 30 min and that the person stays at the boundary for the entire release.

1.102 The ventilation system design criteria call for three complete air changes per hour in the lab. If the ventilation system works as designed, what is the maximum dose the operator could receive?

1.103 If the ventilation system works as designed, what is the maximum dose equivalent that could be delivered to the person at the site boundary?

1.104 Your Design goal for the dose equivalent delivered to the operator during the tritium accident is 500 mrem. How many air changes per hour will be required to limit the operator's dose equivalent to this value?

Scenario 1.27

1.105 Calculate the committed dose equivalent (CDE), the committed effective dose equivalent (CEDE), the annual limit on intake, and the derived air concentration for the inhalation of Cs-137. The Cs-137 decay scheme is illustrated in Figure 1.4.

The following data are given:

Specific Effective Energy (MeV per gram per transformation) of Cs-137.

Targets	**Sources**	
	Lungs	**Total Body**
Gonads	0.0	2.7×10^{-6}
Breast	0.0	2.7×10^{-6}
Red Marrow	0.0	2.7×10^{-6}
Lungs	1.9×10^{-4}	2.7×10^{-6}
Thyroid	0.0	2.7×10^{-6}
Bone surface	0.0	2.7×10^{-6}
SI wall	0.0	2.7×10^{-6}
ULI wall	0.0	2.7×10^{-6}
LLI wall	0.0	2.7×10^{-6}
Uterus	0.0	2.7×10^{-6}
Adrenals	0.0	2.7×10^{-6}

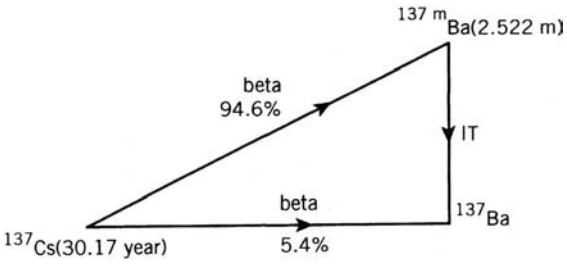


Figure 1.4 Decay scheme for Cs-137.

Specific Effective Energy (MeV per gram per transformation) of Ba-137m.

Targets	Sources	
	Lungs	Total Body
Gonads	5.7×10^{-8}	4.7×10^{-6}
Breast	2.7×10^{-6}	3.9×10^{-6}
Red Marrow	2.5×10^{-6}	4.3×10^{-6}
Lungs	9.5×10^{-5}	4.0×10^{-6}
Thyroid	2.6×10^{-6}	3.9×10^{-6}
Bone surface	2.0×10^{-6}	4.0×10^{-6}
SI wall	5.9×10^{-7}	4.9×10^{-6}
ULI wall	8.0×10^{-7}	4.8×10^{-6}
LLI wall	1.7×10^{-7}	4.9×10^{-6}
Uterus	2.1×10^{-7}	4.9×10^{-6}
Adrenals	4.9×10^{-6}	5.2×10^{-6}

Number of Nuclear Transformations Over 50 Years in Source Organs or Tissues per Unit Intake of Activity (Transformations/Bq) of Cs-137 (U_S).

Organ	Isotope	Oral	Inhalation (Class D)
	Cs-137	$f_1 = 1.0$	$f_1 = 1.0$
	Ba-137m	$f_1 = 1.0$	$f_1 = 1.0$
Lungs	Cs-137		1.9×10^4
	Ba-137m		1.8×10^4
Other tissue	Cs-137	1.2×10^7	7.7×10^6
(whole body 70 000 g)	Ba-137m	1.2×10^7	7.3×10^6

f_1 is the fraction of a stable element reaching the body fluids following ingestion.

Scenario 1.28

Assuming that Tc-99m acts as an insoluble compound, calculate the following for an uptake of 1 μCi of Tc-99m into the stomach:

1.106 The cumulated activity of Tc-99m in μCi-hr in each segment of the ICRP-30 gastrointestinal (GI) tract.
1.107 The dose equivalent in rem to the walls of each segment of the GI tract.
1.108 The maximum permissible uptake rate in μCi/hr and the maximum allowed concentration in water for occupational exposure in μCi/ml. Assume that an intake of 1100 ml/day of contaminated water is consumed during the work day and that the maximum allowed organ dose permitted at your facility is 15 rem.

ICRP-30 GI Tract Parameters.

Section of GI Tract (i)	Mass of walls (g)	Mass of Contents (g)	Mean Residence Time (day)	λ_i (day^{-1})
Stomach (ST)	150	250	1/24	24
Small intestine (SI)	640	400	4/24	6
Upper large intestine (ULI)	210	220	13/24	1.8
Lower large intestine (LLI)	160	135	24/24	1

Absorbed Dose per Unit Cumulated Activity (rad/μCi-hr) for Tc-99m with a Half-Life of 6.03 hr.

Target Organs	Source Organs			
	Stomach Contents	SI Contents	ULI Contents	LLI Contents
GI stomach wall	1.3×10^{-4}	3.7×10^{-6}	3.8×10^{-6}	1.8×10^{-6}
GI SI wall	2.7×10^{-6}	7.8×10^{-5}	1.7×10^{-5}	9.4×10^{-6}
GI ULI wall	3.5×10^{-6}	2.4×10^{-5}	1.3×10^{-4}	4.2×10^{-6}
GI LLI wall	1.2×10^{-6}	7.3×10^{-6}	3.2×10^{-6}	1.9×10^{-4}

Scenario 1.29

Two possible approaches for estimating the risk of cancer induction from exposure to low levels of ionizing radiation are ICRP-26 and the Probability of Causation (PC) Tables published by the Department of Health and Human Services (HHS).

Answer the following questions to demonstrate your understanding of these reports.

1.109 Based upon ICRP risk estimates, what is the probability of developing a radiation-induced fatal cancer over a lifetime for an average occupationally exposed radiation worker who has received 100 000 mrem of uniform, whole-body external exposure?

1.110 Assuming a normal cancer fatality rate of 20%, what would be the total probability of developing a fatal cancer for a group of occupationally exposed workers with a 3 in 1000 probability of contracting a radiation-induced fatal cancer?

1.111 The ICRP-26 risk model for cancer is based on:
- a. An absolute risk model.
- b. A relative risk model.
- c. An absolute and relative risk model.
- d. A stochastic model.
- e. A linear stochastic model.

1.112 The PC Tables are based on:
- a. An absolute risk model.
- b. A relative risk model.
- c. An absolute and relative risk model.
- d. A stochastic model.
- e. A linear stochastic model.

1.113 Which statement is not true regarding the PC tables?
- a. The formulation of these tables was mandated by Congress.
- b. Smoking history is not considered when using the tables to estimate risk.
- c. Resource of data for the table includes: rodent data, *in vitro* cell studies, and human data.
- d. The tables were published to provide scientific evidence to resolve radiation litigation cases.
- e. Prior medical X-ray exposure is not considered when using the tables to estimate risk.

Scenario 1.30

The biological effects of ionizing radiation encompass a broad range of topics. The following questions are designed to indicate your general understanding of this area.

1.114 Equal amounts of tritium as tritiated water and tritiated thymidine (a basic component of DNA) are incorporated into a large volume of cells. Which statement best describes the biological effectiveness of these compounds?
- a. Tritiated water will cause more biological damage to the cell because the cell is principally made up of water.

b. Tritiated water will cause more biological damage to the cell because tritiated thymidine is quickly metabolized by the cell.
c. Tritiated thymidine will cause a greater biological effect than tritiated water because it is incorporated into the cell's nucleus.
d. Both compounds will deliver the same biological effect because they are distributed in equal activities.
e. The biological effect will be the same for both compounds because both emit the same low-energy beta radiation and are equal in activity.

1.115 Match the following inhaled radionuclides with the adult critical organ. The critical organ may be used more than once.

a. Lung ___ Strontium-90 (soluble).
b. Bone ___ Cesium-137.
c. Total body ___ Plutonium-239 (soluble).
d. Liver ___ Uranium-238 (insoluble).
e. Kidney ___ Radon-222.

1.116 Based on the law of Bergonie and Tribondeau, order the following cells from most to least radiosensitive:

a. Mature lymphocytes.
b. Intestinal crypt cells.
c. Mature spermatocytes.
d. Erythrocytes (red blood cells).
e. Nerve cells.

1.117 Ionizing radiation has been directly associated with cataract formation. Select the statement that is incorrect.

a. The cataractogenic dose response is considered a threshold effect.
b. Fast neutrons are more effective at producing cataracts than are other forms of radiation.
c. The cataract effect is dependent on age at the time of irradiation.
d. Occupational exposure to X-rays accounts for approximately 1% of the cataracts observed in X-ray technicians.
e. Radiogenic cataracts are distinct in that they originate on the anterior epithelium of the lens.

1.118 Figure 1.5 can be used to express cell survival under a number of different irradiation circumstances. Which of the following statements is not true?

a. Curve A best represents the response of a cell system to a high dose rate, whereas curve B best represents the response to a low dose rate.
b. Curve B best represents a multitarget cell system response, whereas curve A best represents a single-target system.
c. Curve A best represents the effect in a cell system that is irradiated under hypoxic conditions, whereas curve B best represents the response of the same system under aerated conditions.
d. Curve B best represents the response of a cell system to low LET radiation, whereas curve A best represents the response of the same system to high LET radiation.

e. Curve B best represents the response of a cell system when a radioprotective compound is used, whereas curve A best represents the response of that cell system without a radioprotective compound.

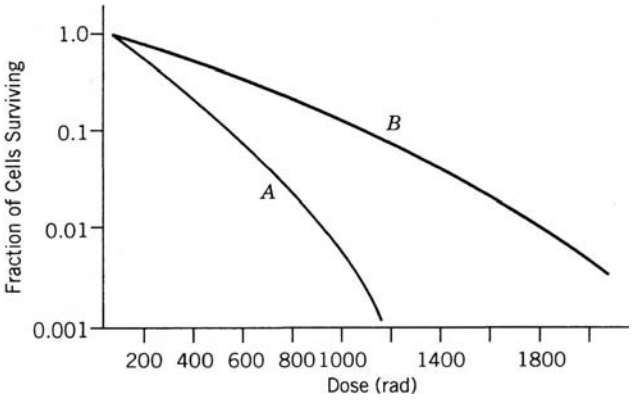


Figure 1.5 Fractional cell survival curves as a function of dose.

Scenario 1.31

The following series of questions relates to dosimetry and dose limits.

1.119 The major pathway by which soluble radioactive material is removed from the body is
   a. Perspiration.
   b. Feces.
   c. Respiration.
   d. Exhalation.
   e. Urine.

1.120 Prior to January 1993, the internal dose assessment methodology used to meet regulatory requirements in Title 10 CFR Part 20 is:
   a. NCRP-84.
   b. NCRP-91.
   c. ICRP-30.
   d. ICRP-2.
   e. ICRP-26.

1.121 The ICRP (ICRP-26) determined that, to prevent nonstochastic effects, the annual dose equivalent limit which must not be exceeded for all tissues except the lens of the eye is:
   a. 30.0 rem
   b. 5.0 rem
   c. 5.0 Sv
   d. 0.5 Sv
   e. 30 rad.

1.122 In order to limit stochastic effects, the dose limit recommended in ICRP-26 is based on the principle that the risk associated with uniform irradiation of the whole body is
   a. equal to
   b. greater than
   c. less than
   d. related to
   e. not related to
   the risk associated with nonuniform radiation.

1.123 ICRP-26 replaces the ICRP-2 concept of the critical organ with the concept of:
   a. Genetic region.
   b. Source region.
   c. Organ equivalent.
   d. Tissue region or target tissue.
   e. Weighted critical organ.

1.124 Most large data sets of measurable occupational annual dose equivalents have been found to fit a:
   a. Poisson distribution.
   b. Normal distribution.
   c. Log-normal distribution.
   d. Binomial distribution.
   e. Weibull distribution.

1.125 In 1980, the ICRP reviewed its annual dose limitation recommendations on the lens of the eye.
   They decided to:
   a. change its recommendations from 0.3 Sv to 0.15 Sv.
   b. change its recommendations from 0.3 Sv to 0.50 Sv.
   c. leave the number unchanged.
   d. drop its recommendations for the lens of the eye.
   e. make the eye limitations 0.50% of that for other tissues.

1.126 NCRP-116 recommends that the occupational cumulative effective dose limit should be:
   a. 10 mSv × age (y).
   b. 20 mSv × age (y).
   c. 50% larger than the NCRP-91 value.
   d. 50% smaller than the NCRP-91 value.
   e. 50 mSv × (age (y) – 18).

1.127 Assuming an average annual dose equivalent of 5 mSv and employing BEIR III methodology, the annual risk estimate (total radiation-induced effects) for occupational radiation workers is considered to be a nominal value of about:
   a. 1×10^{-2}
   b. 1×10^{-3}
   c. 1×10^{-4}

d. 1×10^{-5}
e. 1×10^{-6}.

How would your answer change if BEIR V and BEIR VII were the basis of the risk coefficient?

1.128 The average annual fatal accident rate in safe industries in the United States is approximately:
a. 1×10^{-3}
b. 1×10^{-4}
c. 1×10^{-5}
d. 1×10^{-6}
e. 1×10^{-7}.

Scenario 1.32

The biological effects of ionizing radiation depend upon the tissues involved and the nature of the radiation impinging upon the cellular structures. One of the more radioresilient tissues is the skin of the whole body. This scenario addresses radiation effects on skin.

1.129 Which one of the following, lists the skin response to acute radiation exposure in correct chronological order?
a. Dry desquamation, moist desquamation, erythema, recovery.
b. Moist desquamation, dry desquamation, erythema, recovery.
c. Dry desquamation, moist desquamation, recovery, erythema.
d. Erythema, moist desquamation, dry desquamation, recovery.
e. Erythema, dry desquamation, moist desquamation, recovery.

1.130 Which one of the following factors does not affect the severity of the skin's reaction to radiation?
a. Skin pigmentation.
b. Fractionation of dose.
c. Charged particle equilibrium at the basal cell layer.
d. Dose rate.
e. LET.

1.131 The radiosensitivity of skin is based on the sensitivity of which tissue?
a. Epidermal layer.
b. Basal cell layer.
c. Horny layer.
d. Hair follicle.
e. Fat cells.

1.132 If the skin were contaminated by an isotope with a half-life of 8 days, and assuming an exponential turnover time of the skin of 50% in 5 days, calculate the time to reduce the contaminant to 10% of the initial level. Assume that decontamination has been ineffective.

1.333 ICRP recommends a weighting factor of 0.01 for assessing stochastic risk to the skin.

This means that:

a. Radiogenic skin cancer is a low risk.
b. Radiogenic skin cancer is a high risk.
c. Spontaneous skin cancer is a low risk.
d. Radiogenic skin cancer exceeds spontaneous skin cancer as a risk.
e. Dose equivalent to the whole body is 100 times dose to the skin.

Scenario 1.33

The BEIR VII report gives an analysis for cancer risk assessment from exposure to low levels of ionizing radiation. This report was preceded by a number of studies that addressed human risk estimation. The following questions address biological risk from ionizing radiation.

1.134 There are several important areas for which human data are inadequate for risk estimation. Provide the types of information from animal studies that can be useful in human risk estimation.

1.135 Initiation, promotion, and progression are three distinct stages of experimental radiation-induced carcinogenesis. Identify the radiobiological factors which affect either the onset or the development of malignant tumors in experimental animals.

1.136 For this question, assume that all radiation induced effects lead to death. Based upon BEIR V, the weighted average risk of death following an acute dose equivalent of 0.1 Sv of low-LET radiation to all body organs is estimated to be:

a. 1×10^{-4}
b. 8×10^{-3}
c. 2×10^{-3}
d. 8×10^{-4}
e. 1×10^{-2}.

1.137 Two competing functional forms have been used for describing fatal cancer risks from radiation exposures.

Choose the best statement:

a. The BEIR V multiplicative risk model multiplies the dose by a constant to determine cancer risk.
b. The BEIR IV additive risk model adds a constant times the underlying risk of cancer to the age- and sex-dependent radiation dose to determine cancer risk.
c. The BEIR V relative risk model computes fatal cancer risk for individuals by an age- and sex-dependent factor times the cancer risk in the victim's relatives.
d. The additive risk model has been dropped by the BEIR V Committee in favor of the multiplicative risk model.
e. The BEIR IV absolute risk model holds that fatal cancer risk is a linear function of the absolute value of the radiation dose.

1.138 Consider a general form of a cancer risk estimate from BEIR V:

$$r(d) = r_0\,[1 + f(d)\,g(B)]$$

where
$r(d)$ = total risk
r_0 = background risk
$f(d)$ = function depending on the dose d
$g(B)$ = function of dose-modifying parameters B.

Which of the following statements is not correct?
a. The constant 1 ensures positive values of the excess risk estimate.
b. $g(B)$ may include components which depend on sex and age.
c. $f(d)$ can be a linear or linear-quadratic function.
d. r_0 can vary significantly for different populations at risk.
e. r_0 is not specifically modeled by the Committee.

Scenario 1.34

You are employed as the Radiation Health Manager by the Big Pharma Corporation (BPC) at their Elephant's Ear, AZ production facility. BPC is licensed for radiopharmaceutical production using a variety of isotopes. The BPC license is based on ICRP-26 methodology.

A worker on the production line where radioiodine materials are manufactured appears to have a positive result for I-131 in a spot urine sample. The sample was taken late in the worker's shift, after the majority of the day was spent working with I-131. No air sample results are available, but thyroid counting capability exists. The chemical form of the I-131 is NaI.

From external thyroid counting, the following data are obtained:

Time Post Intake (d)	Thyroid Activity (kBq)	IRF[a, b, c]
1	250	0.133
7	230	0.0995
10	130	0.0751

[a] Intake Retention Function (IRF) for inhalation of Class D I-131.
[b] Fraction of intake expected to be in the thyroid at this time post-intake.
[c] Radioactive decay is included in these values.

For inhalation of Class D I-131, the dose conversion factor for the thyroid is 2.9×10^{-7} Sv Bq^{-1}. Approximately 75% of the I-131 as NaI is excreted from the body in the urine in 1–2 days with an effective halftime of about 6 hours. The remaining 25% of I-131 is trapped in the subject's thyroid, reaching a maximum about 24 hours post-intake, and is excreted with an effective halftime of about 7 days. The ICRP-26 organ weighting factor (w_T) for the thyroid is 0.03.

1.139 Given that you can choose *in vivo* or *in vitro* methods of analysis to perform bioassay, describe and discuss the optimal approach for this case. In your discussion list two advantages and two disadvantages for *in vitro* and *in vivo* methods of analysis as related to this case.

1.140 How might your approach to bioassay change as time goes by, given the metabolic model for iodine?

1.141 Based on the thyroid counting data, what is your best estimate of the subject's intake? For this question, use the NUREG/CR-4884 (1987) methodology.

1.142 Assume that the intake was 5 MBq. What is the committed dose equivalent (CDE) to the thyroid for this intake? What is the committed effective dose equivalent (CEDE) for this intake? Assume that organs other than the thyroid make a negligible contribution to the CEDE. Have any regulatory limits been exceeded?

Scenario 1.35

You are an assistant professor in the Nuclear and Radiological Sciences Department at the Le Chat Institute of Technology. The following questions will be part of an exam in HP303, External Radiation Dosimetry for undergraduate health physics students. This question involves a LiF thermoluminescent dosimeter and the ICRP-60 and NCRP-116 recommendations. Candidate dosimeter configurations follow.

Chip	Material	Thickness (cm)	Cover
1	^{7}LiF	0.38	100 mg/cm^2 copper and 200 mg/cm^2 plastic
2	^{7}LiF	0.38	1000 mg/cm^2 plastic
3	^{7}LiF	0.15	7 mg/cm^2 mylar
4	^{6}LiF	0.38	300 mg/cm^2 plastic
5	^{7}LiF	0.38	300 mg/cm^2 plastic
6	^{6}LiF	0.38	300 mg/cm^2 plastic and Cd filter

1.143 An employee works in a mixed radiation field, which includes beta particles, gamma photons, alpha particles, and thermal and mixed energy fast neutrons. The absorbed dose from external sources in the work environment was reported to be 30 μGy beta, 70 μGy gamma, 90 μGy thermal neutrons, and 25 μGy fast neutrons with an average energy of 10 MeV. Calculate the ICRP-60 equivalent dose in μSv.

1.144 A 26 year old radiation worker had a lifetime effective dose of 0.32 Sv. Compare this worker's lifetime dose to the recommendations of NCRP-116.

1.145 A radiation worker recorded the following effective doses over the past 4 years:

Year	Effective Dose (mSv)
1	10
2	30
3	40
4	20

According to ICRP-60, what is the maximum recommended dose for this worker in year 5?

1.146 Given the information in the problem statement, construct a dosimeter to measure the effective dose for a laboratory worker using a Pu/Be neutron source. State the number of chips to be included in the dosimeter and limit the number of chips to a maximum of four.

1.147 Given the information in the problem statement, construct a dosimeter to measure the effective dose for an X-ray technologist. State the number of chips to be included in the dosimeter and limit the number of chips to a maximum of four.

1.148 A portable meter (i.e., BF_3) could be used to determine the neutron effective dose to an individual with:

a. knowledge of the neutron spectrum so that the proper RBE can be determined.
b. knowledge of the relationship between the neutron energy spectrum and the energy of the neutron calibration source, the ratio of gamma and neutron fluence rates, and the individual's stay-time.
c. knowledge of the magnitude of the effective dose due to photons to be subtracted from the total effective dose (i.e., the meter is 'zeroed') and application of a neutron energy correction.
d. knowledge of how the instrument responds to the spectrum as compared to the neutron calibration source as well as the individual's stay-time.
e. knowledge of near laboratory conditions controlling temperature, humidity, neutron energy, and fluence rate.

Scenario 1.36

You have been retained by the Nuclear Regulatory Commission for developing a revision to 10CFR20, which will be based on the 1990 recommendations of the ICRP. These recommendations are contained in ICRP Publication 60, *1990 Recommendations of the International Commission on Radiological Protection*, and in subsequent ICRP publications.

The NRC has requested the answers to several questions in order to enhance their understanding of ICRP-60 and its supporting documents. A portion of these

questions involves an intake of I-131. The Annual Limit on Intake (inhalation) for I-131 is 1.0×10^6 Bq. The breathing rate of the average worker is 20 L min^{-1}, and the tissue weighting factor (w_T) for the thyroid is 0.05.

1.149 Define the following ICRP-60 dosimetric quantities: a. equivalent dose, b. effective dose and c. committed effective dose.
1.150 What are the ICRP-60 recommended occupational limits for effective dose?
1.151 For a declared pregnant worker, what is the ICRP-60 recommended limit for: a. external exposure and b. intake of radioactive material?
1.152 An individual performs a job in a room that contains airborne radioactive materials. The room concentration of I-131 is 8.3×10^4 Bq/m^3. The job requires 30 minutes to complete. Calculate the committed effective dose to the worker.
1.153 If the individual in Question 1.152 suffers from thyroid disease (for example hyper- or hypo-thyroidism), why do you need additional information to evaluate the committed effective dose? What information do you need to enhance the dose assessment?

Scenario 1.37

Radiation litigation cases frequently require a technical assessment of the biological effects of ionizing radiation. You have been retained by the law firm of Whiplash, Ripov, and Scam to participate in the case preparation for a suit being filed against a nuclear utility for exposure to ionizing radiation during a worker's career. The worker was exposed to both Co-60 and fission neutrons during her career.

You have been requested to explain the dose response curves for chromosome aberrations in human lymphocytes exposed to Co-60 gamma rays (γ) and to fission spectrum neutrons (n). Following ICRU Report No. 40, the number of chromosome aberrations per cell in human lymphocytes (e) when exposed to fission neutrons is $e_n = 0.60\ D_n$ where D is the absorbed dose in Gy. For photons, the number of chromosome aberrations per cell in human lymphocytes when exposed to photon radiation is $e_\gamma = 0.0157\ D_\gamma + 0.05\ D_\gamma^2$.

1.154 What is the primary mode of interaction for the following radiation types in tissue: fast neutrons, thermal neutrons, and Co-60 gamma rays?
1.155 What property of the neutrons and gamma-rays accounts for the difference in shape of the two dose response relationships defined above?
1.156 What is the relative biological effectiveness (RBE) for neutrons for an effect of 0.5 chromosome aberrations per cell?
1.157 What is the maximum value of the RBE for chromosome aberrations for neutrons based on the information provided in the problem statement?
1.158 What value should be used for the quality factor Q for neutrons with unspecified energies? Provide the basis (source) for your answer.

Scenario 1.38

An inhalation incident involving airborne Co-60 and I-131 has occurred at the Alabama National Radiochemistry and Environmental Laboratory (ANREL). The worker immediately showered, changed clothing, and received a whole body count. Analysis of material in the incident area suggests that the Co-60 was a Class Y compound and the I-131 was Class D. Laboratory procedures specify the use of ICRP-26/30 methodology.

Co-60 has a half-life of 5.27 y, a stochastic inhalation ALI (Class Y) of 1×10^6 Bq, and a committed dose equivalent (CDE) in the lungs of 3.4×10^{-7} Sv/Bq. The ($f_{N\text{-}P}, f_{T\text{-}B}, f_P$) values are (0.0, 0.0, 100.0) where $f_{N\text{-}P}, f_{T\text{-}B}, f_P$ are the percentage contributions of the CDE in the three lung regions due to initial depositions in the nasopharyngeal (N-P), tracheobronchial (T-B), and pulmonary (P) lung regions, respectively.

Fractions of Initial Intake Remaining in the Whole Body as a Function of Inhaled Particle Size (μm) and Elapsed Time.

Elapsed Time (d)	Inhaled Particle Size		
	1 μm	5 μm	10 μm
0	0.63	0.91	1.00
1	0.57	0.80	0.87
5	0.18	0.10	0.09
10	0.14	0.06	0.04
15	0.13	0.05	0.04
20	0.12	0.05	0.03

ICRP-26 Recommended 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	0.30

Correction for particle size:

$$\frac{H_{50}\ (\text{AMAD})}{H_{50}\ (1\ \mu\text{m})} = f_{\text{N-P}}\ \frac{D_{\text{N-P}}\ (\text{AMAD})}{D_{\text{N-P}}\ (1\ \mu\text{m})} + f_{\text{T-B}}\ \frac{D_{\text{T-B}}\ (\text{AMAD})}{D_{\text{T-B}}\ (1\ \mu\text{m})} + f_{\text{P}}\ \frac{D_{\text{P}}\ (\text{AMAD})}{D_{\text{P}}\ (1\ \mu\text{m})}$$

Fraction of Intake Deposited in the Lung Compartments.

Particle Size AMAD (μm)	Deposition Probabilities			
	$D_{\text{N-P}}$	$D_{\text{T-B}}$	D_{P}	Sum
1	0.30	0.08	0.25	0.63
5	0.74	0.08	0.09	0.91
10	0.87	0.08	0.05	1.00

1.159 The Co-60 component of the whole body count result was 7.77×10^5 Bq. Assuming that the activity median aerodynamic diameter (AMAD) of the aerosol was 1 μm, estimate the intake, expressed in percent ALI, based on the whole body count result.

1.160 Calculate the committed effective dose equivalent (CEDE) for an inhalation intake of 9.25×10^5 Bq of 1 μm AMAD Class Y Co-60.

1.161 For this part only, assume that the CEDE due to Co-60 was 0.5 mSv (50 mrem). The worker has an I-131 intake that resulted in 6 mSv (600 mrem) committed dose equivalent (CDE) to the thyroid. Assume that the thyroid is the only significantly irradiated organ or tissue. During the same monitoring period, the worker also received 2.5 mSv (250 mrem) due to external radiation exposure from Co-60. What is the total effective dose equivalent (TEDE) to the worker during the monitoring period?

1.162 Another worker inhaled 1.11×10^6 Bq of Class Y Co-60. The AMAD was determined to be 10 μm. Calculate the committed dose equivalent (CDE) to the lungs.

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Part II
Specialized Areas: Theory and Problems

2
Medical Health Physics

The medical use of radiation is the major contributor to man-made radiation exposures to the public. This exposure is deliberate because it is intended to benefit the individual. Other types of public exposure, such as nuclear power plant releases, are generally acceptable because of their low levels and because they benefit society in general.

The Medical Health Physicist is responsible for maintaining the radiation exposures of the hospital staff, patients, and general public to values that are as low as reasonably achievable. In addition, the medical health physicist has radiation safety program management and regulatory compliance responsibilities. He or she will also be involved in facility design and modifications, training, and emergency response. As part of these duties, he or she must assess medical related radiation exposures that arise from a variety of sources.

Medical uses of radiation include diagnostic procedures, radiotherapy, and biomedical research. Diagnostic techniques include X-ray and nuclear medicine procedures that are used to determine noninvasively the absence, presence, and extent of disease. Radiotherapy includes external beam therapy, brachytherapy, and radionuclide therapy which utilize radiation to deposit large amounts of energy in specific tissues.

Brachytherapy, which involves implants of sources into patient tissue, provides an external radiation hazard. Other external hazards include X-ray or imaging procedures. Internal radiation hazards are derived from medical administrations of radioisotopes.

2.1
Historical Perspective

Within months of Wilhelm Roentgen's discovery of X-rays in November 1895, they were being used for medical diagnosis. In conjunction with this diagnostic use, biological effects were observed.

By the turn of the century, the destructive effects of radium on skin were noted. However, radiation was found to have a more destructive effect on diseased cells than on normal cells. By fractionating the exposure, researchers discovered that

the difference was quite significant and that radiation exposures provided a curative effect.

Radiotherapy is based on the differences in cell characteristics between diseased and healthy cells. Normal cells have active repair mechanisms, whereas cancer cells have degraded repair characteristics. Radiotherapy utilizes this difference by applying the radiation in fractionated doses over an extended period of time. During the time between exposures, the normal cells repair themselves and undergo cell division, where the cancer cells either die or are significantly reduced. Radiotherapy fails when the radiation tolerance of normal tissue is reached before all the cancer cells are destroyed.

Radiation therapy is possible because limited areas of the body can tolerate at least 10 times the lethal whole-body radiation dose. The lethal dose is relatively low because certain organs are radiosensitive. In humans, bone marrow is the first system to be destroyed by radiation and it determines the maximum survivable radiotherapy exposures.

The choice of using radiotherapy involves a careful consideration of risks. A cancer patient will normally view the radiation risks as small compared to the risks of not being treated. Often the radiation risks are smaller than the risks of other medical options, including surgery and chemotherapy.

Although radiation is a valuable medical tool, it is also a recognized carcinogen which requires that its use be restricted and controlled. Radiation is almost entirely reserved for the treatment of cancer. For that reason, the term *radiotherapy* is often referred to as *radiation oncology*, which recognizes the broader range of care provided to the cancer patient.

2.2 Medical Accelerator Physics

Radioactive implants are used in treating about 5% of cancer cases, but most treatments are administered by external radiation sources. Over the last 20 years, the electron linear accelerator has become a dominant tool replacing the betatrons and Co-60 devices. Treatment utilizes both photon and electron beams in the 4 to 20-MeV energy range. Linear accelerators are utilized by university medical centers and hospitals as a routine treatment device.

Although accelerators will be discussed in a subsequent chapter, it is appropriate to briefly address the physics of radiotherapy as it relates to accelerator-produced radiations. Photon or X-ray beams have the widest application (80%) in radiotherapy, and electron beams are used in about 10–15% of the cases. Radiotherapy doses are often as large as 60 Gy delivered in 30 or more sessions over six or more weeks. The biological effect of the incident high-energy photons or electrons is to create large electron densities in the irradiated tissue.

For photon beams, divergence and attenuation reduce the photon fluence as a function of the depth in tissue. However, the electron density builds to an equilibrium value inside the tissue. The combination of these two effects produces

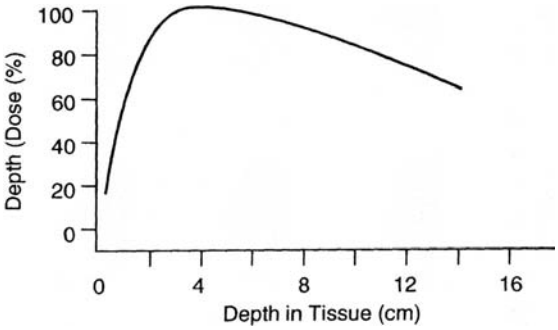


Figure 2.1 Typical depth dose curve as a function of tissue depth for a beam of 15-MeV X-rays filtered for uniformity over a 35-cm × 35-cm area. (From Ford 1993.)

a depth dose curve that rises to a maximum and then decreases with increasing depth in the tissue. Electron backscatter increases the surface dose to a value between 15% and 100% of the maximum dose. The depth of the maximum dose increases with increasing beam energy. For example, the depth dose curve for 15-MeV photons is illustrated in Figure 2.1. The depth dose curve peaks at 2.7-cm depth, and clinically useful radiation is available beyond 10-cm tissue depth.

Curves similar to that in Figure 2.1 can be developed for electron beams. In electron beams, the primary electrons slow down in tissue and produce high ionizations per unit length as they reach their maximum range. For tissue depths beyond the maximum range, the electron dose decreases very rapidly to a value of only a few percent of the maximum dose. The energy loss for high-energy electrons is about 2 MeV cm^{-1} in tissue and about twice this value in bone.

For energies below 1 MeV, the maximum dose occurs near the skin surface. Because most lesions are below the surface of the skin, it is useful to use higher-energy beams, which have a larger dose in the tissue. Figures 2.2 and 2.3 provide depth dose curves for 6-MeV and 18-MeV electrons.

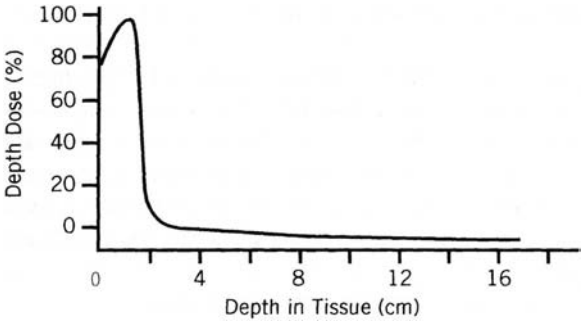


Figure 2.2 Typical depth dose curve as a function of tissue depth for 6-MeV electrons spread by a system of scattering foils to provide for uniformity over a 35-cm × 35-cm area. (From Ford 1993.)

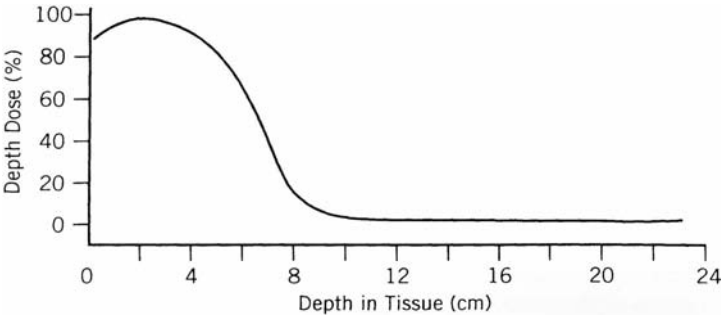


Figure 2.3 Typical depth dose curve as a function of tissue depth for 18-MeV electrons spread by a system of scattering foils to provide for uniformity over a 35-cm × 35-cm area. (From Ford 1993.)

Electron beams are useful when the tumor volume is near the skin surface. By properly selecting the beam energy, the tumor is attacked while the underlying tissue is spared. For example, a chest wall tumor must be treated without damaging the underlying lung tissue. As the electron beam energy increases from 4 to 20 MeV, the shape of the depth dose curve shifts from a surface peak to a broader plateau extending into tissue. Beyond 20 MeV, the plateau expands, and the advantage of sparing healthy tissue at depth is lost. In general, the useful electron energy range is between 4 and 20 MeV.

Neutrons have also been used for therapy applications. These probes deposit significant energy in tissue, but the depth dose profile has a broad energy deposition peak that limits the capability to localize the neutron dose at the tumor site. The broad energy deposition profile irradiates healthy tissue that can produce undesirable side effects.

Neutron therapy is performed using a range of energies including the thermal, epithermal, and fast regimes. Neutron capture therapy uses the thermal and epithermal energy regions and fast neutron therapy incorporates the highest energy regime.

In capture therapy, an isotope with a large absorption cross-section for thermal or epithermal neutrons is introduced into the body usually through injection into the blood. For example, neutron capture therapy is used to activate ^{10}B atoms to irradiate brain tumors (glioblastoma). Boron Neutron Capture Therapy (BNCT) has the potential to preferentially irradiate these tumor cells. This is achieved by selectively delivering ^{10}B to the tumor, and then irradiating the tumor with thermal/epithermal neutrons. The short range of the $^{10}B(n, \alpha)^{7}Li$ reaction products limits most of the dose to the boron-loaded tumor cells.

Fast Neutron Therapy (FNT) uses neutron beams of about 15 MeV. The therapeutic effect is achieved from the recoiling protons and heavier fragments resulting from fast neutron interactions. A variety of tumors, including those of the head and neck, salivary glands, and soft tissue, are treated with fast neutrons. For BNCT or FNT, neutrons are generated through a variety of reactions including $^{2}H(d, n)^{3}He$ and $^{3}H(d, n)^{4}He$.

Proton and heavy-ion beams produce a relatively low constant depth dose that terminates in a narrow peak at the end of the depth dose curve. The dose can be highly localized, which produces high tumor energy deposition and lower tissue doses. Dose localization may have some disadvantages when the tumor geometry is considered. A tumor may be approximated as a central mass with numerous protrusions or microextensions extending radially outward in random directions. In order to destroy the tumor, both the central mass and the microextensions must be destroyed. A highly localized beam could destroy the central tumor mass, but leave the microextensions relatively intact and capable of further growth. Therefore, some spreading in the depth dose profile is desirable.

2.3 Diagnostic Nuclear Medicine

Diagnostic techniques have been used to determine the presence and location or absence of infections, blood clots, myocardial infarctions (heart attacks), pulmonary emboli (blood clots in the pulmonary lung), occult bone fractures, and cancer. These techniques also have the ability to assess whether the organ of interest is functioning properly without disturbing the organ under observation.

To accomplish this, a photon-emitting radionuclide is attached to a biologically active chemical to form a radiopharmaceutical which is administered to a patient by intravenous injection, oral ingestion, or inhalation. Following administration, the radiopharmaceutical is monitored over time using a gamma camera to detect photons escaping from the organ being irradiated. These photons create an image of the tissue distribution of the radionuclide via a technique called *scintigraphy*, which utilizes a gamma camera (sodium iodide crystal) to detect photons emitted from the radiopharmaceutical.

2.3.1 X-rays

Diagnostic X-ray procedures include radiography, fluoroscopy, and mammography. Typical radiography involves the exposure of the body to a beam of X-rays. Attenuation of the X-ray beam depends on the intervening tissue which leads to variable darkening (contrast) of the photographic film. In diagnostic procedures, the attenuation of X-rays by the photoelectric effect is the most important contributor to the film contrast. Contrast materials with high atomic numbers provide for clearer images. Iodine ($Z = 53$) and barium ($Z = 56$) are commonly used contrast materials.

Fluoroscopy procedures take the X-ray beam, after it traverses the patient, and sends it into an imaging system that produces a video output. The video output permits real-time imaging and visual information that exceeds that obtained from snapshot X-ray images.

2.3.2 Nuclear Medicine

Nuclear medicine includes uptake studies, tracer studies, planar imaging, single photon emission computed tomography, and positron emission tomography. Commonly used diagnostic radionuclides are summarized in Table 2.1.

Table 2.1 Diagnostic Radionuclides Used in Nuclear Medicine.

Radionuclide	Organ	Half-life	Gamma Energy (keV)
Tc-99m	Heart and bone	6.0 hr	141
In-111	Labeled blood products	2.8 days	171 245
Tl-201	Heart	73.1 hr	167
I-123	Thyroid	13.2 hr	159
I-131	Thyroid	8.0 days	365
Ga-67	Tumor agent	3.3 days	185 300
Xe-133	Lung	5.2 days	81
C-11	Heart	20.4 min	511
N-13	Heart	10.0 min	511
O-15	Brain	2.0 min	511
F-18	Brain	109.8 min	511

2.3.3 Computed Tomography

Diagnostic procedures include imaging the body's cross-section via a technique known as *tomography*. When compared with planar imaging, tomography has the advantage of being able to resolve underlying organs with the aid of computational techniques. Tomography has been used to investigate abnormal physical shapes or functional characteristics in organs such as the liver, brain, or heart. It has also been used to determine the location and size of tumors in the body.

Tomography is commonly manifested in two forms: single photon emission computed tomography (SPECT) and positron emission tomography (PET). The SPECT method uses a rotating gamma camera and the same radionuclides used in conventional planar imaging. SPECT has been effective in the diagnosis of heart, liver, and brain disorders.

In contrast to the use of a rotating gamma camera, PET cameras are a stationary array of detectors that detect the two 511-keV photons produced by the annihilation

of a positron–electron pair. PET scanners rely on coincidence detectors located 180° apart or time-of-flight electronic circuitry to determine the spatial coordinates of the annihilation event, which is assumed to be at the same location as that of the positron emission. The coincidence events are assembled to produce images of the scanned organs. Currently PET scans are performed on heart patients, but brain and cancer work is also significant.

Table 2.1 provides a listing of commonly used PET diagnostic radionuclides. These radionuclides typically have short half-lives (< 2 hr) and have low atomic numbers. A cyclotron produces the positron emitter and the radionuclide is attached to a chemical compound such as $^{13}NH_3$, $H_2{}^{15}O$, ^{11}C-labeled amino acids, or ^{18}F glucose analogs.

2.3.4
Tracer Studies and Radioisotope Administration

The most commonly used diagnostic radionuclide is Tc-99m because its photon energy of 141 keV is optimal for imaging. Its half-life of 6.0 hr is long enough for the completion of most imaging procedures, but minimizes patient radiation exposures.

2.4
Therapeutic Nuclear Medicine

Therapy applications include external beam therapy via linear accelerators or Co-60 teletherapy units. Brachytherapy using encapsulated sources implanted in tissue is another common therapy application. A final class of therapy procedures includes radionuclide administrations.

2.4.1
Radionuclide Administration

Therapeutic radionuclides have a variety of uses including the suppression of hyperthyroidism, reduction in pain levels associated with metastic cancer, and the treatment of cancers and tumors. Table 2.2 summarizes commonly used beta-emitting radionuclides and alpha-emitting radiotherapy candidates, and the diseases they treat. The alpha-emitting radionuclides are in the research and development phase, and their use is currently limited by the relative difficulty of their production.

Therapeutic doses are significantly larger than those associated with diagnostic radionuclide administrations. In the treatment of thyroid cancer, administrations of up to hundreds of millicuries of I-131 are given to the patient. This quantity of I-131 is very effective in nonsurgically destroying the thyroid gland via the deposition of local thyroid doses of more than 50,000 rad. In contrast, typical diagnostic doses are less than 10 rad to any organ.

Table 2.2 Therapeutic Radionuclides Used in Nuclear Medicine.

Radio-nuclide	Half-life	Radiation Type and Average Beta Energy (keV) or Alpha Energy (MeV)	Disease Treated
P-32	14.3 d	Beta: 695	• Leukemia • Bone Cancer Pain • Pancreatic Cancer • Head-Neck Tumors • Ovarian Cancer • Rheumatoid Arthritis
Cu-67	2.58 d	Beta: 154	• Breast Cancer • Colorectal Cancer • Lymphoma • Rheumatoid Arthritis
Y-90	64.0 h	Beta: 935	• Cancer
I-131	8.0 d	Beta: 192	• Hyperthyroidism • Thyroid Cancer
Sm-153	46.7 h	Beta: 220 and 226	• Bone Cancer Pain • Leukemia • Spinal Cord Tumors
Lu-177	6.68 d	Beta: 47.3	• Bone Cancer Pain • Heart Disease
Re-186	90.6 h	Beta: 309 and 362	• Bone Cancer Pain
Re-188	17.0 h	Beta: 728 and 795	• Bone Cancer Pain
Au-198	2.7 d	Beta: 315	• Limit Spread of Ovarian Cancer
Bi-213	47 min	Alpha: 5.87	• Leukemia
Ra-223	43 min	Alpha: 6.56	• Bone Cancer Pain
Ac-225	10 d	Alpha: 5.8	• Solid Tumors

Source: Stubbs and Wilson (1991).

Beta-emitting radionuclides can destroy a tumor mass, but the radiation also affects healthy tissue. High-energy beta radiation (e.g., P-32 and Y-90) has a range that can extend beyond the tumor boundary. The irradiation of healthy tissue has a negative impact on patient recovery and can also impact their subsequent quality of life.

Healthy tissue irradiation is minimized if alpha-emitting radionuclides are administered. The alpha particles have a much shorter range in tissue than the beta particles and their use has positive advantages in terms of the preferential delivery of dose to cancer cells while minimizing the dose to healthy tissue.

2.4.2 External Beam Therapy

Large therapy doses can be delivered using accelerators and teletherapy units including Co-60 systems. General aspects of accelerator applications were discussed earlier in this chapter and will be further amplified. External beams can be combined with other techniques (e.g., brachytherapy) to enhance patient treatment options.

External beam therapy uses ions having a range of mass, charge, and energy values. The ion characteristics impact the dose deposition profile and the capability to localize dose at the tumor site.

For a tissue volume irradiated by a beam of charged ions of a given energy, the absorbed dose (D) as a function of penetration distance x into tissue is:

$$D(x) = \frac{1}{\rho}\left(-\frac{dE}{dx}\right)\Phi(x) \tag{2.1}$$

where ρ is the density of the material (tissue, tumor, or other structure) attenuating the heavy ion, $-dE/dx$ is the stopping power, and Φ is the heavy ion fluence. The particle fluence varies with tissue penetration according to the relationship:

$$\Phi(x) = \Phi(0)\exp(-\Sigma x) \tag{2.2}$$

where $\Phi(0)$ is the entrance fluence and Σ is the macroscopic reaction cross-section. The stopping power in a uniform medium for a heavy charged particle or heavy ion is given by:

$$-\frac{dE}{dx} = \frac{4\,\pi\,k^2\,z^2\,e^4\,n}{m\,c^2\,\beta^2}\left[\ln\frac{2\,m\,c^2\,\beta^2}{I\,(1-\beta^2)} - \beta^2\right] \tag{2.3}$$

where k is an electric constant (8.99×10^9 N-m^2 C^{-2}), z is the atomic number of the heavy ion, e is the magnitude of the electron charge, n is the number of electrons per unit volume in the medium interacting with the heavy ion, m is the electron rest mass, c is the velocity of light in a vacuum, β is the velocity of the ion relative to the speed of light (v/c), v is the velocity of the heavy ion, and I is the mean excitation energy of the medium interacting with the heavy ion.

The relative velocity is obtained in terms of the ion's kinetic energy (E) and rest mass (M):

$$\beta = \left[1 - \left(\frac{M\,c^2}{E + M\,c^2}\right)^2\right]^{1/2} \tag{2.4}$$

The mean excitation energy I can be represented by the following empirical formulas for an element with atomic number Z:

$$I \cong 19.0 \text{ eV} \qquad Z = 1 \tag{2.5}$$

$$I \cong (11.2 + 11.72\ Z) \text{ eV} \qquad 2 \le Z \le 13 \tag{2.6}$$

$$I \cong (52.8 + 8.71\ Z) \text{ eV} \qquad Z > 13 \tag{2.7}$$

Delivered dose is only part of the therapy protocol. A successful therapy approach must deliver the beam to the desired target volume. Fortunately, the beam can often be tracked using positron emission tomography (PET).

PET monitoring of a heavy ion beam is feasible over the entire Periodic Table. As the ion penetrates tissue, it is stripped of neutrons, which moves the nucleus further from the line of stability and increases the proton-to-neutron ratio. This change fosters the emission of positrons, which annihilate with electrons and the resulting photons are monitored using PET techniques. For example, a ^{197}Au therapy beam dose deposition profile could be monitored using the β^+ decay of ^{192}Au, ^{194}Au, and ^{196}Au.

Table 2.3 Heavy Ion Ranges in Water (cm) for Selected Energies.

Ion	Ion Energy (MeV/n)[a]			
	90	**198**	**270**	**330**
^{4}He	6.42	25.6	43.4	60.5
^{12}C	2.14	8.54	14.5	20.2
^{16}O	1.60	6.40	10.8	15.1
^{20}Ne	1.28	5.12	8.67	12.1
^{40}Ca	0.64	2.56	4.34	6.05
^{63}Cu	0.48	1.92	3.25	4.53
^{92}Mo	0.34	1.34	2.26	3.15
^{107}Ag	0.31	1.24	2.10	2.93
^{142}Nd	0.25	1.01	1.71	2.38
^{172}Hf	0.21	0.85	1.44	2.01
^{184}Os	0.20	0.82	1.38	1.93
^{197}Au	0.20	0.81	1.37	1.91
^{209}Bi	0.19	0.78	1.32	1.83
^{238}U	0.18	0.72	1.22	1.70
^{236}Np	0.18	0.70	1.18	1.65

[a] MeV/nucleon.
Source: Bevelacqua (2005).

Table 2.3 provides the range in water for a number of heavy ions including ^{4}He, ^{12}C, ^{16}O, ^{20}Ne, ^{40}Ca, ^{63}Cu, ^{92}Mo, ^{107}Ag, ^{142}Nd, ^{172}Hf, ^{184}Os, ^{197}Au, ^{209}Bi, ^{238}U, and ^{236}Np. The ions' ranges are evaluated for energies between 90 and 330 MeV/nucleon (n) that are typical of therapy applications. Although ions heavier than $A = 20$ are not routinely used in therapy applications, the results of Table 2.3 illustrate the flexibility and utility of the heavy ion approach.

The results of Table 2.3 illustrate that desired irradiation locations can be achieved by selecting specific ion and energy combinations. This characteristic is highly desirable in providing an effective treatment protocol. The ability to target a specific location by selecting the ion and its energy makes heavy ions an attractive tool for therapy applications.

An extension of the use of heavy-ion therapy is the use of beams of antiparticles and antinuclei. Initial data regarding antiproton therapy has recently been obtained.

The trajectory of antiparticles and antinuclei in matter is similar to conventional particles and nuclei until they reach the end of their range. When antihadrons annihilate, they deposit more energy than corresponding hadrons. The net result is that antihadrons boost the biological effectiveness. The emitted pions and photons from their decay offer the possibility to perform on-line imaging of the antimatter beam during patient treatment. Antinuclei therapy would be a logical extension of the use of antiparticles.

Antihadron probes present practical issues regarding their generation, storage, and delivery to the tumor site. A viable method of antimatter containment in hospital and research settings must be developed before antihadron therapy becomes a viable therapy tool. Obtaining sufficient quantities of antimatter at a reasonable cost is a major near-term challenge of this technique.

2.4.3 Brachytherapy

Brachytherapy involves the placement of radioactive material in direct contact with tumors. It is reserved for tumors that are accessible via natural body cavities (intracavitary applications) or on body surfaces (interstitial applications). I-125, Cs-137, Ir-192, and Au-198 are commonly used brachytherapy materials.

Brachytherapy presents an external radiation hazard to hospital staff, patients, and the public. The unshielded exposure rate to patient rooms in the vicinity to the brachytherapy suite is given by

$$\dot{D}_0 = m_{\mathrm{eqRa}}\, G_{\mathrm{Ra}}\, (1 - P)/r^2 \tag{2.8}$$

where

$\dot{D}_0$ = unshielded exposure rate to adjacent patient (mR/hr)
m_{eqRa} = radium equivalent activity in the brachytherapy patient (mg)
G_{Ra} = 8.25 mR-ft^2/mg-hr
P = radiation attenuation by the brachytherapy patient
r = distance between brachytherapy patient and an adjacent patient (ft)

The use of brachytherapy also exposes the patient to a potential internal radiation dose as a result of using seeds composed of radioisotopes that can be absorbed into the body as a result of seed leaks or breaches in the seed capsules. For this type of situation, MIRD methodology can be utilized to determine the dose consequence of the uptake:

$$D_T = B\, t\, S \tag{2.9}$$

where

D_T = dose to the patient's organ (rad)
B = organ burden (μCi)
T_P = physical half-life of the radionuclide (hr)
T_B = biological half-life of the radionuclide (hr)
T_{eff} = effective half-life (hr)
= $(T_P\ T_B)/(T_P + T_B)$
t = mean life of the radionuclide (hr)
= $T_{eff}/\ln(2)$
S = mean dose per unit accumulated activity (rad/μCi-hr)

The total dose, which is the sum of the external radiation exposure and the weighted internal exposure, is often of interest because it characterizes the total bodily insult from the brachytherapy. The total committed effective dose equivalent (D) from a brachytherapy source can be defined as

$$D = D_{\text{whole body}} + \sum_T w_T\ D_T \tag{2.10}$$

where the sum includes all organs or tissues that have absorbed the radionuclide leaking from the source, and w_T is the organ weighting factor.

In order to perform the calculation dictated by the second term, a variety of approaches may be utilized to calculate the absorbed dose to the organ. One approach utilizes the Medical Internal Radiation Dose (MIRD)-type methodology to calculate the absorbed dose. The MIRD methodology is described in Appendix IV.

2.5 Facility Design

The facility design depends on the characteristics of the radiation sources employed by the facility. These sources include the primary radiation source or beam, scatter radiation, leakage radiation, and radionuclide sources. Primary radiation is that radiation which exits the source after containment by a beam restriction device or shielding structure. Scattered radiation results from the interaction of the primary radiation and the scattering media, and it has an energy distribution similar to that of the primary radiation. Leakage radiation is radiation exiting the source in directions other than the intended beam direction. For X-ray tubes, where the

output is highly filtered as it passes through the beam housing, leakage radiation has a higher energy than that of the primary or scattering radiation.

The primary as well as the scatter or leakage radiation must be shielded to meet the facility design requirements. Shielding requirements depend upon the beam quality, workload, distance from source to target individual, occupancy factor, and utilization.

2.6 Shielding Design

Hospital or medical research environments utilize X-ray or imaging equipment. In order to reduce radiation exposures, shielding becomes an important consideration. In the shield design evaluation a number of input parameters are required to design a shielding configuration that is cost-effective but that meets the basic needs of the facility. One key design parameter is the facility workload (W) given by

$$W = E\, N_{\mathrm{v}}\, N_{\mathrm{p}}\, k \tag{2.11}$$

where

W = weekly workload (mA-min/wk)
E = exposure/view (mAs/view)
N_{v} = number of views per patient
N_{p} = number of patients per week
k = conversion factor (1 min/60 s)

A second key design parameter is the use factor (U_x) for structure x. The structure may be any occupied area such as a control room, office, or waiting room.

$$U_x = V_x / N_{\mathrm{v}} \tag{2.12}$$

where V_x is the number of views directed toward structure x.

A third key parameter is the occupancy factor (T) which is the factor that multiplies the workload in order to correct for the degree or type of occupancy for the area under design evaluation. When historical occupancy data are not available, NCRP-49 occupancy factor values have typically been used. A value of $T = 1$ is assumed for full occupancy areas which include wards, work areas, wide corridors large enough for personnel occupancy at a desk or workstation, restrooms used by radiation workers, play areas, living spaces, and occupied areas in adjacent buildings. Partial occupancy ($T = 1/4$) may be assumed for narrow corridors, utility rooms, restrooms not normally used by radiation workers, elevators requiring an operator, and adjacent public parking lots. Occasional occupancy areas are assigned a $T = 1/16$ value and include stairways, automatic elevators, and closets too small for work-space utilization.

These factors are important design parameters that influence the calculation of the unshielded facility exposure rate:

$$\dot{X}_u(r) = o_{PB}(T)(U)(W)\, k\, (r_0/r)^2 \tag{2.13}$$

where

$\dot{X}_u(r)$ = unshielded exposure rate at the location of interest (r) (mR/wk)
r_0 = location of the primary beam output measurement
o_{PB} = primary beam output @ r_0
T = occupancy factor
U = use factor
W = workload = mA-min/wk
k = conversion factor (60 s/1 min)
r = distance from X-ray source to point of interest

For locations not directly exposed to the primary beam, scattered radiation must be considered. The unshielded exposure rate due to scattered radiation is

$$\dot{X}_u(r)^{scatt} = o_{PB}(W)\, k\, f_{scatt}(T)\, (r_0/r)^2 \tag{2.14}$$

where

$\dot{X}_u(r)^{scatt}$ = unshielded exposure rate at the location of interest (r) in mR/wk due to the scattered radiation
f_{scatt} = ratio of scattered beam to incident beam at 1 meter

The reader should note that for scattered radiation, $U = 1$.

2.7 X-ray Shielding

A common X-ray procedure involves a collimated beam of X-rays directed toward a patient being radiographed. After passing through the patient, it is attenuated to a design value by the primary protective barrier before irradiating other individuals in neighboring offices, hallways, or waiting areas. Leakage and scattered radiations are attenuated by a secondary protective barrier before irradiating other individuals. The following discussion will outline the procedure for calculating the primary and secondary barrier shielding.

The shielding requirements depend upon the maximum beam current and voltage of the X-ray tube, and the workload, use factor, and occupancy factor for the tube and shield structure. The workload measures the use of the X-ray machine, and it is normally measured in units of mA-min/wk. The fraction of useful beam time during which the beam is pointed at the shield in question is defined as the use factor. The occupancy factor corrects for the degree of occupancy of the area in question.

X-ray shielding design requirements are defined by the recommendations of three primary documents. NCRP-49 applies to legacy facilities and is applicable to both imaging and therapy applications. New facilities for imaging follow NCRP-147. NCRP-151 provides the shielding design criteria for new therapy facilities. Each of these reports is addressed in subsequent discussion.

2.8 NCRP-49

2.8.1 Primary Barrier

The maximum exposure rate at an occupied location a distance d from the target in an X-ray tube in R/wk is

$$\dot{X}_{\mathrm{m}} = P/T \tag{2.15}$$

where P is the maximum permissible weekly exposure rate (R/wk) and T is the occupancy factor. NCRP-49 specifies 0.1 R/wk for controlled areas and 0.01 R/wk for uncontrolled areas. The maximum exposure may be used to obtain the exposure at other distances if the source geometry is known. Normally, a point source approximation is applicable.

For point source conditions, the exposure rate at 1 meter $(\dot{X}_1)$ from the X-ray tube target is

$$\dot{X}_1 = d^2\ \dot{X}_{\mathrm{m}} = d^2\ P/T \tag{2.16}$$

If it is assumed that the exposure is due to a workload W and use factor U, the ratio K may be defined:

$$K = \frac{\dot{X}_1}{W\ U} = \frac{d^2\ P}{W\ U\ T} \tag{2.17}$$

where K is the primary beam ratio or transmission factor in units of R-m^2/mA-min, and W is the workload in mA-min/wk.

Values of the transmission factor K have been measured for various X-ray energies and shielding thicknesses. Once the transmission factor is known, the required barrier thickness may be obtained from tabulated K versus shield thickness curves. NCRP-49 provides transmission versus shield thickness curves for common shielding materials such as concrete and lead.

2.8.2 Secondary Barrier

Scattered and leakage radiation are attenuated by the shielding of the secondary barrier. The required shield thicknesses to attenuate scattered and leakage radiation are calculated separately. NCRP-49 provides guidance for determining the required shielding. If the required thicknesses are about the same, an additional half-value layer is added to the larger thickness. If the difference between the calculated shielding thicknesses for scattered and leakage radiation is at least one-tenth-value layer, then the larger of the two values should be selected.

The intensity of the scattered radiation is a function of a number of parameters such as the scattering angle, primary beam energy, and the geometry of the scattering area.

The secondary barrier transmission factor relationship is based on the assumption that the point source approximation is applicable and that the exposure rate from the scattered radiation is directly proportional to the scattering area.

The transmission relationship for scattered radiation is written in terms of a scattering transmission factor (K_{ux}) with units of R-m^2/mA-min:

$$K_{ux} = \frac{400\ P\ (d_{scat})^2\ (d_{sec})^2}{a\ W\ T\ F\ f} \tag{2.18}$$

where P is the maximum weekly exposure (R), d_{scat} is the distance between the X-ray tube's target and the object scattering the X-rays (normally the patient undergoing the procedure), d_{sec} is the distance from the scatterer to the point of interest that is shielded by the secondary barrier, a is the ratio of scattered to incident radiation, and F is the scattering field size (in centimeters squared). NCRP-49 tabulates values of the ratio a based on $F = 400$ cm^2. The F factor is the actual scattering field size. The final factor appearing in Eq. (2.18) is f, which is a factor accounting for the fact that the X-ray output increases with voltage; that is, smaller K_{ux} values require larger shield thicknesses. Values assigned to f are summarized in Table 2.4.

Table 2.4 Values of the Factor f.

Voltage (kV)	f
≤ 500	1
1000	20
2000	300
> 2000	700

2.8.3 Leakage Radiation

The protective tube housing limits the leakage radiation. Once a measured leakage value is determined at a fixed distance, usually 1 m, it is possible to determine the required barrier thickness at other distances if the energy of the leakage radiation is known. The leakage radiation is filtered and hardened in traversing the tube housing and emerges in a narrow, essentially monochromatic, energy region. This leakage property suggests that the half-value layer is strongly dependent on the voltage across the X-ray tube.

For a diagnostic X-ray tube, the leakage is restricted to 0.1 R in an hour at a distance of 1 m from the tube when it is operating at its maximum voltage and current rating. The leakage attenuation factor (B_{Lx}) from a diagnostic X-ray tube is

$$B_{Lx} = \frac{600\ P\ d^2\ I}{W\ T} \qquad \text{(diagnostic)} \tag{2.19}$$

where d is the distance from the X-ray tube target to the point of interest and I is the X-ray tube electron beam current. The other factors appearing in this equation were previously defined.

The leakage radiation relationship for therapeutic exposures depends on the operating tube voltage. For a therapeutic tube with an operating voltage less than or equal to 500 keV, the leakage is limited to 1 R in an hour at 1 m from the target of the X-ray tube. The barrier attenuation relationship for voltages of 500 keV or less is given by

$$B_{Lx} = \frac{60\ P\ d^2\ I}{W\ T} \qquad \text{(therapeutic, V} \leq \text{500 keV)} \tag{2.20}$$

This relationship is altered for potentials greater than 500 keV. For therapeutic X-ray tubes with potentials greater than 500 keV, the leakage is limited to 0.1% of the intensity of the useful beam at 1 m. The attenuation factor for this case is

$$B_{Lx} = \frac{1000\ P\ d^2}{W\ T}(1/\dot{X}_n) \qquad \text{(therapeutic, V} > \text{500 keV)} \tag{2.21}$$

where $\dot{X}_n$ is the exposure rate at 1 meter from the therapeutic x-ray tube when the electron beam current is 1 mA.

The number of half-value layers (n) required to obtain the desired attenuation (B_{Lx}) is obtained from the relationship

$$B_{Lx} = \left(\frac{1}{2}\right)^n \tag{2.22}$$

The barrier thickness as a function of the barrier attenuation is tabulated in NCRP-49 for common materials such as lead and concrete as a function of peak voltage.

2.9 NCRP-147

NCRP-147 addresses structural shielding design for medical X-ray imaging facilities and it supersedes the corresponding parts of NCRP-49. The recommendations of NCRP-147 apply only to facilities designed after November 19, 2004. NCRP-147 is intended for use in planning and designing new facilities and in remodeling existing facilities. Facilities designed before the publication of NCRP-147 and meeting the requirements of NCRP-49 need not be re-evaluated.

In NCRP-147, the recommended quantity for shielding design calculations for X-rays is air kerma (K). Air kerma is defined as the sum of the initial kinetic energies of all the charged particles liberated by the incident uncharged particles or photons per unit mass of air, measured at a point in air. The unit of air kerma is the gray (Gy) defined as a J/kg.

The shielding design goals (P values) in NCRP-147 apply only to new facilities and new construction. These goals do not require retrofitting existing facilities. The NCRP-147 shielding design goals are 0.1 mGy/week (5 mGy/y) air kerma for Controlled Areas and 0.02 mGy/week (1 mGy/y) air kerma for Uncontrolled Areas.

In a shielding calculation, the distance to the occupied area of interest should be taken from the source to the nearest likely approach of the sensitive organs of a person to the barrier. For a wall, this may be assumed to be not < 0.3 m. For a source above potentially occupied spaces, the sensitive organs of the person below can be assumed to be not > 1.7 m above the lower floor. For ceiling transmission, a distance of at least 0.5 m above the floor of the room above the source is generally reasonable.

The workload of a medical imaging X-ray tube is the time integral of the X-ray tube current over a specified period (normally a week) and is normally expressed in units of mA-min/wk:

$$W_{\text{tot}} = N\, W_{\text{norm}} \tag{2.23}$$

where W_{tot} is the total workload per week (mA-min/wk), N is the average number of patients per week (patient/wk), and W_{norm} is the normalized workload or the average workload per patient (mA-min/patient). NCRP-147 tabulates the W_{norm} values for specific types of installations.

2.9.1 Unshielded Air Kerma

As noted earlier, the shielding design goals are defined in terms of air kerma. The unshielded primary air kerma ($K_{\text{P}}(0)$) in mGy/wk is:

$$K_{\text{P}}(0) = \frac{K_{\text{P}}^{1}\, U\, N}{d_{\text{P}}^{2}} \tag{2.24}$$

where K_P^1 is the unshielded primary air kerma per patient at 1 m (mGy-m^2/patient), U is the use factor, and d_P is the distance (m) from the X-ray tube to the occupied area.

NCRP-147 provides a new allowance for preshielding of the primary X-ray beam, and the primary beam intensity is substantially reduced due to attenuation by the patient, the image receptor, and the structures supporting the image receptor. The attenuation provided by the preshielding material is expressed as an equivalent thickness of shielding material. This equivalent thickness is designated x_{pre}.

In NCRP-49, it was recommended to separately calculate the contributions from scattering and leakage radiation. In NCRP-147, all secondary radiation (leakage and scattering) is typically calculated together.

The air kerma ($K_{sec}(0)$) in mGy/wk from unshielded secondary (leakage and scatter) radiation is written as:

$$K_s(0) = \frac{K_s^1 N}{d_{sec}^2} \tag{2.25}$$

where d_{sec} is the secondary distance from the X-ray tube (m) and K_{sec}^1 is the total secondary air kerma including the leakage and scatter contributions (mGy-m^2/patient). Equation (2.25) assumes that the distances relevant for leakage (d_L) and scatter (d_S) radiation are equal. Using the shorter of these two distances for d_{sec} is an acceptable choice for shielding applications.

2.9.2 Shielding Calculations

Shielding calculations determine the thickness of a barrier that is sufficient to reduce the air kerma in an occupied area to a value ≤ P/T (i.e., the weekly shielding design goal modified by an occupancy factor for the area to be shielded). The broad beam transmission function [$B(x)$] is defined in NCRP-147 as the ratio of the design air kerma behind a barrier of thickness x to the air kerma at the same location with no intervening barrier:

$$B(x_{barrier}) = \left(\frac{P}{T}\right)\frac{d^2}{K^1 N} \tag{2.26}$$

where $x_{barrier}$ is the acceptable barrier thickness, d is the distance between the radiation source and the individual beyond the barrier (m), K^1 is the average unshielded air kerma per patient at 1 m from the source (mGy-m^2/patient), and N is the expected number of patients examined per week. Once calculated, the broad beam transmission function yields the shielding thickness from an appropriate transmission curve. In addition to transmission curves, a parameterized, algebraic solution for the barrier thickness is obtained in terms of three fitting parameters α, β, and γ that depend on the barrier composition, workload, and the peak machine voltage.

When evaluating shielding for primary barriers the broad beam transmission function and barrier thickness are:

$$B_{\mathrm{P}}(x_{\mathrm{barrier}} + x_{\mathrm{pre}}) = \left(\frac{P}{T}\right)\frac{d_{\mathrm{P}}^2}{K_{\mathrm{P}}^1\, U\, N} \tag{2.27}$$

$$x_{\mathrm{barrier}} = \frac{1}{\alpha\,\gamma}\ln\left[\frac{\left(\frac{N\,T\,U\,K_{\mathrm{P}}^1}{P{d_P}^2}\right)^{\gamma} + \frac{\beta}{\alpha}}{1 + \frac{\beta}{\alpha}}\right] - x_{\mathrm{pre}} \tag{2.28}$$

The relationships for shielding secondary barriers have a similar form:

$$B_{\mathrm{sec}}(x_{\mathrm{barrier}}) = \left(\frac{P}{T}\right)\frac{d_{\mathrm{sec}}^2}{K_{\mathrm{sec}}^1\, N} \tag{2.29}$$

$$x_{\mathrm{barrier}} = \frac{1}{\alpha\,\gamma}\ln\left[\frac{\left(\frac{N\,T\,K_{\mathrm{sec}}^1}{P{d_{\mathrm{sec}}}^2}\right)^{\gamma} + \frac{\beta}{\alpha}}{1 + \frac{\beta}{\alpha}}\right] \tag{2.30}$$

In Eqs. (2.29) and (2.30) the use factor is normally unity for secondary barriers.

2.10 NCRP-151

NCRP-151 supersedes the recommendations of NCRP-49 pertaining to medical therapy facilities. It applies only to new facilities and new construction designed after December 31, 2005.

The NCRP-151 shielding design goals are defined in terms of the dose equivalent. These goals are 0.1 mSv/week (5 mSv/y) for Controlled Areas and 0.02 mSv/week (1 mSv/y) Uncontrolled Areas.

A Two-Source Rule is applied whenever a location is to be shielded from two different sources of radiation, each passing through the same barrier. The thickness of the barrier should be equal to the greater of the two individual thicknesses if they differ from one another by more than a tenth-value layer (TVL). Otherwise, the shielding thickness should be equal to the greater thickness plus one half-value layer (HVL) as determined by the more penetrating of the two radiation sources. This rule is similar to the NCRP-49 rule for addressing leakage and scatter radiation.

For shielding applications, the workload for radiotherapy equipment is the time integral of the absorbed-dose rate determined at the depth of the maximum absorbed dose, 1 m from the source. The units for W are Gy/wk, and the workload at a distance d_2 is:

$$W_2 = W\left(\frac{1\ m}{d_2}\right)^2 \tag{2.31}$$

NCRP-151 assumes all length dimensions are in m.

2.10.1 Primary Barrier

The transmission factor of the primary barrier B_{pri} that reduces the radiation field to the design goal is:

$$B_{\text{pri}} = \frac{P\ d_{\text{pri}}^2}{W\ U\ T} \tag{2.32}$$

where P is the shielding design goal (Sv/wk) expressed as dose equivalent, d_{pri} is the distance from the X-ray target to the point protected (m), W is the workload or photon absorbed dose delivered at 1 m from the X-ray target per week (Gy/wk), U is the use factor for the barrier under consideration, and T is the occupancy factor for the protected location. The required number (n) of tenth-value layers of shielding is given in terms of the primary barrier transmission factor:

$$n = -\log(B_{\text{pri}}) \tag{2.33}$$

The primary barrier thickness (t_{barrier}) is given by:

$$t_{\text{barrier}} = TVL_1 + (n-1)\,TVL_{\text{e}} \tag{2.34}$$

where the first (TVL_1) and equilibrium (TVL_{e}) tenth-value layers of the desired material are used to account for the spectral changes in the radiation as it penetrates the barrier. When a barrier of thickness (t) is greater than TVL_1, the primary barrier transmission factor (B) is given by:

$$B_{\text{pri}} = (10^{-1})\,10^{-\left[\frac{(t-TVL_1)}{TVL_{\text{e}}}\right]} = 10^{-\left(1+\left[\frac{(t-TVL_1)}{TVL_{\text{e}}}\right]\right)} \tag{2.35}$$

If the material used in the primary barrier is concrete, then the barrier will adequately absorb all photoneutrons and neutron capture gamma rays and no additional shielding is required.

2.10.2 Secondary Barriers

Since leakage and scattered radiation have different energies, the secondary barrier requirements of each are calculated separately when determining the final recommended shielding configuration. Recommended barrier transmission relationships are provided in subsequent discussion.

2.10.3 Scattering

The barrier transmission required for radiation scattered by the patient (B_{ps}) is:

$$B_{ps} = \frac{P}{a\ W\ T} d_{sca}^2\ d_{sec}^2\ \frac{400\ \text{cm}^2}{F} \tag{2.36}$$

where a is the scatter fraction or fraction of the primary beam absorbed dose that scatters from the patient at a particular angle, W is the workload for radiotherapy equipment (absorbed dose delivered to the isocenter in a week, Gy/wk), d_{sca} is the distance from the X-ray target to the patient or scattering surface (m), d_{sec} is the distance from the scattering object to the protected point (m), and F is the field area at mid-depth of the patient at 1 m (cm^2). The use factor is assigned a unit value for scattering.

2.10.4 Leakage

The barrier transmission of leakage radiation (B_L) is essentially the same as the NCRP-49 format:

$$B_L = \frac{P\ d_L^2}{(0.001)\ W\ T} \tag{2.37}$$

where the 0.001 factor is derived from the assumption that leakage radiation from the accelerator head is 0.1% of the useful beam. In this equation, the use factor is 1.0 and d_L is the distance from the isocenter, if the accelerator gantry angles are, on average, symmetric. If this is not the case, then d_L should be taken from the closest approach of the accelerator head to each barrier and actual use factors should be included in the denominator of Eq. (2.37).

2.11 Management of Radionuclide Therapy Patients

NCRP-155 addresses the management of radionuclide therapy patients. For purposes of limiting radiation doses, the patient's family is grouped in a separate category than members of the public. A family member is any person providing support and comfort to a patient on a regular basis and is considered by the patient as a member of their family by birth, marriage, or a close, loving relationship. Selected family members (i.e., pregnant women and children) are limited to the public dose limit recommendation.

Since radionuclide therapies are utilized no more than a few times in a patients lifetime, this frequency of radionuclide administration justifies the classification of a family member as being infrequently exposed with the NCRP-116 recommended annual effective dose limit of 5 mSv. In addition, NCRP Commentary No. 11 recommended that an adult member of the patient's family be permitted to receive 50 mSv/y on the recommendation of the treating physician. With these considerations, NCRP-155 recommends an annual effective dose limit of 5 mSv for family members. Other family members (i.e., children and pregnant women) are limited to 1 mSv/y.

For radiation exposure recommendations, members of the public include other patients, visitors to the medical facility, and facility staff not specifically trained in radiation safety. Other patients being treated at the facility may be exposed to patients receiving radionuclide therapy. This exposure typically occurs from occupancy of a room immediately adjacent to the radionuclide therapy patient. NCRP-155 recommends that members of the public are limited to 1 mSv/y.

The shielding required to meet the NCRP-155 recommendations can be determined in terms of a half-value layer equation:

$$\left(\frac{1}{2}\right)^N = \frac{E_{\text{NCRP-155}}}{E_0} \tag{2.38}$$

where

N = number of half-value layers of shielding needed to meet the NCRP-155 effective dose recommendation
$E_{\text{NCRP-155}}$ = NCRP-155 effective dose recommendation
E_0 = unshielded effective dose

Once N is known, the required shielding thickness (t) is readily obtained from the definition of the half-value layer:

$$t = N\, t_{\text{HVL}} \tag{2.39}$$

where

t = thickness of shielding required to meet the NCRP-155 recommendations
t_{HVL} = half-value thickness

2.12 Ventilation Considerations

Another interesting aspect of exposing a patient to an internal administration of radionuclides is that the patient will begin to excrete the material and becomes a source of exposure to other hospital occupants. Because the patient is normally confined to a room, the room ventilation parameters play a key role in the determination of the extent of this hazard.

With a single pass system and the assumption of uniform mixing of the air, the air concentration may be determined by considering (a) the total volume of air handled in the room in a specified time period such as 1 year and (b) the total activity excreted by the patient. The volume of air passing through the treatment room in a year (ml/year) $(\dot{V})$ is given by

$$\dot{V} = V\, r\, k \tag{2.40}$$

where

V = room volume (ft^3)
r = number of room turnovers per hour (hr^{-1})
k = conversion factor
= (24 hr/day) (365 day/year) (2.832×10^4 ml/ft^3)

The resulting activity released into the room per year $(\dot{A})$ is

$$\dot{A} = A\, N\, f \tag{2.41}$$

where

N = number of subjects treated per year
f = loss factor or the fraction of material injected into the patient that is released into the room
A = average activity burden of the patient (μCi/subject)

The average from room concentration $(\overline{C})$ may be determined from the information derived above:

$$\overline{C} = \dot{A}/\dot{V} \tag{2.42}$$

The average room concentration may be compared to applicable standards or used as the basis for an internal dose assessment.

2.13 Scenarios

Scenario 2.1

Your hospital has decided to dedicate a lead-shielded room for brachytherapy patients to reduce exposure from this treatment to staff, adjacent patients, and visitors. Typical treatments for gynecological implants are Cs-137 sealed sources. Assume that the adjacent patient could remain in her bed for an entire 7-day period during her hospital stay. Also assume that the brachytherapy room is continuously occupied for at least a week by a patient implanted with Cs-137. A diagram is provided in Figure 2.4.

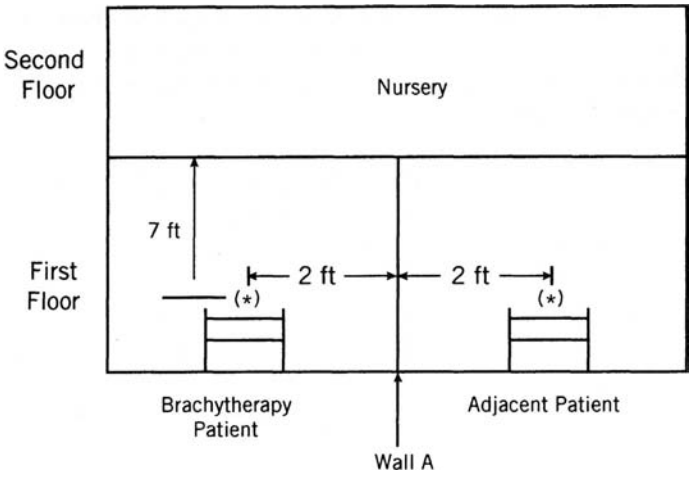


Figure 2.4 Hospital cross-section including the brachytherapy suite and its location relative to the nursery and an adjoining patient room. The symbol "(*)" denotes the patient locations.

Data

Specific gamma ray constant for radium = 8.25 mR-ft^2/mg-hr

Patient attenuation = 30%

Brachytherapy patient is: 2 ft from wall A;
4 ft from adjacent patient's bed;
7 ft from a nursery above;
able to move freely about the room

HVL for Cs-137 in lead = 0.65 cm

Mass attenuation coefficient for concrete = 0.06 cm^2/g

Density of concrete = 2.4 g/cm^3

Physical half-life of I-125 = 60 days

Biological half-life of I-125 = 138 days

S for I-125 = 3.0×10^{-3} rad/μCi-hr

$w_T = 0.03$ for the thyroid

2.1 What are the NCRP-155 dose recommendations regarding the dose a patient receives from other patients during a hospital stay?

2.2 Based on a maximum activity of 70-mg Ra equivalent, calculate the amount of lead in wall A necessary to comply with the NCRP-155 recommendations for adjacent patient exposure. For this question, assume that the adjacent patient only has a single hospital visit during the year, and that the dose at the bed controls implementation of the NCRP recommendations.

2.3 Using the same information as in the preceding question, calculate how much concrete is needed for the ceiling above.

2.4 List differences related to radiation protection for Cs-137 and I-125 brachytherapies.

2.5 A patient who received I-125 brachytherapy using seeds to treat prostate cancer had a confirmed I-125 thyroid burden of 300 nCi detected 3 days after the procedure. Calculate the dose equivalent from the measured uptake to the thyroid using ICRP-26 methodology. Ignore back-decay.

2.6 A week after the initial 300-nCi measurement, a second measurement indicated 330 nCi. As a health physicist, how would you evaluate this situation?

Scenario 2.2

A physician in your hospital wants to assess liver metabolism in diabetics using a C-14-labeled glucose compound. He wants to study a total of 20 patients. Assuming that all of the activity is instantaneously absorbed by the liver, glucose is metabolized into CO_2 with a 67-hr half-life. $^{14}CO_2$ is released from the whole body via the lungs with a 1.2-hr half-life. All 20 subjects will be studied in the same hospital room. The patients will be treated one at a time over a 1-year period. Each patient will be in the room for 6 hr post injection so that blood samples may be obtained. The following information should be evaluated:

Activity administered to each patient = 200 μCi

$S_{\text{liver–liver}} = 5.8 \times 10^{-5}$ rad/μCi-hr

$S_{\text{whole body–whole body}} = 1.5 \times 10^{-5}$ rad/μCi-hr

ICRP weighting factor for the liver = 0.06

Hospital's administrative limit for $^{14}CO_2 = 1 \times 10^{-6}$ μCi/ml

1 $ft^3 = 2.832 \times 10^4$ ml

2.7 Based on this information, calculate the committed effective dose equivalent to the patient.

2.8 Given that the room is 18 ft × 20 ft × 8 ft, uses single-pass air, and has three room-air changes per hour, assess the likelihood that the hospital's administrative C-14 limit for air will be exceeded over the year period in which the project is scheduled. Assume uniform mixing and that 20% of the administered activity will be exhaled into the room as $^{14}CO_2$.

2.9 Would recirculation of the room air affect your answer to question 2.8? Assume that the room represents 1/50 of the total air volume handled, that all of the areas served are turned over three times per hour, and that there is 33% fresh-air makeup.

2.10 Using the following equation, calculate the quantity of $^{14}CO_2$ released into the room per patient. Assume a 6-hr release period.

$$N_G \xrightarrow{\lambda_G} N_{CO_2} \xrightarrow{\lambda_C} N_{Lung}$$

$$N_{CO_2}(t) = \frac{\lambda_G}{\lambda_C - \lambda_G} N_G(0) \left[\exp(-\lambda_G t) - \exp(-\lambda_C t)\right]$$

where G and C label the glucose compound and CO_2, respectively.

Scenario 2.3

You are asked to recommend shielding for the mammographic suite, illustrated in Figure 2.5. It can be assumed that the average kVp for all the views taken is 30 and that the average film requires 120 mA-s in order to get an appropriate density. On average, 40 women per week will be radiographed. Each breast will be imaged once in the cephalocaudal orientation (primary beam directed straight down) and once in the mediolateral view (primary beam directed toward the control booth wall or the outside wall).

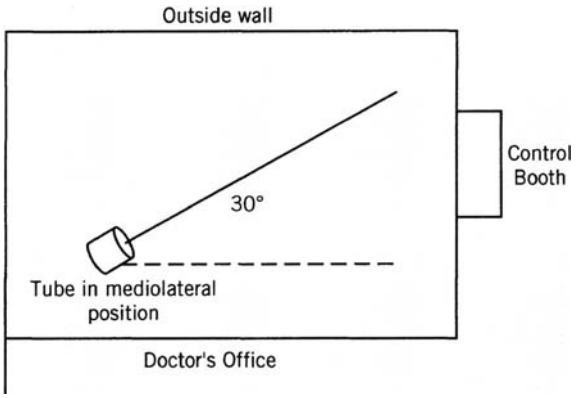


Figure 2.5 Geometry of the mammographic suite and its relationship to the doctor's office, an outside wall, and the control booth.

2.11 Calculate the weekly workload for this room.

2.12 Based upon the guidance give in NCRP-49, the cost of shielding will not increase significantly if the design exposure for the operator's position is based on:

a. 2 mR/wk
b. 1 mR/wk
c. 100 mR/wk
d. 10 mR/wk
e. 5 mR/wk

2.13 Calculate the use factor for the control booth.

2.14 Based upon the recommendations of NCRP-49, the assumed occupancy factor (OF) for the control booth would be:

a. 0.10
b. 0.25
c. 0.50
d. 0.75
e. 1.00

2.15 Based upon the original layout, calculate the required thickness of gypsum board needed to ensure that the operator in the control booth receives no more than 5 mR/wk from the primary beam. Consider the following in performing your estimate:

Data

Primary beam output = 1.0 mR/mA-s at 100 cm

Occupancy factor = 0.5

Use factor = 0.5

Workload = 160 mA-min/wk

Distance from X-ray source to point of interest (mediolateral position) = 200 cm

A transmission factor of 8.3×10^{-3} corresponds to 1.1 cm of gypsum board.

2.16 Based upon the original layout, calculate the required thickness of gypsum board needed to ensure that an individual in the doctor's office receives no more than 5 mR/wk from the scattered radiation. Consider the following in performing your estimate:

Data

Primary beam output = 1.0 mR/mA-s at 100 cm

Scatter to incident ratio at 1 m = 0.0003

Occupancy factor = 0.6

Workload = 160 mA-min/wk

Distance from point of scattering (0.5 m from X-ray source) to point of interest = 150 cm

2.17 Assuming a leakage rate of 100 mR/hr at 100 cm when the unit is operated at 30 kVp and 7 mA continuously, calculate the required thickness of gypsum board to ensure that an individual in the doctor's office receives no more than 5 mR/wk from leakage radiation. Consider the following in formulating your response:

Data

HVL (gypsum) for 30 kVp = 0.5 cm

Occupancy factor = 0.5

Workload = 640 mA-min/wk

Distance from point of X-ray source to point of interest = 200 cm

Scenario 2.4

Consider an Mo-99/Tc-99m generator and radiopharmaceutical production. Answer the following questions after considering the information provided:

Data

1.

Isotope	Half-life
Mo-99	66 hr
Tc-99m	6 hr

2. Linear attenuation coefficient for Tc-99m gammas in lead is 3.25/cm.
3. Gamma constant for Tc-99m = $0.56 \frac{\text{R-cm}^2}{\text{mCi-hr}}$.

2.18 On day 1 at 8 a.m. an Mo-99/Tc-99m generator is milked of all its Tc-99m. The amount of Mo-99 present on day 1 at 8 a.m. is 1000 mCi. How much Tc-99m will be available for milking at 8 a.m. on day 2? Assume that all Mo-99 decays result in Tc-99m.

2.19 A pharmaceutical kit for a sulfur colloid contains 5 cm^3 of appropriate liquid. Assuming that the specific activity of the Tc-99m is 80 mCi/cm^3 at 8 a.m. On day 2, how many cubic centimeters of the Tc-99m milking must be placed with the kit's 5 cm^3 so that the specific activity of the sulfur colloid at 4 p.m. of day 3 is 10 mCi/cm^3?

2.20. If a lead syringe shield of 5-mm thickness is used on a syringe containing 50 mCi, what will the exposure rate be at 50 cm?

It is now 8 a.m. on day 90. This time is applicable for questions 2.21 and 2.22.

2.21 Assuming that the activity on day 1 is 1000 mCi, what activity of Mo-99 will be present?

2.22 A recent manufacturer's bulletin warns that the generator might be contaminated with Ru-106. You survey the spent generator column with a pancake GM probe, and you observe 3700 dpm in contact with the column. Explain this observation.

Scenario 2.5

A 30-keV photon beam with 4.0×10^{11} photons per second is to be used for angiography. The beam is 0.5 mm high by 123 mm wide. The patient is positioned in a chair that moves vertically through the beam.

Data

u/p for Al at 30 keV = 1.12 cm^2/g

u_{en}/p for Al at 30 keV = 0.87 cm^2/g

u_{en}/p for air at 30 keV = 0.15 cm^2/g

Aluminum density = 2.7 g/cm^3

W_{air} = 34 eV/ion pair = 34 J/coulomb

2.23 Calculate the beamline exposure rate.

2.24 Calculate the thickness of an aluminum beam stopper to reduce the exposure rate to 2.0 mR/hr. The stopper is located downstream of the collimator. Assume that the collimator is very thick.

2.25 Calculate the patient's beamline exposure for a constant vertical chair movement of 60 mm/s.

2.26 Discuss the use of a transmission ionization chamber located in front of the patient to provide a direct measurement of the surface dose.

2.27 List parameters that must be monitored to ensure that the patient exposure is maintained below 1.0 R for a single procedure.

Scenario 2.6

During a radiographic study of a small child, it was necessary to restrain the patient. The child's mother was asked to do this by standing alongside the X-ray table while holding the child motionless with her hands. The mother was supplied with a lead apron and lead gloves. No part of the mother's body was in the direct X-ray beam. Subsequently, the woman discovered that she was pregnant. Her physician is concerned about possible effects of the radiation exposure to the woman and fetus.

You are at home and about to leave for a two-week vacation when the physician calls and tells you of the aforementioned situation. He requests that you quickly assess the relevant radiation doses. During the subsequent 30-min phone conver-

sation you will be asked the questions in this scenario. Use your best judgment to assess any information that the physician does not have available.

Data

Voltage = 70 kVp

70 kVp yields 4 mR/mA-s at 40 in.

Filtration = 2.5 mm Al

Time/current = 25 mA-s view^{-1}

Film size = 8 in. × 10 in.

Source image detector distance = 40 in.

Number of views = 5

Scatter radiation at 1 m = 0.1% of the primary beam

Distance between the primary beam and the abdomen = 18 in.

2.28 Excluding the lead apron, estimate the dose to the skin of the woman's abdomen.
2.29 Estimate the dose to the fetus. Assumptions regarding X-ray penetration through the lead apron and percent depth dose at the uterus should be stated.
2.30 What guidelines exist regarding fetal X-ray exposure?
2.31 What other information should you provide to the physician?
2.32 Based upon your calculated dose to the fetus, discuss the practice of not allowing fertile females to hold patients during X-ray procedures.

Scenario 2.7

A patient is to be given a 200-mCi I-131 oral therapeutic dose as an iodide for an inoperable thyroid metastasis. The I-131 is administered in solution form. Thyroid surgery was unsuccessful during a previous hospitalization. This scenario addresses the health physics aspects of the I-131 administration and the following hospitalization of the patient.

Data for I-131

Thyroid dose commitment = 6.6 rem/μCi in thyroid

Gamma constant = 2.2 R hr^{-1} mCi^{-1} cm^2

Radiological half-life = 8.05 days

2.33 Quantify the internal and external radiation hazards associated with I-131.

2.34 What is the contact (1 cm) exposure rate and the exposure rate at 1 m for the container holding the 200-mCi administration? Assume no shielding and that the activity may be treated as a point source.

2.35 What exposure control and health physics practices should be utilized as part of the I-131 administration and subsequent hospitalization of the patient?

Scenario 2.8

You are a medical health physicist at the Packer Memorial Hospital in Green Bay (PMHGB), Wisconsin. A pharmaceutical representative has just completed a presentation on the merits of a new radiopharmaceutical for the study of blood profusion of heart muscles in patients with suspected or known coronary artery disease. The new radiopharmaceutical is labeled with Tc-99m (trade name Tetrolite). The radiopharmaceutical that has been the standard in heart profusion imaging for years is Tl-201Cl (trade name Thallium). One of the selling points that the salesperson uses is the favorable dosimetry of Tetrolite relative to Tl-201Cl.

After the presentation, the nuclear medicine manager asks you to explain why one can inject nearly 10 times more Tetrolite than Thallium into the patient without increasing the patient's effective dose equivalent.

Following PMHGB procedures, specific protocols are defined for the conduct of heart muscle blood profusion studies. For Thallium, the patient is physically stressed and 148 MBq of Tl-201Cl is injected at peak stress. An image is made immediately following the end of stress. About 2 hours later, an image is obtained which is representative of the rest condition. No additional Tl-201Cl is injected.

Energies, Yields, Absorbed Fractions, and Absorption Coefficients in Air of Major Photons Emitted from the Decay of Tc-99m and Tl-201[a].

Radiation Type	Energy (keV)	Yield	Absorbed Fraction[b]	Absorption Coefficient in Air (cm^2/g)
Tl-201				
X-ray $K_{\alpha 1}$	68.9	0.27	0.47	0.0262
X-ray $K_{\alpha 2}$	70.8	0.465	0.47	0.0262
X-ray $K_{\alpha\beta}$	80.3	0.205	0.47	0.0236
Tc-99m				
Gamma	140.5	0.891	0.36	0.0245

[a] For this scenario, ignore contributions of Auger and internal conversion electrons to patient doses.

[b] The fraction of the energy of a photon originating in the whole body (source organ) which is absorbed in the target organ (whole body). This fraction is derived from Snyder et al. MIRD Pamphlet No. 5 (1969).

For Tc-99m, a patient in a rest condition is injected with 370 MBq of Tetrolite. One hour later a rest image is obtained. About 3 hours later, the patient is stressed physically. At peak stress, an additional 740 MBq is injected and an image representative of the patient's physical stress is obtained about 40 minutes later. The injection/handling time for each Tetrolite administration is the same.

From the literature, you discover that both radiopharmaceuticals distribute themselves uniformly throughout the body, and that the half-lives for biological elimination from the body are 10 days. The physical half-lives for Tc-99m and Tl-201 are 6 hours and 73 h, respectively. Reference man weighs 70 kg.

Assume that the distribution and kinetics of elimination of the two radiopharmaceuticals is about the same during both the rest and stress phase of the diagnostic protocol.

2.36 What are the three major factors, which one would use to define the committed effective dose equivalent received by the patient who has been injected with either of these radiopharmaceuticals?

2.37 What are the relative contributions of each factor in determining the nearly ten-fold difference in effective dose equivalent per MBq between the Thallium and the Tetrolite radiopharmaceuticals?

2.38 If the biological half-life of Tetrolite in the gall bladder is 7 days, what is the effective half-life for Tc-99m in the gall bladder?

2.39 The gamma camera used for production of a nuclear medicine image functions in much the same manner as a solid scintillation survey meter. Assume that the injection activities given for the previously described protocols result in 20 times more counts being acquired with Tc-99m-Tetrolite than with the Tl-201Cl. What is the relative variability in the counts, which are acquired using the two radiopharmaceuticals? Variability is defined by the standard deviation of the number of counts divided by the number of counts.

2.40 What is the relative extremity dose to a nuclear medicine technician who is performing injections of Thallium and the Tetrolite for heart perfusion studies? The protocol is the same as that identified previously. The injection process requires about 1 minute of syringe handling time for the Thallium procedure and about 2 minutes of syringe handling time for the Tetrolite procedure. Assume each Tetrolite administration requires 1 minute.

Scenario 2.9

You are a medical health physicist at the University of Lower Ohio Medical Center. A shielding evaluation is needed for two barriers of a dedicated chest unit that is used to image 300 patients per week. There is a wall behind the chest image receptor that is the primary barrier and an adjacent perpendicular wall that is a secondary barrier. Assume that the X-ray beam is always directed horizontally toward a wall-mounted image receptor of area 1,535 cm^2 at 1.83 m source image distance (SID).

The room behind the image receptor wall is a fully occupied, uncontrolled office. Assume the distance from the source to the maximally exposed individual beyond the primary barrier is 3 m. The wall on which the image receptor is mounted serves as the primary barrier to the X-ray beam with a use factor of 1.0. This area is occupied 100% of the time.

In addition, consider a wall adjacent to the wall on which the chest image receptor in the chest room is mounted. This wall is never struck by the primary beam and is to be considered a secondary barrier. Assume a fully occupied, uncontrolled area located at a distance of 2.1 m from both the patient and X-ray tube and that the walls are at an angle of 90°. Applicable data is provided in the following four tables.

2.41 What is the NCRP-147 shielding design goal for controlled areas?
- a. 0.01 mGy/wk
- b. 0.02 mGy/wk
- c. 0.05 mGy/wk
- d. 0.1 mGy/wk
- e. 0.2 mGy/wk

2.42 What is the NCRP-147 shielding design goal for uncontrolled areas?
- a. 0.01 mGy/wk
- b. 0.02 mGy/wk
- c. 0.05 mGy/wk
- d. 0.1 mGy/wk
- e. 0.2 mGy/wk

Fitting Parameters for Transmission of Broad Primary X-ray Beams for a Workload Distribution Representative of a Chest Room.

Shield Material	**α (mm^{-1})**	**β (mm^{-1})**	γ
Lead	2.283	10.74	0.637
Concrete	0.03622	0.07766	0.5404
Gypsum Wallboard	0.01286	0.03505	0.9356

Derived from NCRP-147.

Secondary Barrier Parameters Applicable to a Chest Room.

W_{norm} (mA-min/patient)	0.22
F (cm^2) at d_F (m)	1,535 at 2 m
Unshielded Air Kerma (mGy-m^2/patient) at 1 m	
• Leakage	3.8×10^{-4}
• Side Scatter	2.3×10^{-3}
• Leakage and Side Scatter (K^1_{sec})	2.7×10^{-3}
• Forward/Backscatter	3.2×10^{-3}
• Leakage and Forward/Backscatter (K^1_{sec})	3.6×10^{-3}

Derived from NCRP-147.

2.43 What is the weekly, unshielded primary air kerma?
2.44 What is the transmission required for the primary barrier to meet the shielding design goal for uncontrolled areas?
2.45 What thickness of lead in the primary barrier is required to achieve the transmission value of Question 2.44?
2.46 What is the weekly, unshielded secondary air kerma?
2.47 What is the transmission required for the secondary barrier to meet the shielding design goal for uncontrolled areas?
2.48 A qualified expert is an individual certified by all of the following organizations, except:
   a. American Board of Radiology
   b. American Board of Health Physics
   c. American Board of Medical Physics
   d. American College of Physicists in Medicine
   e. Canadian College of Physicists in Medicine

Unshielded Primary Air Kerma per Patient [(K_p^1 in mGy-m^2/patient)] for the Indicated Workload [W_{norm} (mA-min/patient)] and Workload Distribution, Normalized to Primary Beam Distance $d_p = 1$ m.

Workload Distribution	**W_{norm} (mA-min/patient)**	**K_p^1 (mGy-m^2/patient)**
Rad Room (chest bucky)	0.6	2.3
Rad Room (floor or other barriers)	1.9	5.2
Rad Tube (R&F room)	1.5	5.9
Chest Room	0.22	1.2

Derived from NCRP-147.

Equivalent Thickness of Primary Beam Preshielding (x_{pre}).

Application	**x_{pre} (mm)**		
	Lead	**Concrete**	**Steel**
Image receptor in radiographic table or wall-mounted cassette holder (attenuation provided by grid, cassette, and image-receptor supporting structures)	0.85	72	7
Cross-table lateral (attenuation by grid and cassette only)	0.3	30	2

Derived from NCRP-147.

Scenario 2.10

You are a medical health physicist assigned to the Radiological Quality Assurance Group at the Omega Minus Memorial Hospital in Pine Bluff, PA. A patient with a thyroid condition is given I-131 therapy. The patient receives an oral dose of 3700 MBq of I-131. Assume that 30% of I-131 is taken up instantly by the thyroid and is distributed uniformly in that organ. The patient has an unusual thyroid iodine retention rate so the published I-131 dose conversion factor cannot be used. Assume the thyroid mass is 20 g. The I-131 physical half-life is 8.05 d, its biological half-life (in the thyroid) is 90.0 d, and its average beta energy is 190 keV.

2.49 Using the information in this problem, calculate the absorbed dose to the patient's thyroid over the first year after the procedure.
2.50 Regarding patient room preparation by the hospital radiation protection staff, state four radiation protection concerns.
2.51 Regarding patient room preparation by the hospital radiation protection staff, state four specific measures that could be taken in room preparation.
2.52 What radiation protection measures/controls should be implemented for the protection of the hospital staff?
2.53 What radiation protection concerns would you have regarding allowing the patient to resume activities as a radiation worker at a nuclear power facility?

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3
University Health Physics

A university health physicist is required to demonstrate many of the concepts discussed in the previous chapters. This professional may be required to perform internal dose assessments involving the wide variety of isotopes used within the particular research environment or to evaluate exposures from university medical or accelerator facilities. The university health physicist is also faced with dealing with assessments of effluent releases and for managing radioactive waste.

In a sense, the health physics concerns at a university encompass the wealth of the field. The limits of the scope of university health physics are only imposed by the imagination of the particular research faculty. Therefore, the scope is quite broad. For example, university research activities utilize a variety of radiation sources. These include large fixed gamma sources, X-ray machines, nuclear reactors, particle accelerators, neutron sources, unsealed radioisotopes used in biomedical or chemical applications, and biomedical tracer studies. Because of this breadth, we will address a variety of topics that are representative of the concerns and challenges of the university health physicist.

3.1
Research Utilizing Radionuclides

The initial part of this chapter will focus upon research activities involving specific radioisotopes, their hazards, and the engineering controls that limit their impact upon the researcher. The consequences of the failure of these controls, the resulting release of radioactive material to the work environment, the quantification of this release, and its impact upon the worker will also be addressed.

Research involving radionuclides spans a variety of areas, including the biomedical sciences, engineering, physics, chemistry, biology, and geology. A summary of common university radioisotopes is presented in Table 3.1. A summary of the characteristics of these research isotopes and their health physics considerations are outlined below. In all cases, proper contamination control practices must be utilized.

Table 3.1 Common University Research Radioisotopes.

Isotope	Use
H-3	Biomedical tracer studies
C-14	Biomedical tracer studies
P-32	Biochemical labeling experiments
Co-60	Large sealed sources used in radiation damage studies
I-125/I-131	Medical research/therapeutic treatments
Cf-252	Neutron studies

3.1.1 H-3

Tritium enters the body by inhalation, ingestion, and absorption through the skin and equilibrates with the body's water. H-3, entering the body as tritiated water, will equilibrate with total body water after inhalation and transfer from the lung to the blood. Skin absorption follows transfer to the lymph system and then to the blood. Equilibration may take 2–4 hr.

Tritium uptakes are best detected by urinalysis. Urine samples should be counted for the 0.0186-MeV maximum beta energy using liquid scintillation counting.

3.1.2 C-14

Breath analysis is the most sensitive technique for analyzing intakes of C-14 via the excretion of CO_2 by exhalation. Urinalysis with a liquid scintillation counter may be more practical and is nearly as sensitive as breath analysis.

Engineering controls mitigate the uptake of C-14. The use of local ventilation and fume hoods are reasonable controls for handling C-14. The chemical form of the C-14 material also impacts its uptake.

3.1.3 P-32

P-32 is often administered as a biochemical label in an aqueous solution. The isotope is a pure beta emitter with a maximum energy of 1.71 MeV and a half-life of 14.3 days. The health physics considerations are potential contamination problems and direct beta dose to the researcher. Exposures may be minimized with proper shielding. The use of P-32 is normally performed in a ventilation hood which minimizes the airborne hazard and also protects the worker's eyes from the beta particles emitted by P-32. P-32 will not volatilize under normal circumstances, but a spill or misapplication could lead to P-32 vapor.

Good housekeeping practices and laboratory procedures minimize the potential for contaminating other equipment. Finger rings or wrist badges should be utilized to assess the beta dose to the hands. Eye protection may be necessary depending upon the geometry and the research application. Gloves, shielded vests, shadow shielding, and short tongs could be employed to further minimize the worker's radiation exposure.

Uptakes of P-32 could be detected by either urinalysis or whole-body counting. Whole-body counting would detect the bremsstrahlung radiation produced by the 1.71-MeV beta particles.

3.1.4
Co-60

Radiation damage studies or food irradiation experiments utilize kilocurie sources of gamma isotopes such as Co-60. The irradiation chamber's door and shutter should be interlocked to prevent entry when the source is exposed.

The health physicist's major concern is direct exposure and appropriate training and surveillance of interlocks and warning systems. The source shield and the shutter should be labeled with radiation warning signs. The door should also have appropriate signs, warning lights, and audible alarms that provide an indication that the source is exposed. A whole-body thermoluminescent dosimeter (TLD) or film badge would be the recommended dosimetry.

3.1.5
I-125/I-131

The retention of iodine by the thyroid makes thyroid counting the most sensitive and practical form of bioassay for iodine radioisotopes. Both I-125 and I-131 are airborne hazards and should be utilized under controlled conditions such as in a fume hood or in a well-ventilated area.

I-125 has a 60-day half-life, and it is detected through the 35-keV X-ray emitted as a result of electron capture. Its emissions are weaker than those of I-131 and include X-rays and conversion electrons.

Iodine-131 poses both an external and internal radiation hazard due to the beta and gamma radiation associated with its decay governed by its 8-day half-life. Research studying thyroid dysfunction must consider the impact of radiation because 6.6-rem/μCi uptake occurs in the thyroid. The gamma constant for I-131 is 2.2 R-cm^2/hr-mCi so that 100 mCi of unshielded I-131 leads to an exposure rate of about 22 mR/hr at 1 meter and 220 R/hr at 1 cm.

The administration of radioiodine is normally in a basic (pH > 7) solution to minimize the quantity of iodine that volatilizes. Shielded vials, protective gloves, and protective clothing should be utilized to minimize the technician's exposure. Oral administration should be followed by other fluids to wash the mouth and esophagus by removing residual radioiodine and thus lower the exposure to these tissues. These additional fluids will also serve to reduce the stomach and

gastrointestinal (GI) tract exposures. Intravenous administration would eliminate these exposure concerns.

3.1.6 Cf-252

The radiation hazard from Cf-252 is primarily due to neutrons emitted following spontaneous fission. For Cf-252, the associated gamma dose equivalent is only about 5% of that due to the neutrons. The Cf-252 dose equivalent rate at 1 meter is about 2.4 rem/hr-μg which suggests that personnel monitoring for both neutrons and gammas should be provided.

3.2 Engineering Considerations

A university health physicist will be required to minimize the radiological impact of experiments involving radionuclides that have the potential to become an airborne hazard. These experiments often involve fume hoods or glove boxes that protect the researcher from the airborne hazard.

The university health physicist will become involved in the development of engineering controls to minimize releases of radioactive material into laboratory workspaces. These control measures are not always completely effective, due to researcher errors or equipment malfunctions, and releases may occur. The release has the potential to lead to an uptake of radioactive material, which also requires the health physicist to determine the radiological consequences of the release.

3.2.1 Engineering Controls

The uptakes noted in the previous section may be mitigated through the use of engineering controls. University research environments frequently utilize ventilation systems and other engineering controls (hoods and glove boxes) to minimize the uptake of volatile radionuclides. The design of containment systems for volatile research material should consider the following features:

1. Supply air should enter near the ceiling and exhaust near the floor. This will sweep contamination that leaks from the hood or glove box toward the floor and away from the respirable air.
2. Components should be readily accessible for testing and component change-out.
3. Individual laboratory rooms should have a dedicated air supply and exhaust system.
4. Backup power should be provided to ventilation systems and radiation air monitors.

5. Components should be constructed of stainless steel or other nonreactive materials.
6. Sharp bends, nonlaminar transitions, and long pipe runs should be eliminated for efficient system operation.

The system design should be periodically tested to ensure that the intake, exhaust, and recirculating air flows meet the system specifications and that the system is properly balanced. Room air-flow patterns can be investigated using $TiCl_4$ smoke sticks. The $TiCl_4$ combines with the humidity in the air to produce a dense white smoke (TiO_2). The TiO_2 is carried by the air flow to produce a visible indication of air flow. Air flows around intake and exhaust ducts and in the vicinity of hoods/glove boxes are readily evaluated using the $TiCl_4$ method.

System filters should be tested with DOP (dioctylphthalate) to verify particulate collection efficiency. The standard test using 0.3-μm particles should lead to a collection efficiency of at least 99.95%. DOP testing can also be used to perform leakage tests of various system components by introducing the DOP particles into the duct, glove box, or hood during system operation. The DOP sampling probe can be used to measure DOP concentrations at areas of concern.

Should the engineering controls fail or degrade, it will be necessary to assess the quantity of radioactive material released as well as the radiological impact upon affected workers. The next section addresses the determination of activity from air sample analysis.

3.3 Sample Counting

The concept of counting samples for gamma radionuclides is common to many fields of health physics, including the university environment. Sample counting involves a book-keeping exercise in which an unknown spectrum is compared to radionuclide emissions having known energy emissions. The comparison of known radionuclide to unknown emission is made as a function of energy.

The concept of counting efficiency is essential to a quantitative assessment of the sample's activity. The counting efficiency is given by

$$e(E) = N(E)/S \times t \tag{3.1}$$

where

$e(E)$ = counting efficiency as a function of gamma ray energy
E = gamma ray energy
$N(E)$ = total detector counts at energy E
t = counting time (s)
S = strength of the calibration source (gammas/s) at energy E

The counting laboratory will have the necessary data to determine the activity of the unknown sample. This information will include a curve of counting ef-

ficiency versus gamma ray energy, a listing of peak energies and net detector counts, and a list of possible isotopes and their gamma energies and abundances. With these data, we can calculate the activity of the isotopes in the sample from the relationship

$$A_i = C(E)k/(t)\,[e(E)]\,[Y(E)] \tag{3.2}$$

where

A_i = activity of isotope i
k = conversion factor (μCi/2.2 × 10^6 dpm or Bq/60 dpm)
t = counting time
$C(E)$ = counts at energy E
$Y(E)$ = gamma yield for the gamma ray of energy E

In estimating the activity, unique peaks are used to obtain the activity of the isotopes corresponding to these peaks. This information is then used to obtain the contributions of isotopes having overlapping peaks. An example of this procedure will be illustrated in one of this chapter's scenarios. Beta and alpha counting follow a similar methodology.

A positive air sample indicates that a release of material has occurred. Once radioactive material is no longer contained in its engineered structure, it can be inhaled, ingested, or absorbed through the skin. In the next section, we present a technique for a quick assessment of the radiological consequence of the released material. More detailed assessments using ICRP-30, -60, or -103 methodology will be required, but the quick assessment will serve as a guide to the immediate recovery steps.

3.4 Intake of Radionuclides

If a radioactive release from a glove box, fume hood, or other enclosure occurs, the health physicist has a number of tasks to accomplish. These include investigating the cause of the release as well as assessing its radiological consequences. In assessing the radiological hazards, the quantity of material taken into the body must be determined.

The intake can occur following accident or routine conditions and may be represented by a relationship that assumes that all potential airborne radionuclides have volatilized or evaporated and are uniformly distributed in the room or laboratory. With these assumptions, the intake is given by

$$I_i = (\mathrm{BR})\,t\,A_i/V \tag{3.3}$$

where

I_i = intake of radionuclide i (μCi)
A_i = activity of isotope released into the room (μCi)

BR = breathing rate (liters/min)
t = time the worker resides in the room after the activity is uniformly dispersed (min)
V = volume of the laboratory (liters)

Once a deposition of radionuclides occurs, an assessment of dose is often required. Assuming that the organ of interest may be treated sufficiently within a single compartment approximation, the initial dose equivalent rate is given by

$$\dot{D}_0 = 2.13\, PE(\mathrm{QF})/m \tag{3.4}$$

where
$\dot{D}_0$ = initial dose equivalent rate (rem/hr)
P = initial activity in the organ (μCi)
E = energy deposited in the organ (MeV)
m = organ mass (g)
QF = quality factor for the emitted radiation

3.5 Other Research Activities

In addition to the research activities utilizing the isotopes of Table 3.1, additional university research activities use radioisotopes or produce ionizing radiation. A sampling of these activities include agricultural and environmental research, research reactor operations, nuclear and particle physics accelerator activity, materials research via X-ray diffraction techniques, and fusion energy research. A brief summary of the health physics considerations from these activities will be provided in the remainder of this chapter.

3.6 Agricultural/Environmental Research

Agricultural and environmental research activities encompass the areas of soil fertility, irrigation, crop production, insect and pest control, plant breeding and genetics, and agrochemical usage. This work includes research on the application of isotope techniques to minimize the use of nitrogen fertilizers and studies of the application of radiation-based techniques to control the population of insect pests.

These research studies include techniques for monitoring and assessing environmental pollution, including analysis of radioactive and nonradioactive pollutants, monitoring of pesticide residues in food and in the environment, and climatic studies through the isotopic content of precipitation. Research studies also include techniques for reducing the environmental impact of chemical-induced productivity increases, radiosterilization of insects, studies of nitrogen fixation in

crops and trees, use of tracers to optimize fertilizer use, and nuclear techniques to improve water management practices.

The health physicist must control the isotopes used in the studies. With tracer isotopes, the control of internal exposures and contamination control will be a primary concern. Sterilization of insects may involve use of very-high-activity gamma ray sources, and direct dose will be the major concern. Material analysis using accelerator-activation techniques or proton-induced X-ray emission requires the application of sound accelerator health physics practices.

3.7 Research Reactors

Research reactors have many problems in common with power reactors, but at a smaller scale because of the limited reactor power. Releases of radioactive material and maintenance problems are not as severe, but the neutron and gamma fluences merit an appropriate level of concern.

University reactors are used in a variety of activities, including the production of irradiated materials and special isotopes. The samples pass through high-flux regions and can become highly activated. Production of these materials challenges the health physicist. He or she must balance sound health physics and dose control efforts with the needs of the researcher.

Material radiation damage studies involve the bombardment of materials with large neutron or gamma ray fluences. This research may involve studies of basic material properties and may be related to power reactor nuclear safety. A current nuclear safety topic involves the embrittlement of reactor vessels after years of being exposed to large radiation fluences from the nuclear fuel core. These studies require that the health physicist control high activity sources, and remote handling techniques are often employed.

3.7.1 Operational Characteristics

University reactors do not have the economic constraints of power reactors and operate to accommodate research and isotope production schedules. Since isotope production is an important function of these reactors, neutron activation is a significant health physics consideration in the design of experiments. Neutron fluences on the order of 4.0×10^{13} n/cm^2-s are available for in-core experiments.

Health physics concerns at a research reactor include the loss of water shielding, gaseous releases, control of beam tubes, control of the rabbit system, the insertion and removal of experimental material in the reactor, and pulsing the reactor. In a research reactor, water provides most of the shielding above the reactor core. A loss of this shielding causes an increase in the radiation levels near the research reactor pool. Area radiation monitors and water level instrumentation provide advance warning of a decreasing pool level.

If the pool level decreases, then the research reactor's core could be uncovered. Cladding failures occur with an increase in the fuel's temperature. A failure of the cladding creates a pathway for the release of fission products from the fuel assembly, into the pool water, and to the surrounding atmosphere. Continuous air monitors and radiation monitors signal increases in gaseous or airborne particulate activity.

Beam tubes are pipes or ports that permit neutron and gamma radiation to exit from the reactor core and impinge on a target. Pipe diameters vary, but are usually within the range of 10 to 31 cm. In order to control radiation emission, beam tubes are fitted with remotely operated shields. Shielding is required because radiation levels exiting the beam tube typically reach levels of at least 70 Gy/hr within the irradiation room. Neutron fluence levels are on the order of 5.0×10^{10} n/cm^2-s.

With the potential for significant radiation levels, irradiation rooms are controlled to eliminate inadvertent access. Rooms are locked or made inaccessible during reactor operation. Even following reactor shutdown, significant radiation exists within the core and, if the beam tube shield plugs are not properly seated, elevated radiation levels exist in irradiation rooms.

In addition to beam tube irradiations, research reactor experiments are also performed using the radiation field directly above the reactor core. The above core experiments cause radiation levels to rise above normal levels. If the levels are significant, the experiment will remain submerged until radiation levels decay to acceptable values. Sound radiological controls practices, require that streaming paths from the experiment be eliminated. As an example, pipes or conduit should have free flooding holes to ensure they are filled with water to avoid a streaming pathway. Although this approach minimizes radiation streaming, it does enhance the potential for contamination buildup within the piping or conduit.

Material to be irradiated traverses the research reactor core via a transfer or rabbit system. Rabbit systems permit quick access and removal of material from the core. Short-lived isotope studies are well suited to rabbit systems. Samples prepared for irradiation are usually in either a powdered or liquid form and sealed in glass or plastic tubing.

Health physics controls are required in dealing with irradiated samples. Emergent contact dose rates will be as large as 2 Gy/hr upon exit from the core. The use of tongs, long-handled tools, and shadow shielding are appropriate, to minimize extremity and whole-body doses. The rabbit system as well as sample tubes and containers are also radiation sources since they are activated by the core's neutron flux. Other prudent controls include sample manipulation in a ventilated hood to control the emission of vapors, gases, or particulate material.

The design of research reactors permits the generation of high neutron fluxes by rapidly removing and inserting a control rod from the core. This action is known as pulsing and it generates high instantaneous flux levels on the order of 3.0×10^{16} n/cm^2-s versus the typical 4.0×10^{13} n/cm^2-s level. The increase in flux is accompanied by a corresponding increase in radiation levels. However, the pulse has a short duration on the order of 15 ms.

3.7.2 Reactor Systems and Associated Radionuclide Production

The fuel of a research reactor is an important barrier that minimizes the release of fission products into the moderator. Although fuel designs vary, the fissile material is contained within a cladding material. Both the fissile fuel matrix and the clad are barriers to the release of fission products. Fission products include particulates, radioiodine, and Noble gases.

The type of cladding material also impacts radionuclide production. For example, in aluminum clad fuel, Na-24 is produced from the reaction Al-27(n, α) Na-24.

Most research reactors are light water cooled and moderated. Tritium production occurs at an enhanced level in heavy water (D_2O) systems, and the production of tritium in light water systems depends on the degree of operation (MW-days) of the system.

Most research reactors with open-air pools produce air activation products, including Ar-41, from dissolved gases. N-16 is produced from activation of oxygen via the O-16(n, p)N-16 reaction.

Reflectors surround the reactor core and decrease the loss or leakage of neutrons from the core region. In an undermoderated research reactor, reflectors are used to enhance the neutron fluence available for experimental purposes. Typical reflector materials include beryllium, graphite, and D_2O that enhance the neutron fluences at the experimental location.

Graphite reflectors have been observed to contain a number of activation products including Cl-38, Sc-46, Br-82, and Eu-152. Beryllium reflectors tend to expand and crack as their utilizations exceed 26 000 MW-days. These effects were noted in several research reactors.

3.7.3 Reactor Effluents

In contrast to power reactors that are generally sited in low population or rural areas, non-power reactors are found at universities or research facilities located in urban or suburban areas. The proximity to population centers makes research reactor effluents a topic of public interest. Despite the fact that the annual public dose from research reactor effluents is less than a few μSv, there are continuing efforts to minimize effluents because of public perceptions.

3.7.4 Gaseous Effluents

Ar-41 is the principal airborne effluent of research reactors. Other radionuclides that are detectable in the stack effluent include tritium, fission product gases, and Br-82. Smaller quantities of other radionuclides are present, but these are usually insignificant compared with Ar-41.

A number of reactors are installing loops to produce radioiodine for medical and research applications. These loops use a variety of reactions including using capture reactions in iodine and xenon to produce these isotopes. Releases from iodine loops present an effluent release potential.

3.7.5 Liquid Effluents

Research reactor releases to the sanitary sewer system are typically small fractions of the concentration and total annual activity limits. The total activity, exclusive of tritium, is typically less than 400 MBq even for reactors with thermal power ratings of up to 20 MW. Tritium releases for higher power heavy water cooled reactors are usually less than $4-11 \times 10^4$ MBq. The liquid releases are less significant than the airborne releases.

3.8 Particle Accelerators

University accelerators include a variety of research tools. These include accelerators utilized in low-energy nuclear physics research involving terminal voltages of up to 20 MV that include ions of most nuclear cores. Intermediate-energy machines, with energies up to a few thousand MeV, produce pion and muon beams with an added complexity associated with their dosimetry. The high-energy machines, with maximum energies on the order of several thousand GeV, create a variety of new particles and associated radiations. Specific problems will be addressed in the scenarios and in Chapter 7.

Accelerators are also utilized in medical research applications. These activities are addressed in Chapter 2.

3.9 Materials Research Via X-ray Diffraction Techniques

The output from a research X-ray diffraction source is normally a thin, high-exposure-rate beam of energy on the order of 25 keV or less. The exposure rate in the primary beam may be as large as 10^5 R/hr. Therefore, very high local exposure rates are possible, and appropriate health physics controls are warranted.

The diffracted beam is also of concern, with exposure rates on the order of 100 R/hr possible. Beam leakage through improperly aligned shielding, exposure from fluorescent radiation from the material under evaluation, and exposure to X-rays from high-voltage components and power supplies are additional exposure concerns.

X-ray diffraction devices should be periodically monitored to ensure that the maximum operating voltage and other operating parameters remain unchanged.

The shielding configuration, beam shutters, beam stops, scattered radiation, diffracted beam exposure rate, and fluorescence exposure rates should be verified to fall within the expected ranges. Changes in the experimental configuration or materials should be surveyed before research activity begins.

Personnel dosimetry includes both whole-body and finger rings or wristband dosimetry. Proper training to emphasize the radiation hazard is essential.

From a health physics perspective, regular monitoring of this equipment is required. The machine should have a highly visible warning indicator that signals when the beam is on and when the shutter is open. The device should be enclosed to prevent inadvertent access and be designed with beam interlocks to shut the unit off if the enclosure is breached. The strategic use of signs and warning labels should be utilized to remind the researcher of the inherent hazards of this device.

3.10 Fusion Energy Research

Fusion energy research, using inertial confinement or laser techniques, is concerned with the confinement and fusion of hot plasmas of light nuclei. The fusion process produces a variety of radiation types which must be controlled by the health physicist.

Potential fusion reactions under active investigation include the following:

$$\begin{aligned}
\text{D} + \text{D} &\xrightarrow{50\%} \text{T}(1.01\ \text{MeV}) + p(3.02\ \text{MeV}) \\
&\xrightarrow{50\%} \text{He-3}(0.82\ \text{MeV}) + n(2.45\ \text{MeV}) \\
\text{D} + \text{T} &\longrightarrow \text{He-4}(3.50\ \text{MeV}) + n(14.1\ \text{MeV}) \\
\text{D} + \text{He-3} &\longrightarrow \text{He-4}(3.60\ \text{MeV}) + p(14.7\ \text{MeV}) \\
\text{T} + \text{T} &\longrightarrow \text{He-4} + 2\,n + 11.3\ \text{MeV} \\
\text{He-3} + \text{T} &\xrightarrow{51\%} \text{He-4} + p + n + 12.1\ \text{MeV} \\
&\xrightarrow{43\%} \text{He-4}(4.8\ \text{MeV}) + D(9.5\ \text{MeV}) \\
&\xrightarrow{6\%} \text{He-5}(2.4\ \text{MeV}) + p(11.9\ \text{MeV}) \\
p + \text{Li-6} &\longrightarrow \text{He-4}(1.7\ \text{MeV}) + \text{He-3}(2.3\ \text{MeV}) \\
p + \text{Li-7} &\xrightarrow{20\%} \text{He-4} + \text{He-4} + 17.3\ \text{MeV} \\
&\xrightarrow{80\%} \text{Be-7} + n - 1.6\ \text{MeV} \\
\text{D} + \text{Li-6} &\longrightarrow \text{He-4} + \text{He-4} + 22.4\ \text{MeV} \\
p + \text{Be-11} &\longrightarrow \text{He-4} + \text{He-4} + \text{He-4} + 8.7\ \text{MeV} \\
n + \text{Li-6} &\longrightarrow \text{He-4}(2.1\ \text{MeV}) + \text{T}(2.7\ \text{MeV})
\end{aligned}$$

where D is deuterium and T is tritium. For binary events the particle energy is provided in parentheses. A negative yield or *Q*-value indicates that the reaction is endothermic.

The wide variety of potential reaction products and energies suggests considerable challenge for the health physicist responsible for worker radiation protection at a fusion energy research facility. For example, fusion neutrons will present an external radiation hazard. These neutrons will require shielding, and particular attention must be paid to leakage pathways which will vary with the type of reactor design, the fusion reaction under consideration, and the reactor operating characteristics.

Activation products will be produced by the high neutron fluence impinging on the structural components of the reactor, including the reactor or plasma containment vessel. The specific activation products will depend upon the structural material utilized in the design. Anticipated structural materials include stainless steel, niobium, and ceramic materials such as Al_2O_3. Activation products will include isotopes of Al, Na, Fe, Co, Ni, Mn, Nb, and Zr which decay by beta, positron, and electron capture with associated gamma emission.

As with fission reactors, activation products present a radiation hazard during maintenance activities. Major components require periodic replacement due to the high-energy neutron bombardment. These structural waste materials will require remote handling and processing to minimize worker exposures.

If nuclear fusion becomes a commercial electrical power source, the initial plants will likely use the D + T reaction. The D–T reaction will require that commercial fusion plants retain large tritium fuel inventories.

These tritium inventories will present an internal uptake challenge. Tritium in either molecular form or as tritiated water will diffuse through the structural materials at high operating temperatures. Tritium leakage from the reactor vessel's coolant, through seals, valves, and piping, will require health physics controls. Some tritium will also diffuse into the steam system and be released to the environment. A portion of the tritium will reside in routine work areas where it presents a skin absorption and inhalation hazard. The tritium will appear as surface contamination which can be resuspended into the air or directly contaminate personnel.

Maintenance of activated structural components presents an external as well as internal radiation hazard. The health physics concerns will be similar to those encountered in a commercial fission reactor. In particular, maintenance activities generate particles of a respirable size as a result of cutting, grinding, and repair activities. Additional discussion of health physics concerns associated with maintenance activities is provided in Chapter 5.

Additional radiation hazards are presented by the unique scenarios of a fusion plant accident. Initial designs propose to use liquid metal coolants and heat exchange systems. In a severe accident the liquid metal coolants contacting air, water, or steam may lead to an explosive reaction that produces hydrogen gas. Such an event could lead to a loss of structural integrity with the subsequent transport and deposition of activation products and tritium to offsite locations. These releases

will differ significantly from those of a fission reactor which involve primarily noble gas and radioiodine concerns. The final safety analysis report for the commercial fusion plant will address these and other fusion accident scenarios.

3.11 Overview of an Initial Fusion Power Facility

In view of the current direction in fusion research and the selection of magnetic confinement as the design concept for the International Thermonuclear Experimental Reactor (ITER), subsequent discussion is based on the ITER tokamak design concept that utilizes D–T fusion.

The heart of the magnetic confinement system in a tokamak is the torus, which is a large toroidal vacuum vessel surrounded by devices to produce the confining magnetic field. In addition to its confinement function, the magnetic field is also designed to prevent plasma from striking the inner wall of the vacuum vessel.

A blanket shield supports the first wall by providing neutron shielding for the vacuum vessel. This shielding is a combination of stainless steel and water. The blanket also provides the capability for testing tritium breeding blanket modules and for tritium production blankets.

Production scale facilities build upon the ITER experience, are expected to be physically larger, and have a higher power output. The ITER is a formidable structure with the main plasma parameters and dimensions provided in Table 3.2.

From a safety perspective, the ITER design focuses on confinement with successive barriers provided for the control of tritium and activated material. These barriers include the vacuum vessel, the cryostat, air conditioning systems with detritiation capability, and filtering capability of the containment building. Effluents are filtered and detritiated such that radioactive releases to the environment are minimized.

Table 3.2 ITER Plasma Parameters and Dimensions.

Total Fusion Power	500–700 MW
Plasma Major Radius	6.2 m
Plasma Minor Radius	2.0 m
Plasma Current	15 MA
Toroidal Field @ 6.2 m	5.3 T
Plasma Volume	837 m^3
Plasma Surface Area	678 m^2

Derived from the ITER Final Design Report (2001).

Worker radiation safety and environmental protection are enhanced by the structure housing the vacuum vessel. For worker protection, a biological shield of borated concrete surrounds the cryostat and concrete walls provide additional neutron and gamma shielding.

3.11.1 General Radiological Characteristics

The ITER radiological hazards are representative of those occurring in a production fusion facility. These hazards include tritium, neutron radiation, activation products, and particulates generated by plasma collisions with containment structures.

Tritium in gaseous form (T_2) and as oxides (HTO, DTO, and T_2O) will be present at ITER. The particular chemical form depends on the location within the tritium processing system and the physical conditions encountered during a tritium release scenario.

Neutron radiation is produced in the DT fusion process. The 14.1 MeV neutrons pose a direct radiation hazard, have a significant potential for activation of fusion reactor components, and lead to radiation damage of reactor components. The radiation damage increases maintenance requirements and radioactive waste generation, and increases occupational radiation doses.

Activation products are the largest contributor to the radiological source term. At ITER, the most significant activation products of stainless steel are isotopes of Mn, Fe, Co, Ni, and Mo and the most significant activation products of copper are Cu, Co, and Zn. During ITER's Extended Performance Phase, a reactor inventory of approximately 10^{14} MBq is anticipated. Smaller activation product inventories reside in structures outside the shield blanket or circulating as suspended corrosion products in the first wall, blanket, and divertor coolant streams.

These activation products and their activities present high radiation fields inside the cryostat and vacuum vessel. The radiation fields are sufficiently high to require remote maintenance for systems, structures, and components within the cryostat and vacuum vessel.

Fine particles are produced following ion impacts with plasma facing components. These particles form a fine radioactive dust that could be released during maintenance inside the plasma chamber or during a severe accident.

Tritium, activation products, and toxic materials could be released during an accident or off-normal event. There are power/energy sources associated with fusion reactors that facilitate the dispersal of radioactive and toxic material. For ITER, these sources are fusion power (1.5 GW), plasma energy (2.3 GJ), magnetic energy (120 GJ), decay heat 260 GJ (in the first day), chemical energy following a reaction (800 GJ), and coolant systems energy (300 GJ).

3.11.2
ALARA-Confinement Methods and Fusion Process Types

Confinement methods are intimately related to the specific fusion process and its associated ALARA characteristics. ALARA aspects of the fusion confinement method are presented in Table 3.3 which summarizes ALARA considerations for the selection of fuel type, reaction geometry, and plasma density for inertial confinement (IC) and magnetic confinement (MC) fusion devices.

The impact of the inherent physics of the D–D and D–T fusion processes on selected facility design considerations is summarized in Table 3.4. Specifically, Table 3.4 considers vacuum vessel maintenance and replacement, production of ^{16}N, ^{3}H intakes, and activation product generation.

Excluding all factors except radiation protection would suggest that the ideal fusion facility would not be based on D–T magnetic confinement. Tables 3.3 and 3.4 suggest that ALARA considerations alone would favor a D–D inertial confinement device.

Table 3.3 ALARA Comparison of Fusion Confinement Methods.

Consideration	Comment	ALARA Preference
Fuel Type	MC fusion uses T_2 and D_2 gas and HTO production is more likely than in IC fusion. IC fusion uses a solid D–T pellet.	IC fusion – The solid fuel pellet minimizes the internal intake of tritium.
Reaction Geometry	MC fusion occurs within a toroidal geometry. IC fusion occurs in the small D–T pellet (point source). For equivalent fusion powers and distances from the source, the point source geometry has a higher effective dose rate value. However, the effective dose rates are within about 1 % of each other when the distance from the MC source reaches three times the vacuum vessel diameter.	MC fusion – Near the vacuum vessel, higher effective dose rates occur with IC fusion for equivalent fusion powers. The MC fusion advantage disappears as the point of interest moves further from the reaction volume.
Plasma Density	IC fusion operates at a higher density that softens (i.e., reduces the energy of) the fusion neutron and fusion gamma spectra. The MC fusion spectrum will be harder than the IC fusion spectrum.	IC fusion – The vacuum vessel receives less damage due to the softer neutron spectrum. Reduced neutron damage minimizes the associated maintenance requirements.

Source: Bevelacqua (2008).

Table 3.4 ALARA Comparison of D–D and D–T Fusion Processes.

Consideration	Comment	ALARA Preference
Vacuum vessel maintenance and replacement	The threshold neutron energies from D–D and D–T fusion are 2.45 and 14.1 MeV, respectively.	D–D fusion – The vacuum vessel receives less neutron damage due to the lower energy D–D neutron spectrum. This reduces maintenance requirements and the need for high dose repair activities.
^{16}N activity	The D–D fusion neutron threshold energy lies below the $^{16}O(n, p)^{16}N$ activation reaction threshold. The higher energy D–T fusion neutron threshold lies above the $^{16}O(n, p)^{16}N$ activation reaction threshold.	D–D fusion – Compared to D–T fusion, the D–D fusion neutron spectrum minimizes the ^{16}N source term.
Internal Intake of ^{3}H	D–T fusion uses tritium and deuterium as the fuel source. Tritium and HTO are more hazardous than deuterium. D–D fusion uses deuterium as the fuel source. Tritium is produced inside the vacuum vessel and then burned in the D–T process.	D–D fusion – Deuterium is less hazardous than tritium.
Activation products	The threshold neutron energies from D–D and D–T fusion are 2.45 and 14.1 MeV, respectively.	D–D fusion – Activation products with higher threshold energies are minimized by the lower energy D–D fusion neutron spectrum.

Source: Bevelacqua (2008).

3.12 Scenarios

Scenario 3.1

You are given the assignment of determining the amount of activity found on a charcoal filter used in the stack monitor at a research reactor. The detector you choose to use is a Ge(Li) detector.

Detector calibration, sample counting, and isotopic information are provided as follows:

Detector Calibration: Charcoal Filter Geometry.

Gamma Energy (keV)	Calibration Source (gammas s^{-1}) Decay Corrected	Ge(Li) Detector Net Counts (1000-s count)
60	2000	83762
88	1840	96001
121	1590	79242
344	1490	31257
768	730	8730
963	820	7400
1408	1170	8215

Sample Count: Charcoal Filter Geometry.

Peak Energy (keV)	Ge(Li) Detector (net counts in a 10-min count)
121.5	7266
136.2	8034
265.0	4467
279.2	2279
401.1	629

Possible Isotopes in the Filter.

Isotope	Half-life (day)	Gamma Energy (keV)	Abundance (%)[a]
Co-57	271.8	122.1	85.9
		136.5	10.3
Se-75	119.8	121.1	17.3
		136.0	59.0
		264.7	59.2
		279.5	25.2
		400.7	11.6
Hg-203	46.7	279.2	81.5

[a] Number of gammas emitted per disintegration × 100.

3.1 Calculate the counting efficiency at each gamma photon energy for the charcoal filter geometry.
3.2 Calculate/graph the efficiency curve.
3.3 From the efficiency calculation, estimate the counting efficiency for a gamma photon having an energy of 2 MeV.
3.4 At medium energies (100–300 keV), the best explanation for the decrease in counting efficiency is:
   a. Photoelectric absorption decreases linearly when graphed on a log–log scale.
   b. Compton scattering decreases linearly when graphed on a log–log scale.
   c. Pair production does not become a significant interaction in Ge(Li) until gamma energies are about 5 MeV.
   d. Energy absorption from Compton scattering decreases with increasing energy.
   e. The K-edge for Ge(Li) occurs at 4 MeV.
3.5 The best explanation for the decrease in counting efficiency at low gamma energies (< 80 keV) is:
   a. Attenuation of the gammas by the detector dead layer and housing becomes significant.
   b. These gammas do not interact with the Ge(Li).
   c. Pair production is insignificant at these low energies.
   d. The probability of interaction decreases at energies below the K-edge of Ge(Li).
   e. The signal given off by the Ge(Li) detector is no longer proportional to the energy absorbed in the crystal.
3.6 Identify and calculate the activity (μCi) of all nuclides contained in the charcoal filter sample.

Scenario 3.2

A laboratory contains hoods to provide graduate students with safe areas to use radiochemicals. Unknown to laboratory personnel, the motor serving the exhaust fan for the hoods is turned off for repair for 1 hour. During this time, one graduate student completes a dual labeling experiment using 10 mCi of P-32 (as orthophosphate) and 10 mCi of tritium (as tritium oxide) in a hood. Another graduate student completes a procedure with 5 mCi of I-125 (as sodium iodide solution). As soon as the hood condition is recognized, the Radiation Safety Officer (RSO) is called by the graduate students.

Data

Laboratory dimensions = 20 ft × 20 ft × 10 ft

1 Cubic foot = 28.3 liters

Breathing rate = 20 liters/min

3.7 As the RSO, you want to sample the air of the laboratory. What is the airborne potential for these three isotopes?
3.8 Describe how you would perform air sampling for potential airborne radionuclides.
3.9 Should air sampling for these radionuclides be isokinetic or anisokinetic?
3.10 Assume a maximum credible case: All potentially airborne radionuclides have volatilized or evaporated and are uniformly distributed in the room. The air exchange rate is low. Calculate the maximum intake the graduate students could have received.
3.11 How much radioactivity could be in the critical organs?
3.12 As the RSO, you perform urinalyses on the two students for appropriate sampling periods. Explain what analytical techniques would be used to assess radioactivity in the urine.
3.13 If sodium borotritide was used instead of tritium oxide, how could you assess the airborne potential?
3.14 Describe measures that should be instituted to prevent recurrence of this event.
3.15 List conditions that would affect the amount of radioactivity in the students.

Scenario 3.3

You are the RSO at a large university. A technician in the Physics Department reports that a radioactive source may have ruptured. The source is labeled as 100 mCi of Am-241. Preliminary wipes taken on the source, in the technician's work area, and on the technician's clothing show extensive alpha contamination. The technician believes that the source broke when a sharp object fell on it the previous day. This morning he carried the source to the machine shop in the basement and to two offices in the building. The basement and technician's work areas have linoleum floors, whereas the hallways and offices are carpeted.

Data

Lung mass = 1000 g

Am-241 particle clearance rate: $T_{1/2}$ = 120 days (ICRP-2)

Deposition energy of the alphas and recoil atoms = 5.57 MeV per decay

3.16 Given the following methods available to you, which would be best to identify all radioactive contamination from this incident in the building?
a. Take wipes in all suspected areas, and count on a shielded gas-flow proportional counter/sample changer.
b. Use a pancake probe GM meter to survey all suspected areas.
c. Use a ZnS-coated/photomultiplier-tube-based portable alpha probe to survey all suspected areas.

d. Use a 1.5-in. × 1-in. NaI crystal probe to survey all suspected areas.
e. Use a 5-in. × 0.06-in. NaI crystal probe to survey all suspected areas.

3.17 If the source had instead been Cs-137, which means would be best to identify all contamination in the building?
a. Take wipes in all suspected areas, and count on a shielded gas-flow proportional counter/sample changer.
b. Use a portable pressurized ion chamber to survey all suspected areas.
c. Use a ZnS-coated/photomultiplier-tube-based portable alpha probe to survey all suspected areas.
d. Use a 1.5-in. × 1-in. NaI crystal probe to survey all suspected areas.
e. Use a 5-in. × 0.06-in. NaI crystal probe to survey all suspected areas.

3.18 Which would be the most sensitive means of detecting an intake of Am-241 by the technician who handled the source, assuming the measurement is made within the first 48 hr following the exposure?
a. Count nasal swabs on a shielded gas-flow proportional counter to estimate the activity inhaled.
b. Whole-body counting in a shielded facility via a 3-in. × 3-in. NaI detector/MCA.
c. Whole-body counting in a shielded facility via a coaxial germanium detector/MCA.
d. Analysis of 24-hr urine sample collected on day 2 by liquid scintillation counting.
e. Analysis of 24-hr fecal sample collected on day 2 by germanium detector/MCA.

3.19 If the source had instead been Cs-137, which would be the most reliable means of quantifying the technician's uptake, assuming the measurement is made within the first 48 hr following exposure?
a. Count nasal swabs on a shielded gas-flow proportional counter to estimate the activity inhaled.
b. Whole-body counting in a shielded facility via a 3-in. × 3-in. NaI detector/MCA.
c. Analysis of activity exhaled in the breath.
d. Analysis of 24-hr urine sample collected on day 2 by liquid scintillation counting.
e. Analysis of 24-hr fecal sample collected on day 2 by germanium detector/MCA.

3.20 Assuming that the technician had a deposition of 1 mCi Am-241 in the lung, calculate the initial daily dose equivalent rate to the lung. Assume that the lung behaves as a single compartment.

3.21 If a 1 mCi uptake of Am-241 to the lung occurred due to a chronic dose over a 100-day period versus an acute, 1-day dose, how would this affect the 50-year committed dose equivalent?
a. Dose by chronic exposure would be greater by a factor of ln(100).
b. Dose by chronic exposure would be greater by a factor of log(100).

c. Dose by both exposure periods would be virtually the same.
d. Dose by acute exposure would be greater by a factor of log(100).
e. Dose by acute exposure would be greater by a factor of ln(100).

3.22 ICRP-26 recommends the use of a weighting factor (w_T) to express the proportion of stochastic risk from an irradiated tissue (T) to the total risk when a body is uniformly irradiated. For the lung, this factor has a value of:
a. 0.03
b. 0.12
c. 0.25
d. 0.30
e. 0.50

3.23 Am-241 is used in a variety of applications because of its desirable nuclear properties. The most common mode of production of Am-241 is:
a. from naturally occurring americium in the environment.
b. from alpha decay of Cm-245.
c. from spontaneous fission of Cf-252.
d. from the beta decay of Pu-241.
e. from the Np-237(alpha, *n*) reaction.

Scenario 3.4

You have been employed as the RSO at a university for 1 month. Late on Friday afternoon (when else?) you are contacted by the Student Health Center concerning a 20-year-old female student. The SHC performed an abdominal X-ray examination on the student 5 days earlier. The student has just now reported that she was 2 weeks pregnant at the time of the examination. You must make a quick assessment of the dose to the embryo based on the data you obtained during a survey of the X-ray unit 1 week earlier.

Data from your Survey
Unit set in manual mode with 30 mA-s and 80 kVp.

Ionization chamber readings were taken at a distance of 30 cm above the table top.

Readings from the ion chamber:
0 mm of Al added between the chamber and the tube = 500 mR
2.5 mm of Al added between the chamber and the tube = 280 mR
4.5 mm of Al added between the chamber and the tube = 180 mR
kVp on the control panel of the unit: 80 kVp

Measured kVp: 80 kVp
Patient and X-ray exam information: One anterior-posterior (AP) and one posterior-anterior (PA) film were taken, with a retake needed on each film. The source image distance (SID) used was 40 inches. Technique factors = 35 mA-s and 80 kVp. The

X-ray field was collimated to a 14-in. × 17-in. size at 40 inches. The film (image receptor) was located 2 in. below the table top in the tray. The patient abdominal thickness is 20 cm.

Abdominal Embryo Data

Abdominal-Organ Dose (mrad) for 1-R Entrance Skin Exposure (Free-in-Air).

Beam Quality HVL (mm Al)	=	Dose (mrad/R)[a]					
		1.5	2.0	2.5	3.0	3.5	4.0
Ovaries	AP	97	149	203	258	313	367
	PA	60	100	146	198	255	317
	LAT	18	33	50	70	93	118
Embryo (uterus)	AP	133	199	265	330	392	451
	PA	56	90	130	174	222	273
	LAT	13	23	37	53	71	91

[a] SID 102 cm (40 inches); Film size = field size: 35.6 cm × 43.2 cm (14 inches × 17 inches).

The geometry for an X-ray situation is illustrated in Figure 3.1.

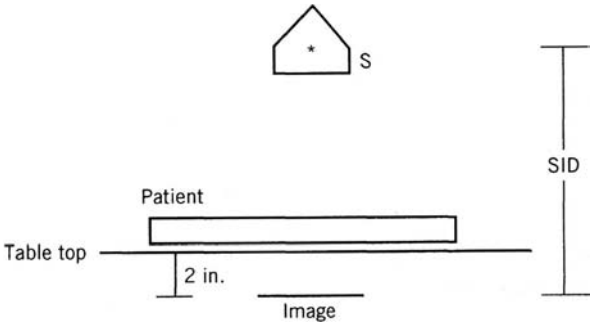


Figure 3.1 Geometry for the unanticipated fetal X-ray situation. The labels S and SID represent the X-ray source and the source image distance, respectively.

3.24 Find the absorbed dose to the embryo based on your survey data and the given tabular data. Clearly state any assumptions used in your determination.

3.25 Identify which of the following best reflects radiation exposure *in utero* and its effect on severe mental retardation of the child. (*Note:* Your numeric answer to question 3.24 is not considered in this question.)

a. The first 2 weeks following fertilization are the most critical with respect to radiation exposure causing severe mental retardation.

b. Prior to the eighth week following fertilization, there is apparently little risk of severe mental retardation.

c. Lowest risk of forebrain damage based on A-bomb data occurred between the 8th and 15th week following fertilization.
d. A-bomb data show that there is no correlation between time after fertilization and exposure of the embryo or fetus resulting in severe mental retardation of the child.
e. A-bomb data indicate that the highest risk of forebrain damage occurred after the 15th week following fertilization.

3.26 The NCRP-91 recommendations for pregnant radiation workers are to limit the dose equivalent to:
a. 100 mrem/month to a pregnant woman over the entire pregnancy.
b. 100 mrem/month to the fetus/embryo over the entire pregnancy.
c. 300 mrem/quarter to a pregnant woman.
d. 450 mrem to a pregnant woman over the entire pregnancy.
e. 500 mrem to the fetus/embryo over the entire pregnancy.

Scenario 3.5

During a bioassay counting session at your university, you discover that a researcher has an uptake of I-125. The following data should be considered:

Mass of thyroid = 30 g

Physical half-life of I-125 = 60 days

Biological half-life of I-125 in thyroid = 130 days

Mean energy of K X-rays = 28 keV (139.8%)

Mean energy of gamma rays = 35 keV (6.7%)

Equilibrium dose constants = 0.0884 and 0.0434 g-rad/μCi-hr for X- and gamma rays, respectively

Absorbed fraction for thyroid (@0.03 MeV) = 0.7

w_T (thyroid) = 0.03

3.27 Thyroid bioassay results show 2.4 μCi of I-125 currently in the thyroid of a researcher on the day of your thyroid bioassay counting. Investigation determines that this uptake occurred during a single iodination procedure 14 days earlier. Consider the uptake as a single exposure and find the thyroid dose to time = infinity based on the initial uptake which occurred 14 days earlier.

3.28 Find the effective dose equivalent to a researcher with the following annual exposures:

Whole-body film badge = 1.25 rem

Internal dose to the thyroid = 25 rem

3.29 Using the annual internal dose equivalent guidance of ICRP-26, determine if a thyroid overexposure (for the dose equivalent reported in 3.28) has occurred. Justify your answer by giving the appropriate limit.

Scenario 3.6

A member of the University Chemistry Department has received a grant from the Nuclear Regulatory Commission to investigate the movement of radioactive material in the vicinity of a low-level waste burial facility. The object of the study is to obtain data on the quantities of radionuclides, particularly the I-129 in low-level radioactive waste from a typical light-water reactor facility. Data are to be obtained from several BWR and PWR reactors. The study involves the collection of a sample by passing a volume of reactor coolant through an ion-exchange column.

Data

Column dimensions	
Inside diameter:	1.0 in.
Active length:	6.0 in.
Wall thickness:	0.0625 in.
Column material:	Plexiglass (density = 1.0 g/cm^3)
Resin collection efficiency:	0.95
Resin density:	1.0 g/cm^3
Cation/anion ratio:	1.0
Sample flow rate:	100 ml/min
Sample volume for this activity:	5000 liters
Normal sampling volume:	1 liter
Reactor coolant	
Radionuclide:	Co-60
Concentration:	1.92×10^{-2} μCi/ml
Specific gamma constant:	13.2 (R-cm^2)/(mCi-hr)
Average beta energy:	0.096 MeV

Assume that activated corrosion products behave as cations. I-129 LLD for a normal reactor coolant sample is 3.0×10^{-10} μCi ml^{-1}.

Radiolytic Hydrogen Generation Rates:

G_H = 0.6 molecules H_2/100 eV (anion bed)

G_H = 0.13 molecules H_2/100 eV (cation bed)

1 rad = 6.242×10^7 MeV/g

3.30 Calculate the exposure rate in mR/hr at a distance of 18 in. from the center of the unshielded ion-exchange column at the end of the sample collection period. Clearly list all assumptions.

3.31 At the end of the sample collection period, the investigator performing this study notifies the university radiation safety officer that there will be a delay of approximately 6 months to 1 year in processing the sample that was just collected. State and determine two potential safety concerns that the RSO might have about storage of this sample over an extended period. Calculations must support your answer.

3.32 State one administrative and two operational radiological controls that should be imposed on this activity.

3.33 Calculate the new LLD obtained for I-129 as a result of this study. Assume that the radiochemical yield and counting efficiency are the same as that used for routine analysis.

Scenario 3.7

You are a university health physicist who has started a small firm manufacturing medical isotope generators. One of your products is an Mo-99/Tc-99m generator packaged within a 5-cm-thick lead shield within a $60 \times 60 \times 60$ cm^3 carton, so that the activity is in the center of the carton. Each package bears a label with the following information:

Radioactive III

Contents Mo-99

Activity 1.5 Ci

Transportation index = 3.0

The half-life of Mo-99 is 67 hr, and the half-life of Tc-99m is 6.0 hours.

3.34 What is the maximum dose equivalent rate at 1 meter from the surface of one of these packages?

3.35 Calculate the maximum dose equivalent rate at the surface of one of these packages.

3.36 Is the package legal for shipment on passenger aircraft in the United States? Why?

3.37 Four such packages comprise a shipment in the cargo hold of a commercial airliner. What is the transport index (TI) of the shipment?

3.38 A small package containing TLDs (being shipped for processing) is centered among the four packages containing radioactive materials. The TLD package is positioned 50 cm from the surface of each of these packages and left in place for 6 hours. The TLD package is separated from the generator packages by cartons of styrofoam pellets. What dose equivalents do the TLDs receive?

3.39 What provisions do commercial dosimetry vendors make to compensate for the transient dose described above and obtain valid dosimetry results?
3.40 If the generator package is returned to the manufacturer for disposal and recycling exactly 2 weeks after it was initially labeled, what will the TI be for the return package?
3.41 An Mo-99/Tc-99m generator package (not the one specified previously) has a surface dose equivalent rate of 15 mrem/hr and a TI of 2.0. What label is required and why?

Scenario 3.8

During a routine laboratory inspection you discover that a 10-Ci polonium-210 source being used by a group of experimenters has been badly leaking for about 2 weeks and that contamination has spread widely. This scenario addresses the actions that you would take in evaluating and correcting the uncontrolled spread of radioactive material.

3.42 List the first four actions you would take to begin evaluating and correcting this situation.
3.43 Based upon your past experience, identify the following pertinent parameters with respect to the possible radiation hazards from this source: primary mode of decay, decay energy, half-life, and hazard type.
3.44 During the investigation, you request measurements for airborne, removable, and fixed contamination. For each of these measurements, list the preferred equipment and locations, as well as precautions and other concerns you would have in evaluating the information.
3.45 Following the evaluation, list the areas you would decontaminate first. Why?
3.46 List items that should be included in the incident investigation report for advising the University President of the follow-up evaluations and corrective actions taken.

Scenario 3.9

You are the RSO at Excited State University where investigators use a variety of radionuclides. The major radiochemicals of concern are inorganic P-32, NaI-125 to label biological compounds, and tritiated water (HTO). All are used in millicurie quantities.

3.47 Characterize the relative hazards of these three radiochemicals in terms of skin dose potential, bioassay requirement, eye hazard, personnel dosimetry requirement, and air sampling requirement.
3.48 Discuss how you would shield 10 mCi P-32 in terms of shielding material to protect the torso when working with it on a bench top for about 3 hr per week. What material and thickness would be warranted?

3.49 If the activity of P-32 being handled is increased to 1000 mCi, how do the shielding requirements change?

3.50 Describe how P-32 radioactive waste should be processed to minimize your department's budget.

3.51 A researcher asks you for advice on whether he should take prophylactically 100 mg of potassium iodide before iodinating proteins with 2 mCi of I-125. What would you advise? Why?

3.52 You perform a wipe test in an experimental area where H-3 and I-125 are being used. If you could only afford one instrument to detect mixed H-3/I-125 contamination, what would it be? Why?

Scenario 3.10

You are the senior health physicist at the Cubs and Bears Research Reactor at the University of Illinois, Chicago. The reactor is equipped with a number of sample irradiation locations, including a pneumatic sample delivery or rabbit system.

Licensed operators are performing extended activation runs with the reactor. They will measure thermal neutron flux levels in the rabbit sample location using gold foil activation analysis. Following insertion of the sample into the reactor, the pneumatic system encountered a mechanical problem that precluded retrieval of the sample. Six hours after insertion of the sample into the reactor, the rabbit problem is repaired and the sample retrieved. The next morning (11 hours after being removed from the reactor), the sample is counted in a NaI(Tl) well detector with a multichannel analyzer.

The rabbit transfer system has an inner diameter of 4 cm. System design requirements specify a 2 second rabbit transit time from reactor to rabbit portal.

The target foil (thin) is Au-197 with a mass of 25 mg. Relevant Au-198 radiological properties include: (1) a thermal activation cross-section for Au-197(n, γ) of 98.8 b, (2) a maximum beta energy of 961 keV @ 0.9865, (3) a 412 keV @ 0.955 photon, and (4) a 2.695 d half-life. Ignore self-shielding in the gold foil.

When the gold foil is counted using a NaI system, the measured net photopeak yields 827 410 counts in 1 minute. The NaI(Tl) detector efficiency at 412 keV is 27.3%.

3.53 Using the measured values for the sample, what is the Au-198 activity in Bq when it was removed from the reactor?

3.54 For this question only, assume the sample activity upon exiting the reactor is 130 MBq. Estimate the gamma dose equivalent rate, in mSv/h, at 10 cm.

3.55 If the gold foil target contains 1.0×10^{24} Au-197 atoms and the incident thermal neutron flux is 1.0×10^{11} n/cm^2-s, what is the saturation activity?

a. 9.88×10^{36} Bq
b. 9.88×10^{12} Bq
c. 1.00×10^{35} Bq

d. 1.00×10^{11} Bq
e. 98.8 Bq

Scenario 3.11

You are the Radiation Safety Officer at the University of Fau Paux that is regulated by the US Nuclear Regulatory Commission. A molecular biologist has a National Academy of Sciences research grant to evaluate the clinical performance of I-125 (60.14 d half-life) in the form of sodium iodide. The CEDE per unit intake of I-125 is 6.53×10^{-9} Sv/Bq.

The biologist has recently initiated experiments involving the iodination of thyroid hormones and their subsequent metabolism in rats. Her laboratory technicians will perform a series of three procedures over a week: iodination, animal sacrifice and tissue preparation, and low-pressure liquid chromatography (LPLC).

Each procedure is performed in a separate room in the laboratory and the iodination is done in a fume hood. Room #3 has a volume of 60 m^3 and an exhaust rate of 15 m^3/min. You place air samplers in each of the rooms and collect a sample for the duration of each procedure. The air sample filter efficiency is 95%.

Facility and dosimetric data:

Air Sampler Data.

Room No.	Sampling Rate (L min^{-1})	Description of Procedure	Duration of Procedure (h)	I-125 Filter Activity (MBq)
1	20	Iodination	5	0.56
2	15	Animal Sacrifice	2	0.15
3	30	LPLC	6	1.76

I-125 Dose Conversion Factors.

Organ	Gonad	Breast	Lung	Red Marrow	Bone Surface	Thyroid	Remainder
CDE[a] (Sv/Bq)	1.84×10^{-11}	9.25×10^{-11}	1.19×10^{-10}	4.41×10^{-11}	4.27×10^{-11}	2.16×10^{-7}	3.33×10^{-11}

[a] Committed Dose Equivalent per Unit Intake.

I-125 Radiological Parameters.

ALI Type	Occupational ALI (MBq)	Occupational DAC (MBq/m^3)
Stochastic	4	0.002
Non-Stochastic	2	0.001

3.56 Calculate the average airborne iodine concentration in each room.

3.57 If one of the technicians was exposed to a room air concentration of 0.192 MBq/m^3 for a three-hour procedure, have any dose limits been exceeded?

3.58 One of the laboratory technicians is a declared pregnant worker. Three weeks after a procedure, she showed a decay-corrected intake of 0.27 MBq, with minimal deep dose equivalent. Based on this information, calculate the CDE and the CEDE and determine if the NCRP's fetal dose recommendation for the entire pregnancy was exceeded.

3.59 List four radiological controls that could be established to reduce personnel exposures during the procedures.

3.60 An air conditioning repairman enters Room #3 for the final one hour of the chromatography procedure and stays in the room for an additional 2 hours to monitor and test the repaired and operating ventilation system. The repairman is an unmonitored member of the public. How many DAC-hours were received and did this intake exceed the dose limit for a member of the public? For this problem, assume the air concentration in Room #3 during the chromatography procedure is 0.10 MBq/m^3.

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4
Fuel Cycle Health Physics

The nuclear fuel cycle includes the mining and processing of ore, the enrichment of that ore to reactor grade material, its use in a power reactor, and subsequent reprocessing. To date, the ore has been uranium- or thorium-based, but other materials may be utilized depending upon the economics of the specific chemical and physical processes of the fuel cycle. Waste disposal and weapons fabrication are also included within the fuel cycle arena.

Environmental issues, including the selection criteria for waste burial sites and radon issues associated with uranium mining, will be addressed in Chapter 6, which specifically addresses environmental issues. The health physics aspects associated with power reactors are discussed in Chapter 5.

4.1
Radiation in Fuel Cycle Facilities

Beta radiation fields are usually the dominant external radiation hazard in facilities requiring work with unshielded forms of uranium. Table 4.1 summarizes the major uranium and decay product emissions. The majority of the U-238 beta dose is derived from the 2.29-MeV beta emitted by Pa-234m daughter radiation.

Gamma radiation from uranium is not normally the limiting radiological hazard. However, low-level (< 5 mrem/hr) gamma fields can exist where large quantities of material are stored. Such fields create ALARA problems, particularly where large numbers of workers can be exposed.

Neutron radiation is also emitted from enriched uranium fluoride compounds. Large storage containers of UF_6, for low enriched (< 5%) and highly enriched (> 97%) material, lead to neutron radiation levels of 0.2 mrem/hr and 4 mrem/hr, respectively. At high enrichments, these neutron emissions will play a dominant role in the whole-body radiological hazards of uranium fuel cycle facilities and can be up to a factor of 2 larger than the gamma fields.

Table 4.1 Major Uranium and Decay Product Emissions.

Nuclide	Alpha Energy (MeV)	Maximum Beta Energy (MeV)	Gamma Energy (MeV)
U-238	4.15 (25%) 4.20 (75%)	–	–
Th-234	–	0.103 (21%) 0.193 (79%)	0.063 (3.5%) 0.093 (4%)
Pa-234m	–	2.29 (98%)	0.765 (0.3%) 1.001 (0.6%)
U-235	4.37 (18%) 4.40 (57%) 4.58 (8%)	–	0.144 (11%) 0.185 (54%) 0.204 (5%)
Th-231	–	0.140 (45%) 0.220 (15%) 0.305 (40%)	0.026 (2%) 0.084 (10%)
U-234	4.72 (28%) 4.77 (72%)	–	0.053 (0.2%)

4.1.1 Occupational Exposure

In 1993, the U.S. Department of Energy (U.S. DOE) amended its regulations to set new occupational radiation dose limits for its employees and contractor personnel. Although the amendments to 10CFR835 have specific criteria unique to the U.S. DOE, radiation workers are normally limited to no greater than 5 rem (0.05 Sv)/year, which is the same limit as workers whose employers are regulated by the U.S. Nuclear Regulatory Commission (U.S. NRC).

DOE regulations also allow for the 5-rem limit to be exceeded by an additional 5 rem for workers involved in planned special exposures (PSEs). A PSE would not be classified as an emergency exposure, but would only occur under "approved, well-justified, well-controlled, highly infrequent, and unusual conditions". A worker would be limited to a total of 25 rem from planned special exposures during his or her career. The U.S. DOE rule is very similar to the 1993 10CFR20 revision promulgated by the U.S. NRC.

The DOE is in the process of implementing a 10CFR835 revision based on ICRP-60 (see Appendix IV). The ICRP-60 based revision to 10CFR835 is planned for full implementation in 2010. It is interesting to note that the NRC currently has no plans to implement a revision to 10CFR20 based on ICRP-60. Neither agency has announced plans for implementing the 2007 ICRP-103 recommendations.

4.2
Nuclear Fuel Cycle

The nuclear fuel cycle describes the path that is followed by nuclear reactor fuel in its successive stages from mining of the uranium or thorium ores to the final disposal of the radioactive wastes derived from the reprocessing of the spent reactor fuel. In its ideal form, the spent fuel removed from a reactor is reprocessed and the U-235 and plutonium are recovered for subsequent reuse as new reactor fuel.

4.3
Uranium Fuel Cycle

Natural uranium consists of three primary isotopes; U-238, U-235, and U-234, whose natural abundances are 99.2739, 0.7204, and 0.0057, respectively. The decay products of these uranium isotopes consist of long decay chains that decay by both alpha and beta radiation. These chains are outlined in more detail in Chapter 6.

Uranium and its decay products are predominantly an internal radiation hazard, and their radiological hazard can be assessed by the methods noted in Appendix IV. Although uranium and its decay products are predominantly an internal radiation hazard, their beta decays produce a significant external hazard as noted in Table 4.2.

Historically, the radiological concerns of uranium have been overshadowed by chemical toxicity concerns. As a heavy metal, uranium is chemically toxic to the kidneys. Although the radiological hazards are becoming an increasing concern, the health physicist must still ensure that the chemical toxicity is properly addressed.

The theoretical cycle for uranium fuels is illustrated in Figure 4.1. The theoretical uranium cycle assumes that the spent reactor fuel is reprocessed in order to recover uranium and plutonium for subsequent recycling as reactor fuel.

Table 4.2 Beta Surface Dose Rates from Equilibrium Thicknesses of Uranium Metal and Compounds.

Source	Surface Dose Rate (mrad/hr)[a]
Natural uranium metal slab	233
UO_2	207
U_3O_8	203
UF_4	179

[a] Beta surface dose rate in air through a 7-mg/cm^2 polystyrene filter.
Source: EGG-2530.

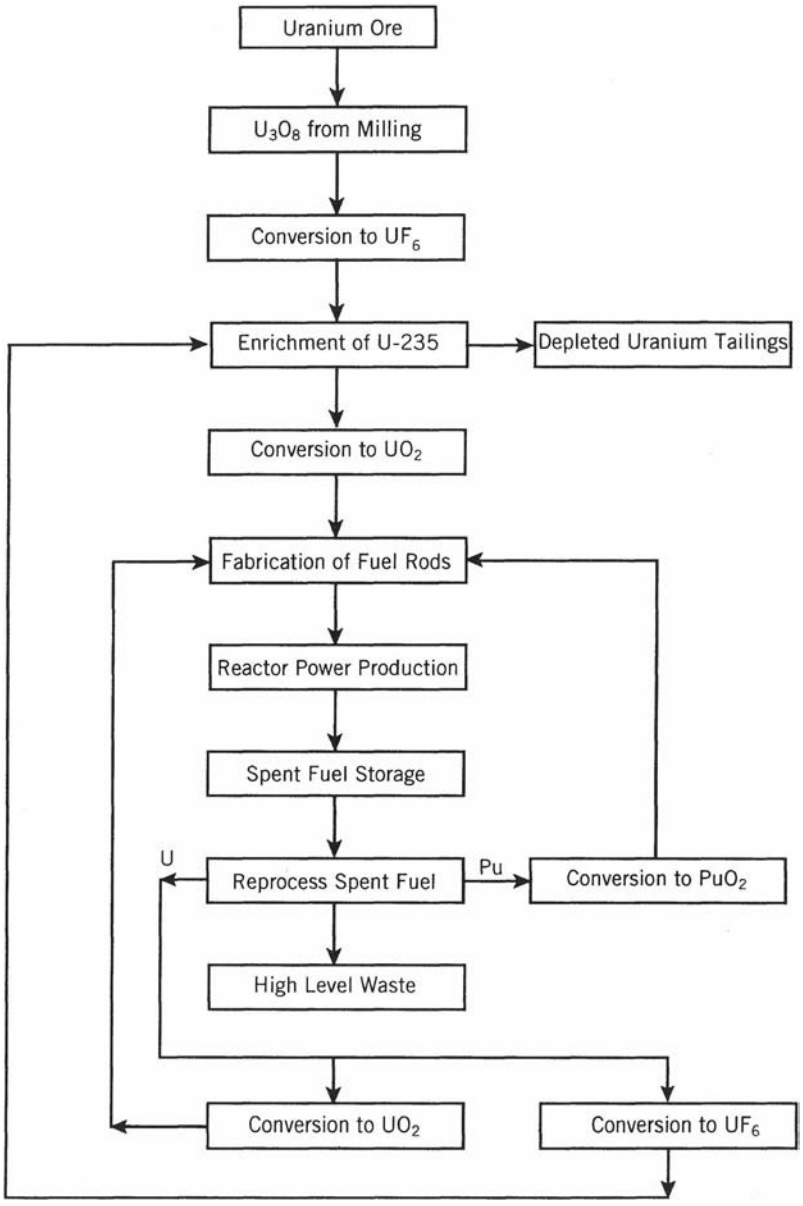


Figure 4.1 Theoretical nuclear fuel cycle for uranium.
[Adapted from the Energy Deskbook (DOE/IR/05114-1).]

4.3.1 Open and Closed Fuel Cycles

Figure 4.1 illustrates the conventional fuel cycle that utilizes single-pass recycling. Spent fuel is reprocessed to recover plutonium and uranium, and this recovered material is used to fabricate fuel for use in a subsequent reactor cycle. Single-pass recycling results in reprocessing waste including residual uranium and plutonium, other actinides, and long-lived fission products. These materials must be stored for an extended time (i.e., 10^4 to 10^6 y) and lead to an open fuel cycle with a long-term high-level waste legacy.

Single-pass recycling is preferable to no reprocessing that treats the spent fuel as waste. However, single-pass recycling only recovers 99.5% of the uranium and plutonium, and recovers none of the minor actinides (e.g., neptunium, americium, and curium) or long-lived fission products. Failure to recover the minor actinides and long-lived isotopes (e.g., ^{99}Tc (0.213 My), ^{129}I (15.7 My), ^{237}Np (2.14 My), and ^{242}Pu (0.375 My)) complicates the licensing basis for fuel repositories, fuel reprocessing facilities, and waste disposal facilities.

The removal of the minor actinides and long-lived isotopes is a step forward, but these isotopes must be destroyed to effectively eliminate their radiological hazard. Eliminating the minor actinides, residual uranium and plutonium, and long-lived fission products, significantly limits the licensing time frame of a fuel cycle facility and closes the fuel cycle. With a closed fuel cycle, the limiting isotopes are Cs-137 and Sr-90 and a facility's licensing basis is reduced from 10^4 to 10^6 y to about 10^3 y.

An attractive option to closing the fuel cycle is to destroy or burn these isotopes in a nuclear reactor. Eliminating these isotopes and closing the nuclear fuel cycle is one of the motivations for the construction and operation of Generation IV Fission Reactors (see Chapter 5).

The Generation IV technologies maximize the utilization of fissile resources and minimize high-level waste. Six candidate Generation IV reactor types are currently under review. These reactor types include: Gas-cooled fast reactors (GFRs), lead-bismuth cooled fast reactors (LFRs), molten salt epithermal reactors (MSRs), sodium-cooled fast reactors (SFRs), supercritical water-cooled reactors (SWCRs), and very high temperature, helium-cooled, graphite moderated thermal reactors (VHTRs).

The Generation IV concepts vary in their ability to close the fuel cycle and to address the disposition of high-level waste. The VHTR and thermal SWCR designs do not advance the long-term high-level waste disposal issue because they utilize an open fuel cycle with no reprocessing or actinide recycle. The long-term management of high-level waste is a significant health physics issue that impacts worker doses, offsite doses, and environmental protection issues. The candidate Generation IV concepts and their deployment projection (optimistic) are summarized in Table 4.3.

Table 4.3 Generation IV Reactor Concepts and Deployment Summary.

Generation IV Concept	Base Case Deployment Date	Type of Fuel Cycle
GFR	2025	Closed with actinide recycling.
LFR	2025	Closed with actinide recycling.
MSR	2025	Closed with actinide recycling.
SFR	2015	Closed with actinide recycling.
SWCR	2025	Closed with actinide recycling or open with once through fuel in the thermal SWCR.
VHTR	2020	Open with once through fuel.

Derived from GIF-002-00 (2002).

4.3.2 Uranium Ore and Chemical Processing

The average uranium content of U.S. ores, expressed as the oxide U_3O_8, is about 0.2 weight percent; that is, 1000 kg of ore will contain the uranium equivalent of 2 kg of U_3O_8. After processing, the raw ore is converted to U_3O_8, also known as *yellowcake*. The U_3O_8 is reduced to UO_2 (brown oxide) by utilizing hydrogen gas:

$$U_3O_8 + 2\ H_2 \rightarrow 3\ UO_2 + 2\ H_2O \tag{4.1}$$

Uranium tetrafluoride (UF_4), also known as *green salt*, is formed by heating uranium dioxide in hydrogen fluoride gas:

$$UO_2 + 4\ HF \rightarrow UF_4 + 2\ H_2O \tag{4.2}$$

The tetrafluoride is subsequently converted into uranium hexafluoride by the use of fluorine gas:

$$UF_4 + F_2 \rightarrow UF_6 \tag{4.3}$$

The mining and chemical processing steps primarily present an internal radiation hazard from ingestion or inhalation of the material. At low enrichments, chemical toxicity will be a more limiting concern than the radiological hazard.

4.3.3 Enrichment

The next step in the uranium fuel cycle is the enrichment of the U-235 content from its nominal value of 0.72% by weight in natural uranium. The traditional methods for enriching uranium are the gaseous diffusion and gas centrifuge technologies. An emerging enrichment technology involves the use of selective laser photoionization of atomic uranium vapor.

4.3.4 Gaseous Diffusion

Isotopic separation by the diffusion process is accomplished by diffusing uranium hexafluoride through a porous membrane. The different molecular weights of U-235F_6 and U-238F_6 and their resulting difference in molecular velocities are used as the basis for separating U-235 from U-238. In a mixture of U-235F_6 and U-238F_6, the average speed of the lighter U-235F_6 molecules is greater than that of the heavier U-238F_6 molecules. When the mixture contacts a porous barrier, the lighter U-235F_6 molecules strike the barrier and pass through it more frequently than the heavier U-238F_6 molecules. Because the velocity difference is small, the enrichment through each gaseous diffusion, chamber or stage is small. Consequently, thousands of stages are required to increase the assay from 0.7% U-235 to the desired enrichment.

A gaseous diffusion stage consists of a motor, compressor, and converter which contains the porous barrier or membrane and a cooler. The uranium hexafluoride is introduced as a gas and is made to flow through the inside of the barrier tube. A portion of the gas, about half, diffuses through the barrier and is fed to the next higher (increased U-235 enrichment) stage. The remaining gas that did not diffuse through the tube is fed to the next lower enrichment stage. The diffused stream is slightly enriched in U-235, and the gas remaining in the tube is slightly depleted in U-235.

A few of the stages of a diffusion cascade are illustrated in Figure 4.2. The stages above the location of feed entry are the enriching section, and the U-235 concentration exceeds that of the nominal feed concentration. In the stripping section, below the feed point, the concentration of U-235 is less than the nominal feed concentration. The enrichment increases (decreases) the further the stage is upstream (downstream) of the feed point.

As noted in the figure, the feed for each stage in the cascade is a mixture of the enriched material from the stage immediately below and the depleted material from the stage immediately above. The enriched material from a stage will be that material that has preferentially diffused through the barrier, and the depleted material is preferentially the undiffused material. The cascade operates continuously with feed material supplied (at the right side of Figure 4.2), enriched product drawn off the top, and depleted tails drawn off the bottom.

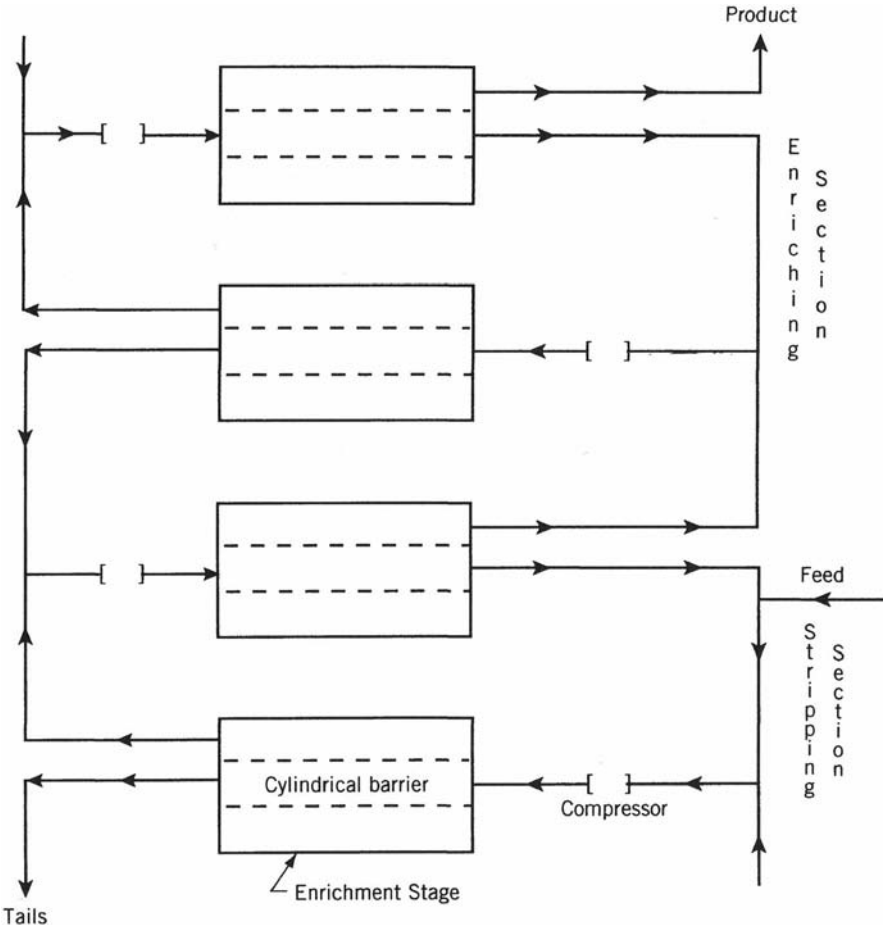


Figure 4.2 Portion of a gaseous diffusion cascade.
Material flowing radially through the cylindrical barrier will be enriched while material flowing axially along the barrier will be depleted in comparison to material entering the enrichment stage.
[Adapted from the Energy Deskbook (DOE/IR/05114-1).]

The number of stages in a cascade will be a function of a number of variables, including the isotopic concentration of the feed material, the desired product and tails concentrations, and the efficiency of the diffusion barrier material. For a typical application of natural uranium feed material, reactor-grade fuel product of 2.5–4.0% U-235 enrichment, and a tails assay of 0.2%, about 2000 stages will be required.

The number of stages could be altered if the product or tails assay were altered. For example, the number of stages would be reduced if the U-235 content of the tails material were increased. However, this change would reject a larger amount of U-235 that would be eliminated as tails material.

The primary hazard in a gaseous diffusion plant is from an acute exposure from a release of uranium hexafluoride from the process equipment. Chronic exposures may arise from routine maintenance or processing operations. The radiological hazard varies with the U-235 enrichment.

The uranium feed materials for the enrichment process may include small quantities of neptunium and plutonium. The radiological controls based upon the uranium hazards will usually be adequate to control the presence of these transuranic contaminants. However, these transuranics can represent a significant internal radiation hazard because their specific activities and Annual Limits on Intake (ALIs) are generally more limiting than those of the uranium isotopes.

For low enrichments, chemical toxicity remains the controlling hazard. At higher enrichments, radiation effects become the primary concern. Criticality must also be considered at higher enrichment stages primarily at stages near the top of the cascade.

Most of the chemical compounds encountered in a gaseous diffusion plant, including uranium hexafluoride and uranyl fluoride, are class D compounds. Interactions of these materials with the process equipment and the environment can produce Class W compounds during normal and abnormal operations.

In a gaseous diffusion plant, or other facility utilizing uranium hexafluoride, the probability of a criticality is minimized by controlling the process parameters to prevent the solidification of the uranium hexafluoride. The integrity of the process stages is also maintained to prevent the inleakage of water or moist air. Radiation monitors located in key locations provide early detection of an accumulation of solidified uranium hexafluoride. For plant components containing uranium solutions or storing uranium compounds, various criticality controls are applied. These controls include geometry and batch control, limitations on the uranium concentrations and enrichment, and administrative or procedural controls.

The primary personnel hazard from a criticality event is to personnel in the immediate vicinity of the event. Timely evacuation of personnel is an effective radiation control measure. Criticality alarms will not prevent an inadvertent criticality, but they will enhance the evacuation of personnel from the immediate area of the criticality.

4.3.5
Gas Centrifuge

Gas centrifuge technology utilizes uranium hexafluoride as its working fluid. Consequently, the health physics considerations are similar to gaseous diffusion.

A centrifuge is defined as a device for whirling an object, in a circular trajectory, with a high velocity. The force imposed on an object whirling with a velocity v is mv^2/r, where m is the object mass, v is its instantaneous velocity, and r is the radius of its circular path. For a given centrifuge design, the heavier objects will be subjected to a larger force and will tend to be moved to a larger radial distance than lighter objects. This difference in trajectories permits the heavier and lighter

objects to be separated, which is the basis for the use of a centrifuge for the enrichment of uranium isotopes.

The centrifuge device is a cylinder that rotates about its long axis. Its enrichment capacity increases with the length of the device, with the radius of the device, and with an increase in its speed. Limits in material properties restrict the available values of these parameters. The actual design of a centrifuge will depend upon the enrichment desired, the technology level of the group developing the device, and the desired end-use of the enriched material.

The working fluid in a gaseous centrifuge is composed of primarily U-235F_6 and U-238F_6. Consequently, when normal uranium hexafluoride is centrifuged, material drawn off from the interior region will be somewhat enriched in the lighter U-235 isotope.

Figure 4.3 illustrates material flow in a gaseous centrifuge. The uranium hexafluoride feed material is introduced at or near the axis of the device. The circulation of the gas is illustrated by arrows. Enriched U-235F_6 product is drawn off at the top of the figure, whereas depleted tails material is removed at the bottom of the figure. The product withdrawal scoop will be located at a smaller radius than the tails withdrawal location.

In order to obtain the desired U-235 enrichment, the gas centrifuge process is operated in a cascade of numerous stages similar to that utilized in the gaseous diffusion technology. However, the degree of enrichment is greater for the centrifuge technology. Therefore, fewer stages are needed to obtain the same enrichment with the centrifuge technology.

4.3.6 Laser Isotope Separation

Laser methods for isotope separation are an emerging technology because they are projected to be more economical than either gaseous diffusion or gaseous centrifuge technologies. Electrical energy utilization is also expected to be significantly less than existing technologies. Because of significantly higher enrichment factors, laser technology could recover the residual U-235 residing in the tailings from either diffusion or centrifuge plants.

Laser isotope separation techniques rely on the property that different isotopic species, in either an atomic or molecular form, exhibit small differences in their atomic and molecular spectra. That is, equivalent transitions from one energy level to another require a different energy to induce the transition. Thus, selective excitation is possible, and this property is a significant factor in separating isotopes.

In order to utilize the selective excitation property, an excitation source that can be selectively tuned to the desired excitation energy is required. Lasers offer a useful tool for this selective excitation.

A laser is a source of radiation that can be designed to operate at a specified frequency and intensity. Therefore, it is possible to preferentially excite one isotopic species via a precisely tuned laser and leave other isotopic species in their ground states.

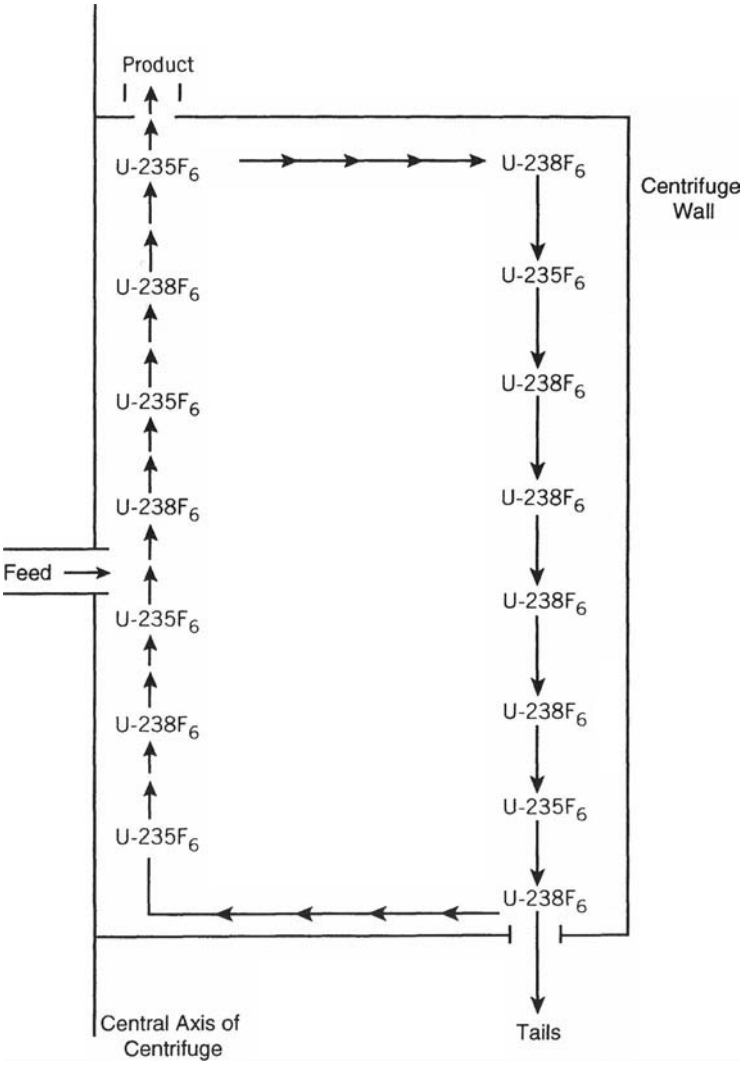


Figure 4.3 Gaseous centrifuge isotope separation technology. The product (tails) will be slightly enriched (depleted) with respect to the feed enrichment.
[Adapted from the Energy Deskbook (DOE/IR/05114-1).]

Two general laser techniques are under evaluation for the enrichment of U-235. One technique involves the use of uranium vapor, and it is based on the selective photoionization of atomic uranium atoms. A second method of laser enrichment is based on the photodisintegration of U-235 hexafluoride molecules. The molecular laser isotope separation (MLIS) and atomic vapor laser isotope separation (AVLIS) technologies will be briefly addressed.

4.3.6.1 MLIS

In the molecular process an infrared laser is utilized to preferentially excite the U-235F_6 vibrational states until the U-235 hexafluoride molecule dissociates:

$$\text{U-235F}_6 + h\nu \rightarrow \text{U-235F}_5 + \text{F} \qquad \text{(dissociation)} \tag{4.4}$$

$$\text{U-238F}_6 + h\nu \rightarrow \text{U-238F}_6 + h\nu \qquad \text{(no reaction)} \tag{4.5}$$

The excitation process is based upon the inherent assumption that the UF_6 molecule is its ground state. Thus, it may be necessary to cool the molecules via flow through an expansion nozzle in order to ensure that all molecules reside in their vibrational ground states.

The UF_6 dissociation may be enhanced with other lasers types. For example, an ultraviolet laser could be utilized to cause electronic excitation of the vibrationally excited molecule. The electronically excited state may then dissociate immediately. Once formed, the U-235F_5 molecule precipitates as a solid, leaving the unaffected U-238F_6 molecule in a gaseous form. Preferential collection of the U-235F_5 solid is the basis for enrichment.

The health physics concerns associated with uranium hexafluoride vapor also apply to the molecular separation process. The use of laser components with high-voltage power supplies introduce X-ray hazards that need to be addressed. Other concerns associated with laser technology will be addressed in a subsequent scenario.

4.3.6.2 AVLIS

In the United States, laser enrichment methods will probably utilize uranium vapor, instead of uranium hexafluoride, as the working fluid. The use of metallic uranium will impact the fuel cycle chemical processing both prior to and post enrichment. The extent of these changes will be governed by the manner in which this technology is implemented on a production scale.

The laser technology that was under development by the United States Department of Energy is the atomic vapor laser isotope separation (AVLIS) process. In the AVLIS process, the uranium metal is fed into a large vacuum vessel where it is melted and then vaporized by the impingement of an electron beam. The uranium vapor is illuminated by laser radiation which is tuned to selectively ionize only the U-235 atoms. Collection of the U-235 ions is accomplished by electromagnetic alteration of the ion's trajectory via an electromagnetic field. The un-ionized U-238 atoms pass through the collection region and are separately collected on a tails collector.

Laser enrichment presents an internal as well as external radiation hazard. Internal exposure is due to the alpha decay of U-235 and U-238 and their daughter products. In contrast to UF_6-based enrichment technologies that use class D compounds, the AVLIS technology will produce primarily class W material (UO_2, UO_4, and U_3O_8) because the uranium metal will oxidize during maintenance activities. Class D UO_3 may also be produced during the wide variety of maintenance required in an AVLIS facility.

An external hazard is presented from photons and X-rays generated from the electron impingement on the uranium metal, from the various high-voltage equipment utilized in the laser and electron beam components, and by the possibility of an inadvertent criticality event following the enrichment process. As noted in Table 4.2, uranium metal and its compounds also present a beta radiation hazard. Examples of these hazards will be addressed in one of the scenarios.

4.3.7
Spent Power Reactor Fuel

Following its use in power production, uranium-based nuclear fuel is stored in an on-site spent fuel pit. In 1977, the United States decided to postpone the reprocessing of nuclear fuel from light-water reactors. Moreover, there are no commercial plutonium breeder reactors in operation, thereby precluding the need for the fuel reprocessing step. The current U.S. fuel cycle is therefore incomplete, and it fails to operate in the most efficient or environmentally sound manner.

4.4
Thorium Fuel Cycle

A thorium-based fuel cycle is summarized in Figure 4.4. The thorium cycle requires the fuel reprocessing operation because there is no natural source of U-233.

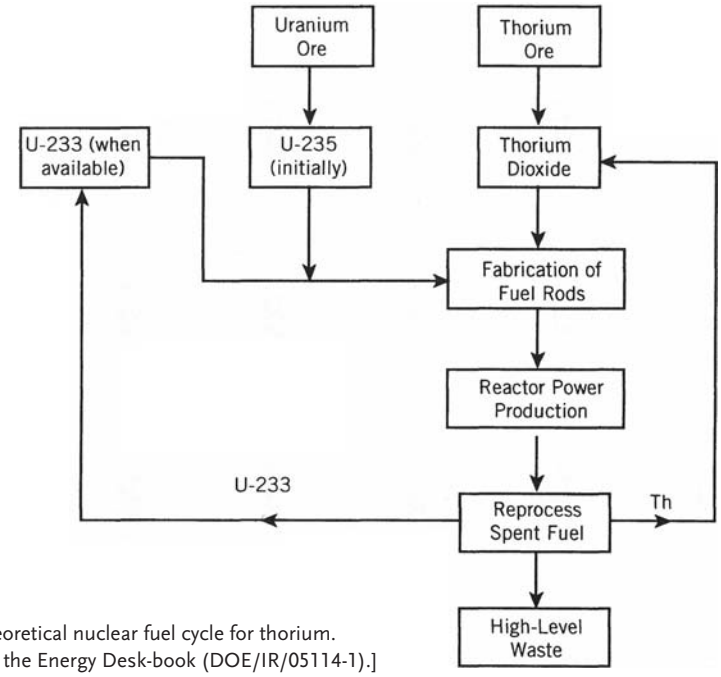


Figure 4.4 Theoretical nuclear fuel cycle for thorium. [Adapted from the Energy Desk-book (DOE/IR/05114-1).]

In the early stages of use of the thorium cycle, some U-235 would be used to ensure an adequate fissile content of the fuel. Thorium represents an alternative to uranium, but it has not been as widely utilized as the uranium fuel cycle. However, the health physics concerns of the uranium and thorium fuel cycles will be similar.

4.5 Radioactive Waste

One of the consequences of power production in a nuclear reactor or the end product of the utilization of medical radioisotopes is the generation of radioactive waste or effluents that may be released from the facility. A summary of these waste materials and effluent isotopes typically associated with the uranium fuel cycle is presented in Table 4.4. This listing is not necessarily complete, but it is indicative of the isotopes that will be encountered in the various fuel cycle activities. The isolation of radioactive waste to prevent access by people, and potential releases to the environment, are key health physics concerns.

Table 4.4 Radioactive Waste and Effluent Isotopes from the Fuel Cycle.

Operation	Waste Form		
	Gaseous	**Liquid**	**Solid**
Mining and milling	Rn-222 Po-218 Bi-214 Po-214	–	U Ra-226 Th-230 Pb-210
Refining	–	U-238 Th-234 Pa-234 Ra-226	–
Fuel fabrication	–	–	U Pu Th
Reactor operation	A-41 Kr-87, Kr-89 Xe-135, Xe-I38	H-3	Co-58, Co-60 Fe-59 Cr-51
Chemical reprocessing	H-3 Kr-85 Xe-133 I-129, I-131	Fission products dissolved in acid solutions	Pu Cm Am Np Fission products

Radioactive wastes are loosely characterized as either high-level or low-level. Low-level wastes include contaminated articles of disposable protective clothing, spent ion-exchange resins, trash, animal carcasses, or other items commonly used in power reactor, medical, research, or industrial environments. Low-level wastes are typically addressed by the methods summarized in Tables 5.7 and 5.8.

Another broad waste category is transuranic (TRU) wastes. TRU wastes contain appreciable quantities of elements heavier than uranium (plutonium, americium, and curium). These wastes are produced in the nuclear weapons program and in the fabrication of fast reactor fuel derived from recycled light-water reactor spent fuel.

4.5.1 High-Level Wastes

High-level wastes include spent nuclear fuel and the wastes associated with the reprocessing of spent nuclear fuel. Both of these waste streams contain fission products with their associated beta–gamma activity. As part of the reprocessing, the spent fuel is dissolved and the fissile material is removed. The remaining waste is a liquid containing high levels of fission fragments; residual uranium and plutonium; minor actinides including neptunium, americium, and curium; and long-lived nuclides including ^{99}Tc, ^{129}I, ^{237}Np, and ^{242}Pu. As noted previously, open fuel cycles that generate high-level waste create significant health physics challenges.

4.6 Criticality

Criticality is defined as the process of achieving a self-sustaining chain reaction. Although criticality is anticipated within the nuclear fuel core of a power reactor, its occurrence in fuel storage areas or reprocessing equipment is neither anticipated nor desired. The energy produced by the unanticipated criticality may be sufficient to terminate the criticality event by destroying the process equipment or by rearranging material in its immediate vicinity. Although the event may be short-lived, 10^{16} or more fission events can occur.

An unanticipated criticality is one of the most serious radiation hazards that a health physicist must confront. It may occur during the processing or handling of enriched uranium or plutonium, and its consequences for personnel and equipment are very severe. An accidental criticality generally requires the violation of a facility's physical and administrative controls.

Critically, safety may be affected by the following factors:

a. Quantity and type of fissile material.
b. Geometry or physical arrangement of the fissile material.

c. Enrichment of the fissile isotope.
d. Presence of a neutron moderator, reflector, or absorber materials.

Before addressing these factors in more detail, their general impact on a recent event is examined. This event is the Tokaimura criticality.

4.6.1 Tokaimura Criticality

On September 30, 1999, a criticality accident occurred at the uranium conversion facility in Tokaimura, Japan. The Tokaimura facility supported fuel manufacturing for commercial reactors (3–5 wt% U-235) and research reactors (~19 wt% U-235). The event occurred during production of fuel material for a research reactor.

A solution of enriched uranium (18.8% wt% U-235) was poured directly into a precipitation tank, which bypassed a dissolution tank and buffer column designed as part of the process controls to avoid a criticality event. Homogenization of uranium oxide was performed by mechanical stirring in the precipitation tank instead of in the mass-control equipment specified as part of the approved process.

A process-flow violation occurred by pouring uranyl nitrate solution directly from a steel bucket into the precipitation tank. The tank was not designed with a critically safe geometry. This tank was also equipped with a cooling water jacket. The water jacket provided a reflector that scattered neutrons back into the tank and sustained the neutron reaction.

When the precipitation tank solution volume reached about 40 L, a critical mass was achieved and a self-sustaining fission reaction occurred. At the point of criticality, intense gamma and neutron radiation was emitted. The poured mass was several times larger than the safe mass limit established for the precipitation tank. This action was in violation of the approved criticality control measures.

Violations of established process requirements and procedural controls significantly influenced the criticality event. In addition to procedure and process violations, the workers involved with this task were not explicitly trained for the performance of this task, and no clear and specific qualification and training requirements were established.

All of the aforementioned factors represented barriers to the criticality event. Each of these barriers was breached during the sequence of events leading to the criticality.

The criticality resulted in three workers in proximity to the tank suffering acute radiation syndrome and a number of workers and members of the public receiving radiation doses. Two of the three workers received absorbed doses estimated in the range of 10–18 Gy and died of multiple organ failure following months of intensive care. The third worker received a dose on the order of 3 Gy and survived.

4.6.2 Critical Mass

The minimum mass of material which sustains a nuclear chain reaction for a given set of conditions is called the *critical mass*. The critical mass will depend on the fissile isotope, the isotope's enrichment, its geometry, and the presence and type of moderator and reflector material. Examples of critical masses for selected fissile isotopes for moderated and unmoderated conditions are summarized in Table 4.5.

Table 4.5 Minimum Critical Mass Parameters (kg).[a]

Isotope	Moderated	Unmoderated
U-235	0.82	22.8
U-233	0.59	7.5
Pu-239	0.51	5.6

[a] Moderation is by water. Water reflection is assumed for all values.
Source: U.S. AEC, *Nuclear Safety Guide*, TID 7016 (1961).

Fissile material may also be dissolved in a solvent. For solutions, the key parameter is the density or concentration defined in terms of mass per unit volume. For example, a uniform U-235 solution of less than 12.1 g U-235/liter will be subcritical at any volume. However, larger concentrations could become critical under the proper conditions. Table 4.6 summarizes the minimum critical densities and volumes for various fissile isotopes.

Table 4.6 Minimum Critical Concentration and Volume Parameters.[a]

Isotope	Moderated Conditions	
	Concentration (aqueous) (g/liter)	Volume of Solution (liters)
U-235	12.1	6.3
U-233	11.2	3.3
Pu-239	7.8	3.5

[a] Moderation is by water. Water reflection is assumed for all values.
Source: U.S. AEC, *Nuclear Safety Guide*, TID 7016 (1961).

4.6.3 Geometry or Shape

The leakage of neutrons from a system depends on its shape and on the properties of the moderating and reflecting material. Shapes that minimize the surface-to-volume ratio (i.e., spheres) will have a smaller critical mass than other fissile material configurations. For example, minimum U-235 critical properties for a variety of geometries are summarized in Table 4.7.

Table 4.7 Minimum Critical Parameters for U-235.[a]

Geometric Shape	Critical Value
Mass of sphere	820 g
Diameter of infinite cylinder	5.4 in.
Thickness of infinite slab	1.7 in.

[a] Moderation is by water. Water reflection is assumed for all values.

Source: U.S. AEC, *Nuclear Safety Guide*, TID 7016 (1961).

4.6.4 Enrichment of the Fissile Isotope

Higher enrichment values of the fissile isotope will result in smaller values of the critical mass and volume. Enrichment is required to sustain a critical reaction with a reasonable size and volume of fissile material. Table 4.8 summarizes experiments used to illustrate the impact of enriched U-235 on the critical volume of an equilateral cylinder in water-moderated lattices. The lattices are formed using enriched metal rods whose spacing is optimized to provide the minimum critical volume. As expected, higher enrichments lead to smaller critical volumes.

4.6.5 Moderation and Reflection

A moderator is a material which thermalizes, or slows down, fast neutrons. The most effective moderators include materials containing hydrogen. The hydrogen concentration is usually expressed as the ratio of the number of hydrogen atoms to the number of fissile atoms. This ratio varies from zero for pure fissile material metal or dry unhydrated salt to several thousands for dilute aqueous solutions. The critical mass may vary from a few tens of kilograms for small ratios to a few hundred grams for an optimum moderated configuration. The presence of a moderator allows a smaller mass to become critical.

A reflected system is an assembly of materials containing a fissile material that is wholly or partly surrounded by another material having a neutron-scattering

Table 4.8 Lattices of Slightly Enriched Uranium Metal Rods in Water.

U-235 Enrichment (%)	Rod Diameter (cm)	Average U-235 Density (g/cm^3)	Critical Volume of Equilateral Cylinder at Optimum Lattice Spacing (liters)
1.027	0.98	0.055	524
	1.52	0.06	430
	1.90	0.065	393
1.143	0.98	0.055	274
	1.52	0.065	238
1.299	0.98	0.06	175
	1.52	0.075	155
2.0	1.52	0.095	58.2
	2.35	0.12	56.6
3.063	0.445	0.09	32.0
	0.762	0.105	29.8
	1.52	0.15	30.1
	2.35	0.175	35.1

Source: LA-10860-MS (1987), p. 46.

cross-section that is larger than that of air. Therefore, the presence of a reflector causes a fraction of the neutrons escaping from the fissile material to be reflected back into the material rather than escape from the system. Good reflector materials have large neutron-scattering cross-sections and small neutron-absorption cross-sections. Water, concrete, polyethylene, graphite, and stainless steel are good reflector materials.

4.6.6 Neutron Absorbers or Poison Material

Neutron absorbers or poisons are nonfissionable materials which absorb neutrons and reduce the number of neutron-induced fissions. These materials are characterized by large neutron-capture cross-sections and include material such as cadmium, hafnium, and boron. Boron is used as a soluble poison in commercial pressurized water reactors. Cadmium is a key component in power reactor control rods.

4.6.7
Consequences of a Criticality Event

In addition to direct radiation exposure, a criticality results in numerous fission events that produce a variety of gaseous and particulate fission fragments. These materials may be released into the environment. The following sections of this chapter will address methods used to characterize a radioactive release that are independent of their initiating mechanism.

4.7
Dispersion of Radioactive Gas from a Continuous Source

Estimates of the dispersion of gases into the atmosphere are based on mathematical models that consider the meteorological characteristics of the atmosphere. One of the most commonly used models for estimating the ground-level concentration of a gaseous effluent is the Gaussian plume model. In this model, the plume is assumed to travel in a straight-line trajectory and the contaminant is assumed to be normally distributed around the central axis of the plume. The downwind dispersion characteristics depend upon the atmospheric stability, wind speed, and release height according to the Pasquill–Gifford equation:

$$X(x, y) = \frac{Q}{\pi\, a_y\, a_z\, u} \exp\left[-(1/2)\left(y^2/a_y^2 + H^2/a_z^2\right)\right] \tag{4.6}$$

where

$X(x, y)$ = ground-level concentration in Bq(Ci) per cubic meter at the location (x, y)
x = downwind distance on the plume centerline (m)
y = cross-wind distance (m)
Q = release rate Bq(Ci)/s
a_y = horizontal standard deviation of the contaminant concentration in the plume (m)
a_z = vertical standard deviation of the contaminant concentration in the plume (m)
u = mean wind speed (m/s) at the plume centerline
H = effective release height (m)

Equation (4.6) assumes total reflection of the gas by the ground.

If the released material has a significant exit velocity or it exits at a high temperature, then it rises to an elevation that is greater than the physical release height. The effective release height is given by

$$H = h + d(v/u)^{1.4}\,(1 + \Delta T/T) \tag{4.7}$$

where

h = actual release elevation or stack height (m)
d = stack exit diameter at the release point (m)

v = release velocity of the gas (m/s)
T = absolute temperature of the released gas (K)

Equation (4.7) requires that ΔT be defined

$$\Delta T = T - T_0 \tag{4.8}$$

where T_0 is the absolute ambient temperature (K).

The dispersion of the plume as it moves downwind is determined by the atmospheric stability, wind speed, and downwind distance. Pasquill proposed atmospheric stability categories to facilitate calculations of ground-level concentrations. These stability classes, denoted as A, B, C, D, E, F, and sometimes G, are described in Table 4.9.

Class A represents the least stable atmospheric conditions, and it results in a broad plume with the lowest ground-level concentration. The most narrow plumes are associated with stable atmospheric conditions (F and G) and yield the largest ground-level concentrations. A qualitative assessment of the Pasquill atmospheric stability classes in terms of observable meteorological conditions is summarized in Table 4.10.

Table 4.9 Description of Stability Classes.

Stability Class	Description
A	Extremely unstable conditions
B	Moderately unstable conditions
C	Slightly unstable conditions
D	Neutral conditions[a]
E	Slightly stable conditions
F	Moderately stable conditions
G	Extremely stable conditions

[a] Neutral conditions are applicable to heavy overcast conditions during both day and night.
Source: ORO-545.

Table 4.10 Qualitative Description of Atmospheric Stability Classes.

Surface wind speed (m/s)	Daytime Insolation			Cloud Cover[a]	
	Strong	Moderate	Slight	< 1/2	≥ 1/2
< 2	A	A–B	B	F	F–G
2	A–B	B	C	E	F
4	B	B–C	C	D	E
6	C	C–D	D	D	D
> 6	C	D	D	D	D

[a] The degree of cloudiness is defined as that fraction of the sky above the local horizon which is covered by clouds.
Source: ORO-545.

4.8
Dispersion of Radioactive Particulates from a Continuous Source

The dispersion of a continuous particulate release is similar to a gaseous release, but the ground-level concentration is smaller than the gas release because some particulates may be retained in the ground. For a particulate release the dispersion is given by the approximation

$$X_P(x, y) = \frac{Q}{2\,\pi\, a_y\, a_z\, u} \exp\left[-(1/2)\,(y^2/a_y^2 + H_P^2/a_z^2)\right] \tag{4.9}$$

For particulates (P), the depletion of activity in the plume is due to gravitational settling, impaction on the surface or surface structures, precipitation settling, humidity, particulate solubility, and particle size. Therefore, calculations of the deposition of particulates are more difficult than for gaseous releases. Gravitational settling is taken into account through the effective height (H_P):

$$H_P = H - x\, v_t / u \tag{4.10}$$

where v_t is the terminal settling velocity.

Another common parameter is the ground deposition (Bq/m^2-s) of particulates (w) at the point (x, y) which is obtained from the product of the ground-level particulate concentration $X_P(x, y)$ and the deposition velocity v_g:

$$w(x, y) = X_P(x, y)\, v_g \tag{4.11}$$

Experimental values of the deposition velocity range from about 0.001 to 0.10 m/s, with an average of about 0.01 m/s.

4.9
Fuel Cycle Facilities

Fuel cycle facilities include the associated infrastructure needed to support the processes in Figure 4.1. Fuel cycle facility design and operation will require health physics controls to protect the worker from both internal and external radiation hazards. Table 4.11 summarizes the hazards associated with the uranium fuel cycle in terms of the material's enrichment, chemical form, and physical form. The information provided in Table 4.11 was derived from ICRP-30 methodology and its model-specific parameters. The use of other internal dosimetry models, including those supporting the ICRP-60 and ICRP-103 methodologies, would lead to different tabulated values for the chronic and acute internal radiological hazards.

Table 4.11 Health Physics Impact of Uranium Fuel Cycle Material Characteristics.

Characteristic	Material	Radiological Hazard	
		External	**Internal**
Enrichment	Depleted, natural, and slightly enriched uranium	Minimal penetrating (gamma) radiation.	Inhalation results in chemical toxicity.
		Moderate beta hazard from bare uranium.	Inhalation intake leads to exposure to bone surfaces, lungs, and kidneys.
	Moderate to highly enriched uranium which leads to increased U-234 and U-235 enrichments in centrifuge and diffusion processes	Penetrating radiation increases with enrichment.	Higher enrichments imply higher doses per unit mass.
		Moderate beta hazard from bare uranium.	For chronic exposures, doses become more significant and chemical toxicity less significant as enrichment increases.
		Criticality hazards increase with enrichment.	
Chemical form	Class D: UF_6, UO_3, UO_2F_2, and $UO_2(NO_3)_2$	Chemical form does not significantly affect hazard.	For acute exposures to any enrichment, chemical toxicity is more limiting.
		Neutron levels increase in homogeneous fluorine compounds.	For chronic exposures, chemical toxicity is more limiting up to 15% enrichment.
			Beyond 15% enrichment, the nonstochastic limit for bone surfaces is limiting.
	Class W: U_3O_8, UF_4, UO_4, and UO_2	Chemical form does not significantly affect hazard, except for fluorine mixtures and compounds.	For acute exposures, chemical toxicity is limiting up to 39% enrichment. Beyond 39%, the effective dose equivalent becomes limiting.
			For chronic exposures, chemical toxicity is more limiting up to 1.3% enrichment. Beyond 1.3%, the effective dose equivalent becomes limiting.
	Class Y: UC_2, UAl_x, UZr, and UO_2 (high-fired)	Same as Class W.	Chronic and acute inhalation are limited by the effective dose equivalent due primarily to lung dose.

Table 4.11 (continued)

Characteristic	Material	Radiological Hazard	
		External	**Internal**
Physical Form	Fixed	Beta dose rates are slightly higher from metal than from compounds. Penetrating dose rates are not significantly affected by the physical form.	Do not pose an internal hazard.
	Loose	Same as fixed.	Dispersible hazard as noted under enrichment and chemical form.

Source: EGG-2530.

4.10
Detection of Fuel Cycle Facility Activity

Routine swipe and air samples and direct radiation surveys are used to quantify the uranium and daughter activity within the facility. These surveys consist of direct measurement of gamma radiation along with the alpha and beta counting techniques.

Alpha particles can be counted using ionization, proportional, scintillation, or solid-state detectors. Care must be taken to account for the self-shielding of the alpha particles by the filter medium or by dust loading. The energy spectrum of the collected alpha particles can be used to identify the collected alpha emitter. Typical spectroscopy applications include the use of semiconductor detectors and membrane filters or other surface collecting filters with low dust loading.

A typical problem in counting alpha samples involves distinguishing between natural short-lived alpha activity and the alphas derived from the fuel cycle facility. The naturally occurring radionuclides – radon and thoron and their decay products – will be present in widely varying concentrations. These radionuclides will normally be at higher concentrations than the uranium isotopes in the fuel cycle facility. Moreover, radon, thoron, and their daughters will interfere with the sample analysis unless they are given sufficient time to decay after sample collection.

Radon daughters, usually more abundant than the thoron daughters, decay with an effective half-life of about 30 minutes, whereas the effective thoron daughter half-life is about 10.6 hours. The presence of natural products can be determined by counting the sample several times at intervals of a few hours.

Beta particles are counted using thin-window Geiger-Mueller, ionization, proportional, or solid-state detectors. The wide range of potential beta energies

requires care in calibrating the detector. Beta counting is less dependent upon self-absorption in the filter medium and dust loading. Beta spectroscopy can be obtained from tissue-equivalent plastic detectors.

Gamma emitters can be identified by using NaI and GeLi detectors. Direct dose rates may be determined from a variety of hand-held survey instruments.

4.11 Scenarios

Scenario 4.1

You are the lead health physicist at a plutonium processing facility and are investigating an exposure incident. A worker has reported hearing a hissing sound coming from a nearby glove box. About 1 min after hearing the hiss, the continuous air monitor (CAM) in the room alarmed. The CAM gamma count rate meter was noted by the worker to have pegged high. The worker then walked over to the hissing glove box and noticed that an extensive rupture of process vessels within the box had occurred, that material was spread throughout the internals of the box, and that one of the gloves had ruptured. He estimates that 20 s elapsed between the CAM alarm and the time he exited the room. The worker was not wearing a respirator.

The on-duty health physics technician determined that the worker was contaminated (hair, face, and clothing), and a positive indication of alpha contamination was found on a nasal smear.

The following information should be considered:

Isotope	Activity Fractions in the Mixture
Pu-238	1.2×10^{-2}
Pu-239	1.4×10^{-1}
Pu-240	3.1×10^{-2}
Pu-241	8.1×10^{-1}
Am-241	1.5×10^{-3}

The CAM data suggest that the concentration is 0.74 Bq/ml (alpha + beta)

Particle size = 1-μm activity median aerodynamic diameter

Breathing rate of worker = 20 liters/min

Inhalation Dose, First Year, 1.0 μm AMAD (Sv/Bq intake).

Nuclide	Lung	Liver	Bone Surface	Red Marrow
Pu-238(Y)	7.0×10^{-5}	9.7×10^{-7}	3.5×10^{-6}	3.0×10^{-7}
Pu-239(Y)	6.7×10^{-5}	9.2×10^{-7}	3.5×10^{-6}	2.7×10^{-7}
Pu-240(Y)	6.7×10^{-5}	9.2×10^{-7}	3.5×10^{-6}	2.7×10^{-7}
Pu-241(Y)	7.0×10^{-9}	1.1×10^{-9}	3.8×10^{-9}	2.7×10^{-10}
Am-241(W)	1.8×10^{-5}	1.5×10^{-6}	5.4×10^{-5}	4.3×10^{-6}

Weighting Factors.

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 (five highest other organs)	0.30

4.1 Calculate an estimate of the annual (first year) dose equivalent to (1) the lung, (2) red bone marrow, (3) bone surfaces, and (4) the liver of the worker. Assume that the CAM data represent the average concentration to which the worker was exposed during the incident prior to leaving the room.

4.2 Calculate the annual effective dose equivalent (first year) assuming that the organs identified in question 4.1 above are the only important target organs for internal plutonium exposure and that any external gamma exposure during this incident was negligible.

4.3 What are the relevant ICRP-26/30 stochastic and nonstochastic exposure limits appropriate for this circumstance? Was either limit exceeded?

4.4 What additional follow-up would you recommend to assess more accurately the dose to this worker? What clinical symptoms are expected for the doses calculated in question 4.1?

Scenario 4.2

As a result of process-flow problems at a depleted uranium facility, two maintenance workers are assigned a task to cut into an overhead process line and replace the pipe section suspected of being clogged. The maintenance workers note that the radiation work permit (RWP) for the job specifies that the utility services department was to relieve all line pressure prior to starting the job, and

they assume that this was done. Worker "A" is standing on a ladder to make the cut while Worker "B" is holding the base of the ladder at ground level. Both workers are wearing coveralls, gloves, booties, and lapel-type personnel air samplers as required by the RWP. Worker "A" is also wearing a half-face respirator. Worker "B" has no respiratory protection.

As Worker "A" initially cuts into the line, a heavy mist begins to be evolved from it, indicating that line internal pressure had not been shut off as assumed. Worker "A" nonetheless continued the cutting operation, while Worker "B" went to look around the process module to try and find the shut off valves. Failing to do so, Worker B then rejoined Worker "A" at the work location and they decided to call health physics via the telephone located within the module. The workers later indicated that approximately 4 minutes had passed between the time the release first occurred and the time they called health physics.

Immediately upon arriving, health physics technicians removed the workers from the work area and confirmed contamination of the workers via facial surveys and positive nasal swipes. It took approximately 6 min for health physics to arrive following the phone call from the workers. As the Senior Health Physicist, you are asked to perform an initial dose assessment for this incident. The following data should be utilized in your assessment:

Flow rate of the lapel air samplers = 2 liter/min

Breathing rate (both workers) = 20 liters/min

Gross alpha count on filter:
Worker A = 30 000 cpm
Worker B = 20 000 cpm

Alpha counting efficiency = 0.3

Respiratory protection factor (half-face) = 10

Average particle size for the mist evolved from the process line = 1 μm

The mist or aerosol is enriched uranium as highly insoluble metallic uranium/UO_2

Uranium isotopic composition (alpha activity fractions):
U-234 0.10
U-235 0.02
U-238 0.88
(*Note:* For dosimetric purposes, daughter activity can be ignored.)

Weighting factor (w_T) = 0.12 for the lung

For a 1-μm particle size, the pulmonary deposition factor is 0.25.

Inhalation lung dose equivalent conversion factors (rem/μCi for 1 μm AMAD particles) are listed below:

For the First Year.

Nuclide	D	W	Y
U-234	1.1	59	230
U-235	1.1	54	210
U-238	1.0	52	200

For 50 Years.

Nuclide	D	W	Y
U-234	1.2	59	1100
U-235	1.1	55	1000
U-238	1.0	53	980

The mathematical model used to describe clearance from the respiratory system is given below. The values for the removal half-times (T) and compartmental fractions (F) are tabulated below for each of the three classes of retained material. The values given for D_{NP}, D_{TB}, and D_P (left column) are the regional depositions for an aerosol with an AMAD of 1 μm. Figure 4.5 identifies the various clearance pathways from compartments a–j in the four respiratory regions, NP, TB, P, and L.

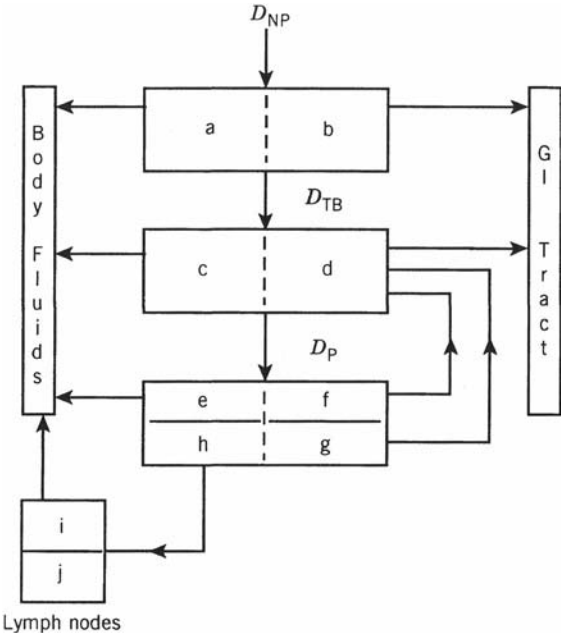


Figure 4.5 Inhalation model for the uranium uptake. (From ICRP-30, Part 1.)

Region	Compartment	Class					
		D		*W*		*Y*	
		T (day)	*F*	*T* (day)	*F*	*T* (day)	*F*
NP ($D_{NP} = 0.30$)	a	0.01	0.5	0.01	0.1	0.01	0.01
	b	0.01	0.5	0.40	0.9	0.40	0.99
TB ($D_{TB} = 0.08$)	c	0.01	0.95	0.01	0.5	0.01	0.01
	d	0.2	0.05	0.2	0.5	0.2	0.99
P ($D_P = 0.25$)	e	0.5	0.8	50	0.15	500	0.05
	f	n.a.	n.a.	1.0	0.4	1.0	0.4
	g	n.a.	n.a.	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	n.a.	n.a.	n.a.	n.a.	∞	0.1

n.a., not applicable.

4.5 Calculate the total uranium intake (gross alpha activity) for each worker.

4.6 Estimate the committed dose equivalent to the lung for the long-term retention component resultant from the deep pulmonary deposition for each worker. Consider as trivial (ignore) the dose from the short-term elimination component (biological half-life) ≤ 1 day), the dose to other regions of the respiratory tract, and any external exposure. Assume that the maximum deposition occurs for the particle size considered in the problem.

4.7 For a uranium inhalation incident such as this, what percentage of the committed dose equivalent to the lung is represented by the first year's lung dose?

4.8 Evaluate if applicable ICRP dose limit recommendations have been exceeded for either worker.

4.9 What follow-up actions would you recommend to assess more completely the long-term pulmonary dose to these workers?

Scenario 4.3

You have been selected as the Health Physics Manager for the atomic vapor laser isotope separation (AVLIS) facility at Moose River, Idaho. This facility represents the U.S. DOE's most advanced technology for the enrichment of uranium. The AVLIS technology is based upon the selective photoexcitation and subsequent U-235 ionization of the uranium vapor by laser irradiation. The photo-ions formed by the interaction of the laser beam and the uranium vapor are extracted from the vapor stream by the combined action of magnetic and electric fields. Enriched and

depleted streams are collected on separate surfaces. The process unit is a large vacuum chamber into which solid uranium feed and laser light are admitted. The internal surfaces of the process unit, particularly those that contact uranium, will be removed periodically from the vacuum chamber for cleaning and refurbishment. The uranium vapor is generated by the impingement of an electron beam upon the surface of the solid uranium source.

Use your experience with components similar to those described above to answer the following questions. Clearly state all assumptions about the process equipment and their corresponding parameters.

Data

The lasers utilize 30 000-V power supplies.

The AVLIS technology has the potential to enrich uranium to 90%.

The electron beam and associated components utilize 50 000-volt power supplies.

Cleaning of the vacuum chamber internals will be performed manually.

Internal components will be contaminated with uranium dust.

The electromagnetic field strength is high enough to warrant consideration.

The intensity of the laser radiation warrants personnel protection.

Assume that each fission from a criticality would produce eight prompt gammas with an average energy of 1 MeV and three prompt neutrons with an average energy of 2.5 MeV.

Ignore delayed radiation from fission products.

Assume that the average energy of the neutrons and gamma rays are representative of the entire spectrum.

The density of polyethylene is 1.4 g/cm^3.

The gamma fluence rate = 5.5×10^5 gammas/cm^2-s to give 1 R/hr.

The mass attenuation coefficient for polyethylene is 0.0727 cm^2/g.

The dose attenuation factor for 2.5-MeV neutrons passing through 4 inches of polyethylene = 0.3.

The flux to dose conversion factor for 2.5-MeV neutrons:
2.5 mrad/hr = 20 neutrons/cm^2-s.

4.10 List ionizing radiation hazards and their sources, nonionizing radiation hazards and their sources, and other physical hazards and their sources that can occur in this facility.

4.11 List engineering design features which could be included in the AVLIS facility to reduce the ionizing radiation hazards.

4.12 List engineering design features which could be included in the AVLIS facility to reduce the nonionizing radiation hazards.

4.13 Assume that the facility has been in operation for six months. In the process of removing enriched material from the unit, a technician violates standard operating procedures and stacks enriched uranium into a critical geometry. Following additional procedure violations, a criticality results in which 1×10^{16} fissions occur. What is the approximate dose to the technician (in rad) who is standing behind a 4-inch-thick polyethylene shield located 20 feet from the source?

Scenario 4.4

You have been selected as the health physics manager at the U.S. DOE's Frostbite Falls Fuel Reprocessing Facility (FFFRF), which consists of two large processing buildings. Building A includes a tank farm used to process highly enriched uranium. During a batch-processing operation in Building A, a technician violates standard operating procedures, which leads to a critical geometry in a small tank. A criticality results in a burst of energy in which 1.0×10^{16} fissions occur. The plant manager is standing behind a 12-in. polyethylene shield, and he is 10 ft from the center of the tank.

FFFRF Building B contains a plutonium processing tank but contains no shield walls. It is unoccupied and operated by remote control.

Data for FFFRF Building A

The density of the polyethylene shield is 1.4 g/cm^3.

Each fission event produces three neutrons and eight gammas.

The neutron spectrum for the criticality is represented by a dose conversion factor of 2.5 mrem/hr per 20 neutrons/cm^2-s.

The mean neutron and gamma energies are 2.5 MeV and 1.0 MeV, respectively.

The neutron dose attenuation factor for 2.5-MeV neutrons through 12 in. of polyethylene is 0.005.

The mass attenuation coefficient for polyethylene for the fission gamma spectrum is 0.0727 cm^2/g.

The gamma spectrum is represented by an exposure rate conversion factor of 5.5×10^5 gammas/cm^2-s per 1 R/hr.

4.14 What is the neutron dose equivalent in rem received by the plant manager during the Building A criticality?

4.15 What is the gamma dose equivalent in rem received by the plant manager during the Building A criticality?

4.16 The Building B criticality monitor is a gamma response instrument with an alarm set point of 500 mR/hr. If the detector responds to 1/2500 of the actual gamma exposure rate during a short transient in Building B, what is the maximum distance over which the device will be effective in signaling an unshielded, 1-ms criticality of 1.0×10^{16} fissions? Neglect air absorption. Assume that 1.0×10^{15} fissions yield a gamma exposure of 2.5 R at 6.0 feet.

Scenario 4.5

At nuclear facilities, noncontaminated and very-low-level radioactive materials present unique regulatory, economic, and radiological health concerns. The following questions involve establishing a reasonable and practical approach to dealing with these materials.

Miscellaneous containers of waste oil from within the radiation control area of an operating nuclear facility have been consolidated into a single 500-gallon tank.

4.17 Give two major characteristics of this raw (unprocessed) waste that make collecting a representative sample particularly difficult.

4.18 Describe two acceptable methods for obtaining representative samples from the 500-gallon tank.

4.19 Give two general categories of analyses that should be performed on the waste oil prior to the determination of the most appropriate disposal method.

As a nuclear facility health physicist, you have been asked to establish the survey protocol for the release of clean trash (paper, glass, and plastic) from within the radiation control area. Assume that alpha contamination is not a problem.

4.20 Give two general survey techniques which trash should undergo.

4.21 State the regulatory accepted limits for release as nonradioactive waste.

As a nuclear facility health physicist, you have been assigned the responsibility of developing an application for alternative disposal of slightly contaminated lagoon sludge. You have decided that because of the large volume (10 000 ft^3) and low specific activity (1.0×10^{-8} $\mu Ci/cm^3$), land farming on facility controlled property is the appropriate disposal scenario.

4.22 List four main radiation exposure pathways which must be evaluated in the application.

4.23 What level of exposure to the maximum exposed individual has been found acceptable for alternative disposal methods? You may use Title 10 of the Code of Federal Regulations (10 CFR 20.302) as a guide.

4.24 List six general environmental categories which must be evaluated for impact on waste stability and transport when submitting a request for approval of alternative disposal methods (10 CFR 20.302).
4.25 List six other types of information which must be included in the application.

Scenario 4.6

The following data are to be utilized to prepare an incident report concerning a 27-year-old male radiation worker at a pressurized water reactor:

Period of Year (Quarter)	External Deep Dose Equivalent (rem)	Intake-Inhalation Class D (μCi)
1	0.7	Cs-137: 80.0 I-131: 19.5
2	1.2	None
3	0.8	None
4	0.3	None

Nuclide	Inhalation ALI-ICRP-30 (μCi)
Cs-137(D)	200
I-131(D)	50 nonstochastic thyroid (200 stochastic)

4.26 What is the dose equivalent to the thyroid from the I-131 intake?
4.27 What is the committed effective dose equivalent from the internally deposited radionuclides? State all assumptions.
4.28 Have any of the occupational dose equivalent limit recommendations in NCRP Report No. 116 (1993) been exceeded? State which ones and justify your responses.

Scenario 4.7

You are the senior health physicist at a facility that is involved in the treatment and solidification of high-level radioactive waste that was generated from the reprocessing of spent reactor fuel. You receive a telephone call at home late one evening from the shift supervisor at the plant. He informs you that an hour ago it was discovered that a process line had ruptured, spilling high-level waste sludge into a processing area.

Upon your arrival at the plant (30 min later), you are informed that a worker had entered the process area when the leak was first discovered. He turned off the processing pump, which terminated the spill, and participated in the cleanup

of the spill event. You instruct the shift supervisor to immediately remove the worker from the contaminated area and to have him report to the decontamination and survey station.

Upon removing the worker's anticontamination clothing (booties, coveralls, and gloves) and lapel air sampler, it was determined that the worker had facial contamination and a positive nasal swab. An initial survey of the air sample was performed, and it showed considerable beta/gamma activity. Prior to permitting the cleanup activity to continue, you must assess the radiological aspects of the spill event and subsequent contamination. Using the data provided, answer the following questions.

Major Radionuclide Components of the High-Level Waste.

Isotope	Waste Concentration (Ci/L)	Gamma Constant R-m^2/Ci-hr	DCF[a] (1 year)	DCF[a] (50 year)	DAC[b] (μCi/ml)
Cs-137	0.2	0.33	2.9×10^{-2}	3.2×10^{-2}	7×10^{-8} (D, W, Y)
Sr-90	0.01	–	0.32	1.3	2×10^{-9} (Y)
Pu-238	0.001	0.0	30.0	310	3×10^{-12} (W)

[a] Dose conversion factor expressed as effective dose equivalent, rem per μCi inhaled, 1-μm particle size.
[b] DOE Order 5480.11, Table 1, Derived Air Concentration (DAC) for Controlling Radiation Exposures to Workers at DOE Facilities.

Data

Breathing rate of worker: 20 liters/min

The worker was wearing a functional lapel-type air sampler throughout the period of time he was in the contaminated area. You had the filter removed and analyzed immediately after you arrived on the scene. Assume all activity deposited on the filter resulted from resuspension of the spilled high-level waste.

Spill dimensions: Circular spill with a diameter of 5 m.

Spill volume: 100 liters

Air Sampling Data:
Flow rate of lapel sampler = 2 liters/min
Sampling time = 90 min

Total "long-lived" activity on lapel air sampler filter
(corrected for radon progeny decay):
Gross alpha = 5×10^3 dpm
Gross beta = 5×10^3 dpm (excludes Cs-137 and Y-90 contribution)
Gross gamma = 1×10^5 dpm (i.e., photons/min)

4.29 Determine the external dose equivalent from gamma rays to the worker. Assume that the worker was located at the center of the spill during the entire 90-min exposure period and that the point of interest is 1 m above the spill. Assume that 1 R = 1 rem for the external dose equivalent calculation.

4.30 Calculate the effective dose equivalent to the worker from radionuclide intakes that would be received during the first year immediately following the incident.

4.31 Calculate the 50-year committed effective dose equivalent to the worker from radionuclide intakes.

4.32 Name a currently implemented, occupational regulatory dose equivalent limit, and state whether it has been exceeded as a result of this incident. Explain your answer and state the applicable regulatory limit (i.e., 10 CFR 20, DOE Orders, etc.). Ignore considerations of any other occupational exposures independent of the incident.

4.33 Assume that you are the first person who is contacted by the shift supervisor. List four instructions that you should provide to the shift supervisor.

4.34 For Cs-137, the 1-year and 50-year dose conversion factors given above are nearly equal, while for Pu-238 they differ by more than a factor of 10. Explain the characteristics of these radionuclides that lead to this difference.

Scenario 4.8

You are the radiation protection manager at the Leakey Valley Fuel Fabrication Facility. A ten ton net weight (9071 kg) cylinder of 3% UF_6 is punctured during heating inside the fuel fabrication building. The facility building ventilation is released to the atmosphere via a 25 meter tall stack. The building has a volume of 2.83×10^4 m^3 and a stack flow rate of 2.83×10^2 m^3/min. The uranium isotopes (half-lives) of interest are: U-234 (2.5×10^5 y), U-235 (7.1×10^8 y), and U-238 (4.5×10^9 y) and the atomic mass of fluorine is 19 amu. Assume a Gaussian Plume Model is applicable and the following Class D Stability Meteorological Parameters are applicable: σ_y (500 m) is 40 m and σ_z (500 m) is 20 m.

4.35 Calculate the release rate (in MBq/s) for U-238 assuming the instantaneous release of the contents of the UF_6 cylinder and instantaneous uniform mixing of the UF_6 within the fuel fabrication building. Assume a negligible mass contribution for U-234.

4.36 Calculate the activity concentration of U-238 at 500 meters downwind at ground level at the plume centerline for class D stability conditions and 5 m/s wind speed using the Gaussian plume model equation.

4.37 State two assumptions that may contribute to the inaccuracy in the use of the Gaussian plume model in the previous question.

4.38 Will the Gaussian plume model equation tend to overestimate or underestimate the ground level concentration? Why?

Scenario 4.9

You have recently accepted a position as a Senior Health Physicist at the Swampy Flats Facility (SFF), a DOE weapons production facility. Your initial assignment involves the assessment of the adequacy of plutonium handling practices and recovery procedures.

During your assessment, a plutonium fire occurs in a glove box following air leakage into that structure. The glove box contains 500 grams of Pu-239 in the form of a fine powder and combustible solvents. The fire burns for 20 minutes immediately causing a breach of the integrity of the glove box and smoke fills the 6 m × 6 m × 3 m surrounding room.

Normal ventilation is automatically secured and emergency room ventilation starts due to the high airborne radioactivity in the room. The emergency ventilation is exhausted to the atmosphere via a single-stage high-efficiency particulate air (HEPA) filter through a 10-meter high stack. The HEPA filter maximum penetration is 0.05% and the emergency ventilation flow rate is 7 m^3/min.

During the event, the wind speed is 7.0 m/s and the meteorology stability class is C. For these conditions, the horizontal and vertical standard deviations for the plume at the site boundary (1000 m) are 110 m and 65 m, respectively.

Pu-239 has a half-life of 24 100 years. The DAC for Pu-239 is 0.08 Bq/m^3.

4.39 How many MBq of Pu-239 are contained within the glove box?

4.40 Assume 1.85×10^6 MBq of Pu-239 is initially present in the glove box. Estimate the Pu-239 concentration in air in MBq/m^3 in the room 20 minutes after the start of the fire. Assume complete combustion and release of the plutonium at a constant rate.

4.41 Assume an air concentration of 11.1 MBq/m^3 exists in the room when the fire ceases. SFF emergency procedures permit re-entry if the air concentration is equal to 1.0 DAC with respiratory protection. How much time must elapse before entry can be permitted for an inspection team whose members wear self-contained breathing apparatus respirators operating in a pressure demand mode? Assume radioactive material removal occurs by the emergency ventilation flow only.

4.42 Emergency ventilation is exhausted to the atmosphere via a single-stage high-efficiency particulate air (HEPA) filter through a 10-meter high stack. Assuming a constant air concentration of 11.1 MBq/m^3 in the room, what is the air concentration at ground level at the site boundary (on the plume centerline), which is 1000 m downwind?

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5
Power Reactor Health Physics

The field of power reactor health physics is concerned with protecting the health and safety of the public, including the plant's workers, from a wide variety of radiation environments associated with the use of the fission process to produce electricity. These environments include external radiation sources (gamma, beta, and neutron) as well as internal sources of beta and alpha radiation. The situation is often complicated by the occurrence of mixed radiation fields consisting of combinations of the previously noted radiation types.

5.1
Overview

All power reactors have a fuel core in which fissions occur, and most of the fission energy appears as heat. The nuclear fuel core contains fissile material (i.e., U-233, U-235, or Pu-239) and fertile material (U-238 or Th-232). Most commercial reactors use uranium fuel enriched to about 2–5% in U-235. The fuel consists of uranium dioxide pellets contained within zirconium rods that are arranged in a fuel bundle. The core is composed of 100–200 or more fuel bundles or assemblies.

The fuel core is cooled by water which also serves to moderate the fission neutrons to enhance reactor performance. The coolant that directly cools the core, the primary coolant, contains radioactive material which is of concern to the health physicist.

The primary reactor coolant contains a variety of dissolved and suspended radionuclides that present an external as well as an internal hazard. The primary system consists of a reactor vessel, containing the reactor core, cooling water piping, heat exchanger equipment (including a pressurized water reactor's steam generators), and a multitude of support pumps, valves, and system control components.

The various components containing primary fluid may be approximated by point, line, disk, or slab sources. For example, primary piping behaves like a line source, and hot particles are approximated by point sources. The dose rates above demineralizer beds and surface contamination are often approximated by the use of either a thin disk or slab source.

The specific isotopes that comprise the reactor sources include both fission and activation sources. Their relative importance will depend upon the type of activities performed and their timing relative to reactor shutdown.

Commercial power reactors are of two basic design types: pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). In PWRs, the pressure within the reactor vessel, containing the core and primary coolant, is high enough to prevent the water from boiling. After flowing through the core, the primary coolant is heated, pumped through a steam generator (heat exchanger), and then returned to the core region. Heat is transferred from the primary coolant to the lower pressure feedwater surrounding the steam generator tubes. These tubes form a boundary between the radioactive primary coolant and the clean feedwater or secondary coolant. Because the feedwater is at a lower pressure, it boils and produces steam that is used to drive the turbine generator to produce electricity. The exhaust steam from the turbine is condensed and returned as feedwater to the steam generator.

The second type of commercial power reactor is the BWR. The pressure of the primary coolant in a BWR is lower than that in a PWR and the primary water boils. Steam is produced directly by the fission heat, and there is no separate steam generator. The BWR steam is radioactive, which adds to the concern of the health physicist.

5.2
Generation I, II, III, and IV Reactors

The progression of fission reactor development is characterized in terms of four generations. The Generation I reactors were developed during the 1950s–60s and none are operational today. Generation II reactors are 1970s – mid 1990s facilities and include all US commercial reactors and most reactors currently operating in the world. Generation III are advanced reactors with the first becoming operational in 1996 in Japan and others under construction or ready to be ordered. Generation IV designs are still in development and their operation is optimistically projected to occur during the 2015–2025 timeframe.

Generation II reactors provide reasonably reliable power and safety performance. However, their safety performance has been criticized with the March 28, 1979, Three Mile Island Unit 2 accident serving as the focal point of that criticism. Generation III reactors are envisioned to improve the safety performance and economic viability of fission reactors.

The Generation II reactors are characterized by a variety of design concepts within the various reactor types (e.g., PWRs). These varied design concepts complicated the licensing process and the lack of standardization hindered the effective communication of operating experience. As illustrated by the Three Mile Island Unit 2 accident, Generation II reactors could be vulnerable to off-normal operating conditions and utilized active safety systems. Active systems require either electrical or mechanical actuation to occur in order for the system

to operate during an off-normal event, and these systems are potentially vulnerable to mechanical or electrical failures. A more reliable safety system operates passively and uses inherent physical properties as the basis for their design. These inherent properties include physical phenomena such as gravity or convection as the basis for their functionality.

Incorporating the lessons learned from operating Generation II reactors, Generation III reactors utilize standardized designs to facilitate the licensing process, reduce capital cost, and reduce construction time. Generation III designs are also less complex, have larger safety margins, and should be more reliable than Generation II reactors. As a result, Generation III reactors minimize maintenance requirements, simplify operations, and are less vulnerable to operational upsets. These characteristics should lead to higher capacity factors, longer operating lifetimes, and lower collective doses.

The Generation III reactors represent incremental enhancements of the basic Generation II design. Therefore, their overall operating characteristics and accident consequences are similar to the Generation II designs. The major improvements come in enhanced safety performance and gains in operability and maintainability.

The Generation IV International Forum (GIF), initiated in 2000, represents countries having a stake in nuclear energy. The Forum, including the United States, Argentina, Brazil, Canada, France, Japan, South Korea, South Africa, Switzerland, and the United Kingdom, are committed to the joint development of the next generation of nuclear technology. The characteristics of these reactors are summarized in Table 5.1. Some of these technologies operate at higher temperatures than the Generation II and III reactors, and four are designated for hydrogen production.

The six design concepts represent the potential for improved economics, safety, reliability and proliferation resistance. These Generation IV technologies maximize the utilization of fissile resources and minimize high-level waste. The specific Generation IV reactors addressed by the GIF are: gas-cooled fast reactors, lead-bismuth cooled fast reactors, molten salt epithermal reactors, sodium-cooled fast reactors, supercritical water-cooled reactors, and very high temperature, helium-cooled, graphite moderated thermal reactors. An advantage of the Generation IV design is the capability for full actinide recycling using a closed fuel cycle concept. Full actinide recycle has the potential for dose savings, but the radiological performance of the Generation IV systems has yet to be demonstrated.

Most of the Generation IV designs are variants of established Generation II and III systems and have three fission product barriers (i.e., fuel/clad, primary coolant system, and containment structure) to prevent the release of radioactive material to the environment. The Generation IV molten salt reactor (MSR) dissolves the fuel in the salt coolant and removes one of these barriers (i.e., the fuel/clad). Therefore, the MSR concept only has two fission product barriers. The health physics issues associated with the MSR design are unique and depend on the ability of the salt-fuel eutectic to retain fission products over a range of operating conditions and transients.

Table 5.1 Generation IV Reactor Concept Characteristics.

Reactor Technology	Power Rating (MWe)	Operating Temperature (°C)	Fuel	Economic Justification
Gas-cooled fast reactors	288	850	^{238}U, other fertile materials, and fissile materials	Electricity and hydrogen are produced.
Lead-bismuth cooled fast reactors	50–1200	550–800	Fuel is ^{238}U metal or nitride	Electricity and hydrogen are produced.
Molten salt epi-thermal reactors	1000	700–800	The uranium fuel is dissolved in a salt coolant.	Electricity and hydrogen are produced
Sodium-cooled fast reactors	150–1500	550	Metal or mixed oxide	Electricity is produced
Supercritical water-cooled reactors (thermal and fast versions)	1500	510–550	UO_2	Electricity is produced
Very high temperature, helium-cooled, graphite moderated thermal reactors	250	1000	UO_2 Can also utilize Pebble Bed Modular Reactor and Gas Turbine-Modular Helium Reactor fuel	Electricity and hydrogen are produced

Derived from Uranium Information Centre, Briefing Paper # 77 (2005).

In view of the status of commercial power reactors, this chapter focuses on Generation II systems. The discussion of Generation II systems is initiated with common health physics hazards.

5.3 Health Physics Hazards

The power reactor health physicist must deal with a wide variety of health physics issues. Issues, such as internal and external exposure control, are not unique to the power reactor field, but their application is unique to the reactor environment. In order to address the power reactor health physics concerns, we will present a sampling of exposure concerns associated with a commercial reactor.

Table 5.2 Power Reactor Health Physics Concerns.

Activity	Concern		Hazard
	Internal	External[a]	
Primary component maintenance during an outage	Yes	Yes	Activation products and fission products, depending on the fuel integrity. Hot-particle controls are warranted.
Primary component maintenance during power operation	Yes	Yes	Activation and fission products, hot particles, neutrons, and N-16.
Steam generator eddy current surveillance and repair (outage)	Yes	Yes	Activation and fission products and hot particles.
Spent-fuel-pool fuel movements	Yes	Yes	Hot particles and criticality.
Containment at power inspections	Yes	Yes	Noble gas, skin exposure, tritium, iodine, neutrons, and N-16.
Radioactive waste processing	No[b]	Yes	Activation and fission products and hot particles.

[a] Direct radiation exposure.
[b] Unless personnel error or procedure violations occur.

A summary of contamination and direct dose radiation hazards from typical power reactor activities are summarized in Table 5.2. Examples of these hazards include:

1. The buildup of activity on a demineralizer bed or filter.
2. Activation of reactor components.
3. Fuel element cladding failures.
4. Reactor coolant system leakage.
5. Hot-particle skin dose.

These hazards are typical of the challenges that are part of a power reactor environment. They are, however, not a complete listing of the challenges faced by nuclear power plant health physics personnel.

5.3.1 Buildup of Filter or Demineralizer Activity

A common power reactor problem involves the calculation of the buildup of activity on a filter element or demineralizer bed. Demineralizers remove radionuclides by an ion-exchange mechanism. The buildup of activity (A) on a filter or in a demin-

eralizer bed can be determined by knowing the system properties and the isotopes present. Important parameters that impact the buildup of the activity include the concentration of the isotope entering the demineralizer, the system flow rates, and the time the demineralizer is operating. The demineralizer activity can be defined in terms of these parameters through the following relationship:

$$A = \frac{C\,F\,e}{\lambda}\left[1 - \exp\left(-\lambda\, t\right)\right] \exp\left(-\lambda\, t_{\text{decay}}\right) \tag{5.1}$$

where

A = activity of the isotope at time t
C = influent activity (μCi/ml)
F = flow rate (ml/min)
e = efficiency of the filter/demineralizer for removal of the isotopes of interest
λ = decay constant
t = time demineralizer/filter is on line
t_{decay} = decay time or time demineralizer/filter is isolated following loading

Equation (5.1) is an application of a production equation (see Appendix VII) with the production term given by the product $C\,F\,e$.

This activity can be used in dose rate estimates. Estimates of streams involving multiple isotopes require the application of (5.1) for each nuclide present in the influent stream.

5.3.2 Activation of Reactor Components

Another source of activity is the direct irradiation of reactor components and the activation of corrosion products. Corrosion or wear material, dissolved or suspended in the coolant, passes through the core region. Within the core region, it is exposed to the core's neutron fluence. Activation occurs by a variety of neutron-induced reactions, and its magnitude depends upon the neutron fluence impinging upon the material in the core region.

The activity derived from the irradiation is given by

$$A = N\,\sigma\,\phi\left[1 - \exp\left(-\lambda\, t_{\text{irrad}}\right)\right] \exp\left(-\lambda\, t_{\text{decay}}\right) \tag{5.2}$$

where

N = number of atoms of a given isotope that are activated
σ = cross-section for the reaction induced by the flux ϕ (barns/atom)
ϕ = fluence rate or flux (neutrons/cm^2-s) inducing the activation reaction
λ = decay constant of the activated material
t_{irrad} = time the sample was irradiated or exposed to the core flux
t_{decay} = decay time or time the sample was removed from the reactor's core region or activating flux

Equation (5.2) represents another example of a production equation (see Appendix VII) with the quantity $N \sigma \phi \lambda$ representing the production term for activation.

Once the activity of a source is known, its dose rate impact can be determined from a knowledge of its basic geometry properties. Common geometries include the point, line, disk, and slab sources.

For example, the dose rate at a distance r from a small particle source can be obtained from the point source approximation. The point source approximation is applicable whenever the distance from the source is at least three times the largest source dimension.

A second useful approximation encountered in a power reactor environment is the line source approximation. The line source equation is often useful when assessing the dose from sample lines or piping carrying primary coolant or other radioactive fluids.

The third useful relationship for estimating the dose rate from typical power reactor components is the thin disk source approximation. A disk source provides a reasonable approximation to the dose rate from a radioactive spill or from a demineralizer bed.

Slab sources can be useful in approximating the dose rates from contaminated soil or contaminated pools. Dose rates from a spent fuel pool whose coolant activity is known, or from contaminated concrete floors or walls, can be addressed with reasonable accuracy with a slab source approximation.

5.3.3 Cladding Failures

A nuclear reactor contains a number of barriers designed to prevent fission products from escaping from the reactor core to the environment. These barriers include the fuel element cladding, the reactor coolant system and included piping, and the containment building. A breach of any of these barriers warrants serious attention in order to prevent the release of radioactivity to the environment.

A cladding failure immediately releases fission products contained within the fuel element and increases the primary coolant activity. The gaseous activity that is released to the primary coolant is released to containment via leakage paths and directly to off gas systems. These gaseous fission products are an early indication that a fuel cladding failure has occurred. BWRs normally detect fuel failure by detection of fission gases in the off-gas system. However, PWRs normally monitor the primary coolant line or letdown filter lines for these fission products, or monitor the containment atmosphere for released xenon and krypton and their daughter products. The analysis of primary coolant samples by gamma spectroscopy is a routine follow-up action in either type of reactor.

5.3.4
Reactor Coolant System Leakage

Leakage from the primary coolant system is an undesirable but inevitable problem at a power reactor. Leaking valves, pump seals, valve packing, or instrument lines provide pathways for small leaks that contaminate the vicinity of the leak. This contamination must be controlled in order to limit station external and internal exposures. In addition to leaks directly from the primary system, PWR health physicists must address leakage from the primary to secondary systems.

Leakage from the primary system steam generator tubes to the secondary system presents a health physics concern because additional plant areas may become contaminated. Because the secondary components are considered as clean systems, the presence of contamination can have a negative impact on facility operation and expand the areas requiring stringent radiological controls at the facility. In addition to direct leakage, a pathway to the secondary system permits the release of fission gases to the clean side of the facility.

The secondary side gaseous activity can be calculated by assuming an activity balance. Assuming that all gaseous activity released from the primary system, via the leaking steam generator tubes, is released without holdup to the atmosphere through the condenser air ejector, the secondary activity may be determined from the activity balance relationship

$$A_p^i \, L \, k_1 = A_s^i \, \dot{F} \, k_2 \tag{5.3}$$

where

A_s^i = the air ejector activity for isotope i ($\mu Ci/cm^3$)
A_p^i = primary coolant activity of isotope i ($\mu Ci/cm^3$)
L = primary to secondary leak rate (gal/min)
$\dot{F}$ = air ejector flow rate (ft^3/min)
k_1 = conversion factor (3785 cm^3/gal)
k_2 = conversion factor (28 317 cm^3/ft)

Activity from fission or activation products can lead to significant buildup of surface contamination from the primary to secondary leakage. If the leak is not repaired in a timely manner, an area (A) of the plant will be contaminated to a level S at the end of a year's time. The secondary plant surface contamination is given by

$$S = A_s \, K \, L / A \tag{5.4}$$

where

S = surface contamination level (dpm/100 cm^2-year)
A_s = secondary system activity ($\mu Ci/cm^3$)
K = $\dfrac{(3785 \text{ cm}^3/\text{gal})\,(365 \text{ days/year})\,(2.2 \times 10^6 \text{ dpm}/\mu\text{Ci})}{929 \text{ cm}^2/\text{ft}^2\,(100)} \times (60 \text{ min/hr})\,(24 \text{ hr/day})$

L = secondary system leak rate (gal/min)
A = area of secondary system exposed to the secondary system activity (ft^2)

Equation (5.4) is an application of a production equation (see Appendix VII) with the production term given by the product A_s L. Contamination buildup over a shorter period of time may be obtained by adjusting the conversion factor K.

PWR secondary coolant contamination has a number of negative health physics aspects. The leakage activity tends to concentrate in components such as the main steam isolation valves and high-pressure turbine piping such that locally higher surface contamination areas and local hot spots result. Secondary ion-exchange resins and filters will become contaminated, which adds to the unit's contamination problems and increases the volume of radioactive waste generated. Steam generator cleanup systems will also become contaminated, and the reassessment of the plant's effluent releases may be required.

5.3.5
Hot-Particle Skin Dose

Particulate matter is produced by a variety of power reactor activities. Normal maintenance of pumps, valves, and piping create small particles in the process of cutting, grinding, and welding. The operation of valves and pumps leads to the wearing of active surfaces, and this wear often produces small particulate material. Cladding failures or erosion of control rod surfaces contribute additional matter to the reactor coolant system. This material is often too small to be removed by the reactor coolant filters. Therefore, it passes through the core and is activated in the core's neutron fluence. The result of this activation is the creation of highly activated, microscopic material that is called a "hot particle".

Hot particles are very small highly radioactive particles with high specific activities. The particles may be composed of activation products and possibly fission fragments depending upon the core's fuel clad integrity. Particles may contain either single isotopes or a large number of radioisotopes. Skin exposure is the primary radiation hazard from these particles. Beta radiation is the dominant contribution to the skin dose, but the gamma component can contribute as much as 30% to the total hot-particle skin dose.

Table 5.3 summarizes the types of hot particles that are found in power reactor environments and classifies them into three broad categories: fission products, activation products, and zirconium/niobium particles. The fission product hot particles consist of fission nuclides that have escaped from the fuel cladding. Activation products represent corrosion or wear products that have been subsequently activated by the core's neutron fluence. The zirconium/niobium particles are derived from cladding material wear.

Table 5.3 Representative Radionuclide Distribution in Hot Particles at a PWR for Three Types of Particles.

Nuclide	Fraction of Total Activity		
	Fission Product	Activation Product	Niobium/Zirconium
Na-22	2.7×10^{-3}	–	–
Sc-47	–	–	1.3×10^{-2}
Cr-51	–	4.8×10^{-1}	8.8×10^{-3}
Mn-54	–	4.0×10^{-3}	1.5×10^{-3}
Co-57	–	1.0×10^{-3}	–
Co-58	8.5×10^{-4}	4.3×10^{-1}	2.0×10^{-3}
Fe-59	–	5.0×10^{-3}	4.1×10^{-4}
Co-60	5.6×10^{-4}	6.9×10^{-2}	1.3×10^{-3}
Nb-95	–	1.0×10^{-3}	5.2×10^{-1}
Zr-95	–	–	4.0×10^{-1}
Ru-106	1.2×10^{-1}	–	–
Rh-106	1.2×10^{-1}	–	–
Sn-113	–	2.0×10^{-3}	1.4×10^{-2}
In-113	–	2.0×10^{-3}	1.4×10^{-2}
Sb-125	9.2×10^{-3}	–	1.5×10^{-2}
Te-125	9.2×10^{-3}	–	1.5×10^{-2}
Cs-134	5.8×10^{-2}	–	–
Cs-137	2.9×10^{-1}	–	–
Ce-144	1.9×10^{-1}	–	–
Pr-144	1.9×10^{-1}	–	–
Hf-181	–	–	2.8×10^{-3}
Am-241	2.6×10^{-3}	–	–

Source: W. W. Doolittle, R. S. Bredvad, and J. J. Bevelacqua (1992).

Table 5.3 summarizes typical particulate compositions which are classified according to their dominant constituent isotopes. The reader should note that each particle may contain isotopes from any of the three classifications.

A common problem encountered in a power reactor involves the assessment of skin exposures from a hot particle. The hot particle may reside directly on the skin, on protective clothing, or on personal clothing. Table 5.4 provides calculated beta and gamma dose rates to the skin from the particles of Table 5.3. Dose rates are provided for a variety of particle locations. The skin doses are provided for unit activity and assume an exposure area of 1 cm^2 at a depth of 7 mg/cm^2.

Table 5.4 Skin Exposure Rate per μCi for Various Protective Clothing Configurations.

PCC[a]	Skin Dose Rate for Particle Type (mrad/hr to 1 cm^2)						
	Density Thickness (g/cm^2)	Fission Product		Activation Product		Niobium/Zirconium	
		Beta	Gamma	Beta	Gamma	Beta	Gamma
Bare skin	0.0070	5.2×10^3	1.7×10^2	7.8×10^2	3.2×10^2	2.4×10^3	4.0×10^2
One pair of PCs	0.0278	2.2×10^3	–[b]	1.3×10^2	–	3.6×10^2	–
Two pair of PCs	0.0530	1.4×10^3	–	2.4×10^1	–	5.6×10^1	–
One pair of rubber gloves	0.0567	1.3×10^3	4.0×10^1	1.9×10^1	7.3×10^1	4.3×10^1	9.2×10^1
Two pairs of rubber gloves	0.1028	8.9×10^2	–	1.3	–	3.5	–
Three pairs of rubber gloves	0.1487	7.0×10^2	2.0×10^1	9.9×10^{-2}	3.7×10^1	8.7×10^{-1}	4.6×10^1
Boots-I	1.6822	2.0	–	1.2×10^{-9}	–	1.0×10^{-10}	–
Boots-II	1.9050	8.9×10^{-1}	5.2×10^{-1}	0.0	9.7×10^{-1}	0.0	1.2

Source: W. W. Doolittle, R. S. Bredvad, and J. J. Bevelacqua (1992).
[a] Protective clothing configuration.
[b] Not calculated.

The use of the data of Table 5.4 would require a knowledge of the total particulate activity. Obtaining this information requires counting in a calibrated geometry which may take an hour or more. In order to obtain a quick estimate of the dose to the exposed worker, it would be desirable to be able to convert the survey instrument reading used to detect the particle into an exposure rate. Table 5.5. provides factors that are useful in obtaining quick estimates of the hot-particle dose rate from open-window ion-chamber readings.

Follow-up action by the health physicist will require a more refined dose assessment. These assessments are required from both regulatory and legal perspectives and may be performed by a variety of methods.

The skin dose from a hot particle residing on the skin is given by

$$D = \sum_i A_i (DF_i)\, t / S \tag{5.5}$$

where

D = hot-particle skin dose (rad)
A_i = particle activity (μCi) for radionuclide i
DF_i = dose factor for radionuclide i (rad-cm^2/μCi-hr)
t = residence time on the skin (hr)
S = area over which the dose is averaged (cm^2)

Following NCRP-130, the skin dose is generally averaged over 10 cm^2 and evaluated at a distance of 7 mg/cm^2 at the basal cell layer depth. Because the dose from a point source falls off rapidly as 1 over *r*-squared, the dose from a hot particle is highly localized.

Table 5.5 Ratio of Skin Dose Rate to Ion-Chamber Survey Instrument Reading for Various Protective Clothing Configurations (RSO-50).

Protective Clothing Configuration	Ratio of Skin Dose Rate to Observed Instrument Reading for Particle Type		
	Fission Product	Activation Product	Niobium/Zirconium
Bare skin	2.2×10^{2}	3.0×10^{2}	1.3×10^{2}
One pair of PCs	9.2×10^{1}	7.4×10^{1}	2.5×10^{1}
Two pairs of PCs	5.8×10^{1}	3.1×10^{1}	7.8
One pair of rubber gloves	5.4×10^{1}	2.6×10^{1}	6.4
Two pairs of rubber gloves	3.7×10^{1}	1.6×10^{1}	3.4
Three pairs of rubber gloves	2.9×10^{1}	1.0×10^{1}	2.2
Boots-I	1.0×10^{-1}	3.1×10^{-1}	3.8×10^{-1}
Boots-II	5.6×10^{-2}	2.8×10^{-1}	3.3×10^{-1}

Source: W. W. Doolittle, R. S. Bredvad, and J. J. Bevelacqua (1992).

5.4 NCRP-130 Hot Particle Recommendations

Since hot particles are small and mobile, they can attach to and irradiate various tissues. In addition to the possibility that a hot particle attaches to the skin, it can become lodged in eyes or ears. Hot particles can also be inhaled and irradiate the lungs or be ingested and irradiate the organs comprising the gastrointestinal tract. NCRP-130 provides guidance for addressing hot particle doses to each of these tissues.

Hot particle doses can be substantial. Since large doses are possible, guidance is needed for dose assessments and medical review. For hot particles that reside on skin, ears, eye, hair, or clothing, NCRP-130 recommends absorbed dose limits that ensure the skin's barrier function is maintained. Preserving the skin's integrity is important because it maintains the primary barrier to contaminants entering the body.

For the skin and ear, an observation period of 4–6 weeks is recommended when 0.1 Gy is exceeded. Absorbed doses should be evaluated at a depth of 70 μm and the dose should be averaged over the most highly exposed 10 cm^2 area. NCRP-130 recommends an absorbed dose limit of 0.5 Gy for the skin and ear. This is a per-particle limit as long as the areas exposed by the hot particles do not overlap. If two or more hot particles irradiate the same area, then the limit applies to the calendar year, rather than to individual events.

For the eye, the NCRP-130 absorbed dose limits ensure no loss of visual function, breaching of the eye (e.g., cornea damage), or breakdown of eye-related skin (e.g., eyelid). An observation period of 4–6 weeks is recommended when 1 Gy is exceeded. Eye doses should be evaluated at a depth of 70 μm and averaged over the most highly exposed 1 cm^2 area. An absorbed dose limit of 5 Gy/y is recommended by NCRP-130. Because of the small size of the eye, this value should be viewed as an annual limit for each eye, even if hot particles are believed to have been present at different locations and/or times.

Respiratory system absorbed dose limits are based on currently applicable effective dose limits, with the effective dose determined using general respiratory system models (i.e., ICRP-66) and residence times for insoluble material, unless *in vivo* bioassay data indicates that particle locations and residence times deviate significantly from the default model values. Medical referral is made if *in vivo* bioassay data indicate sequestration of a hot particle and respiratory symptoms occur. For the special case of hot particle sequestration in the anterior nasal compartment, the evaluation depth is 70 μm and the dose is averaged over the most highly exposed 1 cm^2 area. The absorbed dose limit is 5 Gy/y. Because of the small size of the anterior nasal compartment, this should be viewed as an annual limit, even if hot particles are believed to have been present at different locations and/or times.

Gastrointestinal (GI) System absorbed dose limits are based on currently applicable effective dose limits, with the effective dose determined using general GI system models (i.e., ICRP-100) and residence times for insoluble material,

unless *in vivo* bioassay data indicate that particle locations and residence times deviate significantly from the default model values. Medical referral is made if *in vivo* bioassay data indicate sequestration of a hot particle and GI symptoms occur.

5.5 Health Physics Program Elements

In addition to addressing radiation hazards, the health physics program at a power reactor must consider a variety of other factors. The radiation protection program at a power reactor includes the goal of maintaining radiation exposures as low as reasonably achievable (ALARA), the assessment of plant effluent releases, assessment of internal and external exposures, radiation surveys, personnel monitoring, radiological environmental surveillance, radioactive waste management, and control of work activities. These program elements that are unique to a power reactor environment will be addressed in more detail below.

5.5.1 ALARA

New regulatory requirements and professional ethics suggest the need to minimize radiation exposures of plant workers. An effective ALARA program requires management commitment and cooperation between the various facility work groups, particularly health physics, maintenance, and operations. Examples of this cooperation include:

1. Development of pre-job ALARA briefings and post-job ALARA critiques for high-dose activities. These activities include reactor vessel head removal and undressing, steam generator surveillance and repair activities, reactor coolant pump seal replacement, reactor vessel head installation, and reactor cavity work.
2. Development of outage pre-planning and post-outage critiques from a radiation exposure perspective. Personnel involved with the work activities must take an active part in these activities.
3. Establishing exposure goals for each outage activity and major nonoutage evolutions.
4. Development of a system for tracking exposure by work request to identify improvement areas.

In addition to these cooperative activities, the radiation protection group can significantly reduce worker's exposures by minimizing the use of protective clothing. The utilization of respiratory protection should focus on minimizing the total worker radiation exposure. Because the total dose is a sum of both internal and external exposure, respirators should not be used to reduce only the internal dose.

Contamination control is another key to an effective ALARA program. For example, minimizing contamination levels in the refueling cavity minimizes direct dose and reduces the protective clothing requirements. The use of remote technology, including cameras, state-of-the-art radiation monitors, and wireless headsets, is also effective in minimizing exposures.

Finally, the support of plant management is required for a successful ALARA program. Plant management support will be required in the following types of areas:

1. Budgetary support to upgrade health physics equipment.
2. Improving preventive maintenance of plant equipment.
3. Training personnel in ALARA techniques.
4. Training and utilizing dedicated crews for specific high-dose tasks.
5. Providing sufficient training for contractor outage personnel.
6. Chemical decontamination of highly contaminated primary system components.
7. Reduction of filter sizes to remove suspended particulates.

5.5.2 Effluents

Although off gas systems are designed to trap most gaseous effluents, quantities of Kr-85, H-3, and C-14 and smaller amounts of Xe, I, and Br isotopes may also be released. The Kr, Xe, and I isotopes are generated in the fission process. Their release is facilitated by breaks in the fuel clad. Tritium arises from the neutron

Table 5.6 Typical Power Reactor Activation Products.

Radionuclide	Neutron Energy Region	Source	Reaction
H-3	Thermal	Lithium hydroxide[a]	$^{6}Li(n, \alpha)^{3}H$
	Thermal	Boric acid[a]	$^{10}B(n, 2\alpha)^{3}H$
	Fast	Primary coolant	$^{2}H(n, \gamma)^{3}H$
Mn-54	Fast	Stainless steel[b]	$^{54}Fe(n, p)^{54}Mn$
Fe-59	Thermal	Stainless steel[b]	$^{58}Fe(n, \gamma)^{59}Fe$
Co-58	Thermal	Stainless steel[b] and stellite	$^{57}Co(n, \gamma)^{58}Co$
	Fast	Stainless steel[b] and stellite	$^{58}Ni(n, p)^{58}Co$
Co-60	Thermal	Stainless steel[b] and stellite	$^{59}Co(n, \gamma)^{60}Co$
Zr-95	Thermal	Zirconium fuel cladding	$^{94}Zr(n, \gamma)^{95}Zr$

[a] Primary coolant chemistry control.
[b] Corrosion or wear products and structural material.

activation of the primary coolant, from neutron capture in B-10 [B10(n, 2α)H-3], and from tertiary fission. C-14 is produced from the N-14(n, p)C-14 reaction.

Liquid effluents include fission products and activation products as well as tritium. Liquid-waste cleanup systems, including filtration and demineralization, remove much of these radionuclides which are then processed for burial as low-specific-activity waste.

Fission product radionuclides generated from binary fission include Kr-85, Kr-87, Kr-88, Xe-133, Xe-135, Xe-137, I-131, Cs-137, Ba-137, Ce-141, Ce-144, Ru/Rh-103, Ru/Rh-106, and Sr/Y-90. Activation products are produced by neutron capture by materials in the vicinity of the nuclear core. These include (a) chemical control agents dissolved in the primary coolant (Li-6 and B-10), (b) stainless steel or stellite corrosion or wear products resulting from system maintenance, (c) primary coolant system piping and the reactor vessel, and (d) the core structural material. Examples of these activation products are summarized in Table 5.6.

5.5.3 Radioactive Waste

Power reactor waste occurs in solid, liquid, and gaseous forms. Solid waste includes spent reactor fuel, ion-exchange resins, filters, evaporator bottoms, articles of protective clothing, glassware, tools, and materials used in the control of contamination (paper, plastic, and absorbant materials). Liquid waste includes the primary reactor coolant, cooling water in the spent-fuel pool, contaminated solvents and pump oil, spent chemical reagents, and scintillation fluids. Sources of gaseous radioactive waste are fission product gases and gases due to neutron activation of the primary coolant.

Processing of these wastes is summarized in Table 5.7. With rising burial costs, volume reduction and source term control are becoming important considerations in the economics of plant operation.

The use of incinerable or dissolvable materials leads to significant volume reduction. Following use, the incinerable material is shipped to an offsite vendor and burned. The vendor returns a small volume of ash. The ash is disposed as low-level waste with significant cost savings. Similar cost savings are achieved using dissolvable materials. Following their use, the materials are shipped to an offsite vendor and processed. The residual activity is removed using filtration and demineralization. The residual filter or resin volumes become low-level waste and are disposed at a fraction of the cost of the original volume of material.

5.5.4 Decontamination

Radioactive waste minimization is important for cost and ALARA considerations. One key aspect of minimization is the effective decontamination of material to preclude their disposal as radioactive waste.

Table 5.7 Radioactive Waste Processing Methods.

Solid	Liquid	Gas
Compaction of low-specific-activity material into boxes or drums.	Concentrate by evaporation. The remaining material is usually solidified or stabilized.	Dilution with large air volumes, monitoring, and stack discharge.
On-site storage to allow decay prior to burial.	Dilution of liquid followed by monitoring and discharge to large bodies of water or rivers.	Hold up gases in tanks to permit decay prior to release.
Solidification of resin and evaporator bottoms.	Ion exchange to remove dissolved radioactivity.	Off-gas treatment (activated charcoal beds, silica, gel, and alumina). These media are then treated as solid waste.
Burial of filters in high-integrity containers with concrete overpacks.	Filtration	
Demineralization via ion-exchange resins as an alternative to evaporation and solidification. Resins are then dewatered and placed in high-integrity containers with concrete overpacks.	Holdup lines or tanks to permit decay of short-lived isotopes.	

The first priority should be the partial or total elimination of the source of the waste stream. If waste cannot be eliminated, it should be recycled in an environmentally safe manner. If elimination or recycling is not feasible, the waste should be treated to reduce its hazard and volume. Finally, a disposal method should be selected that minimizes cost and dose.

All levels of management must support the waste minimization program for it to be effective. Definitive and realistic goals must be established. Employee awareness, incentives and training are essential to ensure that each individual involved in the waste minimization program is an active participant.

Table 5.8 provides examples of decontamination technologies and their potential applications. The use of any of these technologies must incorporate waste minimization techniques when it is applied to a specific facility. Each technique will have positive and negative attributes that must be reviewed in terms of the facility-specific operations and configuration.

Table 5.8 Decontamination Techniques and Principle Uses.

Decontamination Technology	Description/Principle Uses
Application of chemical gels containing sodium hydroxide	Used to decontaminate piping interiors.
Stabilization with various agents	Applied as coatings on contaminated residue. Coatings may be peelable.
Stripping, contaminant-specific removal or treatment	Removal of coatings containing radioactive material or toxic species.
Ultrasonic cleaning	Cleaning nonporous surfaces or components having non-fixed contamination.
Low-pressure wet treatment	This is the most frequently used decontamination method It entails flushing surfaces with water, usually hot water. Detergents or other additives may be used to enhance effectiveness.
High-pressure washing (hydroblasting)	Used for stripping and decontamination of large areas. Can be used to remove coatings or layers of concrete.
Steam cleaning	Can be used to decontaminate complex shapes.
Sponge blasting (cleaning surfaces by blasting with sponges made from various grades of foam-cleaning media such as water-based urethane)	On contact with the surface (e.g., a condenser tube), the sponges expand and contract creating a scrubbing effect.
Carbon dioxide blasting	Method uses solids that vaporize during the cleaning process.
Wet grit blasting	A liquid abrasive decontamination method using water, abrasive material and compressed air. Adopted by many facilities for removing smearable and fixed contamination from metal surfaces such as structural steel, metal components, and hand tools.
Grit blasting (sand blasting or abrasive jetting)	Well-established method for the uniform removal of surface contamination. This technique uses abrasive materials (e.g., minerals, steel pellets, or glass beads) propelled by water, compressed air, or another mode of force to impact the surface being decontaminated.
Needle scaling	Used for both concrete and steel surfaces. Uniform sets of 2, 3, or 4 mm copper beryllium needles are used in a reciprocating action to chip contamination from a surface.

Table 5.8 (continued)

Decontamination Technology	Description/Principle Uses
Milling	Shaves layers of material of varying thicknesses from metal or concrete. In metal milling, a machine shaves off a layer of material (up to 0.32 cm) from a surface using rotating cutters. Concrete milling is similar to concrete scabbling or scarifying except that it may be applied to a much larger surface area.
Scarification	Removes the outer layers of a contaminated surface down to the depth of sound, uncontaminated material.
Scabblers	Utilize pneumatically operated piston heads designed for simultaneous strikes against a concrete surface. Most effective for removing thin layers, up to 2.5 cm thick, of contaminated concrete and cement.
Dry application ice	Used primarily for the removal of contaminated floor tiles.
Drilling and spalling	Used to remove contaminated concrete surfaces without demolishing the entire structure. This technique entails drilling 3.8 cm diameter holes approximately 7.6 cm deep and then inserting a hydraulic spalling tool.
Sorting	Nonradioactive solid wastes are separated from low level radioactive waste (LLRW) and sorted into different categories to facilitate volume reduction.
Dewatering	Water is removed from rinsates and other wastes by means of centrifuges and filtration systems. Also used for volume reduction of resins prior to shipment for off-site disposal.
Catalytic extraction	Toxic waste and LLRW are broken down into recyclable and reusable components in a molten metal bath.
Biological treatment	Contaminants are removed or stabilized by microbially mediated processes. These methods include activated sludge technology, acid leaching, extended aeration, contact stabilization, pure oxygen aeration, use of trickling filters, and rotating biological disks.

Source: NCRP-143 (2003).

5.6 Outages

Outages occur at commercial power reactors for scheduled refueling and maintenance and for unscheduled maintenance. Scheduled outages occur at a frequency of between 12 and 24 months depending upon the core enrichment and the philosophy of the operating utility.

In both PWRs and BWRs, the primary sources of radiation exposure during scheduled outages are activation products derived from corrosion and wear products. Co-58 and Co-60 are significant contributors to the radiation exposure. PWR outage doses are dominated by steam generator work (eddy current testing of generator tubes, tube plugging, and the installation/removal of nozzle dams), reactor coolant pump maintenance, refueling cavity work, and reactor vessel head installation/removal and associated activities. Other outage activities include valve and pump repair and maintenance, fuel transfer system maintenance, plant modifications, control rod drive repairs, in-core instrument replacement, and radioactive waste processing and packaging.

BWR outage doses are spread over similar activities, excluding steam generator work. In addition, BWR exposures include steam turbine and condenser work.

5.7 Major Radiation Instrumentation Systems

Since two out of three Generation II reactors are PWRs, the radiation instrumentation focus is on these reactor systems. Radiation instrumentation monitors the radioactive material in the primary and secondary systems. These instrumentation systems provide real-time radiation data regarding the integrity of major systems and are used to evaluate off-normal events including accident conditions. PWR radiation instrumentation systems are also designed to monitor the fuel and primary coolant system piping fission product barriers.

5.7.1 Primary System Monitors

The primary system is monitored using containment instrumentation including direct radiation and air monitors. Containment instrumentation provides information regarding the integrity of the primary system and the integrity of the fuel cladding. Any fuel cladding damage leads to an increase in primary coolant activity. This increase in activity results in increased radiation levels. Since the primary system has nominal leakage, the air monitors record radioactive gas that leaks from the primary system.

Any fuel damage increases the primary coolant activity and that increase is detected by the letdown monitoring system. The letdown system removes a portion of the primary coolant (120–400 L/min) for radiation monitoring and

subsequent removal of radioactive material through demineralizers, gas strippers, and filters.

In addition to fuel damage, increased letdown radiation levels arise from activation products that dislodge from the primary system. Reactor transients such as rapid power changes and reactor trips can dislodge trapped activation products.

Loss of coolant accidents also lead to fuel damage and are detected with the letdown monitors. A loss of coolant event is detected by coincident increases in the containment and letdown monitors.

If primary coolant activity increases, it is important to quickly determine if the increase is attributed to corrosion products or fuel damage. A fuel damage event leads to an increase in primary coolant fission gas activity. Reactor coolant samples are also analyzed to ascertain the isotopic nature of the increase in letdown radiation levels. A fuel-damage event will have elevated fission products, which are not present following a release of activated corrosion products.

5.7.2 Secondary System Monitors

Secondary system radiation monitors are used to detect increases in secondary coolant activity. Since the primary system pressure is about twice the secondary system pressure, any damage to steam generator tubes leads to leakage of the contaminated primary coolant into the clean secondary coolant.

Secondary system monitors detect an increase in radiation levels following damage to steam generator tubes. The three major secondary system monitors are the blowdown monitors, steam-line monitors, and condenser air-ejector monitors.

The blowdown monitors detect any activity in the steam-generator effluent. Blowdown is steam generator water that is discharged to flush chemical contaminants from the generator to minimize tube corrosion.

Steam-line monitors detect N-16 activity. The N-16 is created from the O-16(n, p) N-16 reaction that occurs in the primary system. N-16 is transferred to the secondary system during primary-to-secondary leakage.

The final secondary system monitor is the condenser air ejector. Air ejectors are eductors that remove noncondensable gases from the condenser section of low-pressure turbines. The air-ejector effluent passes through a radiation monitor that detects noble gas activity.

Primary-to-secondary leakage is confirmed using the three secondary system monitors. Normally, multiple secondary system monitors detect an increase in activity or radiation levels. A steam generator tube leak event normally exhibits a coincident increase in blowdown, air-ejector, and steam-line radiation monitors.

5.8 Radiological Considerations During Reactor Accidents

Reactor accidents may take a variety of forms, but the most radiologically significant events will involve core damage that could lead to the potential of radioactive releases to plant areas and to the environment. Other events (namely, failure of waste-gas decay tanks or spent-fuel element breaches) are less severe, but more likely, scenarios. Reactor accidents are classified into broad categories. These accidents vary in their severity and are included in the design basis of a commercial nuclear power reactor. The accident categories include:

1. *Loss of Coolant Accidents (LOCAs)*. In a LOCA, the reactor cooling water is reduced or lost and the nuclear fuel begins to heat up. The event may be caused by a piping rupture, seal failure, instrument line failure, and/or leaks in piping, valves, or components. If the LOCA is severe, the fuel will eventually melt. Fuel cladding degradation may occur even without fuel melting. Breaches in the clad will release fission radionuclides into the reactor coolant. Subsequent breaches in the RCS or containment building will offer a release path to the environment.
2. *Steam Generator Tube Ruptures (SGTRs)*. In a PWR, the steam generator tubes form a barrier between the primary and secondary coolants. If a tube rupture or leak occurs, a pathway is created that mixes the primary (radioactive) and secondary (nonradioactive) fluids. As a minimum the secondary (clean) part of the plant will be become contaminated and its radiation levels will show a significant increase. Failures of atmospheric or steam generator safety values or of other secondary system piping, valves, or components, provide a direct release pathway to the environment.
3. *Fuel-Handling Accidents (FHAs)*. The nuclear fuel residing in the spent-fuel pool contains fission products that have been decaying since the fuel assembly was removed from the core. These fuel elements are periodically moved within the pool, and accidents during these evolutions can damage the clad and lead to a release of radionuclides into the radiologically controlled plant areas.
4. *Waste Gas Decay Tank Ruptures (WGDTRs)*. Gas decay tanks store fission gases and permit their decay prior to release to the environment. Failures of the tank structure, valves, or associated components will release fission gases into the plant.

The extent to which these accidents lead to radiological consequences depends largely on the condition of the reactor core. If the fuel cladding remains intact, the releases will be characterized by the steady-state activity of the primary coolant. The radiological hazards increase proportionally with the degree to which the fuel cladding degrades and releases fission products into the primary coolant.

In each of these events, radioactive gas can be released to plant areas and then to the environment. The gamma dose rate from nuclide K in the radioactive gas cloud is often assessed using the semi-infinite cloud model:

$$\dot{D}_K^{\gamma} = k\,\overline{E}_K^{\gamma}\,X_K(r,t) \tag{5.6}$$

where

$\dot{D}_K^{\gamma}$ = dose rate (rad/s) from nuclide K in the cloud

k = conversion factor for a semi-infinite cloud

$$= \frac{1\text{ rad}}{100\text{ erg/g}}\,\frac{1.6\times10^{-6}\text{ erg}}{\text{MeV}}\times\frac{3.7\times10^{10}\text{ dis/s}}{\text{Ci}}\times\frac{\text{m}^3}{1293\text{ g}}\times 0.5$$

$$= 0.23\,\frac{\text{rad}}{\text{s}}\,\frac{\text{dis}}{\text{MeV}}\,\frac{\text{m}^3}{\text{Ci}}$$

$\overline{E}_K^{\gamma}$ = average gamma energy per disintegration for nuclide K (MeV/dis)

$X_K(r, t)$ = air concentration of nuclide K (Ci/m^3) at a distance r from the release point and at time t

It is often more convenient to measure the source term at the plant rather than at the receptor location. This is particularly true in the early stages of an accident when field measurements are not available. The air concentration can be related to the source term:

$$X_K(r,t) = Q_K\,\frac{X}{Q} \tag{5.7}$$

where

Q_K = source term or release rate for nuclide K (Ci/s)

X/Q = atmospheric dispersion parameter (s/m^3)

With this relationship, the dose rate equation becomes

$$\dot{D}_K^{\gamma} = k\,\overline{E}_K^{\gamma}\,Q_K\,\frac{X}{Q} = Q_K\,\text{DRCF}_K\,\frac{X}{Q} \tag{5.8}$$

where the dose rate conversion factor (DRCF_K) is

$$\text{DRCF}_K = k\,E_K^{\gamma} \tag{5.9}$$

If more than a single isotope is released, the total dose rate is the sum of the dose rates from the individual radionuclides:

$$\dot{D}^{\gamma} = \sum_K \dot{D}_K^{\gamma} = \sum_K Q_K\,\text{DRCF}_K\,\frac{X}{Q} \tag{5.10}$$

The semi-infinite cloud model assumes that the release rate is constant and that the atmospheric conditions, as described by the dispersion parameter, are also constant. Accident events are not likely to meet either of these conditions for extended periods of time. Another assumption is that the plume dimensions are large compared with the distance which the gamma rays travel in air. This

assumption is not valid close to the source, but it is more easily achieved further from the source.

Additional model shortcomings include exclusion of the air attenuation of gamma rays. The model also does not account for radiation buildup factors caused by the Compton scattering of the gamma-ray photons.

These shortcomings are easily overcome with computer codes that include these factors and perform the requisite numerical integration. However, knowledge of the semi-infinite cloud models is invaluable for quick assessments during the initial stages of accident conditions.

This general formulation also applies to thyroid, bone, and other organ doses. The dose-conversion factors will change to reflect the type of dose calculation being performed.

5.9 Mitigation of Accident Consequences

The health physicist must maintain worker and public radiation exposures ALARA during an accident. This includes critical reviews of plant repair activities to terminate the radiological release. The use of plant safety systems should also be considered. These safety systems include the use of sodium hydroxide spray within containment to reduce the iodine source term, the use of emergency core cooling systems (ECCS) to cool the core, and the use of filters to reduce the particulate and iodine source terms.

Table 5.9 summarizes the various power reactor accident types, the types of radiological releases that could occur, plant systems that could be used to mitigate the release, and methods that could be utilized to mitigate the release.

Table 5.9 Reactor Accident Mitigation.

Accident Type	Release Type	Mitigation	Termination
LOCA	Iodine Noble gas Particulate	NaOH spray Filtration ECCS	In-plant repairs Re-establish core cooling
SGTR	Iodine Noble gas	Filtration Release via condenser ECCS	Cool and depressurize the RCS In-plant repairs
FHA (< 1-year-old fuel)	Iodine Noble gas	Filtration	Fuel assembly depressurization
FHA (> 1-year-old fuel)	Noble gas	Filtration	Fuel assembly depressurization
WGDTR	Noble gas Iodine (if fuel defects)	Filtration	Tank depressurization In-plant repairs

5.10 Scenarios

Scenario 5.1

A pressurized water reactor demineralizer is loaded with resin to reduce the activity of Co-60 in the primary coolant. During the current 200-day cycle, the demineralizer processed reactor coolant. Specific operational characteristics of the demineralizer are provided below.

The following data may be useful:

The demineralizer is a vertical cylindrical vessel that is 4 ft in diameter and 12 ft high.

Its wall thickness is $\frac{3}{8}$ in., and the demineralizer is made of stainless steel. The demineralization bed is 4 ft in diameter.

Assume that the dose rate from the resin bed can be approximated by a thin disk source which is described by the equation

$$\dot{D}(h) = 3.14 \times G \times C \times \ln \frac{h^2 + R^2}{h^2}$$

C = source strength in Ci/m^2

Attenuation Coefficients (Lead).

***E* (MeV)**	0.6	0.7	0.8	1.0	1.25	1.50	2.75
***u* (1/cm)**	1.36	1.12	0.97	0.78	0.65	0.58	0.47

Demineralizer and Radiation Characteristics.

Nuclide	**Half-life**	**Radiation Energy (MeV)**	**Percent Yield**	**RCS Activity (μCi/cm^3)**
Co-60	5.26 years	1.173 (gamma)	100.0	6.0×10^{-4}
		1.332 (gamma)	100.0	
		1.480 (beta)	0.12	
		0.314 (beta)	99.0	

Dose Buildup Factors for a Point Isotropic Source.

Material	MeV	ux[a]						
		1	2	4	7	10	15	20
Lead	0.5	1.24	1.42	1.69	2.00	2.27	2.65	2.73
	1.0	1.37	1.69	2.26	3.02	3.74	4.81	5.86
	2.0	1.39	1.76	2.51	3.66	4.84	6.87	9.00
	3.0	1.34	1.68	2.43	2.75	5.30	8.44	12.3

[a] $u\,x$ = mass attenuation coefficient (u/density) × shield thickness (cm) × shield density (g/cm^3).

Other Data

Flow rate through the demineralizer: 350 liters/min

Demineralizer efficiency for Co-60 removal: 100.0%

Co-60 specific gamma-ray emission (gamma constant):

$$G = 1.3\,\frac{\text{R-m}^2}{\text{hr-Ci}}$$

1 R = 87.7 erg/gram in air.

Assume the dose (rads) to air equals the dose (rads) to tissue.

5.1 Calculate the gamma dose equivalent rate in rem/hr at a point 1 ft above the centerline of the demineralizer bed immediately after the 200-day run. For the purposes of this problem, assume that the demineralizer contains no water above the resin bed at the end of the run and that all activity is distributed uniformly over the top surface of the demineralizer bed.

5.2 A valve located 25 ft above the demineralizer bed required repair. When isolated, the demineralizer was loaded with 80.0 Ci of Co-60. Assume that the Co-60 is uniformly distributed over the top surface of the bed. The repair must be performed by a pipe fitter whose remaining annual dose equivalent is 300 mrem. The valve repair will take 3 hr. The only shielding available is a previously installed lead mat which is composed of 2 in. of equivalent lead. Will the installed shielding keep the worker below his annual limit? State all assumptions and calculate the total dose equivalent in rem received by the worker. Assume that the repair occurs 6 months after isolation, that there is no water above the resin bed, and that the valve resides on the demineralizer centerline. For simplicity, only consider the gamma-dose contribution.

Scenario 5.2

As a health physicist at a power reactor, you have been assigned the lead role in evaluating hot-particle contamination of station personnel. Hot particles are very small highly radioactive particles with high specific activity. These particles have recently been detected with increasing frequency at your facility. Isotopic analyses indicate that the particles are composed of fuel or neutron-activated corrosion and wear products.

On Tuesday morning, you receive a phone call about a hot particle which was removed from under a pipe-fitter's fingernail. Upon reporting for work, the pipe fitter alarmed the entrance portal contamination monitor. The contamination was detected during a follow-up frisk, and it was successfully removed on the first attempt using sticky tape. Although not visible to the naked eye, the particle was analyzed for its constituent radionuclides.

Data

An analysis of the contamination on the sticky tape led to the following results:

Radionuclide	Activity (μCi)	Dose Factor[a] (rad-cm^2/μCi-hr)
Co-60	3.9×10^{-1}	4.13
Zn-65	1.0×10^{-6} (LLD)	0.106
Zr-95	1.0×10^{-6} (LLD)	4.87
Nb-95	1.0×10^{-6} (LLD)	0.865

[a] Beta dose factors applicable for skin dose calculations averaged over 10 cm^2.

Based upon a review of the event, you determine the following:

Time of contamination: 3:45 p.m. on Monday
Time of removal: 7:30 a.m. on Tuesday (following day)
Source of contamination: Unknown, although the worker's protective clothing is strongly suggested.

Using the above data and your experience, choose the single best answer to the following questions:

5.3 In general, for a hot particle on the skin, the principal contributor to the dose to the skin in the vicinity of the particle is:
   a. The neutron radiation emitted from the particle.
   b. The gamma radiation emitted from the particle.
   c. The alpha radiation emitted from the particle.
   d. The conversion electron emitted from the particle.
   e. The beta radiation emitted from the particle.

5.4 The most plausible explanation for the increasing frequency of detection of hot particles at nuclear power plants is:
   a. The increased use of more sensitive instrumentation for detecting and measuring contamination by these particles.
   b. An increase in the rate of production of these particles at nuclear power plants.
   c. The changes in plant chemistry which have enhanced fuel reliability but have increased the corrosion rate of other components.
   d. A decrease in the average corrosion particle size as plants have aged.
   e. The trend toward increasing the time between refueling outages with a subsequent decrease in preventive maintenance.

5.5 Prior to 1988, explicit recommendations on limits for radiation exposure of skin by hot particles were provided by:
   a. The ICRP, but not the NCRP.
   b. Both the NCRP and the ICRP.
   c. The NCRP but not the ICRP.
   d. Neither the NCRP nor the ICRP.
   e. The ICRU, NCRP, and ICRP.

5.6 Radiobiological evidence suggests that, when compared to more uniform irradiation by the same quantity of radioactive material, highly localized beta irradiation of skin, such as from a particle on the skin, is:
   a. Less likely to cause skin cancer.
   b. More likely to cause skin cancer.
   c. About equally likely to cause skin cancer.
   d. Likely to cause an erythema within a few hours.
   e. Likely to cause a small necrotic lesion on the skin of an individual after only a few hours of exposure.

5.7 For a typical beta–gamma survey meter with a 5-cm pancake probe, what approximate instrument efficiency is appropriate for a 2.0-cm-diameter stainless steel disk of Sr-90 if the source detector distance is 1 cm?
   a. 1%
   b. 5%
   c. 35%
   d. 50%
   e. 65%

5.8 The beta dose equivalent to the skin of the worker's finger is:
   a. 1.6 rem
   b. 2.0 rem
   c. 2.5 rem
   d. 25 rem
   e. 33 000 rem

5.9 For this scenario, assume that the calculated dose for regulatory purposes, beta and gamma, was 60 rem. Which one of the following statements best describes compliance with U.S. Nuclear Regulatory Commission regulations?

a. No limit was exceeded. All tissues received less than the yearly allowable regulatory limit of 75 rem for the extremities.
b. The whole-body exposure was less than the quarterly allowable regulatory limit of 3 rem.
c. The skin of the whole-body limit of 15 rem was exceeded.
d. The dose limit of 50 rem for the extremities was exceeded.
e. The annual whole-body dose limit of 5 rem was exceeded.

5.10 Which one of the listed follow-up actions would not be appropriate if you calculated a 60-rem total dose for this scenario?
a. Survey of the worker's home, car, and girlfriend's home and of the local restaurant where he ate the previous evening.
b. Review, recalibrate, and evaluate the station's laundry monitoring systems.
c. Interview the worker to advise him of his rights under federal law and to answer the worker's questions regarding biological effects from his exposure.
d. Evaluate the sensitivity of portal monitors and other contamination detection instrumentation.
e. Initiate thyroid monitoring and increased urinalysis frequency.

5.11 Which one of the following detectors would provide the best sensitivity for detecting these particles at the exit station from the plant?
a. An energy compensated G-M probe.
b. Air proportional detector.
c. Zinc sulfide scintillation probe.
d. Intrinsic germanium crystal with single channel analyzer.
e. Ionization chamber.

5.12 In 1989, NCRP-106 established the following limit for hot particles:
a. 40 μCi-hr
b. 75 μCi-hr
c. 150 μCi-hr
d. 250 μCi-hr
e. 500 μCi-hr

Scenario 5.3

You are a member of the professional health physics staff at a nuclear power plant and have been asked to prepare a briefing for management, on occupational radiation exposure in the nuclear power industry. The briefing is to include recent trends in collective doses, means of dose reduction and maintaining doses ALARA, and potential effects of changing regulatory requirements. Your briefing will recognize that, in the United States, collective occupational exposure (person-rem per reactor per year) increased to a maximum in 1980. Since then, the general trend has been down. However, when compared to other countries, the collective dose in the United States is still among the highest, indicating a substantial potential for improvement.

In addition to these facts, your briefing will include points covered by the following questions:

5.13 All of the following factors, with one exception, have contributed significantly to the downward trend in annual occupational exposure per reactor in the United States since 1980. That exception is:
   a. Replacement of highly contaminated major components of the reactor system.
   b. Completion of NRC-mandated safety actions (such as fire protection, seismic upgrading, etc.).
   c. Industry actions including ALARA programs.
   d. New plants going into service.
   e. Higher capacity factors for U.S. plants.

5.14 In general, the most important source of occupational radiation exposure at nuclear power plants is:
   a. Gamma radiation from the core.
   b. Long-lived fission product activity.
   c. Neutron activation product activity.
   d. Short-lived fission product activity.
   e. Neutron radiation from the core.

5.15. The radionuclide that is the main cause of shutdown radiation fields at nuclear power plants is:
   a. N-16
   b. Cs-137
   c. Co-58
   d. Co-60
   e. Xe-135

5.16 Comparative assessment of U.S. and foreign nuclear power plant occupational dose experience has indicated that reductions in out-of-core radiation fields can result in substantial reductions in occupational doses. Of the following actions, the one that is the least effective in reducing out-of-core radiation fields is:
   a. Hydrogen water chemistry.
   b. Control of impurities in the reactor coolant.
   c. Preconditioning of out-of-core surfaces (polishing, prefilming).
   d. Reduction or elimination of cobalt in reactor system components.
   e. Chemical decontamination.

5.17 Studies of radiation exposure incidents have identified six radiation protection errors that are common to most incidents of unplanned, unnecessary radiation exposures at nuclear power plants. One error that is not included among these six errors is:
   a. Inaccurate or incomplete radiation surveys.
   b. Inadequately prepared radiological work permits.
   c. Failure of the radiological protection technician to react to changing or unusual conditions.

d. Failure of workers to follow procedures or good radiological work practices.
e. Lack of involvement on the part of supervisors or foremen.
f. A neutral or negative attitude toward radiological protection on the part of the plant operators and overall plant management.
g. Lack of knowledge of NRC regulations by the radiation protection technician.

5.18 All of the following, with one exception, have been reported to be useful in monitoring the performance of the radiation protection program at a nuclear power plant. That exception is:
a. Number of Certified Health Physicists on the staff.
b. Collective radiation exposure.
c. Skin/clothing contamination events.
d. Solid radwaste volumes.
e. Radiological incident reports.

5.19 If personnel with doses less than 100 mrem/year are excluded, the average annual exposure for nuclear reactor workers is typically:
a. 400–800 mrem.
b. 750–1500 mrem.
c. 1200–2400 mrem.
d. 2000–3000 mrem.
e. 2700–4000 mrem.

5.20 Which of the following requirements of the January 1, 1993 revision of 10 CFR Part 20 is least likely to result in a significant change in the radiation protection programs at nuclear power plants?
a. Imposition of a skin dose limit of 50 rem/year.
b. Evaluation of the skin dose at a depth of 7 mg/cm^2 and evaluation of the eye lens dose at a depth of 300 mg/cm^2.
c. Imposition of an eye (lens) dose limit of 15 rem/year.
d. Imposition of a dose limit for the embryo/fetus of 0.5 rem during pregnancy.
e. Changes in the air concentration limits.

5.21 In which of the following situations will the 10 CFR20 dose limit for protection of the embryo/fetus apply?
a. A female worker is known to her employer to be pregnant.
b. A female worker is suspected by her employer of being pregnant.
c. A female worker has voluntarily made her pregnancy known to her employer.
d. It would apply to all female workers of child-bearing age.
e. It would apply to all female workers.

5.22 As a result of re-evaluations of the radiation doses received by the Japanese atomic bomb survivors, the ICRP and other authoritative advisory groups revised their recommendations on occupational dose limits. The recommended ICRP annual dose limit is:

a. 0.5 rem
b. 2.0 rem
c. 3.0 rem
d. 4.0 rem
e. 6.0 rem

Scenario 5.4

As the station health physicist, you are assigned the task of performing the dose-rate analysis for an upcoming outage in which maintenance work is planned on a primary coolant sampling system. Prior to the maintenance work, the primary system will be decontaminated. In support of this effort, you are asked to provide exposure-rate estimates for a number of source configurations. In calculating exposure rates, assume that an individual can be positioned at points *P*, *Q*, and *X*. Ignore any shielding provided by the individual.

Data

The scrap material was originally 100% Co-59 and weighed 10 g (question 5.23).

The Co-59(n, γ)Co-60 cross-section is 37 barns.

The Co-60 half-life is 5.27 years.

Air is the medium between the sources and points *P*, *Q*, and *X*.

The activity of the sources in questions 5.24, 5.25, and 5.26 are each 3.0 Ci, and the activity is uniformly distributed within each source.

Each source is Co-60 which has a specific gamma-ray emission or gamma constant (G):

$$G = 1.3\,\frac{\text{R-m}^2}{\text{hr-Ci}}$$

Neglect gamma-ray self-shielding within the source regions and in containing piping.

Avogadro's number = 6.02×10^{23} atoms per gram atomic weight.

5.23 Six months ago, a small scrap of material was removed from the reactor vessel. Plant records indicate that it had been irradiated for 10 years. The material is natural cobalt and was subjected to an average thermal neutron fluence rate (flux) of 1.0×10^{10} n/cm^2-s and a fast neutron fluence rate (flux) of 5.0×10^{10} n/cm^2-s. What exposure rate in R/hr is expected at a point that lies 2.0 m from the material? The scrap's dimensions are 2 cm × 3 cm × 0.2 cm.

5.24 Calculate the exposure rate in R/hr at point *P*, 2.0 m from a spherical particle of Co-60 that has a radius of 0.5 mm. The particle's activity is 3.0 Ci. The geometry for the exposure assessment is illustrated in Figure 5.1.

5.25 Calculate the exposure rate in R/hr at point *Q*. Point *Q* is a distance of 2.0 m from the end of a sample line containing a uniform distribution of Co-60. The sample line, illustrated in Figure 5.2, is 10.0 m long and has a diameter of 0.5 cm. The total activity contained within the sample line is 3.0 Ci.

5.26 Calculate the exposure rate in R/hr at point *X* defined in Figure 5.3. Point *X* is 2.0 m above the center of the spill which contains a uniform distribution of Co-60. The spill is in the shape of a thin disk (0.2 cm thick) which has a diameter of 20.0 m. The total spill activity is 3.0 Ci.

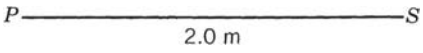


Figure 5.1 Point-source geometry for computing the exposure rate. *P* is the point of interest for calculating the exposure rate, and *S* is the small particle source.

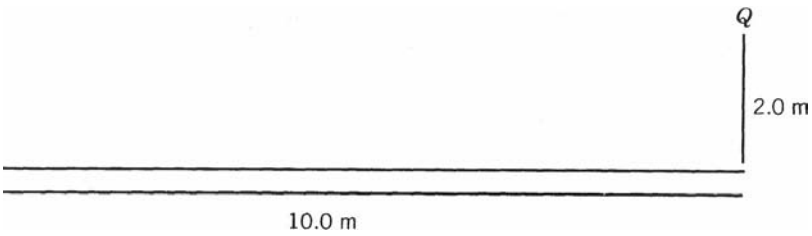


Figure 5.2 Line-source geometry for computing the exposure rate. The sample line is 10.0 m long. Point *Q* is located above the end of the sample line at a perpendicular distance of 2.0 m.

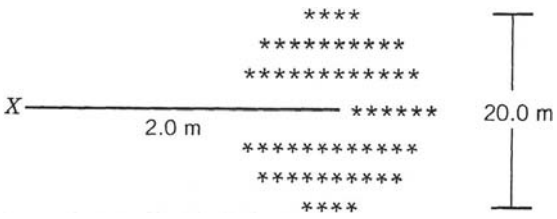


Figure 5.3 Disk source geometry for computing the exposure rate. The spill has a diameter of 20.0 m. Point *X* is 2.0 m above the spill along the central axis of the disk.

Scenario 5.5

You have been requested to perform an ALARA evaluation for the first spent-fuel pool cleanup campaign. The facility is 10 years old, and a variety of materials have accumulated. The irradiated hardware inventory consists of 24 control rod blades that were removed during the last refueling outage, 96 incore neutron flux detectors removed during various refueling outages, and six Sb(Be) startup neutron sources that were removed during the first refueling outage. Other materials that were left in the pool include 18 pleated paper underwater vacuum cleaner filters that are approximately five years old.

The processing of irradiated hardware will require the use of several tools. For each of the following tools, list and briefly discuss one potential radiological problem and possible engineering solution.

5.27 Underwater manipulator pole, consisting of one hollow 30-ft. length that can be used with a variety of small tools such as hooks and grapples.

5.28 Hydraulic cutter used to cut neutron and startup sources into approximately 6-in. lengths. The processing vendor predicts that the cutter jaws will require replacement after each 100–150 cuts.

5.29 Temporary jib crane set up to lower control rods into a crusher/shearer assembly.

Figures 5.4 and 5.5 illustrate the configuration of the installed area radiation monitor in the spent-fuel pool area. The pool is 30 ft × 80 ft and the detector is at a height of 6 ft above the floor on a wall 20 ft from the edge of the pool. The normal spent-fuel pool water level is 18 in. below the floor, and the railing is at a height of 42 in. above the floor.

5.30 What is the minimum line-of-sight distance from the edge of the pool that a source on the surface of the water could be detected by the installed area radiation monitor (ARM) without being shielded by the refueling floor.

5.31 The alarm setpoint of the installed ARM is to be adjusted so that it will alarm if a point radiation source at the center of the pool produces an exposure rate of 1 R/hr at the top of the pool railing. What is the appropriate ARM setpoint?

A major task in the spent-fuel pool cleanup is the radiological characterization of neutron-activated hardware. Two characterization techniques are available.

The first of these involves collecting actual specimens of the activated hardware for radiochemical analysis. This technique requires the use of a small abrasive grinding wheel, glass fiber filters, and a small pump to impinge samples on the filter media.

The second characterization technique involves neutron activation calculations and requires underwater surveys to be performed by health physics personnel. This technique requires the use of an underwater survey instrument.

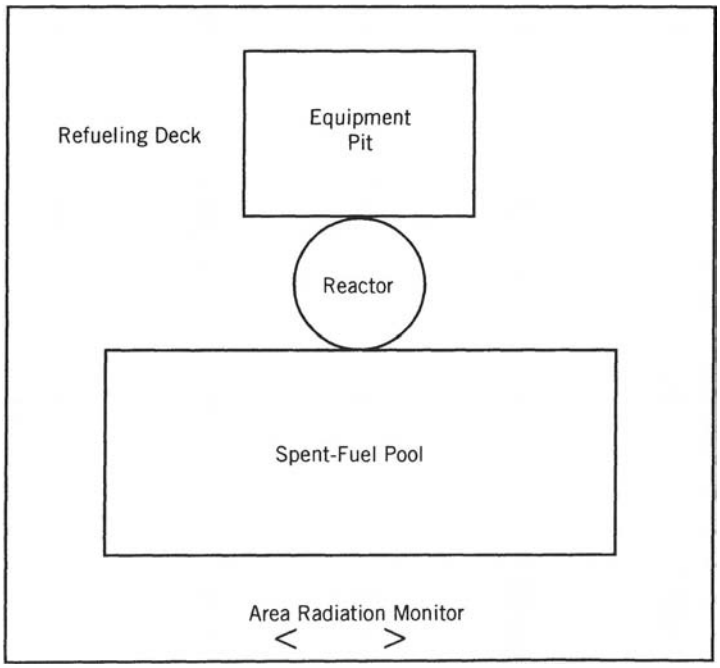


Figure 5.4 Refueling deck floor plan illustrating the location of the reactor cavity, equipment pit, the spent-fuel pool, and the area radiation monitor.

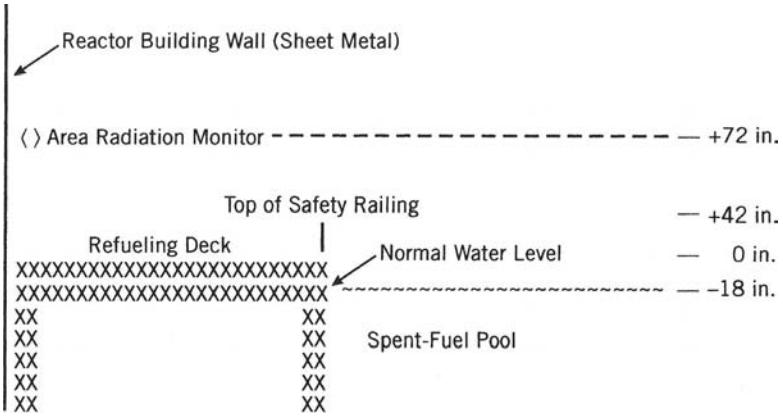


Figure 5.5 Cross-section of the refueling deck illustrating the elevation of the area radiation monitor, spent fuel pool water level, and personnel safety railing relative to the top of the refueling deck.

5.32 For each characterization technique, state and explain one advantage and disadvantage from an overall health physics perspective.
5.33 For each characterization technique, state the major source of uncertainty. Clearly identify any assumptions that you make.

The underwater vacuum cleaner filters need to be removed from the pool for disposal. This task involves moving the vacuum cleaner filters underwater from a storage location to a disposal liner at the opposite corner of the pool.

5.34 State the principal hazard associated with movement of the filters and two major radiological consequences associated with this evolution.
5.35 For each hazard cited above, describe two appropriate radiological controls to maintain personnel doses ALARA.

Scenario 5.6

You are a health physicist at a nuclear utility operating a single pressurized water reactor. The utility has assigned you the responsibilities of the Radiological Control Manager (RCM) at the off-site Emergency Operations Facility (EOF) during a declared emergency in which a radioactive release to the environment is possible.

Data

Applicable parameters prior to the event:

Letdown radiation monitor reading (primary reactor coolant system activity):	0.05 μCi/cm^3
Blowdown radiation monitor reading (secondary system activity):	5×10^{-5} μCi/cm^3
Steam generator "A" radiation monitor reading:	< 0.1 mR/hr
Atmospheric relief valve flow rate:	< 100 cm^3/s
Containment pressure:	0.1 psig

Applicable parameters following the event:

Letdown radiation monitor reading (primary reactor coolant system activity):	7.3×10^2 μCi/cm^3
Blowdown radiation monitor reading (secondary system activity):	2.4 μCi/cm^3
Steam generator "A" radiation monitor reading:	42 mR/hr

Iodine partitioning factor:	0.015
Atmospheric relief valve flow rate:	1.4×10^7 cm^3/s
Containment Pressure:	0.1 psig
Wind Speed:	15 mph
Pasquill stability class:	E
I-131 dose conversion factor:	(77.2 rem/s)/(μCi/cm^3)
1 mile = 1609.36 m	

Steam Generator "A" Blowdown Sample Isotopic Results.

Radionuclide	Concentration (μCi/cm^3)
I-131	6.3×10^{-1}
Xe-133	9.1×10^{-1}
Xe-135	9.3×10^{-1}
Cs-134	7.6×10^{-1}
Cs-137	1.4×10^{-1}

Pasquill Class E Atmospheric Dispersion Factors.

Distance (miles)	Xu/Q (1/m^2)
1	1.57×10^{-2}
2	2.69×10^{-3}
5	1.56×10^{-3}
10	6.19×10^{-4}

5.36 List three of the primary responsibilities of the RCM during activation of the EOF.

5.37 List the three fission product barriers that protect the public from a release of radioactivity.

5.38 List and define the three categories of fission product barrier status used in determining off-site protective action recommendations.

5.39 Based on the data provided, what is the status of the fission product barriers?

5.40 List four factors that can affect off-site dose calculations during a declared emergency.

5.41 At this time, a release has not occurred, but you have been asked to provide an assessment of off-site doses. What is the projected thyroid dose equivalent rate 2.0 miles downwind from the facility?

5.42 Based upon your projected thyroid dose calculation and information from the plant indicating that the reactor has been stabilized and that a release of radioactivity is no longer likely to occur, describe the Protective Action Recommendation you would recommend to downwind sectors within 2.0 miles of the plant.

5.43 A small town with a population of 2500 is located 2.7 miles downwind from the plant. If an atmospheric relief valve, off the main steam line, were opened for 15 min and then closed, what Protective Action Recommendations would you make?

Scenario 5.7

A particulate air sample is to be used in estimating Kr-88 concentrations in a PWR containment building. The filter paper is counted for a 30-min period to detect the Rb-88 beta particles. The Kr-88 source input has been continuous and the Kr-88 concentration unchanged for 2 days. Assume that the collection of the Kr-88 on the filter paper is negligible.

Given:

$^{88}\text{Kr} \rightarrow {}^{88}\text{Rb} + \beta^-$

Kr-88 half-life = 2.84 hr

Rb-88 half-life = 17.7 min

Net counts in 30 min = 1.50×10^5 counts

Beta efficiency = 0.1 counts/disintegration

Transit time = 10 min
(time from the air sampler being turned off to the start of the count)

Air sample time = 30 min

Sample flow rate = 30 liters/min

Filter retention = 100%

5.44 What is the Rb-88 filter activity at the beginning of the counting interval (μCi)?

5.45 What is the Rb-88 activity concentration?

5.46 What is the Kr-88 activity concentration?

Scenario 5.8

You are responsible for establishing the radiation protection requirements for an upcoming maintenance task at a 1 200 MW_e Generation II pressurized water reactor. The task will be performed in an area of high contamination levels and dose rates. The survey data for the work area are shown below.

Each worker will wear the protective clothing defined in Radiation Work Permit No. 08-170. Relevant data include:

Work area survey data (taken at 30 cm):

Beta dose rate (at 7 mg/cm^2) = 0.60 Sv/hr

Gamma dose rate = 0.10 Sv/hr

Radiation Work Permit No. 08-170 Clothing Requirements:

1 bubble hood	2 pairs of coveralls
1 plastic rain suit	2 pairs of rubber gloves
1 pair of glove liners	2 pairs of plastic booties

Protective Clothing Characteristics.

Protective Clothing[a]	Density Thickness (mg/cm^2)	Provides Protection for:
Bubble Hood	63.2	Head and Neck
Glove Liners	8.6	Hands and Wrists
Rubber Gloves	39.9	Hands and Wrists

[a] Assume that the coveralls and rain suit do not add any extra skin protection to the face or hands.

Fraction of Shallow Dose Equivalent (Beta Only, No Absorber) versus Added Absorber Density Thickness.

Fraction of Shallow Dose Equivalent	Density Thickness of Added Absorber (mg/cm^2)
1.0	0
0.3	28
0.2	40
0.08	63
0.05	80
0.04	95
0.03	104
0.01	148

5.47 Comparing the face and the hands, which part of the worker's body will receive the higher shallow dose equivalent? Explain your answer.
5.48 Calculate the dose equivalent rate to the skin of the face.
5.49 List five deterministic effects of irradiation of the skin.

Scenario 5.9

You are the radiation protection manager at the Two Mile Island Nuclear Plant, a 1000 MWe Generation II pressurized water reactor that has shut down for a refueling and maintenance outage. Since previous outages had significant contamination control issues, the Nuclear Regulatory Commission is very interested in this outage's radiological performance.

Isotopes of interest and air sample information are provided in the following tables.

Derived Air Concentrations (DACs) and Inhalation Classes for Isotopes of Interest.

Nuclide	Class	DAC (Bq/m^3)
Co-58	Y	1.0×10^4
Co-60	Y	5.0×10^2
Cs-137	D	2.0×10^3
Mn-54	W	1.0×10^4
Cr-51	Y	3.0×10^5
Fe-59	W	8.0×10^3

Air Sample Data.

Co-60 ALI (inhalation class Y)	1.0 MBq
Air flow rate	0.85 m^3/min
Air sample collection time	10 min
Count time	1 min
Counter efficiency	0.32 c/d
Total counts on air filter	4500 (beta)

5.50 In preparation for removing a man-way cover from a steam generator, you collect an air sample in the surrounding tent. All isotopes of interest are known to contribute to the airborne activity. Assuming the most conservative DAC, calculate the average airborne concentration in terms of fraction of DAC.

5.51 Station Radiation Protection Procedures call for the steam generator to be depressurized prior to opening. However, as the workers were loosening the man-way cover bolts, a vapor spray was released. The workers were not wearing respirators and all immediately exited the tent. The worker closest to the cover required extensive decontamination. Which of the following methods might be used for effective decontamination after the use of soap and water? Justify your answer.
   a. Detergent.
   b. Diethylenetriaminepentaacetate (DTPA).
   c. Nitrile triacetic acid (NTA).
   d. Potassium permanganate.
   e. Sodium iodide.

5.52 No breathing-zone concentration data are available for the most highly contaminated individual. However, a whole-body count was performed after decontamination was completed approximately 3 hours after the intake. The measured Co-60 activity was 4.07×10^3 Bq. Assume the inhaled material is Class Y. About 63 percent of the intake is expected to remain in the body shortly after the incident. Calculate the dose due to this intake.

5.53 Seventy-two hours later, a follow-up body count is performed on the worker in question 5.52. The activity remaining is only 296 Bq. Assume that there was no external contamination at the time of the first body count. What can account for this reduction?

5.54 An isotopic measurement is performed on an air sample after the incident. The Co-60 concentration is 3.63×10^3 Bq/m^3 and the Cr-51 concentration is 2.66 MBq/m^3. A worker wearing a respirator (PF = 50) will spend 3 hours in the environment. What is the total number of DAC-hr to be assigned to the worker?

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6
Environmental Health Physics

Environmental Health Physics deals with the assessment of relatively low levels of radioactivity. Problems in environmental health physics include (a) the assessment of effluent releases from facilities utilizing or producing radioactive material and (b) radon measurements, dose assessments, and mitigation measures. An important aspect of effluent release calculations involves the characterization of the facility site prior to its commercial operation, the continuing assessment of effluent releases during its operation, and characterization of releases during its closure and decommissioning. Increasingly, these estimates are being tied to population risks.

The environmental health physicist is also concerned with the measurement and characterization of naturally occurring radioactive material (NORM). Radon and thoron are particularly important naturally occurring radioactive materials. The impact of radon in homes and public buildings, calculations of radon exposures, and the mitigation of elevated radon levels are becoming increasingly important aspects of environmental health physics.

6.1
Naturally Occurring Radioactive Material

Natural radiation and naturally occurring radioactive material in the environment provide the principal source of radiation exposure to the general public. For this reason, natural radiation is commonly used as a basis for comparison with man-made exposures received by radiation workers and members of the public.

NCRP Report No. 93, *Ionizing Radiation Exposure of the Population of the United States*, assesses population exposures from natural background, consumer products, and other man-made sources. Among the conclusions reached in NCRP-93, none is as significant as the importance of radiation exposure from radon and its daughters. The average annual exposure to radon in the United States is assessed at a dose equivalent of 2400 mrem to the bronchial epithelium. By means of the lung weighting factor, this tissue dose is converted to an effective dose equivalent of 200 mrem. When added to the annual effective dose equivalent of 100 mrem from other sources [contributed by cosmic (27 mrem), terrestrial

Table 6.1 U-238 Series (Radon).

Nuclide	Half-Life	Dominant Decay Mode
U-238	4.51×10^9 years	Alpha
Th-234	24.1 days	Beta
Pa-234m	1.17 min	Beta
U-234	2.47×10^5 years	Alpha
Th-230	8.0×10^4 years	Alpha
Ra-226	1602 years	Alpha
Rn-222	3.823 days	Alpha
Po-218	3.05 min	Alpha
Pb-214	26.8 min	Beta
Bi-214	19.7 min	Beta
Po-214	164 μs	Alpha
Pb-210	21 years	Beta
Bi-210	5.01 days	Beta
Po-210	138.4 days	Alpha
Pb-206	Stable	–

Table 6.2 Th-232 series (Thoron).

Nuclide		Half-Life	Dominant Decay Mode
Th-232		1.41×10^{10} years	Alpha
Ra-228		6.7 years	Beta
Ac-228		6.13 hr	Beta
Th-228		1.91 years	Alpha
Ra-224		3.64 days	Alpha
Rn-220		55 s	Alpha
Po-216		0.15 s	Alpha
Pb-212		10.64 hr	Beta
Bi-212		60.6 min	a. Alpha (36%)
	a. Tl-208	3.10 min	Beta
	Pb-208	Stable	–
			b. Beta (64%)
	b. Po-212	304 ns	Alpha
	Pb-208	Stable	–

[a] Bi-212 decays by alpha emission (36%): $\text{Bi-212} \xrightarrow{\alpha} \text{Tl-208} \xrightarrow{\beta^-} \text{Pb-208}$
and by beta emission (64%): $\text{Bi-212} \xrightarrow{\beta^-} \text{Po-212} \xrightarrow{\alpha} \text{Pb-208}$.

Table 6.3 U-235 Series (Actinon).

Nuclide	Half-Life	Dominant Decay Mode
U-235	7.1×10^8 years	Alpha
Th-231	25.5 hr	Beta
Pa-231	3.25×10^4 years	Alpha
Ac-227	21.6 years	Beta
Th-227	18.2 days	Alpha
Ra-223	11.43 days	Alpha
Rn-219	4.0 s	Alpha
Po-215	1.78 ms	Alpha
Pb-211	36.1 min	Beta
Bi-211	2.15 min	Alpha
Tl-207	4.79 min	Beta
Pb-207	Stable	–

(28 mrem), and internal radionuclides (39 mrem)], natural radiation from all sources yields a combined annual effective dose equivalent of 300 mrem/year.

Because radon and thoron and their daughters contribute two-thirds of the natural background radiation, it is a significant dose contributor and an important topic in environmental health physics. Tables 6.1, 6.2, and 6.3 describe the radon, thoron, and actinon natural decay series, respectively.

6.2 Radon

Radon is a gaseous radioactive element produced from the decay of radium, thorium, and uranium. Its isotopes are short-lived, with a mean half-life of about 30 min. Radon-222 is an alpha emitter, having a half-life of 3.8 days. Radon is heavier than air and is soluble in water; and although it is chemically inert, it will attach to dust or other particulates dispersed in the air. These properties play an important role in the behavior of radon and influence its impact upon the environment.

Naturally occurring uranium-238 and thorium-232 decay to gaseous radon-222 (radon) and radon-220 (thoron), respectively. Gaseous radon enters the atmosphere and ground water by diffusion through the soil. Radon is present in ground water, granite, pumice, clay, brick, and other construction materials such as concrete made from fly ash and industrial slag.

Radon diffuses into the soil and into a building or structure and will increase in concentration, eventually reaching equilibrium. For a given structure, the

radon concentration will vary significantly with the ventilation rate. For example, structures with low ventilation rates are more likely to exhibit elevated indoor radon air concentrations. For these conditions, increased ventilation will reduce the building's radon levels.

The radiological hazard associated with radon is the inhalation of the element and its alpha-emitting daughters that have collected on dust particles suspended in the air. Radon-inhalation effects have been established from studies of uranium mine workers. Exposure to radon and its decay products at elevated concentrations has resulted in several hundred excess lung cancers among uranium mine workers in the western United States. These statistics indicate that there is a direct correlation between radon exposure and the occurrence of lung cancer. However, excess cancers have only been seen in association with exposure to radon concentrations that are two or three orders of magnitude larger than those found in normal indoor environments. Extrapolating the uranium miner data to the case of the general population is complicated by factors including cigarette smoking and silica dust exposure among the miners.

Radon concentrations are often expressed in terms of picocuries per liter (pCi/liter) of air, where 1 pCi/liter = 37 Bq/m^3. The concentration of radon daughters is often expressed in working level units (WL). A working level is any combination of the short-lived radon daughters in 1 liter of air that has a potential alpha energy release of 1.3×10^5 MeV.

The concentration of radon, in equilibrium with its daughter products, that is equal to 1 WL is 100 pCi/liter. Unfortunately, the short-lived daughters are not necessarily in equilibrium with their radon parent. Conversion of radon concentrations to daughter concentrations utilizes a factor of about 0.7 for uranium mines and 0.4 for homes. Therefore, 1 pCi/liter in a home is about 0.004 WL.

Based on available data, the average radon daughter exposure for homes in the United States is estimated to be about 0.004 WL. Outdoor radon concentrations average about 0.001 WL.

The occupational limit for radon exposure in the United States is 4 WLM/year. The Environmental Protection Agency has established a 0.02-WL limit for homes. In order to assess the health effect in homes, an evaluation of the radon concentration will be required.

A common problem in radon assessments is the assessment of the number of working level months of exposure in 1 year that an individual receives from living in a home with a known annual average potential-alpha-energy working level concentration. The occupancy factor of the home will be a required input parameter. The number of working level months (NWLM) is given by

$$\mathrm{NWLM} = \frac{C_{\mathrm{WL}}\, F_{\mathrm{OCC}}}{k} \tag{6.1}$$

where

NWLM = number of working level months of exposure per year
F_{OCC} = occupancy factor
C_{WL} = working level concentration (WL)

k = conversion factor

$$= \frac{170 \text{ WL-hours}}{\text{WLM}} \frac{1 \text{ day}}{24 \text{ hours}} \frac{1 \text{ year}}{365 \text{ days}}$$

The radon progeny equilibrium factor for the home may be determined from careful measurements. With this value, the average potential alpha-energy air concentration in units of working level for this home can be estimated from the relationship

$$C_{WL} = C(EF)\, k \tag{6.2}$$

C_{WL} = average alpha-energy air concentration (WL)
C = average radon air concentration (pCi/liter)
EF = radon progeny equilibrium factor
k = conversion factor (1 WL/100 pCi/liter)

The effective dose equivalent for a member of this household can be determined if you are given the average indoor radon concentration in WLM. To calculate the effective dose equivalent, the average number of working level months per year must be calculated:

$$\text{AWLM} = (F_{OCC})\,(C_{WL})\, k_1 / k_2 \tag{6.3}$$

where
AWLM = average number of working level months per year (WLM/year)
F_{OCC} = occupancy factor for the home
C_{WL} = average working level concentration in the home
k_1 = time conversion factor (365 days/year × 24 hr/day)
k_2 = 170 hr/month for WLM estimates

The committed dose equivalent rate to the tracheobronchial (TB) region of the lung is obtained from the AWLM:

$$D_L = (DCF_{TB})\,(AWLM)\,(QF) \tag{6.4}$$

where
D_L = committed dose equivalent rate to the TB region of the lung (rem/year)
DCF_{TB} = tracheobronchial dose conversion factor
QF = quality factor for alpha particles (rem/rad)

The committed effective dose equivalent (H) is derived from the ICRP Methodology:

$$H = w_L\, D_L \tag{6.5}$$

where
w_L = weighting factor for the lung

6.2.1 Buildup of Radon from Inleakage

The mitigation of radon in a detached, single-family home with a full basement is a common radon problem. Assuming that radon is removed only by outside air infiltration and radioactive decay, one can estimate the annual average radon source strength or entry rate into the structure (dC/dt) in units of pCi of Rn-222 per hour with the relationship

$$dC/dt = S/V - \lambda C - I C \tag{6.6}$$

where
C = radon concentration (pCi/liter) in the air
t = time
S = radon source strength (pCi/hr) which could be determined by a suitable measurement technique such as an extended alpha-track measurement.
V = free space volume of home (liters)
λ = decay constant for radon (1/hr)
I = outside air infiltration rate (1/hr)

For steady-state conditions, $dC/dt = 0$, the radon source strength is defined by

$$S = V C (\lambda + I) \tag{6.7}$$

This expression may be used to determine the radon evolution from a variety of steady-state situations.

6.2.2 Evolution of Radon from the Household Water Supply

An important radon problem is the assessment of radon air concentration as a result of the household water supply. The contribution to the air radon concentration from the water utilization is given by

$$S_w = N U C_w f_w \tag{6.8}$$

where
S_w = radon source strength (pCi/hr) from the water
N = number of family members residing in the home
U = average water usage per member per day (liters/member-day)
C_w = radon concentration in the water (pCi/liter)
f_w = fraction of radon released from the water into the air

6.2.3
Radon Risk Assessments

In 2003, the Environmental Protection Agency (EPA) developed a revised risk assessment for exposure to radon in homes to complement the results of BEIR VI. In BEIR VI, two preferred risk models were utilized to predict that there are 15 400 to 21 800 radon-related excess lung cancer deaths per year.

The EPA modified and extended the BEIR VI approach by constructing a single risk model that yielded a numerical result between the BEIR VI predictions. The EPA model also included a Monte Carlo analysis, which evaluated sources of uncertainty that were not fully evaluated in BEIR VI. In its model, the EPA used more recent mortality and smoking data, and modified the BEIR VI calculations to include all radon-induced lung cancer deaths (rilcd). The BEIR VI excess risk definition effectively omitted premature deaths in individuals who otherwise would have died from lung cancer.

Based on its analysis, EPA estimated that nationally in 1995 about 21 000 (8 000–45 000) lung cancer deaths were radon related. The values in parenthesis represent the 90% confidence interval.

The EPA's projected risk per unit exposure is about 5.4×10^{-4} rilcd/WLM for the US population, with smokers at higher risk from exposure to radon. For smokers (non-smokers), the projected risk is about 9.7×10^{-4} rilcd/WLM (1.7×10^{-4} rilcd/WLM).

6.3
Environmental Monitoring Programs

Environmental monitoring is an important aspect of a facility's radiation protection program. The environmental monitoring program is often mandated by the facility's license or by Federal regulations. The purpose of environmental monitoring is to measure the radiation exposure to the general public from the facility's effluent releases.

Prior to a facility's operation, the monitoring program will determine the location of any radiation anomalies, document local radiation levels, identify local radionuclide concentrations, and document the annual meteorological conditions. The preoperational program provides a baseline set of radiological conditions and parameters that can be used to assess the impact of the facility on the environment and the population living in the vicinity of the facility.

After the facility is in operation, the focus of the environmental monitoring program shifts to ensure compliance with all applicable regulations. Periodic reports will address plant releases by nuclide and pathway. Air, water, vegetation, fish, animal, plant samples, direct radiation measurements, and environmental air samples will further characterize the facility's impact upon the environment. A properly functioning environmental monitoring program will also foster a positive and credible public image.

6.4 Environmental Releases

Effluent releases from nuclear facilities are often characterized in terms of the committed effective dose equivalent (CEDE) from various isotopes and their release pathways. This requires knowledge of the concentration of the radionuclides released via their associated pathways. The following equation outlines the conceptual methodology utilized to assess the effect of effluent releases upon downstream population groups:

$$D = \sum_{i}^{N} \sum_{j}^{P} C^{ij} \; I_a^{ij} \; D_{\mathrm{F}}^{ij} \; \mathrm{CF}^{i} \tag{6.9}$$

where

D = committed effective dose equivalent (rem)
N = total number of different radionuclides released
i = index labeling the i^{th} radionuclide
j = pathway label index
P = total number of pathways (air, water, and food stuffs)
C^{ij} = average concentration of radionuclide i in pathway j (μCi/cm^3)
I_a^{ij} = annual intake of isotope i in the environmental medium in pathway j to an individual in population group a (cm^3 or kg of material)
a = population group composed of infants, children, teenagers, and adults
D_{F}^{ij} = dose factor for the particular nuclide and pathway (rem/μCi)
CF^{i} = conversion factor to ensure the CEDE is in mrem or μSv

More specific pathway equations will be provided in a subsequent section of this chapter. In particular, both gaseous and liquid effluent releases will be outlined in terms of Federal regulatory guidance. Although these equations will be more detailed than Equation (6.9), they will still contain its essential elements.

6.5 Accumulation of Activity in Ponds and Surfaces

Gas or liquid effluents and their accumulation on flat surfaces (ponds or fields) are of concern in assessing facility releases. Assuming a constant release rate, the rate of buildup of activity on a flat surface is given by

$$r_{\mathrm{d}} = (V)\,(Q)\,(X/Q) \tag{6.10}$$

where

r_{d} = surface deposition rate (Bq/m^2-s)
V = deposition rate (m/s)
Q = source term (Bq/s)
X/Q = relative concentration (dispersion factor) (s/m^3)

In addition to surface deposition, the accumulation of activity in a body of water, such as a pond, must often be assessed as part of an environmental monitoring program. The steady-state concentration of a soluble isotope in a pond assuming a constant daily input rate is given by an application of the production equation (see Appendix VII) with $r_d\ S$ being the production term:

$$C_{eq} = (r_d)\,(S)/(\lambda_e)\,(V) \tag{6.11}$$

where

C_{eq} = steady-state concentration in the pond (Bq/m^3)
λ_e = effective disintegration constant
= $\ln(2)/T_e = \ln(2)\,(1/T_p + 1/T_b)$
T_p = physical half-life of the released radionuclide
T_b = biological half-life of the radionuclide in the pond
T_e = effective half-life
r_d = daily input rate (Bq/m^2-day)
S = surface area of the pond (m^2)
d = average pond depth (m)
V = pond volume (m^3)
= $S \times d$

The accumulation of activity in a pond requires the assessment of the intake of radionuclides by organisms living in that environment. The concentration of a radionuclide in an organism living in the pond is given by

$$C_f = (C_{eq})\,(I)/(\lambda_e) \tag{6.12}$$

where

C_f = concentration of activity in the organism (Bq/kg)
C_{eq} = steady-state concentration in the pond (Bq/m^3)
λ_e = effective disintegration constant (1/day)
I = daily intake of pond water by the organism per its unit mass (m^3/kg-day)

Equation (6.12) represents another application of the production equation (see Appendix VII) with $C_{eq}\ I$ being the production term per unit mass.

Ultimately, the effluent results are characterized in terms of dose which can be related to an impact upon humans. Currently, a common approach is to quantify this dose in terms of risk.

The risk (r) expressed in excess cancer deaths is given by

$$r = c\ D \tag{6.13}$$

where c is the risk coefficient (excess cancer deaths per person-rem) and D is the dose equivalent (person-rem) received by the population at risk. Appendix V provides a summary of risk coefficients from major reports.

6.6
Pathways Associated with Open and Closed Fuel Cycles

In Chapter 4, radiological issues associated with open and closed fuel cycles were outlined. Open fuel cycles require the long-term storage of high-level waste in a facility such as a geological repository. Since closed fuel cycles recycle all actinides and long half-life fission products, the storage requirements are significantly reduced.

6.6.1
High-Level Waste

High-level radioactive wastes are the highly radioactive materials produced as a byproduct of the reactions that occur inside nuclear reactors. High-level wastes include spent reactor fuel when it is accepted for disposal and waste materials remaining after spent fuel is reprocessed.

6.6.2
Open Fuel Cycles

Based on studies of the Yucca Mountain Repository, a geological disposal facility is designed to function for periods exceeding 10 000 y. Demonstration of geologic repository performance for 10^4–10^6 years represents a significant scientific challenge.

For these long timeframes, the following radionuclides are important for assessing the environmental performance of a geological repository: U-238 (4.47×10^9 y), U-235 (7.04×10^8 y), U-233 (1.59×10^5 y), Pu-239 (2.41×10^4 y), Pu-242 (3.75×10^5 y), Np-237 (2.14×10^6 y), I-129 (1.57×10^7 y), Tc-99 (2.13×10^5 y), Se-79 (2.9×10^5 y), and Cl-36 (3.01×10^5 y). The relative importance of these radionuclides depends on the repository design lifetime, characteristics of the waste form, and repository specific parameters. This means that the relevant processes and issues may be quite different for various repository designs, fuel forms, and licensing requirements.

Pathways to the environment require degradation of the high-level waste package (e.g., waste form and high-level waste container). Over extended periods, the facility's engineered barriers degrade and the waste form is a primary barrier to the release of radioactivity. The degradation of the engineered barrier, allows radionuclide migration to the surface and its water sources. Once the radionuclides enter the water, dissolved ions and colloids are available for release. The specific release pathways mirror the liquid effluent pathways noted in the subsequent discussion.

6.6.3
Closed Fuel Cycles

Since a closed fuel cycle eliminates the minor actinides and has the potential to eliminate long-lived fission products, the storage time requirements are significantly less than those required for an open fuel cycle facility. Assuming the actinides and long-lived fission products noted previously are eliminated, the radionuclides of interest are Cs-137 (30.1 y) and Sr-90 (28.8 y). The licensing basis for a closed fuel cycle waste disposal facility would be < 1 000 y and would have characteristics similar to those for shallow-land burial facilities. Example burial facility requirements and associated pathways are addressed in Scenario 6.3.

6.7
Regulatory Guidance for Effluent Pathways

The Federal Government promulgates regulatory guidance for the calculation of environmental releases. Liquid, gaseous, particulate, and radioiodine pathways are considered. Herein, we will illustrate the guidance provided to facilities regulated by the Nuclear Regulatory Commission (NRC). The NRC approach is similar to other regulatory approaches and is outlined in Regulatory Guide 1.109, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I*. The dose limits of 10 CFR 50, Appendix I are summarized in Table 6.4. All limits apply to the highest off-site dose calculated for a maximum exposed individual within the model framework.

The NRC considers a population group composed of infants (0–1 year), children (1–11 years), teenagers (11–17 years), and adults (17 years and older). For the purpose of evaluating dose commitment, the maximum exposed infant is assumed

Table 6.4 Summary of NRC Effluent Dose Limits (10 CFR 50, Appendix I).

Dose	Effluent Type	Limit per Unit
Total body from all pathways	Liquid	3 mrem/year
Any organ from all pathways	Liquid	10 mrem/year
Gamma dose in air	Noble gas	10 mrad/year
Beta dose in air	Noble gas	20 mrad/year
Dose to total body of an individual	Noble gas	5 mrem/year
Dose to the skin of an individual	Noble gas	15 mrem/year
Dose to any organ from all pathways	Radioiodine and particulates (including H-3 and C-14)	15 mrem/year

to be newborn, the maximum exposed child is assumed to be 4 years old, the maximum exposed teenager is assumed to be 14 years old, and the maximum exposed adult is assumed to be 17 years old.

The "maximum"-individuals are characterized as "maximum" with respect to food consumption, occupancy, and other usage of the region in the vicinity of the plant site. As such, a maximum exposure is realized from the model's assumptions.

Because the radiation dose commitment per unit intake of a given radionuclide varies as a function of age, four sets of internal dose conversion factors have been calculated. These dose factors are appropriate for the four different age groups defined above. Specifically, these dose conversion factors are based upon continuous intake over a 1-year environmental exposure period. The associated dose commitment extends over a 50-year period following the intake.

6.8 Doses from Liquid Effluent Pathways

The NRC requires the assessment of radiation doses from the use of potable water, aquatic food, shoreline deposits, and irrigated foods by members of the general public. The following dose assessment models are recommended by the NRC guidance in order to evaluate liquid effluent pathways for the maximum exposed individual and the population within 50 miles of the facility.

6.8.1 Potable Water

One of the most basic pathways is the ingestion of water. The water may contain a variety of radionuclides that are utilized or generated in the nearby nuclear facility. Water may be released from the facility through a variety of pathways such as release to wells, lakes, or streams. The dose equivalent from potable water from these pathways is given by

$$R_{apj} = k_1 \frac{U_{ap}\, M_p}{F} \sum_i Q_i\, D_{aipj} \exp\left(-\lambda_i\, t_p\right) \tag{6.14}$$

where

R_{apj} = the total annual dose equivalent to organ j for individuals of age group a from nuclide i in pathway p (mrem/year)

a = label for age group
- = 1 for infants
- = 2 for children
- = 3 for teenagers
- = 4 for adults

i = label for i^{th} nuclide of interest

j = label for organ of interest
p = label for pathway
U_{ap} = usage factor that specifies the exposure time or intake rate for a specific individual and pathway (hr/year, liters/year, or kg/year)
M_p = mixing ratio (or reciprocal of the dilution factor) at the point of exposure (or the point of withdrawal of drinking water or point of harvest of aquatic food). It is dimensionless.
F = flow rate of the liquid effluent (ft^3/s)
Q_i = release rate of nuclide i (Ci/year)
D_{aipj} = dose factor for a specified age group, radionuclide, organ, and pathway, which can be used to calculate the radiation dose from an intake of a radionuclide (in mrem/pCi) or from exposure to a given concentration of a radionuclide in sediment, expressed as a ratio of the dose rate (mrem/hr) and the areal radionuclide concentration (pCi/m^2).
λ_i = i^{th} radionuclide's decay constant (1/hr)
t_p = average transit time required for a radionuclide to reach the point of exposure. For internal dose, it is the total time elapsed between release of the nuclide and ingestion of food or water (hours).
k_1 = conversion factor of 1100, the factor to convert from (Ci/year)/(ft^3/s) to pCi/liter

6.8.2 Aquatic Foods

Aquatic foods include biota such as fish, crabs and mussels that are consumed by the nearby population. The pathway will include all likely routes by which the facility's radionuclides are released, transported to the water body, incorporated into the water body, sediment, plants, or other food chain elements, consumed by the fish, and then consumed by humans. The dose equivalent to humans from the ingestion of aquatic foods is given by

$$R_{apj} = k_1 \frac{U_{ap}\ M_p}{F} \sum_i Q_i\ B_{ip}\ D_{aipj} \exp\left(-\lambda_i\ t_p\right) \tag{6.15}$$

where
B_{ip} = the equilibrium bioaccumulation factor for a given nuclide and pathway expressed as the ratio of the concentration in biota (pCi/kg) to the radionuclide concentration in water (pCi/liter), in liters/kg

The other quantities were previously defined in Equation (6.14).

6.8.3 Shoreline Deposits

The calculation of an individual's dose equivalent from shoreline deposits of radionuclides is complex. It involves the estimate of sediment radionuclide loading, transport, and concentration of radionuclides associated with suspended or deposited radionuclides released from the facility. The calculation of radiation dose from these shoreline deposits is given by

$$R_{apj} = k_2 \frac{U_{ap}\, M_p\, W}{F} \sum_i Q_i\, T_i\, D_{aipj} \exp(-\lambda_i\, t_p) \cdot [1 - \exp(-\lambda_i\, t_b)] \quad (6.16)$$

where

W = shoreline width parameter (dimensionless)
T_i = i^{th} radionuclide's half-life (days)
t_b = period of time for which the sediment or soil is exposed to the contaminated water (hours)
k_2 = conversion factor of 110 000, the factor to convert from (Ci/year)/(ft^3/s) to pCi/liter and to account for the proportionality constant used in the sediment radioactivity model.

6.8.4 Irrigated Foods

Liquid effluents can migrate or be directly released to adjacent water bodies that may supply water to agricultural crops. This contaminated water will provide a source of radioactivity that is taken up by the plant. Once the plant is consumed by humans, its radionuclides will be available for transport within the human body. The dose from the consumption of irrigated foods from tritium and from other radionuclides will be presented in the next two sections.

6.8.4.1 Irrigated Foods (Tritium)

The dose equivalent from the consumption of irrigated foods from tritium is given by

$$R_{apj} = U_{ap}^{\text{veg}}\, C_{\text{v}}\, D_{apj} + U_{ap}^{\text{animal}}\, D_{apj}\, F_{\text{A}}\, (C_{\text{v}}\, Q_{\text{F}} + C_{\text{Aw}}\, Q_{\text{Aw}}) \quad (6.17)$$

where

U_{ap}^{veg} = ingestion rate of vegetables for a specific age group and pathway (kg/year)
C_{v} = tritium concentration in vegetables (pCi/kg)
U_{ap}^{animal} = ingestion rate of meat and poultry for a specific age group and pathway (kg/year)
F_{A} = the stable element (hydrogen) transfer coefficient that relates the daily intake rate by an animal to the concentration in an edible

portion of an animal product, in pCi/liter (milk) per pCi/day or pCi/kg (animal product) per pCi/day.

Q_F = consumption rate of contaminated feed or forage by an animal in kg/day net weight

C_{Aw} = tritium concentration in water consumed by animals (pCi/liter)

Q_{Aw} = consumption rate of contaminated water by an animal (liters/day)

6.8.4.2 Irrigated Foods (Radionuclides Other than Tritium)

The dose equivalent from the consumption of irrigated foods from radionuclides other than tritium is given by

$$R_{apj} = U_{ap}^{veg} \sum_i d_i \exp(-\lambda_i t_h) D_{aipj} \tag{6.18}$$
$$\times \left[\frac{r\,[1-\exp(-\lambda_{Ei} t_e)]}{Y_v \lambda_{Ei}} + \frac{f_I B_{iv} [1-\exp(-\lambda_i t_b)]}{P \lambda_i} \right] + U_{ap}^{animal} \sum_i F_{iA} D_{aipj}$$
$$\times \left\{ Q_F d_i \exp(-\lambda_i t_h) \times \left[\frac{r\,[1-\exp(-\lambda_{Ei} t_e)]}{Y_v \lambda_{Ei}} + \frac{f_I B_{iv} [1-\exp(-\lambda_i t_b)}{P \lambda_i} \right] + C_{iAw} Q_{Aw} \right\}$$

where

d_i = deposition rate of a radionuclide (pCi/m^2-hr)

t_h = holdup time that represents the time interval between harvest and consumption of food (hours)

r = fraction of deposited activity that is retained on crops

λ_{Ei} = effective removal rate constant for a radionuclide from crops (1/hr)
= $\lambda_i + \lambda_w$

λ_w = removal rate constant for physical loss by weathering (1/hr)

t_e = time period that crops are exposed to contamination during the growing season (hours)

Y_v = agricultural productivity (yield) in kg (net weight)/m^2

f_I = fraction of the year's crops that are irrigated

B_{iv} = concentration factor for the uptake of a radionuclide from soil by the edible portion of crops, in pCi/kg (wet weight) per pCi/kg dry soil

P = effective "surface density" for soil, in kg (dry soil)/m^2

F_{iA} = stable element transfer coefficient that relates the daily intake rate by an animal to the concentration in an edible portion of animal product, in pCi/liter (milk) per pCi/day or pCi/kg (animal product) per pCi/day

C_{iAw} = concentration of a radionuclide in water consumed by animals in pCi/liter

These model equations contain numerous parameters that are defined in Regulatory Guide 1.109. The models are necessarily qualitative in nature, and the parameters and assumptions are chosen to ensure that the Federal effluent release limits are conservatively achieved.

6.9 Doses from Gaseous Effluent Pathways

The NRC effluent pathway analytical model permits the calculation of doses from exposure to noble gases discharged to the atmosphere. Separate models are given for air and tissue doses due to gamma and beta rays. With the exception of the case for a noble gas released from an elevated stack, all models assume immersion in a semi-infinite cloud.

Only a sample of the possible release pathways and dose models will be presented. This selection will be sufficient to illustrate the calculational methodology and the associated input parameters.

6.9.1 Annual Gamma Air Dose from Noble Gas Releases from Free-Standing Stacks Higher than 80 Meters

The term gamma air dose refers to the components of the air dose associated with photons emitted during nuclear and atomic transformations. Although gamma rays and X-rays will be the dominant dose contributor, annihilation and bremsstrahlung photon radiations are also contributors to the gamma air dose. The annual gamma air dose from noble gas releases from a high stack is

$$D^{\gamma}(r,\theta) = \frac{k_1}{r(\Delta\theta)} \sum_n \frac{1}{u_n} \sum_s f_{ns} \sum_k u_a(E_k)\, E_k\, I(H, u, s, \sigma_z, E_k) \cdot \sum_i Q_{ni}^{D}\, A_{ki} \tag{6.19}$$

where

$D^{\gamma}(r, \theta)$	=	annual gamma air dose at a distance r and in the sector at angle θ (mrad/year)
r	=	distance from release point (meters)
θ	=	angle of plume relative to a fixed direction. Normally, north = 0°, east = 90°, etc.
k_1	=	conversion factor = 260 to obtain $D^{\gamma}(r, \theta)$ in mrad/year and has the units of mrad-radians-m^3-dis/s-MeV-Ci
Δθ	=	sector width over which atmospheric conditions are averaged (radians)
n	=	label for the wind speed class
s	=	label for the stability class
k	=	label for the photon energy group
i	=	label for the radionuclide
u_n	=	mean wind speed of wind speed class n (m/s)
f_{ns}	=	joint frequency of occurrence of stability class (s) and wind speed class (n) for sector θ (dimensionless)

$u_a(E_k)$ = air energy absorption coefficient for a specified photon energy group (1/m)
E_k = energy of the k^{th} photon group (MeV/photon)
$I\,(H, u, s, \sigma_z, E_k)$ = dimensionless integration constant accounting for the distribution of radioactivity according to the meteorological conditions
H = effective stack height
u = wind speed
σ_z = vertical plume standard deviation
Q_{ni}^{D} = radionuclide release rate corrected for decay during plume transit to a specified distance under a fixed wind speed (Ci/year)
A_{ki} = photon yield for gamma-ray photons in energy group k from the decay of radionuclide i (photons/dis)

6.9.2 Annual Air Dose from All Noble Gas Releases

The next two sections will outline the methodology to calculate the annual air dose from noble gas releases. Specific expressions for gamma and beta doses will be provided. Plumes of gaseous effluents are considered semi-infinite in the case of ground-level noble gas releases.

6.9.2.1 Annual Gamma Air Dose from All Noble Gas Releases

The equation for annual gamma air dose is as follows:

$$D^{\gamma}(r, \theta) = k_2 \sum_i Q_i\,[X/Q]^D(r, \theta)\,\mathrm{DF}_i^{\gamma} \tag{6.20}$$

where
$D^{\gamma}(r, \theta)$ = annual gamma air dose at the point (r, θ) (mrad/year) with respect to the discharge point
k_2 = conversion factor 3.17×10^4, the number of pCi/Ci divided by the number of seconds per year
Q_i = release rate of the specified radionuclide (Ci/year)
$[X/Q]^D(r, \theta)$ = annual average gaseous dispersion factor (corrected for radioactive decay) at a given point (r, θ)
DF_i^{γ} = gamma air dose factor for a uniform, semi-infinite cloud of the specified radionuclide (mrad-m^3/pCi-year)

6.9.2.2 Annual Beta Air Dose from All Noble Gas Releases

Beta air dose refers to the component of the air dose associated with particle emissions during nuclear and atomic transformations. These transformations include beta decay, positron decay, and conversion electrons. The beta air dose is given by

$$D^{\beta}(r, \theta) = k_2 \sum_i Q_i \, [X/Q]^D(r, \theta) \, \mathrm{DF}_i^{\beta} \tag{6.21}$$

where

$D^{\beta}(r, \theta)$ = annual beta air dose at the point (r, θ) (mrad/year)
DF_i^{β} = beta air dose factor for a uniform, semi-infinite cloud of the specified radionuclide (mrad-m^3/pCi-year)

6.9.2.3 Annual Total Body Dose Equivalent from Noble Gas Releases from Free-Standing Stacks More than 80 Meters High

The calculation of the total body dose from a high stack is similar to the calculation for air dose. The whole-body dose equivalent calculation includes a factor that includes the ratio of tissue to air energy absorption coefficients and is given by

$$D^{\mathrm{T}}(r, \theta) = k_3 \, S_{\mathrm{F}} \sum_k D_k^{\gamma}(r, \theta) \exp\left[-u_{\mathrm{a}}^{T}(E_{\mathrm{k}}) \, t_{\mathrm{d}}\right] \tag{6.22}$$

where

$D^{\mathrm{T}}(r, \theta)$ = annual total body dose equivalent evaluated at 5 g/cm^2 tissue depth at the point (r, θ) (mrem/year)
S_{F} = attenuation factor that accounts for the dose reduction due to the shielding provided by residential structures (dimensionless)
k_3 = conversion factor = 1.11, the average ratio of tissue to air energy absorption coefficients
$D_k^{\gamma}(r, \theta)$ = annual gamma air dose equivalent associated with a specified photon energy group at the point (r, θ) in mrem/year
$u_{\mathrm{a}}^{T}(E_{\mathrm{k}})$ = tissue energy absorption coefficient (cm^2/g)
t_{d} = product of the tissue density and depth used to determine a total body dose (g/cm^2)

6.9.2.4 Annual Skin Dose from Noble Gas Releases from Free-Standing Stacks Higher than 80 Meters

The annual skin dose is given by

$$D^{\mathrm{S}}(r, \theta) = k_3 \, S_{\mathrm{F}} \, D^{\gamma}(r, \theta) + k_2 \sum_i Q_i \, [X/Q]^D(r, \theta) \, \mathrm{DFS}_i \tag{6.23}$$

where

$D^{\mathrm{S}}(r, \theta)$ = annual skin dose at a depth of 7 mg/cm^2 in tissue at the point (r, θ) in mrem/year
DFS_i = beta skin dose factor for a semi-infinite cloud of the specified nuclide which includes the attenuation by the outer "dead" layer of the skin in mrem-m^3/pCi-year

6.10
Annual Doses from All Other Noble Gas Releases

Equations (6.22) and (6.23) provided the annual total body and skin dose from noble gas releases from free-standing stacks higher than 80 m. The following two sections provide models for total body and skin doses for all other types [ground-level and shorter (< 80 m) stacks] of noble gas releases.

6.10.1
Annual Total Body Dose Equivalent from All Other Noble Gas Releases

The annual total body dose equivalent from noble gas releases, evaluated at a depth of 5 cm into the body, is given by

$$D_{\infty}^{T}(r, \theta) = S_F \sum_i X_i(r, \theta)\, DFB_i \tag{6.24}$$

where

$D_{\infty}^{T}(r, \theta)$ = annual total body dose equivalent due to immersion in a semi-infinite cloud at point (r, θ) in mrem/year

$X_i(r, \theta)$ = annual average ground-level concentration of a specified radionuclide at the point (r, θ) in pCi/m^3

DFB_i = total body dose factor for a semi-infinite cloud of radionuclide i including the attenuation of 5 g/cm^2 of tissue in mrem-m^3/pCi-year

6.10.2
Annual Skin Dose from All Other Noble Gas Releases

The calculation of skin dose at 7 mg/cm^2 from immersion in a noble gas cloud is given by the sum of gamma and beta contributions

$$D_{\infty}^{S}(r, \theta) = k_3\, S_F \sum_i X_i(r, \theta)\, DF_i^{\gamma} + \sum_i X_i(r, \theta)\, DFS_i \tag{6.25}$$

where

$D_{\infty}^{S}(r, \theta)$ = annual skin dose due to immersion in a semi-infinite cloud at point (r, θ) in mrem/year

6.11
Doses from Radioiodines and Other Radionuclides Released to the Atmosphere

NRC limits also apply to radioiodine and other radionuclides, not including noble gases. Doses due to particulate releases, as well as carbon-14 and tritium intakes from terrestrial food chains, are also included within this category. The ground deposition and inhalation pathways are specifically considered. Equations (6.26) through (6.28) describe doses due to radioiodine, particulates, and tritium. Noble gas exposures where included in (6.19) through (6.25), but not included in (6.26) through (6.28).

6.11.1
Annual Organ Dose from External Irradiation from Radionuclides Deposited Onto the Ground Surface

The annual organ dose resulting from direct exposure to the contaminated ground plane from radioiodine, particulates, and tritium released from the facility is given by

$$D_j^G(r, \theta) = k_4 \, S_F \sum_i C_i^G(r, \theta) \, DFG_{ij} \tag{6.26}$$

where

$D_j^G(r, \theta)$ = annual dose to organ j at point (r, θ) in mrem/year
j = labels organ of interest
i = labels radionuclide of interest
k_4 = conversion constant = 8760, the number of hours in a year
S_F = shielding factor that accounts for the dose reduction due to shielding provided by residential structures during occupancy
$C_i^G(r, \theta)$ = ground plane concentration of radionuclide i at the point (r, θ) in pCi/m^2
DFG_{ij} = open field ground plane dose conversion factor for organ j from radionuclide i, in $mrem\text{-}m^2/pCi\text{-}hr$

6.11.2
Annual Organ Dose from Inhalation of Radionuclides in Air

The annual dose associated with the inhalation of radionuclides to organ j of an individual in age group a is given by

$$D_{ja}^A(r, \theta) = R_a \sum_i x_i(r, \theta) \, DFA_{ija} \tag{6.27}$$

where

$D_{ja}^A(r, \theta)$ = annual dose equivalent to organ j of an individual in age group a at location (r, θ) due to inhalation, in mrem/year

a = labels the age group
R_a = annual air intake for individuals in age group a, in m^3/year
$x_i(r, \theta)$ = annual average concentration of radionuclide i in air at location (r, θ), in pCi/m^3
DFA_{ija} = inhalation dose factor for radionuclide i, organ j, and age group a, in mrem/pCi

6.11.3 Annual Organ Dose from Ingestion of Atmospherically Released Radionuclides in Food

The annual dose equivalent to organ j of an individual in age group a that results from ingestion of atmospherically released radionuclides in produce, milk, meat, and leafy vegetables is given by

$$D_{ja}^{D}(r, \theta) = \sum_i DFI_{ija} \cdot [U_a^v \; f_g \; C_i^v(r, \theta) + U_a^m \; C_i^m(r, \theta) + U_a^F \; C_i^F(r, \theta) + U_a^L \; f_l \; C_i^L(r, \theta)] \quad (6.28)$$

where

$D_{ja}^{D}(r, \theta)$ = annual dose equivalent to organ j of an individual in age group a from the ingestion of produce, milk, leafy vegetables, and meat at location (r, θ), in mrem/year
DFI_{ija} = ingestion dose factor for radionuclide i, organ j, and age group a, in mrem/pCi
U_a^v = annual intake of produce (non-leafy vegetables, fruits, and grains) for individuals of age group a, in kg/year
U_a^m = annual intake of milk for individuals of age group a, in liters/year
U_a^F = annual intake of meat for individuals of age group a, in kg/year
U_a^L = annual intake of leafy vegetables for individuals of age group a, in kg/year
f_g = fraction of the produce ingestion rate that is produced in the garden of interest
f_l = fraction of the leafy vegetable ingestion rate that is produced in the garden of interest
$C_i^v(r, \theta)$ = concentration of radionuclide i in produce at location (r, θ), in pCi/kg
$C_i^m(r, \theta)$ = concentration of radionuclide i in milk at location (r, θ), in pCi/liter
$C_i^L(r, \theta)$ = concentration of radionuclide i in leafy vegetables at location (r, θ), in pCi/kg
$C_i^F(r, \theta)$ = concentration of radionuclide i in meat at location (r, θ), in pCi/kg

6.12
Pathway Selection

The pathway models are sufficient to estimate radiation exposure for maximum individuals and the population within 50 miles of the facility. These pathways appropriately describe exposure routes routinely applicable to nuclear facilities. However, other pathways that arise either from site-specific features or from changes in facility operation should be evaluated if they contribute at least 10% of the total dose from the routine pathways previously defined.

6.13
Model Parameters

Regulatory Guide 1.109 provides recommended values for the various model parameters. Specific values of environmental data, human data, dose factors, and miscellaneous model parameters are tabulated. The NRC licensee may take into account facility-specific exposure conditions. These conditions include actual values for agricultural productivity, dietary habits, residence times, dose attenuation by structures, measured environmental transport or bioaccumulation factors, and other model parameters determined for a specific site.

6.14
Intentional Dispersal of Radioactive Materials

Radioactive materials can be dispersed by natural means such as the diffusion of radon gas, by inadvertent means such as a power reactor accident, and intentionally through acts of sabotage or terrorism. Following the September 11, 2001 attacks; concerns for the intentional dispersal of radioactive material have increased.

The intentional dispersal of radioactive materials would occur primarily through either a radiological dispersal device (RDD) or an improvised nuclear device (IND). An IND is a nuclear weapon that is fabricated by an adversary state or terrorist group from illicit nuclear material. An improvised nuclear weapon produces mass casualties and spreads radioactive materials over a wide area.

A radiological dispersal device is a device or mechanism that is intended to spread radioactive materials from the detonation of conventional explosives or other means. RDDs include dirty bombs and devices that disperse radioactive material over an area to create economic and social impacts. The physical effects of a RDD can be addressed using engineering solutions, but the social and psychological impacts require careful attention.

The health physics hazards associated with an IND or a RDD are addressed in terms of three phases. These are the early phase, intermediate phase, and late phase. The early phase (or emergency phase) is the period at the beginning of the incident when immediate decisions for the effective use of protective actions

are required and actual field measurement data is generally not available. First responders to the scene of a terrorist event may not recognize its radiological aspects unless the attack is directed toward a known nuclear facility. Since it is unlikely that all first responders have received radiological training, it is necessary to establish a mechanism to ensure these individuals do not receive an unacceptable radiation exposure while permitting them to perform critical activities during the early phases of a radiological event.

Exposure to the radioactive plume, short-term exposure to deposited materials, and inhalation of radioactive materials are generally included when considering protective actions during the early phase. Actions taken during the early phase include initial emergency response actions to protect public health and welfare. Priority is given to lifesaving and first aid actions.

The intermediate phase follows the early phase response. This phase begins after the source and release are under control and protective action decisions are made based on measurements of exposure and radioactive materials that were deposited following the initiating event. Activities in the intermediate phase typically overlap with early and late phase activities, and may continue for weeks to many months, until protective actions are terminated.

The late phase is the period when recovery and cleanup actions designed to reduce radiation levels in the environment to acceptable levels are commenced, and it ends when all the recovery actions have been completed. With the additional time and increased understanding of the situation, there will be opportunities to involve key stakeholders in providing sound, cost-effective recommendations. Generally, early (or emergency) phase decisions are made directly by elected public officials, or their designees, with limited stakeholder involvement, due to the need to act within a short timeframe. Long-term decisions should be made with stakeholder involvement, and include incident-specific technical working groups to provide expert advice to decision makers on impacts, costs, and alternatives.

The dose to emergency responders should be limited, if possible, to the occupational limits. However, doses beyond these levels may be authorized during a severe disaster because prompt, well-considered actions can potentially save many lives and avert significant harm to the public. Even in these situations, the principles of ALARA apply for emergency responders. Table 6.5 summarizes recommendations for dose limitation and guidance during an event involving radioactive materials.

In order to protect the public, protective actions are usually required to regain control during or after a radiological emergency. Because all protective actions have an associated risk, the use of a particular action involves an assessment of risk/benefit. Table 6.6 lists available protective actions for a variety of exposure pathways including IND and RDD events.

Table 6.5 Dose Limitation and Guidance During a Radiological Event.

Classification or Action	Applicability	Limit or Guidance[a]
Full mitigation	General public dose limitation	[b]
Sheltering	Avert dose to general public	5–50 mSv[c]
Evacuation	Avert dose to general public	50–500 mSv[c]
Administer stable iodine	Avert dose to children and pregnant women	50–500 mSv[d]
Any single food category[e]	Avert dose to general public	10 mSv/y[c]
Relocation	Avert dose to general public	10 mSv/month, 1000 mSv[c]
Annual Limit	Recovery Workers (non-emergency work)	50 mSv/y[c]
Guidance for emergency action	Recovery Workers (emergency work)	500 mSv[c]

[a] If two values are given, the lower value represents the lowest effective dose at which the countermeasure is likely to be justified. The larger value represents the effective dose at which the countermeasure is usually justified.
[b] Values are obtained through the process of justification and ALARA. The results may be higher or lower than the limit of 1 mSv/y for individual members of the public.
[c] Effective dose.
[d] Equivalent dose.
[e] The Food and Drug Administration (FDA) has provided guidance for intervention in the ingestion pathway based on the total diet. (Federal Register 1998).

Source: NCRP-138 (2001).

Table 6.6 Exposure Pathways and Associated Protective Actions.

Exposure Pathway	Protective Action
External radiation exposure from nuclides in the plume.	Sheltering, evacuation, and control of access.
Internal contamination due to nuclides in the plume.	Sheltering, *ad hoc* respiratory protection[a], administration of stable iodine or other radioprotective agents, evacuation, and control of access.
External contamination from surface deposited radioactive material and activation products.	Sheltering, evacuation, control of access, and decontamination.
External radiation from surface deposited radioactive material and activation products.	Sheltering, evacuation, relocation, control of access, and decontamination.
Internal contamination due to resuspension.	Evacuation, relocation, control of access, and decontamination.
Internal contamination due to personnel contamination.	Control of access and decontamination.
Internal exposure due to ingestion of contaminated water and foodstuffs.	Control of food and water and use of stored animal feeds.

[a] Ad hoc respiratory protection includes actions such as covering the nose and mouth with a dry or wet handkerchief or cloth.

Source: NCRP-138 (2001).

6.15 Protection of the Environment

Public awareness and government regulations have increased interest in protection of the environment. ICRP-103 noted that there is a need for policy advice and guidance in relation to radiological protection that arises from a lack of consistency at the international level.

Previous ICRP policy was based on the premise that protection of the public would ensure that other species were not placed at risk. As noted in ICRP-103, the Commission continues to believe this is valid. However, the ICRP now believes that it is necessary to consider a wider range of environmental situations irrespective of any human connection. Accordingly, the ICRP is developing a small set of reference animals and plants (e.g., Reference Duck) and their relevant databases for species that are typical of the major environments. However, the ICRP does not propose to recommend dose limits for environmental protection. The intention is to compare environmental radiation doses to doses having known or expected biological effects.

The protection of nonhuman species as part of environmental protection will be developed in publications following ICRP-103. It is expected that the concept of protection of nonhuman species will evolve, as the recommendations of ICRP-103 are refined.

6.16 Scenarios

Scenario 6.1

You have been placed in charge of an environmental measurements section for a major health physics firm that has been providing indoor radon measurements. Your involvement in providing these services to the public requires that you understand the principles and technology supporting your services. A reporter has submitted the following list of questions for your interview next week. Based upon published reports from NCRP, EPA, and the Health Physics Society, answer the following questions.

6.1 Radon and radon progeny measurements can be categorized into three types: instantaneous (grab), integrated, and continuous. Define each type and give two examples of a method or instrument that exemplifies each type.

6.2 List three passive, integrating measurement methods or detectors used to measure radon in homes and briefly describe the principle or theory of each.

6.3 The short-lived radon (Rn-222) decay products are:
- a. Po-226, Bi-214, Po-218, and Bi-216
- b. Pb-214, Po-214, Bi-216, and Po-218
- c. Bi-214, Pb-214, Po-218, and Po-214
- d. Po-214, Ra-226, Rn-220, and Pb-214
- e. Po-216, Pb-212, Bi-212, and Po-212

6.4 Indoor unattached radon decay products are commonly measured using all of the following except:
- a. Diffusion battery.
- b. Cyclone precollectors.
- c. Electrostatic collectors.
- d. Diffusion tubes.
- e. Screen samples.

6.5 The fraction of unattached radon decay products in the air depends on all the following except:
- a. Condensation nuclei concentration.
- b. Particle size distribution.
- c. Radon concentration.
- d. Diffusion coefficient of Po-218.
- e. Room surface plateout rate.

6.6 How many working level months of exposure in 1 year will an individual receive living in a home with an annual average potential-alpha-energy concentration of 0.09 working levels and an occupancy factor of 0.75?

6.7 How does the tracheobronchial dose per unit radon concentration depend on (1) breathing rate, (2) equilibrium factor, and (3) unattached fraction? Sketch a graph for each.

Scenario 6.2

From initial emissions to final exposure to humans, numerous factors act to disperse or concentrate emitted radioisotopes. To properly assess the net impact requires familiarity with general concepts and the ability to solve specific problems.

6.8 What mechanisms reduce the concentration of airborne radioactive materials during atmospheric transport?

6.9 A person continuously ingesting radioactive material at a constant rate will eventually build up to a maximum internal dose rate depending on the effective half-life of the material. Draw a graph showing internal dose rate as a function of time in days (with one as the maximum dose on the y-axis). Assume an effective half-life of 1 day, and also assume a continuous intake for six effective half-lives followed by no intake for six effective half-lives.

6.10 Reproduce the previous graph, this time assuming no cessation of intake (i.e., the curve becomes/remains flat). Label this curve 1. Evaluate and superimpose on the graph the dose rate from a second isotope which has an effective half-life of 2 days. Assume that all other parameters remain the same.

6.11 Assume that I-131 settles onto a pond at a steady-state rate. Calculate the input rate to the pond's surface using the following information:

Deposition rate = 1×10^{-2} m/s

Source term = 1×10^{8} Bq/s

Relative concentration (X/Q) = 1.8×10^{-7} s/m^3

6.12 If the daily input rate to the pond is 0.5 Bq/m^2, calculate the maximum steady-state concentration of I-131 in the pond assuming:

Pond surface = 100 m by 10 m

The pond depth = 1 m (average)

I-131 physical half-life = 8 days

I-131 biological half-life (pond) = 15 days

6.13 Calculate the average concentration of radioactive iodine expected in fish using the following information:

Daily intake = 8×10^{-5} m^3/kg-day

Biological half-life (fish) = 21 days

Equilibrium activity in the pond = 2 Bq/m^3

6.14 In question 6.13, the assumption was that all I-131 that entered the pond was available for concentration in the fish tissue. List factors that may contribute to the inaccuracy of that model.

Scenario 6.3

You are the health physicist for an engineering firm that is designing a new low-level radioactive waste disposal facility which will operate for 30 years. The Compact Region which needs this facility has chosen the site and has decided against using traditional shallow land disposal technology (used at Barnwell, Beatty and Hanford). The Compact Commission wants to use an enhanced disposal technology in hopes of lessening public opposition to the facility. Your firm is to provide comprehensive licensing and design services for three conceptual designs to the Compact Commission.

The three designs are for (1) above-ground vaults, (2) earth-mounded bunkers with large above-ground concrete canisters for Class A waste and vaults below ground for Class B and C waste, and (3) below-ground vault disposal for all classes of waste. Vaults are to be constructed of reinforced concrete, and all waste is to be

Table 6.7 Dose Estimates.

Parameter	Disposal Options		
	Below Ground	Above/Below Ground	Above Ground
Average worker dose (person-rem/year)	3	3	5
Number of occupationally exposed employees	30	30	23
Maximum surface dose rate (mrem/year), 100 years post closure	7.7×10^{-7}	4.9×10^{-7}	1
Committed dose equivalent to maximally exposed individual (mrem/year), 500 years post closure	1.5	1.3	4
Average population committed dose, equivalent from 100 to 500 years post closure (person-rem/year)	10	9	30

grouted in place. The concrete canisters are 6 feet in diameter and over 7 ft high (large enough to contain large high-integrity containers and liners). The waste is also to be grouted in place in the canisters.

Preliminary dose estimates for the three options are presented in Table 6.7. The State and Compact Commission agree that the site meets all the technical requirements for stability and all draft licensing criteria.

6.15 Outline a generic preoperational environmental monitoring plan that will cover the three possible site designs. Include media to be sampled, locations, sample frequencies and types, and analyses to be performed.

6.16 Identify four major exposure pathways which could be expected during the institutional care period.

6.17 Discuss any potential health effects you might expect during the post-closure phase of each facility. Use a risk coefficient of 1.0×10^{-4} excess cancer deaths per rem in your assessment and assume a 400-year period of evaluation.

6.18 The Compact Commissioner has reviewed the three designs and asks you which disposal facility you recommended. As a health physicist, which option do you recommend to the Commissioner and why? Assume that all regulatory requirements can be met by each of the three designs.

Scenario 6.4

XYZ Radon Abatement Company has hired you as a consultant to help mitigate a detached single-family home with full basement that has a free-space volume of 100 m^3. A 1-year alpha track measurement was performed in this house, with a result of 25 pCi/liter. SF_6 tracer measurements have determined that outside air infiltrates the home at an average rate of 0.2 air changes per hour.

The following data apply to this question:

Decay constant (λ) for Rn-222 = 0.00755/hr

Number of persons in the household = 5

Occupancy factor = 0.7

Daily water usage per person = 200 liters

Fraction of radon released from water usage = 0.7

Tracheobronchial dose conversion factor = 0.7 rad/WLM

Quality factor (QF) for alpha = 20

ICRP organ weighting factor for the tracheobronchial (TB) region of the lung = 0.06

6.19 Assuming that radon is removed only by radioactive decay and outside air infiltration, estimate the annual average radon source strength or entry rate into the structure in units of μCi of Rn-222 per hour.

6.20 Radon measurements of the household water supply indicate an Rn-222 concentration of 2000 pCi per liter of water. Could the water supply alone account for the elevated radon levels in the home?

6.21 The radon progeny equilibrium factor for the home was determined to be 0.30 (unitless). Estimate the average potential alpha-energy air concentration in units of working level for this home.

6.22 Calculate the annual effective dose equivalent for a member of this household if the indoor radon concentration is 0.1 WL.

6.23 Assuming that the radon contribution from building materials used in constructing the home is negligible and that the household water is an insignificant source, what is the likely source of the elevated indoor radon? Name and briefly describe three mitigation techniques that you might recommend to reduce the home's radon levels.

Scenario 6.5

As an environmental health physicist at a national laboratory, you must evaluate your facility's radioactive releases and their impact upon surrounding population groups. This evaluation is complicated by the existence of natural sources of radiation. At your facility, the following natural sources are of concern: K-40, cosmic radiation, uranium series, and thorium series.

For each health physics task or decision noted below, briefly state how natural radiation sources may affect it.

6.24 Air monitoring.

6.25 Sample counting.

6.26 *In vivo* counting.

6.27 Radiation background measurements.

6.28 Calibration of low-level instruments.

6.29 Materials for construction and shielding of low-level counting facilities.

6.30 Radiochemical analyses including materials and equipment used.

Scenario 6.6

You are the health physicist at a facility that utilizes a large inventory of Kr-85. A mechanical failure has occurred, and Kr-85 gas is escaping into the environment via a 10-m stack.

The release began at 11:00 p.m. and is expected to last for 4 hr. On-site and off-site sampling teams are staged and available for use. The facility also has an environmental monitoring network that surrounds the plant site.

The release is occurring on a cloudless night, and the wind is blowing at 2.0 m/s to the east. The nearest resident in that direction lives 1000 m away.

Data

Ventilation system monitor (gas channel):
Normal value: 50 cpm
Current value: 1.0×10^7 cpm
The gas channel detector has a volume of 50 cm^3 and an efficiency of 0.3 cpm/dpm.

Ventilation system monitor (particulate channel):
Normal value: 50 cpm
Current value: 60 cpm

Ventilation system flow rate: 100 000 cfm

Sampling system flow rate: 10 cfm

$X\,\bar{u}/Q = 5.0 \times 10^{-4}/m^2$ @ 1000 meters

Dose factors for Kr-85 (rem m^3 Ci^{-1} s^{-1}):
Skin dose: 6×10^{-2}
Whole body: 4.7×10^{-4}

1 μCi = 2.2×10^6 dpm

1 ft^3 = 28.32 liters

6.31 Calculate the total body and skin dose to the nearest resident.
6.32 The shift supervisor is concerned about the radiological effects of the off-site doses. What are the off-site impacts and the actions that should be taken by the shift supervisor?
6.33 What would you recommend relative to activating the on-site and off-site monitoring teams?
6.34 What action would you initiate relative to the environment monitoring network, if any?

Scenario 6.7

Enormous States Power and Light has decided that it will build an advanced boiling water reactor (BWR) on the shores of Lake Erie. You have been asked to prepare a preliminary environmental impact assessment for the advanced BWR. The following questions are part of that report:

6.35 List 10 pathways for off-site exposure.
6.36 The BWR will utilize a once-through secondary cooling water system whose effluent is directly released to the lake, along with a gaseous effluent system equipped with a 30-min delay line, but will utilize no charcoal absorbers. Which pathway will be the dominant dose contributor to the population residing within 50 miles of the facility?
6.37 Outline some of the information that you should obtain in order to perform a more detailed evaluation of the relative importance of each pathway.

Scenario 6.8

You are the Environmental Compliance Manager at Omega Nanoelectronics (ON), an NRC licensed facility whose radiation protection program is based on the ICRP-26 methodology. The ON facility uses fission products in its manufacturing operations that produce nanoaccelerators for medical applications. Recently, a discharge line was discovered that allowed contaminated liquids to drain into Sammy Pond. The pond resides within the ON site boundary, and it is uniformly contaminated with Ru-106. Numerous raccoon tracks are present along the pond boundary.

Sample results yield a pond water concentration of Ru-106 of 9.25 Bq/L. The radiological half-life of Ru-106 is 1.02 year. Ru-106 decays yield a 0.039 MeV (max) beta. The Ru-106 ingestion ALI is 7 MBq.

Parameters associated with wildlife utilizing the pond's ecosystem are summarized in the following table.

Animal	Parameter	Value
Raccoon	Body (fresh) weight	8 kg
	Water intake rate	0.08 L/day per kg of raccoon body weight
	Effective loss rate constant for Rh-106 (k_{eff}) which includes biological and radiological losses	0.069/day
	Mouse consumption rate	0.002 kg of mouse/day per kg of raccoon body weight
	Rh-106 in body	30% in soft tissue 30% in bone 40% free
Mouse	Body (fresh) weight	20 g
	Body burden	0.74 Bq

6.38 The raccoon lives at the pond for a period of 1 year. Assuming the raccoon's only source of water is the pond and that he eats his daily fill of mice, calculate the Ru-106 concentration in the raccoon at the end of the year.

6.39 Calculate the steady state activity of Ru-106 if the raccoon lives at the pond for the rest of his life.

6.40 A house located adjacent to the site uses a well, which draws water from the pond. An adult consumes 2.2 L per day for 1 year from the pond. Calculate the committed effective dose equivalent to the adult. How does this number compare with natural background?

6.41 Calculate the biological half-life of Ru-106 in raccoons.

Scenario 6.9

You are a senior health physicist at the Lower Kaliforniastan People's Environmental Facility (LKPEF). The United Nations is assisting the LKPEF in recovering from a release of radioactive materials, and the environmental czar is concerned about previous environmental damage. She requests answers to the following questions addressing various aspects of environmental monitoring principles.

The LKPEF defines the Minimum Detectable Amount (MDA) by the following equation:

$$\text{MDA} = \frac{4.65\,\sigma_b}{K\,T} + \frac{3}{K\,T}$$

where

σ_b = the standard deviation of the background count
T = sample and background count time
K = counting efficiency

6.42 Assume that a Gaussian dispersion model applies to a gaseous release. For the following conditions, would the centerline airborne radioactivity concentration on the ground at the site boundary (1 km from the release point) increase, decrease, or remain the same? Justify your answer for each.

a. The effective release height is increased.
b. The wind speed increases.
c. The ground temperature, which initially exceeds the temperature at 100 meters above the ground at the beginning of the release, decreases below the temperature at 100 meters. For this question, assume that only the dispersion coefficients are affected by the lapse rate.
d. The gaseous effluent is heated above the ambient temperature and ejected from a vertical stack with a velocity exceeding that of the existing wind speed.

6.43 Assume that a release occurs through a pipe to the bottom and in the middle of a wide, shallow, straight river. For the following release conditions, will the downstream centerline radioactivity concentrations in the river increase, decrease, or remain the same. Justify your answer for each.

a. What is the effect on liquid radioactivity concentration if the temperature of the discharge is increased?
b. What is the effect on liquid radioactivity concentration from an increased velocity of the discharge?
c. What is the effect on the liquid radioactivity concentration from an increase in the river current?
d. What effect does time have on the concentration of radioactive material in the sediment on the river bottom?

6.44 A counting laboratory needs to establish the counting time of sediment samples for a given procedure and stated MDA. The laboratory is char-

acterized by a counting efficiency of 0.2 c/d, a background count rate of 85 cpm, a desired MDA of 3.7×10^{-3} Bq/g, and a sediment sample size of 500 g. If the counting time for both the background and the samples is the same, is a count time of 4 minutes sufficient to meet the desired MDA?

6.45 Laboratory analysis shows that a holding tank with 5.68×10^4 L of water contains Cs-137 at a concentration of 3.26×10^5 Bq/L. It is to be released over a 2-day period to a stream with a flow rate of 5.66 m^3/s. Calculate the dose to an individual who consumes 250 g of fish from the stream after this release if the bioaccumulation factor is 2000 and the ingestion ALI based on ICRP-30 guidance is 4 MBq.

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7
Accelerator Health Physics

Particle accelerators present unique challenges to the radiation protection professional. The primary beam can produce large exposure rates in localized areas, and the beam constituents often have large linear energy transfer (LET) values. There is a wide variety of beam constituents that range from conventional particles (protons and electrons) at extreme relativistic energies to heavy ions with enormous LET values.

The secondary radiations from these facilities consist of a variety of radiation types including bremsstrahlung, neutrons, scattered particles, electrons, and other leptons, hadrons, and spallation products. These produce large dose rates over large areas of the accelerator facility.

Accelerators include (a) electron linacs used in radiation therapy, (b) cyclotrons for producing Positron Emission Tomography (PET) radionuclides, and (c) the sophisticated machines used in basic nuclear and particle physics research. Each of these accelerator types has unique properties and associated sources of radiation. Three popular and illustrative accelerator types are the proton, electron, and heavy ion machines. These machine types generally illustrate the particle and radiation characteristics encountered in most accelerator facilities. A fourth accelerator type is a planned muon collider that has a number of unique characteristics including the need to evaluate neutrino contributions to the dose equivalent.

7.1
High-Energy Interactions

Particle accelerators are designed for a variety of purposes including basic research, production of radioisotopes, generation of bremsstrahlung for use in radiotherapy, fusion ignition, pumping of research lasers, and generation of synchrotron radiation. Each purpose requires a particular energy range and particle requiring acceleration. For a particular primary particle beam, the health physicist must understand the radiation fields produced by the beam and its associated interactions, because doses can exceed the lethal range in a short period of time.

Table 7.1 Properties of Selected Low-Energy Particles.

Particle	Mass (MeV)	Mean Lifetime	Dominant Decay Mode
ν_e	< 0.000002	> 300 s/eV[b]	[a]
$\overline{\nu}_e$	< 0.000002	> 300 s/eV[b]	[a]
ν_μ	< 0.19	> 15.4 s/eV[b]	[a]
$\overline{\nu}_\mu$	< 0.19	> 15.4 s/eV[b]	[a]
ν_τ	< 18.2	Not yet determined[b]	[a, c]
$\overline{\nu}_\tau$	< 18.2	Not yet determined[b]	[a, c]
e^-	0.511	$> 4.6 \times 10^{26}$ y	Stable
e^+	0.511	$> 4.6 \times 10^{26}$ y	Stable
μ^-	105.7	2.2×10^{-6} s	$\mu^- \rightarrow e^- + \nu_\mu + \overline{\nu}_e$
μ^+	105.7	2.2×10^{-6} s	$\mu^+ \rightarrow e^+ + \overline{\nu}_\mu + \nu_e$
τ^-	1777	2.9×10^{-13} s	Multiple decay modes
τ^+	1777	2.9×10^{-13} s	Multiple decay modes
π^-	139.6	2.6×10^{-8} s	$\pi^- \rightarrow \mu^- + \overline{\nu}_\mu$
π^0	135.0	8.4×10^{-17} s	$\pi^0 \rightarrow \gamma + \gamma$
π^+	139.6	2.6×10^{-8} s	$\pi^+ \rightarrow \mu^+ + \nu_\mu$
K^-	493.7	1.24×10^{-8} s	$K^- \rightarrow \mu^- + \overline{\nu}_\mu$
K^0	497.6	[d]	$K^0 \rightarrow \pi^+ + \pi^-$
K^+	493.7	1.24×10^{-8} s	$K^+ \rightarrow \mu^+ + \nu_\mu$
p	938.3	$> 2.1 \times 10^{29}$ y	Stable
$\overline{p}$	938.3	$> 2.1 \times 10^{29}$ y	Stable
n	939.6	885.7 s	$n \rightarrow p + e^- + \overline{\nu}_e$
$\overline{n}$	939.6	885.7 s	$\overline{n} \rightarrow \overline{p} + e^+ + \nu_e$

[a] Dependent on the degree of neutrino mixing.

[b] The Particle Data Group (2004) quoted specific lifetime values or noted that the lifetime was not yet determined. In 2006, the Particle Data Group did not quote specific lifetime values, but did note that the measured quantities depend upon the Standard Model's mixing parameters and to some extent on the experimental conditions (e.g., energy resolution).

[c] Decay mode not yet determined.

[d] The K^o particle is a superposition of two states K_S^0 and K_L^0;
$K^0 = \frac{1}{\sqrt{2}}(K_S^0 + K_L^0)$ with lifetimes of $K_S^0 = 8.95 \times 10^{-11}$ s and $K_L^0 = 5.11 \times 10^{-8}$ s.

7.2
Radiation Types

Particle accelerators exhibit considerable diversity in terms of the particles accelerated, their energy, and the energy and radiation types produced. A sample of the types of radiation produced in electron, proton, heavy ion, and muon accelerators are summarized in Table 7.1. These radiation types govern the properties of accelerators considered in this chapter.

Table 7.1 provides a summary of the properties of selected low-energy (< 2000 MeV) particles that are a potential health physics concern. These properties include the particle mass, mean lifetime, and dominant decay mode, and are provided for neutrinos (electron, muon, and tau), the electron (e^-) and its antiparticle (e^+), the muon (μ^-) and its antiparticle (μ^+), three pions (π^+, π^0, and π^-), three kaons (K^+, K^0, and K^-), the proton (p) and its antiparticle ($\overline{p}$), and the neutron (n) and its antiparticle ($\overline{n}$).

7.3
Proton Accelerators

If the accelerated particle is a proton, the physics of the interaction can be discussed in terms of energy regions including elastic scattering, inelastic scattering, and particle production. The particles produced by protons in each energy region and their associated health physics concerns are summarized in Table 7.2.

Elastic proton (p) scattering involves reactions of the type $p + X \rightarrow p + X$, where X is the target nucleus. In elastic scattering, the incident proton does not have enough energy to penetrate the target nucleus' Coulomb barrier and hence nuclear reactions do not occur.

Table 7.2 Proton Accelerators.

Beam Energy (MeV)	Region	Radiation/Particles Produced	Health Physics Concerns
< 6–8	Elastic scattering	Protons (range < 1 mm in most solids and < 1 m in air).	Direct exposure to the beam or scattered protons.
6–100	Inelastic scattering	Neutrons and nuclear fragments.	Neutrons dominate the shielding requirements.
> 100	Particle production	Pions (> 140 MeV); muons from pion decay; neutrons and protons.	Most particles are produced in the beam direction. As accelerator energies increase, muons become more important at increasingly larger angles.

As the energy increases, the proton will begin to penetrate the Coulomb barrier of the target. Binary reactions of the type $p + X \rightarrow b + Y$ can be generated, where b is an ejected particle or cluster and Y is the exit channel or residual nucleus. Tertiary production and multiple exit channel particles are also possible. Generally, reactions of the type $X(p, n)Y$ dominate the health physics concerns. Neutrons, having a range of energies, will need to be evaluated by the health physicist.

At proton energies beyond 100 MeV, reactions become more complex with multiple particle production becoming more common. At about 140 MeV, pion production begins. Because much of the radiation is in the beam (forward) direction, shielding must be more extensive along the beam direction. Because pions decay into muons, muon radiation must be evaluated for its impact on the shielding design.

Additional comments regarding the radiation characteristics of high-energy proton accelerators are provided in the subsequent discussion. This discussion focuses on the Large Hadron Collider that began operation in 2008.

7.4 Electron Accelerators

If the accelerated particle is an electron, it will also interact in a variety of ways depending upon its incident energy. Table 7.3 summarizes the health physics concerns for electron beams as a function of their incident energy. The radiation and particles produced by the interactions of the incident electron beam as a function of energy are also summarized in Table 7.3.

One of the more common electron interactions is the activation of accelerator components and structures. At electron accelerators, activation occurs through the production of secondary particles. For example, electrons produce photons and these photons produce neutrons through (γ, n) reactions. Neutrons and other particles are also produced as the electromagnetic radiation advances through a medium. As the energy increases, electron accelerators produce a variety of radiation types of health physics concern including photons, neutrons, heavy ions, pions, kaons, and muons. Prominent processes producing these particles include bremsstrahlung, synchrotron radiation, and the electromagnetic cascade.

7.4.1 Bremsstrahlung

Bremsstrahlung or braking radiation occurs when an ion's velocity changes. It is a general term applied to the radiation from an accelerated charged particle. Radiation is produced from either a positive or a negative change in velocity. A change in velocity or acceleration occurs when an electromagnetic field alters an electron's trajectory. During the change in trajectory, electron energy is lost, and its velocity decreases. Energy is conserved during this process through the

Table 7.3 Electron Accelerators.

Beam Energy (MeV)	Region	Radiation/Particles Produced	Health Physics Concerns
< 6	Low energy	Ionization and bremsstrahlung (X-rays).	The primary electron beam and photons it produces must be shielded. Backscattered electrons may be important within shielded enclosures.
6–50	Giant resonance (GR)	Electrons; photons produced by electrons excite the nucleus and produce neutrons.	Bremsstrahlung is the dominant source of radiation, but neutron radiation requires hydrogenous shielding such as concrete. As the energy increases, the bremsstrahlung is increasingly forward peaked. At threshold, giant resonance neutrons are produced isotropically. For exposures inside the shield, electrons are a major concern.
30–150	Intermediate energy	Neutrons and bremsstrahlung.	The neutron production cross-section is lower than that in the GR region, and bremsstrahlung still dominates the shielding considerations. For exposure inside the shield, electrons and bremsstrahlung will be the major contributors to the dose.
> 150	High energy	Pions, muons, and neutrons.	For shields thicker than about 120 cm, neutrons become the primary design concern. Bremsstrahlung is the major source of radiation inside the shield.

emission of photon radiation as the electron decelerates. The spectrum of the bremsstrahlung photons is a continuous function of energy.

The total instantaneous power (P) radiated by the accelerated electron is:

$$P = \frac{2\,e^2\,a^2}{3\,c^3} \tag{7.1}$$

where e is the electron's charge, a is the instantaneous acceleration of the electron, and c is the speed of light. Equation (7.1) provides no information regarding the distribution of photon energy relative to the direction of the electron's velocity. If θ is defined as the angle between the electron's initial velocity and the final electron direction resulting from the trajectory change, the variation of the radiated power as a function of the spherical solid angle Ω is:

$$\frac{dP}{d\Omega} = \frac{e^2\, a^2 \sin^2\theta}{4\,\pi\, c^3\,(1 - \beta \cos^5\theta)} \tag{7.2}$$

where

$$\beta = \frac{v}{c} \tag{7.3}$$

and v is the initial electron velocity.

At low energies ($\beta << 1$), the radiated power distribution has a $\sin^2\theta$ dependence, which peaks near 90° relative to the direction of motion. However, at high energies ($v \rightarrow c$), the angular distribution of radiated power is tipped in the beam direction and increases in magnitude. The angle θ_{max} for which the radiated power intensity is a maximum is:

$$\theta_{max} = \cos^{-1}\{[(1 + 15\,\beta^2)^{1/2} - 1]/3\,\beta\} \tag{7.4}$$

For electrons with $\beta = 0.1$, 0.5, 0.9, and 0.999, θ_{max} is 76.0°, 38.2°, 13.4°, and 1.28°, respectively. Equation (7.4) illustrates the fact that the direction of emitted particles moves toward the beam direction (0°) as the incident particle's energy increases. This general result is important in accelerator shield design.

7.4.2 Synchrotron Radiation

When discussing photon sources, synchrotron radiation is the term normally applied to the radiation emitted from a relativistic electron $\gamma >> 1$,

$$\gamma = \frac{1}{\sqrt{1 - \beta^2}} \approx 1957\, E\,(\text{GeV}) \tag{7.5}$$

where γ is the Lorentz factor and E is the electron energy in GeV. If the electrons are moving at close to the speed of light, two effects alter the nature of the radiation. First, a particle moving with a Lorentz factor γ toward an observer emits radiation into a cone of opening angle θ:

$$\theta \approx \frac{1}{\gamma} \tag{7.6}$$

Second, if a source moves at a velocity near c and emits photon pulses, the photon emitted at the end of the pulse almost overtakes the photon from the start of the pulse. This shortens the pulse from a single accelerated particle. The net result of these two effects is the production of very high frequency synchrotron radiation with a continuous spectrum emitted into the narrow cone.

7.4.3
Electromagnetic Cascade

An electromagnetic cascade or shower induced by an electron or positron is a sequence of bremsstrahlung and pair production processes. The cascade originates when bremsstrahlung is produced from the deceleration of an electron or positron. The bremsstrahlung photons initiate pair production followed by additional bremsstrahlung that rapidly disperses the incident electron's kinetic energy into an array of photons, electrons, and positrons. A photonuclear reaction is initiated by a cascade photon and produces secondary neutrons through (γ, n) reactions.

Once an electron's energy falls below a critical energy (E_{crit}), it no longer plays an important role in the electromagnetic cascade shower. Values of E_{crit} are characteristic of the medium traversed by the particle and are given by the approximation:

$$E_{\text{crit}} \approx \frac{800}{Z + 1.2}\,\text{MeV} \tag{7.7}$$

where the critical energy is expressed in MeV and Z is the atomic number of the attenuating medium.

Electromagnetic cascade showers are an important health physics consideration in high-energy electron accelerators. Showers ensure that the electron/positron energy is dissipated within a reasonable distance.

7.5
Light Sources

The term light source is used to describe the collection of photon-generating devices. Light sources probe the structure of matter over a broad frequency range, and their applications continue to expand as the available intensity and range of wavelengths increase.

Both synchrotron light sources and free electron lasers (FELs) are based on the concept that an accelerated charge radiates photons. The output from these photon sources is intense and presents an external radiation hazard that must be carefully managed.

Photon light sources have evolved considerably since the discovery of X-rays. Light sources are based on applications of electromagnetic field acceleration of electrons. Photon light sources include bending magnets, wigglers, undulators, and free electron lasers. Bending magnets, wigglers, and undulators are often associated with storage rings, and free electron lasers are based on linear accelerators. Free electron lasers are addressed in Chapter 8.

7.6 Heavy-Ion Accelerators

The term *heavy ions* refers to nuclei heavier than hydrogen that are to be accelerated. Ion beams as heavy as uranium have been accelerated in nuclear physics applications. Beams composed of Li-7, O-16, N-16, or S-32 ions have been utilized in a wide variety of research activities.

In a heavy-ion accelerator, neutrons dominate the radiation field outside the biological shielding for beam energies above the Coulomb barrier. Unlike proton and electron machines, there does not yet exist a large body of experimental source term data to compile a simple table of health physics concerns for heavy-ion accelerators. For light ions (such as H-2, H-3, or He-4) of several hundred MeV per nucleon, the dose equivalent outside a thin shield or at forward angles may be dominated by neutrons of energy greater than 20 MeV.

In the initial phase of facility design, the accelerator health physicist must provide input into the shielding design. In order to shield a low-intensity heavy-ion beam, it may be acceptable, in the absence of a complete knowledge of the source term, to calculate the neutron shielding requirements based on the average nucleon energy in the beam. However, a high-intensity heavy-ion beam will require a much thicker shield. The neutrons of lower energy will be attenuated in the first few layers of the shield so that the high-energy neutron component will govern the required shielding thickness.

For low-energy heavy-ion beams, the health physicist must determine if the incident heavy ion has sufficient energy to penetrate the target nucleus' Coulomb barrier and to produce neutrons. For ions of mass greater than that of the proton, the energy (in MeV) below which only Coulomb interactions can occur is given by Adler's relationship:

$$E_{\text{Coul}} = \frac{Z_1 \, Z_2 \, (1 + A_1 / A_2)}{A_1^{1/3} + A_2^{1/3} + 2} \tag{7.8}$$

where Z_1 and Z_2 are the charge of the heavy-ion beam and target nucleus, respectively, and A_1 and A_2 are their respective mass numbers. Adler also provides an approximate relationship between the charge and mass number for stable nuclei:

$$Z = 0.487 \, \frac{A}{1 + A^{2/3}/166} \tag{7.9}$$

7.7 Large Hadron Collider

The Large Hadron Collider (LHC) is the world's highest energy accelerator. It began initial operation in 2008, and collides protons in a 27 km circumference tunnel. Each proton beam has an energy of 7 TeV leading to a total collision energy of 14 TeV. The LHC also collides heavy ions such as lead with a collision energy of about 2.76 TeV/nucleon. These energies are sufficient to produce a variety of radiation types including antiprotons, protons, neutrons, muons, hadronic cascade particles, heavy ions, and synchrotron radiation.

7.7.1 Antiprotons

Although the LHC is not currently configured to accelerate antiprotons, these particles are included in the discussion for completeness. However, only those aspects of the radiation field emitted by proton–antiproton or nucleon–antiproton interactions that are important for radiation protection purposes are emphasized.

Momentum conservation requires the emission of at least two particles from each annihilation event. For antiproton–proton annihilations occurring at rest, the total energy available is twice the rest energy of the proton or 1876.5 MeV. From a health physics perspective, the dominant species that escape from the shield or scatter in air are protons, neutrons, pions, and photons. Muons also result from the decay of charged pions.

The radiation field from pion decay is composed of two components. There is a photon component from the decay of neutral pions, and a charged particle component due to the decay of charged pions. In addition to the radiation components derived from pion decay, there is also proton and neutron radiation. Neutrons dominate the radiological and shielding considerations.

The radiation dose is directly related to the number of annihilation events. Since antiprotons are more difficult to create and store, the number of antiprotons governs the annihilation rate, and hence the radiological considerations.

7.7.2 Proton Reactions

Above about 500 MeV, proton reactions produce secondary radiation types including pions, kaons, muons, positrons, and electrons. At facilities with multiple beam lines, care must be taken to ensure the beam only enters active areas configured for beam acceptance. Therefore, safety interlocks and beam-path control are needed to ensure that the proton beam does not enter areas occupied by personnel.

7.7.3 Neutrons

Normally, neutrons are the dominant prompt radiation hazard at proton accelerators above 10 MeV. In view of the interaction mechanisms of the electromagnetic and hadronic cascades, the shield preferentially removes photons and charged particles so they usually account for only a small contribution to the total effective dose outside the shield. Neutrons normally dominate the effective dose.

For proton energies between 200 MeV and 1 GeV, an increased number of nuclear reactions occur. Highly excited compound nuclear states decay by the emission or evaporation of neutrons that lead to the development of hadronic cascades. Other particles and light nuclei are also emitted through evaporation, but these radiation types are readily absorbed in accelerator shielding. Protons and neutrons are produced in roughly equal numbers. As their energy increases, protons are of increasing importance from a radiation protection standpoint.

Above energies of 1 GeV, neutron production still occurs. In addition, enhanced numbers of secondary particles are produced. Both neutrons and other particles initiate hadron cascades in shielding that produce radiation sources extending to a larger spatial volume than the trajectory volumes of either the primary or secondary particles. In the energy region of a few tens of GeV, measurements of the angular distribution of hadrons (principally neutrons, protons, and pions) confirm that, as the energy of the incident particle increases, particle production in the beam direction becomes more pronounced.

7.7.4 Muons

Above proton energies of about 300 MeV, the production of charged pions becomes important. Above 1 GeV, charged kaons are produced. As the proton energy increases beyond these thresholds, pions and kaons are produced by both the primary particles and by energetic secondary particles present in the hadronic cascade.

Both pions and kaons have short half-lives and rapidly decay into muons and neutrinos. Muons and other leptons do not interact via the strong interaction. To first order, charged leptons are attenuated through ionization energy loss mechanisms. For high energies, the muon range becomes quite large, and this range precludes the construction of reasonably sized shields unless they are constructed using soil.

7.7.5 Hadronic (Nuclear) Cascade

The hadronic cascade is an important consideration in determining the shielding of high-energy machines. In many cases, the nuclear cascade is the most important means of transporting radiation through the shield.

At a proton accelerator, the hadronic cascade is initiated when the beam interacts with accelerator or extraction system components. At electron accelerators, high-energy electrons produce hadrons from photon-induced reactions. These reactions include the photodisintegration of (*n*–*p*) pairs within the nucleus and photoproduction of pions that are then reabsorbed within the nucleus. The emitted neutrons and protons initiate a hadronic cascade. The six processes of the hadronic cascade are illustrated in a tabular form in Table 7.4 and include muon production, the electromagnetic cascade, the intranuclear cascade, the extranuclear cascade, evaporation of nucleons and nuclear fragments, and induced activity.

Table 7.4 Processes of the Hadronic Cascade.

Process	Initial Particles Produced as a Result of the Incident Hadron Collision	Time Scale (s)	Typical Energy per Particle (MeV)	Percentage of Energy Deposition
Muon Production	$\pi^+ \rightarrow \mu^+ + \nu_\mu$ $\pi^- \rightarrow \mu^- + \overline{\nu}_\mu$ $K^+ \rightarrow \mu^+ + \nu_\mu$ $K^- \rightarrow \mu^- + \overline{\nu}_\mu$	10^{-8}	any	10
Electromagnetic Cascade	$\pi^0 \rightarrow \gamma + \gamma$ $\gamma \rightarrow e^+ + e^-$ $\gamma \rightarrow \mu^+ + \mu^-$ (γ, n) reactions	10^{-16}	any	20
Intranuclear Cascade	Spallation products dominated by protons, neutrons, pions, and kaons.	10^{-22}	< 200	30
Extranuclear Cascade	Nuclear collisions produce a variety of radiation types including protons, neutrons, pions, and kaons.	10^{-23}	> 200	30
Evaporation of Nucleons and Nuclear Fragments	Evaporation products are dominated by protons, neutrons, deuterons, and alpha particles.	10^{-19}	< 30	10
Induced Activity	Activation reactions lead to the emission of radiation types dominated by photons, beta particles, and alpha particles.	Seconds to years	< 10	< 1

Derived from NCRP-144 (2003) and ICRU (1978).

In Table 7.4, a cascade or sequence of reactions is initiated by an incident hadron. Energetic protons that strike a target make multiple collisions with nucleons in a nucleus, causing spallation. This process is the intranuclear cascade and dominantly produces protons, neutrons, pions, and kaons.

High-energy particles such as neutrons and protons, emitted in the course of the hadronic cascade process, collide with other nuclei, causing additional reactions. This process is referred to as an extranuclear cascade. The residual nuclei of the cascade are in an excited state, and yield additional neutrons (evaporated neutrons). In this process, about ten fast neutrons are emitted for each proton initiating the cascade.

7.8 Muon Colliders

Muon colliders are conceptual accelerators that have unique radiation characteristics. A muon collider provides a platform for colliding beams of muons (μ^-) and antimuons (μ^+). The muon collider facility energy is usually expressed as the sum of the muon and antimuon energies. For example, a 100 TeV accelerator consists of a 50 TeV muon beam and a 50 TeV antimuon beam.

In a muon collider, neutrinos are produced from the decay of muons:

$$\mu^- \rightarrow e^- + \nu_\mu + \overline{\nu}_e \tag{7.10}$$

$$\mu^+ \rightarrow e^+ + \overline{\nu}_\mu + \nu_e \tag{7.11}$$

Since muon colliders produce large muon currents, neutrinos are copiously produced from the decay of both muons and antimuons. Muon colliders can be constructed using either a pair of linear accelerators with intersecting beams or a storage ring that circulates the muons and antimuons in opposite directions prior to colliding the two beams.

7.8.1 Bounding Neutrino Effective Dose – Linear Muon Collider

The bounding neutrino effective dose from a linear muon collider is based on a limiting condition from a circular accelerator with a number of straight sections as part of the facility. Using the narrow-beam approximation, the effective dose is independent of distance (L) for $L < 5\ E_0$ where L is expressed in km and E_0 in TeV. Using this approximation and assuming that the muon beam is collimated, the annual neutrino effective dose (mSv/y) is:

$$H = \frac{K}{2}\frac{N}{g} X(E_0/2)\, E_0^2 \tag{7.12}$$

where

K = conversion factor = 6.7×10^{-21} mSv-GeV/m-TeV^2
N = number of muon decays per year at the accelerator
g = accelerator gradient (GeV/m)
$X(E_0/2)$ = dimensionless cross-section factor as parameterized in Bevelacqua (2008)
E_0 = muon beam energy (GeV)

As a practical example, consider a 1000 TeV muon linear accelerator assuming $E_0 = 500$ TeV (i.e., two, 500 TeV linear muon accelerators) and $N = 6.4 \times 10^{18}$ muon decays per year. Using these values in Equation (7.12) with a $g = 1$ GeV/m value leads to:

$$H\,(2 \times 500\ \text{TeV}) = 1.4\ \text{Sv/y} \tag{7.13}$$

which is a significant neutrino effective dose value that cannot be ignored. Neutrinos exhibit a long, thin conical radiation plume that presents a radiation challenge well beyond the facility boundary.

7.8.2 Bounding Neutrino Effective Dose – Circular Muon Collider

The calculation of neutrino effective doses from a circular muon collider is dependent upon the assumed physical configuration and beam characteristics (E_0 and N). An estimate of the neutrino effective dose for a circular muon collider is:

$$H = \frac{10^{-4}\ E_0^4\ N}{6\ \pi\ r^2}\ \frac{\text{mSv-cm}^2}{\text{TeV}^4} \tag{7.14}$$

where N is the number of muon decays per year, E_0 is the muon energy in TeV, r is the distance from the point of muon decay, and H is the annual neutrino effective dose in mSv.

For consistency with the linear muon collider assumptions, 6.4×10^{18} muon decays per year are assumed in the following example. A 25 TeV circular muon collider produces a neutrino effective dose of 37 mSv/y at a distance of 1500 km from the facility. Although the neutrino effective dose plume only has a diameter of 12 m at 1500 km, it presents a radiation challenge for muon collider health physicists and management. The magnitude of the neutrino effective dose merits attention and warrants emphasis of the ALARA Principle.

7.8.3 ALARA Impacts of Muon Colliders

Neutrinos are electrically neutral and usually interact through the weak interaction. Their small, but nonzero, interaction cross-section creates a unique situation in terms of the behavior of the neutrino effective dose, particularly in terms of the shape and energy dependence of their radiation profile. These properties lead to a modification of the ALARA concepts when they are applied to muon colliders.

The basic ALARA Principle suggests the effective dose at a given location is reduced if the exposure time is reduced, the distance from the source is increased, or shielding is added between the source and the point of interest. This ALARA Principle must be modified at a TeV energy muon collider. The ALARA Principle of time is still valid for muons and neutrinos. Decreasing the exposure time reduces the neutrino and muon effective doses.

The ALARA Principle of distance is ineffective when neutrinos are involved. Since neutrinos interact very weakly, relatively long distances are not effective in significantly reducing the neutrino effective dose. In fact, the neutrino beam remains a hazard for hundreds of kilometers. However, distance is still effective for reducing the muon effective dose.

Unlike other radiation types, shielding neutrinos is anti-ALARA. The magnitude of the particle showers produced by neutrino interactions is governed by the quantity of shielding material between the neutrino beam and the point of interest. However, shielding muons is an effective ALARA measure.

From the standpoint of TeV energy neutrino radiation, a linear muon collider has advantages over circular muon colliders. The radiation is confined to two narrow beams that can be oriented to minimize the interaction of the neutrinos. A simple ALARA technique would be to orient the linear accelerators at an angle such that the neutrino beams would exit the accelerator above the ground. This would minimize the residual neutrino interactions with the Earth or man-made structures. In addition, the spent muons can be removed from the beam following collisions or interactions before they decay into high-energy neutrinos.

7.9 Radiation Types of Concern

Previous discussion addressed electron accelerators, proton accelerators, heavy ion accelerators, and muon colliders. At each of these accelerator types, the radiation environment results from a range of physical phenomena and consists of several radiation types distributed over a broad range of energies. The radiation fields are also time dependent and this dependence is affected by the accelerator duty cycle, details of the accelerator system, and beam extraction system. Accelerator radiation fields are complex, but their description can be simplified in terms of three basic rules:

1. If muons are produced, neutrons are also produced. However, the locations of muon production and neutron production do not necessarily coincide.
2. Very high-energy neutrons are accompanied by high-energy (fast), intermediate-energy, and thermal neutrons.
3. Neutrons are accompanied by photons.

7.9.1 Residual Radioactivity

Residual radioactivity includes the activity induced within the target or within accelerator structures or components. It includes the activation of water, air, soil, and any oils or greases utilized in the vicinity of the beam or its scattered radiation.

Residual radioactivity may make up an important part of the radiation field inside the shield. Whenever the electron energy is greater than the binding energy of a nucleon, or when a proton can penetrate the Coulomb barrier of the nucleus, a radioactive nucleus may form. Residual radioactivity is not as great in electron machines as in proton accelerators because most of the primary electron energy goes into ionization and production of photons and electron–positron pairs. Examples of various residual radioactivity at accelerator facilities are summarized in Table 7.5.

Table 7.5 Residual Activity at Accelerator Facilities.

Accelerator Component or Material	Accelerator Type	Isotopes Produced
Iron magnet yokes	Proton	Mn-54 Co-60 (for certain alloys)
Copper coils	Proton	Zn-65, Co-57, Co-58, and Co-60
Iron structures and components	Electron	Fe-55 via the (γ, n) reaction
Greases and oils	Proton or electron	H-3
Water	Proton or electron	O-15, N-13, C-11, Be-7, and H-3
Air	Proton or electron	O-15, N-13, and C-11
Soil	Proton or electron	H-3 and Na-22

7.9.2 Activation of Water

Water exposed to high-energy radiation will become radioactive. The major source of activated water will be the water used to cool beam dumps, collimators, targets, magnets, and other components requiring cooling water. Commonly produced radionuclides (half-lives in parentheses) include O-15 (2 min), N-13 (10 min),

C-11 (20.4 min), H-3 (12.3 years), and Be-7 (53.3 days). These radionuclides are produced by both electron and proton machines.

O-15, N-13, and C-11 are all short-lived positron emitters. If they are released to the air, they do not present an inhalation hazard, but only an external dose hazard due to their annihilation radiation. Being positron emitters, their 0.511-MeV gammas present a potent external concern. These annihilation photons can produce high radiation levels around cooling water systems,

Be-7 is longer-lived (53.3 days) and may be removed by ion-exchange resins. In some cases the resin beds, used to trap the Be-7, become sources of exposure concern and may require shielding.

Circumstances may arise that require cooling water to be immediately sent to a heat exchanger. High-power levels or component-cooling restrictions may warrant such an operational requirement. Normally, heat exchangers are shielded with lead or iron. This design approach may present an exposure problem if N-17 (4.17 s) is produced because it is a neutron emitter. Concrete shielding should be utilized if N-17 production becomes a design consideration.

H-3 is the only long-lived radionuclide produced in water that cannot be removed by ion-exchange resins. Therefore, tritium will continue to build up in a water system at a rate governed by the intensity of the radiation producing it, natural decay, and water leakage which removes it. The health physicist should provide guidance regarding the tritium-sampling frequency and criteria for disposal of the contaminated water.

7.9.3 Activation of the Soil

High-energy (> 10 MeV) particles produce both H-3 and Na-22 in soil. Tritium will become part of the ground water and can be monitored by liquid scintillation counting. The transport of any leached radionuclide from the soil can be a lengthy process. Therefore, it is important to monitor the soil around accelerator facilities. Na-22 may be detected in soil samples by gamma-ray detectors such as Ge(Li). Facility design can significantly reduce soil activation by adequately shielding the target and beam dump areas with concrete and steel.

7.9.4 Activation of Air

Spallation reactions from high-energy beams passing through air produce radionuclides similar to those created in water (O-15, N-13, and C-11). Where air activation is a concern, it is a good practice to delay entry into areas traversed by the primary or secondary beam for a time appropriate to the half-lives and concentrations of these radionuclides. Because these nuclides are all positron emitters, the hazard is primarily an external one due to the 511-keV annihilation photons.

7.9.5
Buildup of Radioactive and Toxic Gases in an Irradiation Cell

Accelerator-induced gaseous activity will build up during facility operation. The concentration of the gaseous radioactivity will decrease after the beam current is reduced or after accelerator shutdown. The activity in the vicinity of the irradiation cell will vary after shutdown following a relationship that depends on the ventilation rate and time after beam shutdown:

$$C(t) = C(0) \exp\left[-(v/V + \lambda)\, t\right] \tag{7.15}$$

where

$C(t)$ = concentration of the radioactive gas at time t
$C(0)$ = concentration of the radioactive gas at time $t = 0$
v = exhaust velocity (m^3/s)
V = cell volume (m^3)
t = time after shutdown (min)
λ = $0.693/T_{1/2}$ = disintegration constant of the radioactive gas
$T_{1/2}$ = half-life of the radionuclide in the chamber

Similarly, the concentration of the toxic gas varies after shutdown according to the relationship

$$Z(t) = Z(0) \exp\left[-(v/V + 1/T)\, t\right] \tag{7.16}$$

where

$Z(t)$ = concentration of the toxic gas at time t
$Z(0)$ = concentration of the toxic gas at time $t = 0$
T = mean lifetime of the toxic gas (min)

Toxic gases include ozone, nitrous oxides, or sulfur oxides that arise from the air or from materials in the irradiation cell.

7.9.6
Other Radiation Sources

The aforementioned sources of radiation at accelerator facilities are not the only sources of concern to the health physicist. Other sources of radiation, such as klystrons, radiofrequency (RF) equipment, high-voltage power supplies, or experimental equipment in adjacent areas, should be evaluated for their radiological hazard. These sources are more difficult to control than the primary or scattered accelerator radiation because the health physicist may not be aware of their use, the experimenter utilizing this equipment may not be aware of the hazard, or the radiation source is at least partially masked by the accelerator output.

Whenever high-voltage or RF equipment is utilized, there is a strong possibility of X-ray production. The X-ray source can exist even if there is no heated filament

or other obvious source of electrons. Because the physics of these X-ray fields is not completely understood, examples from accelerator facilities will illustrate the magnitude of these radiation hazards.

In an accelerator test, high RF power was applied to a 25-cm section of a standing wave accelerator. A dose rate of 6100 rad/hr (17 mGy/s) resulted at 140 cm from the centerline of the accelerator section when 35 MW of RF power at 120 pulses/s was applied. The resultant X-rays produced a dose of 10 mrad/hr (28 nGy/s) at a distance of 2 m after penetrating through 4 inches of lead and 4 ft of concrete.

On the axis of the accelerator section, the dose rate was much higher. At RF fields greater than 20 MV/m, stray electrons can be continuously accelerated. In the experiment, the accelerated beam melted a hole in a stainless steel plate at the end of the accelerator section.

At another facility, a particle beam separator containing a pair of 400-kV, 1-mA high-voltage units required maintenance due to voltage difficulties. The separator is normally interlocked to exclude personnel entry. However, the maintenance work required the interlock to be temporarily bypassed. During the maintenance troubleshooting at 50 kV, a radiation survey revealed no X-ray production.

During the repair work, a wiring error caused the separator's Cockcroft Walton high-voltage stack to operate at about 400 kV instead of the indicated 50 kV. This error was identified by the sound of sparking from the separator. A follow-up radiation survey showed dose rates of about 1500 rad/hr (4.2 mGy/s) at about 1 ft from the separator's surface centerline.

SLAC-327 provides these and other examples of X-ray hazards which are summarized in Table 7.6.

Table 7.6 Unanticipated X-ray Sources at Accelerator Facilities.

Device	Power/Voltage	Radiological Hazard
RF cavity at a storage ring	200 kW	500 mrad/hr[a] @ 1 m (1.4 μGy/s)
	350 kW	8 rad/hr[a] @ 1 m (22 μGy/s)
Secondary emission test device	110 kV dc	160 rad/hr @ 10 cm from glass viewing port (444 μGy/s)
Doubler RF cavity	65 kW	5 rad/hr @ 1 foot (14 μGy/s)
Three klystrons with end caps removed	50 MW	1700–3600 rads/hr @ 8 cm from end cap (4.7 to 10 mGy/s)
Resonating microwave waveguide driven by a klystron	17.5 MW	300 rad/hr @ 6 cm from the waveguide (833 μGy/s)
A 20-kJ KrF ultraviolet laser	Pumped by a 1.7-MeV, 40-kA, 20-ns pulsed electron source	300 mrad per pulse @ 15 ft (3 mGy)

[a] Dose increased as the fifth power of the RF power.
Source: SLAC-327 (1988).

The optimum way to control these X-ray exposures is to educate the experimenters of the potential hazards of their high-voltage and RF equipment. This can be accomplished through facility initial and periodic training. However, the best approach is for the health physicist to be cognizant of activity in the workplace that can generate a radiological hazard. He or she should frequently tour facility areas and become a useful partner in assisting the researcher in safely conducting his or her experiments.

7.10 Shielding

Shielding is a common design tool utilized to reduce radiation exposures to facility workers and the general public. Shield design is a complex task, frequently utilizing computer models, and this subject will only be addressed in general terms in this chapter. The shielding requirements will depend on a number of factors, including the time radiation workers or the public are exposed to the radiation sources, accelerator radiation fields, induced activity, and miscellaneous radiation sources.

The Department of Energy recommends that shielding for new facilities limit worker exposure from leakage radiation to 1 rem/year (10 mSv/year). Moreover, the facility's boundary dose is limited to 100 mrem/year (1 mSv/year).

For most accelerators, concrete is the preferred shielding material. Concrete is often selected because it has a reasonable density, high hydrogen content, and low cost, is a good construction material, and has good structural properties. Typical concrete densities are in the range of 2.2 to 2.4 g/cm^3.

Heavy concrete can be utilized in special circumstances by using an aggregate of iron, magnetite, or barite. Densities as high as 6.5 g/cm^3 have been attained. The use of heavy concrete yields an increased linear attenuation for photons, charged particles, and high-energy neutrons. However, heavy concrete normally contains less hydrogen, which leads to a decreased linear attenuation for neutrons with energies below a few MeV.

As a shielding rule of thumb, neutron energy equilibrium is achieved and then remains constant after one or two attenuation lengths of shield material. Therefore, for shields thicker than a few attenuation lengths, the dose equivalent outside concrete and iron shielding will be attenuated with attenuation lengths of 120 g/cm^2 and 145 g/cm^2, respectively.

Neutron energy loss by elastic scattering requires a hydrogenous shield to maximize the energy transfer as the neutrons slow down. At energies above 10 MeV, inelastic processes are effective in attenuating the neutrons. Iron can also be used to shield the higher-energy (> 10 MeV) neutrons if it is followed by a hydrogenous material. The iron degrades the neutron spectrum by reducing its energy, while the hydrogenous material will remove the lower-energy neutrons that result from inelastic scattering with the iron nuclei. Because there is no effective removal mechanism by lower-energy (< several hundred keV) neutron interactions in iron, an iron shield will be nearly transparent to these lower-energy neutrons.

Table 7.7 Qualitative Radiation Characteristics Outside Accelerator Shielding.

Accelerator	Shielding	Dominant Shielding Consideration
High-energy proton	Very thick	Neutrons at large angles Muons at small angles
High-energy electron	Thin to moderately thick	Photons
	Very thick	Neutrons at large angles Muons at small angles

A qualitative description of the radiation characteristics dominating the accelerator shielding design is summarized in Table 7.7. Although the beam and its scattered radiation control the shielding design, this source is not the dominant dose contributor to personnel.

This situation is analogous to the power reactor shielding design. The most intense source of radiation, the nuclear fuel core, is heavily shielded, but the greatest dose is derived from exposure to activation sources during maintenance operations.

In an accelerator, the radioisotopes Cu-64, Co-56, Co-57, Co-58, Co-60, Fe-59, Mn-54, Mn-56, Na-22, and Na-24 produce most of the personnel dose, regardless of whether the source is a proton or neutron machine. All these activation products are induced in the accelerator's components and support systems.

Detailed measurements outside thick shields are difficult to perform. Table 7.8 presents the composition of radiation fields above thick shields at the CERN Proton Synchrotron. The CERN measurements are quite complex because the generated neutrons have energies from the eV to the GeV range.

Table 7.8 Composition of Radiation Fields above Thick Shields at CERN.

Radiation	Percentage of Dose Equivalent (%)	
	Above Concrete Shield Bridge	Above Target Through Earth Shield
Thermal neutrons	11–12	< 1–3
Fast neutrons (0.1–20 MeV)	50–70	10–37
High-energy particles (E > 20 MeV)	2–25	52–89
Gamma rays and ionization from charged particles	2–19	1–13

Source: Rindi and Thomas (1973).

7.11
Accelerator Beam Containment

The accelerator beam, like the core of a nuclear reactor, must be controlled to ensure that the facility is operated in a radiologically safe manner. Specifically, the beam must arrive at its designated target location and deposit its energy there. If the beam strikes another location or burns through the target and strikes another area, the facility design assumptions are challenged and unanticipated radiological conditions may be created. These radiological conditions may include very high radiation levels in unprotected areas. Examples of such errors at accelerator facilities are summarized in Table 7.9.

The events of Tables 7.6 and 7.9 actually occurred and produced radiation levels up to the lethal range. The health physicist must be on the alert for these types of situations, must ensure that their occurrence is detected in a timely manner, and must institute corrective actions to preclude their recurrence.

Problems of beam containment are a joint responsibility shared by the health physics, accelerator operations, and beam line design groups. The health physicist must ensure that these groups clearly communicate to avoid the types of errors noted in Tables 7.6 and 7.9.

Table 7.9 Accelerator Beam Containment Errors.

Design Condition	Error/Radiological Consequences
A lightly shielded experimental hall contains a target that will absorb about 1% of the beam power. The remaining beam energy passes through the target and is absorbed in a shielded beam dump.	Because of an alignment error, the beam strikes the target's housing. Radiation levels increase by a factor of 10–100.
An experiment is designed for the beam to strike the target and to produce secondary particles or photons. The beam is then directed by a magnet into a beam dump.	As a result of a magnet failure, the beam passes out of the design area to an area occupied by personnel.
The beam is intended to dump its energy into a beam stop after its interaction with the target.	Because of a personnel error, the beam strikes an inadequate beam stopper, bums through the stopper, and then enters an occupied area.
A new beam line is designed to include a magnet as part of its beam confinement package.	On initial startup, the magnet was incorrectly connected. The 30-W beam struck the outer shielding wall and produced a dose rate > 360 rad/hr (1 mGy/s) outside the shield wall.

Source: SLAC-327 (1988).

The containment of accelerator beams is normally accomplished by utilizing a combination of mechanical and electronic interlock systems. Mechanical components include slits, collimators, magnets, beam stoppers, and beam dumps.

7.12 Dose Equivalent Rate from the Accelerator Target

The neutron dose equivalent rate at a distance r away from the target along the beam centerline can be formulated if the distribution is isotropic and the target is considered to be a point source of radiation:

$$\dot{H} = I\,K\,k_1\,P\,(\mathrm{DCF})/(4\,\pi\,r^2) \tag{7.17}$$

where

$\dot{H}$ = dose equivalent rate (rem/hr)
I = proton beam current (A)
k = charge/proton = 1.602×10^{-19} coulombs/proton
K = $1/k = 6.24 \times 10^{18}$ protons/A-s
k_1 = time conversion factor (3600 s/hr)
P = neutron production rate (neutrons/proton)
DCF = dose conversion factor (rem-cm^2/n)
r = distance from the target (cm)

7.13 Beam Current

A key health physics parameter that is utilized as an input to dose assessments is the beam current. It may be determined from the more readily available parameters such as the power and accelerator voltage. The beam current in an accelerator can be determined from the relationship

$$P = I\,V \tag{7.18}$$

where

P = beam power (watts)
V = accelerator voltage (volts)
I = beam current (amperes)

The production rate of various particles is often related to the beam current.

7.14 Pulsed Radiation Fields

The term "pulse" has different meanings in discussions of accelerator radiation fields. "Pulse" may refer to either beam or detector characteristics. The time that the accelerated beam interacts with the target will be denoted as the "beam pulse". It is also referred to as "pulse", "burst", and "spill". "Detector pulse" refers to instruments when used to detect discrete events and electronically process them.

Measurements in pulsed radiation fields will be influenced by the instantaneous intensity of the field, its duration, cycle time, and characteristics of the detector and its associated electronic circuitry. The amount by which the instantaneous or peak intensity of the radiation field exceeds its time-average value depends upon the accelerator repetition rate or cycle time and the length of time the beam interacts with the target. The peak radiation intensity (I_p) during the beam pulse is related to the average radiation intensity (I) by

$$I_p = I/\mathrm{DF} \tag{7.19}$$

where DF is the duty factor of the accelerator.

Radiation instruments that function accurately during steady-state conditions may yield inaccurate results when subjected to a pulsed radiation field. However, pulsed fields do not affect all radiation detectors in the same manner.

The combination of pulsed radiation fields and high dose rates offer challenges to the measurement of accelerator radiation fields. Ion chambers, proportional counters, and scintillation/photomultiplier detectors can be utilized to measure accelerator fields. Instruments should be tested prior to routine use to ensure that they accurately measure the actual radiation field. Comparison of instrument readings and thermoluminescent dosimetry measurements may be helpful in ensuring accurate results. Modifications, such as changes in the detector volume, may be necessary to obtain sufficiently accurate instrument readings.

7.15 Scenarios

Scenario 7.1

Pions are produced in an accelerator when protons with energies of several hundred MeV or more strike a suitable target. Pions have a rest energy of 140 MeV, about 270 times that of an electron. A pion can carry one unit of charge of either sign, or it can be neutral. Beams of negative pions, extracted from accelerators, have been used for radiation therapy at several institutions. Like all charged particles, negative pions entering the body slow down. By properly selecting the incident energy, a beam of negative pions can be made to come to rest at the site of a tumor to be treated. When it stops in matter, a negative pion is captured by a

positively charged atomic nucleus. The negative pion is annihilated upon capture, releasing its rest energy of 140 MeV inside the nucleus. The highly excited nucleus literally explodes, giving off energetic neutrons, protons, and heavier fragments. The average distribution of emitted particles and energies for capture by an oxygen nucleus is noted below. Similar data describe capture by carbon and nitrogen, the other principal constituents of soft tissue in addition to hydrogen, which does not effectively capture pions.

Reaction Products and Average Energies from Capture of a Stopped Negative Pion by an Oxygen Nucleus.

Emitted Particle	Average Number per Pion Capture	Average Kinetic Energy per Capture (MeV)
Neutrons	2.9	61.0
Protons	1.3	20.0
Heavier fragments	2.0	17.0
Gamma photons	~ 3	6.0
		104.0

7.1 What would be the chief potential advantage of using a negative-pion beam versus using a Co-60 beam for treating a tumor?
   a. More accurate assessment of dose rate at the tumor site.
   b. The generation of neutrons at the tumor site.
   c. Lack of dose buildup in intervening healthy tissue.
   d. Delivery of a localized dose at high LET.
   e. Shorter patient exposure time at higher dose rate.

7.2 Which of the following has the greatest effect in causing an initially parallel beam of charged pions to "spread out" as it penetrates tissue?
   a. Multiple Coulomb scattering of the particles by atomic nuclei.
   b. The generation of delta rays along the particle paths.
   c. Energy-loss straggling.
   d. Range straggling.
   e. Collisions of the particles with atomic electrons.

7.3 What types of radiation are the most important considerations in the shielding design for a negative-pion therapy installation?
   a. Neutrons, prompt photons, muons, and pions.
   b. Neutrons, prompt photons, and pions.
   c. Neutrons, prompt and residual photons, and muons.
   d. Neutrons, prompt and residual photons.
   e. Neutrons and residual photons.

7.4 When the accelerator is not operating, which of the radiation types would most likely contribute the greatest dose equivalent to a technician working in the immediate area where the patients are exposed in a negative-pion radiotherapy facility?

a. Gamma rays and beta rays.
b. Gamma rays and neutrons.
c. Neutrons, gamma rays, and beta rays.
d. Gamma rays, muons, and pions.
e. Muons, beta rays, and gamma rays.

7.5 The difference between the pion rest energy (140 MeV) and the average total kinetic energy released per capture (104 MeV) shown in the table is:
a. Carried away by undetected neutrinos.
b. Spent in overcoming nuclear binding energies.
c. Not zero, because the table gives only average values.
d. Lost by the pion when captured.
e. Emitted as bremsstrahlung during rearrangement of the atomic electrons about the nuclear fragments produced.

7.6 A negative-pion beam from an accelerator will also likely contain:
a. Negative muons and electrons as well as some neutrons.
b. Negative muons only.
c. Neutrons only.
d. Electrons and photons.
e. Electrons only.

7.7 The range of a 50-MeV negative pion in a material of low atomic number can be expressed as 9.1 g/cm^2. What is the range in centimeters, in soft tissue, having a density of 0.95 g/cm^3?

7.8 From the data given in the table, estimate the average absorbed dose in a 1.0-cm-radius sphere of water surrounding the site of capture of a stopped negative pion by an oxygen nucleus. State the assumptions you use in making the estimate.

7.9 In addition to the data in the table, what other information would you need in order to make a more accurate calculation of the average absorbed dose in the last problem?

Scenario 7.2

You are the health physicist at a 4-MeV, 200-mA, electron linear accelerator used for experimental and testing purposes in an industrial setting.

7.10 An aluminum-walled water-filled box is used in the beam as a beam monitor and stopper. The 1.78-MeV gamma ray of the 3-min half-life Al-28 was seen on a spectrometer being used in the target area. What is the most likely explanation for this?
a. Neutron activation of Al-27 resulting from a gamma-neutron reaction in the beam.
b. Neutron activation of Al-28 resulting from a gamma-neutron reaction with deuterium in the beam stopper.
c. Isomeric transition of Al-28, initiated by a gamma-neutron reaction with tritium present in the beam stopper.

d. Neutron activation of Al-27 resulting from a gamma-neutron reaction with deuterium in the beam stopper.
e. Electron–positron pair formation with subsequent release of a gamma ray from Al-28.

7.11 The energy of the beam after the first 90° scatter of a 4-MeV bremsstrahlung beam is represented by which of the following equations?

a. $E = T_e + T_p + 2\, m_0\, c^2$

b. $E = \dfrac{E_0\, m_0\, c^2\, (1 - \cos\theta)}{m_0\, c^2 + E_0}$

c. $E = E_0\, m_0\, c^2\, (1 - \cos\theta)$

d. $E = E_0\, m_0\, c^2\, (\sin\theta)$

e. $E = \dfrac{E_0\, m_0\, c^2}{m_0\, c^2 + E_0\, (1 - \cos\theta)}$

where

E = energy of the scattered beam
E_0 = initial energy of the beam
T_e = kinetic energy of the electron
T_p = kinetic energy of the positron
m_0 = initial mass of the electron
c = speed of light
θ = scattering angle

7.12 One experimenter at your facility directs an electron beam into a copper target. The beam has been running for 4 hr using a water-cooled magnet. The water coolant is stopped and the accelerator scrams. The experimenter wants to rush in to fix his setup. You, as the health physicist, assess the primary hazard to be:

a. O-17.
b. N-16.
c. There is no radiation hazard which would prevent the researcher from taking care of his experiment.
d. Activated dust.
e. Residual scatter.

7.13 An experimenter wishes to test a beryllium oxide ceramic to determine its properties under electron bombardment at high beam powers. What radiological controls would be appropriate for this application?

7.14 The accelerator experiment committee is considering the modification of the existing facility to house a 40-MeV LINAC, capable of a 2-A peak current, 0.5-μs pulse duration, and 250 pps. List the type of information you would require to evaluate the shielding design for this upgrade.

7.15 If both the old machine and the new machine had equal dose rate outputs and lead were used as the shielding material, which machine would require less lead to shield the bremsstrahlung?

a. The 4-MeV machine because 4-MeV electrons are easier to shield.

b. The 4-MeV machine because 4-MeV gammas are easier to shield.
c. The 40-MeV machine because 40-MeV electrons are easier to shield.
d. The 40-MeV machine because 40-MeV gammas are easier to shield.
e. The requirements are essentially the same because the broad-beam tenth-value layer for the 4-MeV machine is nearly the same as that for the 40-MeV machine.

7.16 What is the qualitative relative importance of the neutron source for the 4-MeV and 40-MeV machines?
a. There is essentially no neutron production with the 4-MeV machine, but significant neutron production with the 40-MeV LINAC.
b. There is some neutron production with the 4-MeV machine and essentially none with the 40-MeV machine.
c. The neutron production with the two machines is approximately the same.
d. Both machines produce significant neutrons, with considerably more neutron production with the 4-MeV machine than with the 40-MeV machine.
e. Neutron production with both machines can be neglected.

7.17 What is the qualitative relative importance of the bremsstrahlung production as compared to the neutron production for the two machines if the shielding were constructed of concrete?
a. The bremsstrahlung production is sufficiently high for both machines to control the shielding design.
b. The bremsstrahlung production controls shielding design for the 4-MeV machine, but neutron production controls shielding design of the 40-MeV machine.
c. The bremsstrahlung production controls shielding design for the 40-MeV machine, but neutron production controls shielding design of the 4-MeV machine.
d. The neutron production is sufficiently high for both machines to control shielding designs.
e. Both neutron production and bremsstrahlung production control shielding thickness equally for both machines.

Scenario 7.3

A small plastics manufacturing company is planning to install a 50-MeV, 100-kW electron LINAC for the radiation processing of their product. The electron beam will be used to maximize the dose rate on the product. The facility layout has been drawn and a proposed operation procedures manual has been prepared. You have been hired as a consultant to review the facility design and manual. The following questions are among the many factors that you must consider.

7.18 Both radioactive and chemically toxic gases can result from the irradiation of air by electron or X-ray beams from this accelerator. List the products

that can be expected from this accelerator. Include both radioactive and toxic gas products.

7.19 How do the atomic number of the target and the beam current affect the induced radioactivity and toxic gases produced in the irradiation chamber?

7.20 As the photon energy is raised above the photonuclear reaction threshold to about 35 MeV, the rate of production of induced radioactivity increases very rapidly with the increase in energy. From about 35 MeV up to 100 MeV there is little increase in induced activity, and that increase is only proportional to the relative increase in beam power. Explain this observation.

7.21 A "Rule of Thumb" says that "under certain conditions, if an electron accelerator is properly shielded for bremsstrahlung, it is automatically shielded sufficiently for the neutrons produced". What are those conditions?

7.22 If the beam energy is kept constant but the beam current is reduced in half, the expected induced activity production rate would:

a. be reduced to 1/2.
b. be reduced to 1/4.
c. be reduced by exp (–0.693).
d. remain the same.
e. be reduced by exp (–2.303).

Scenario 7.4

You have been retained to evaluate a Department of Energy accelerator facility's operating practices. The delay time before the irradiation chamber may be safely entered after shutdown of the beam depends on the decay of the induced radioactivity, the rate of destructive reaction of the toxic chemicals, and the clearance of the chamber by exhaust. The following information should be considered in your assessment:

Cell volume = 560 m^3

Exhaust velocity = 4 m^3/s

Toxic gas mean-life = 25 min

G value for toxic gas = 6.0 molecules/100 eV

Beam path length in air = 2.0 m

Toxic gas concentration in the cell at equilibrium = 3.5 ppm

Radioactive Gas Concentration in the Cell at Equilibrium.

Radionuclide	Half-life	Concentration
A	10 min	6.3×10^4 Bq/cm^3
B	2 min	2.9×10^4 Bq/cm^3

7.23 Calculate the time required after beam shutdown for the concentration of the radioactive gas to be reduced to 2 Bq/cm^3.

7.24 Calculate the time required after beam shutdown for the toxic gas concentration to be reduced to 0.1 ppm.

Scenario 7.5

A linear accelerator (LINAC) bombards a tritium target with a 25-μA beam of 2.5-MeV protons. This produces 1.2-MeV neutrons via the T(*p*, *n*) reaction.

Data

Production rate = 1.8×10^{-6} neutrons/proton

6.24×10^{18} protons/amp-s

Neutron removal cross-section for concrete = 0.08 cm^{-1}

Dose equivalent rate = 3.5×10^{-8} rem cm^2 neutron^{-1}

7.25 Calculate the neutron dose equivalent rate at a point 40 cm away from the target along the beam centerline. State all assumptions.

7.26 An operator is located 4.0 meters from the target and is shielded by a 50-cm-thick concrete wall. Compared to the dose equivalent rate calculated in question 7.25, the dose equivalent rate at the operator's console will be reduced by a factor of:

a. 1.8×10^{-2}
b. 1.8×10^{-4}
c. 6.3×10^{-4}
d. 1.6×10^{-5}
e. 6.3×10^{-5}

7.27 Which of the following instruments would have good sensitivity to neutrons while providing the best discrimination against gammas?

a. BF_3 proportional counter in a polyethylene moderator.
b. Geiger-Müller (GM) tube at greater than 2 atmospheres in a polyethylene moderator.
c. Silver-wrapped GM tube inserted in a polyethylene moderator.
d. LiI(Eu) scintillator inserted in a polyethylene moderator.
e. Cadmium-wrapped LiI(Eu) scintillator.

7.28 Which of the following statements best describes the neutron distribution as viewed in the laboratory coordinate system?

a. Isotropic fluence rate, but energy peaked in the forward direction.
b. Isotropic energy distribution, but fluence rate peaked in the forward direction.
c. Both energy and fluence rate peaked in the forward direction.
d. Energy and fluence rate are peaked at 90° to the incident proton beam.

e. Fluence rate peaked at 90° to the incident proton beam, isotropic energy distribution.

Scenario 7.6

You are the health physicist at an electron LINAC processing facility that has the characteristics given below. The laboratory director is concerned that residual ozone (O_3) and oxides of nitrogen (NO_x) may delay access to the processing cell by personnel after the beam is turned off. The director requests that you provide answers to the following questions on O_3 and NO_x levels.

Data

Energy	10 MeV
Peak power	5 MW
Duty Factor	0.01
Irradiation cell volume	75 m^3
Cell exhaust rate	5 m^3/s
Beam path in air	2 m

In order to respond to your boss' request, you have found an empirical expression for ozone production:

$$\text{Production rate (molecules cm}^{-3}\text{ s}^{-1}) = (600\text{ eV cm}^{-4}\text{ A}^{-1}\text{ s}^{-1}) \times G\,I\,d$$

where

G = 10.3 molecules/100 eV for ozone
I = average beam current (A)
d = length of air path traveled by the beam in air (cm)

7.29 What is the ozone production rate in molecules cm^{-3} s^{-1}?
7.30 For an NO_x production rate of 100 molecules cm^{-3} s^{-1}, calculate the steady-state concentration in the irradiation cell. Assume that the mean lifetime of NO_x is 1800 s.
7.31 Assume that the ozone concentration in the cell achieves a steady-state value of 10 ppm. Calculate the delay time after beam shutdown for it to reach the TLV of 0.1 ppm. Assume that the mean lifetime of ozone is 2000 s.

Scenario 7.7

An experimental physicist calls you to report that he believes he has accidentally placed his right arm in the beam of a 6-GeV proton synchrotron for approximately 1 min. The accelerator produces 10 pulses per minute at an intensity of 1.0×10^{12} protons per pulse.

Data

Beam size = 1 cm^2.

Production cross-section for C-11 for protons in oxygen = 20 mb.

C-11 half-life = 20.4 min.

Efficiency of detector including geometry effects = 10%.

Time of measurement = 1 hr after suspected exposure.

The thickness of the arm is 10 cm, and its composition is H_2O.

The NaI detector yields 400 cps when subjected to an exposure rate of 10 μR/hr due to Ra-226 gamma rays.

7.32 You have available a 3-in. × 3-in. NaI scintillation counter. Calculate the counting rate you will observe from C-11 in activated body tissue with such a counter as a result of the exposure.

7.33 The measurement of activity induced in tissue is to be taken in the radioactive environment of the accelerator where the gamma background is about 1 mR/hr. Will the induced activity be detected?

7.34 What other actions would you initiate in investigating this incident?

Scenario 7.8

You are a health physicist at the Pipe Dream National Accelerator Laboratory (PDNAL). PDNAL is limited to an electron beam energy of 100 MeV as a result of previous operating issues associated with the 6 m × 8 m × 2.5 m irradiation chamber. The distance traversed in air by the electron beam is 2 m. This facility operates under the funding of the US Department of Energy.

The DOE has informed you that they will soon audit PDNAL's radiation safety program. As part of this assessment, the USDOE will assess your radiation safety performance based, in part, on your ability to answer the following questions.

Previous measurements at the PDNAL determined that the density of air under operating conditions is 0.001205 g/cm^3 and the collision mass stopping power for 10 MeV electrons in air 1.98 MeV cm^2/g.

Relevant data regarding ozone production in electron-beam facilities and shielding calculations follow.

Ozone production in electron-beam facilities

$$C(O_3) = \frac{3.25 \text{ L-ppm}}{\text{keV-mA-s}} \frac{S_{coll}\, I\, \chi\, t}{V}$$

where $C(O_3)$ is the ozone concentration in ppm, S_{coll} is the collision stopping power of electrons in air in keV/cm, I is the electron beam current in mA, χ is the distance traversed in air by the electron beam in cm, t is the irradiation time in s, and V is the volume of the irradiation chamber in liters.

Shielding transmission

$$B_x \leq 1.67 \times 10^{-5} \frac{\text{Gy-h}}{\text{mSv-min}} \left(\frac{\dot{H}_m \, d^2}{\dot{D}_o \, T} \right)$$

where B_x is the shielding transmission, $\dot{H}_m$ is the maximum permissible dose equivalent rate in mSv/h, d is the distance between the source and the reference point in meters, $\dot{D}_o$ is the absorbed dose index rate in Gy/min, T is the area occupancy factor, and 1.67×10^{-5} is a unit-specific constant. The absorbed dose rate can be parameterized in terms of the beam current (I):

$$\frac{\dot{D}_o}{I} = 4.7 \times 10^2 \text{ Gy-m}^2\text{/mA-min (forward direction)}$$

Broad-Beam Transmission through Concrete of X-rays Produced by 0.5–176 MeV Electrons.

Slab thickness (cm)	Transmission
100	2.2×10^{-3}
75	1.0×10^{-2}
50	4.7×10^{-2}
25	1.7×10^{-2}

7.35 You are inspecting the accelerator facility before it resumes operations when you encounter a beam line interlock. What is an interlock and where should it be installed?

7.36 Upon inspection of the beam dump, you find that the depth of the cavity is greater than the diameter of the aperture. Is this condition acceptable? Describe why or why not.

7.37 List four kinds of radiation produced following an interaction between a particle beam and the material it strikes in an accelerator. Describe the method of production of each.

7.38 A scattering experiment produces an X-ray beam using a 1.0 cm diameter, 10 MeV electron beam incident on a thick W target. The experimenter is located behind a 75-cm thick concrete shield at a point in the controlled area that is on the beam-line and 10 meters from the target. The dose rate is limited to 0.025 mSv/h at the experimenter's location. Calculate the peak current value for the beam.

7.39 Using information in the problem statement and a beam current of 0.05 mA, calculate the concentration (ppm) of ozone in the irradiation chamber following 1 minute of continuous operation. Assume a ventilation rate of 2 m^3/min and an ozone molecule mean life of 50 minutes.

Scenario 7.9

You are employed by Tachyon Enterprises, an international communications company that utilizes electron accelerators in production operations. Senior management assigns you the task of designing the shielding for a 20 MeV electron accelerator facility.

The machine operates with a peak current of 1 A, a beam pulse length of 1 μs, and a beam pulse frequency of 10 Hz. The target is a thick, tungsten beam dump. Data relevant to your design task follows.

Miscellaneous Parameters.

Z_{tungsten}	74
Z_{copper}	29
The X-ray emission rate from a high Z target at 20 MeV incident electron energy for a sideward direction (90°) from the beamline	$\frac{\dot{D}_{I_o}}{I} = 20$ Gy-m^2/mA-min

The Equivalent Electron Energy for Analysis of the Transmission of X-rays Emitted in the 90° Direction from Very Thick High-Z Targets as a Function of the Incident Electron Energy.

Incident Electron Energy (MeV)	Equivalent Incident Electron Energy (MeV)
1.0	0.7
10.0	6.0
20.0	10.0
60.0	40.0

Broad-Beam Transmission of X-rays Through Concrete.

Beam Energy (MeV)	Transmission	Concrete Slab Thickness (cm)
10	1.0×10^{-5}	191
	1.0×10^{-4}	152
	1.0×10^{-3}	114
	1.0×10^{-2}	76
20	1.0×10^{-5}	229
	1.0×10^{-4}	183
	1.0×10^{-3}	140
	1.0×10^{-2}	94

Broad-Beam Transmission Through Concrete of X-rays (the concrete density is 2.35 g/cm^3).

Incident Electron Energy (MeV)	Concrete Thickness (cm)	Transmission
10	76	0.010
20	76	0.025

Tenth Value Layers for Broad-Beam X-rays in Lead.

Incident Electron Energy (MeV)	TVL (cm Pb)
10	5.6
20	5.3

7.40 Assume that the dose equivalent rate in an office, which is 90 degrees from the beam line and 5 m from the target (perpendicular distance from the beam line), can be no greater than 0.005 mSv/hr. Calculate the minimum thickness of shielding required for the concrete wall (density = 2.35 g/cm^3) between the target and the office.

7.41 Assume that in question 7.40 the required transmission factor is 1.0×10^{-4} and the existing concrete wall is 76 cm thick. Calculate the additional lead thickness required to complement the concrete wall.

7.42 List and describe five parameters of significant importance to estimate the emission of radiation from an accelerator.

7.43 List five considerations for selecting shielding materials for an accelerator.

7.44 For each of the following accelerator types, complete the table by identifying two principle radiations of concern for occupied areas:

Accelerator	Particle Accelerated	Beam Energy (MeV)	Principle Radiations
a. Potential Drop	Protons/Deuterons	1–10	
b. Electron Linear	Electrons	1–10	
c. Electron Linear	Electrons	> 10	
d. Cyclotron	Protons/Deuterons	10–50	
e. Betatron	Electrons	1–50	

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8
Nonionizing Radiation Health Physics

Nonionizing radiation (NIR) corresponds to the long-wavelength (> 100 nm), low-energy (< 12.4 eV) portion of the electromagnetic spectrum. NIR propagates as an electromagnetic wave that is characterized in terms of its electric and magnetic fields. As photon radiation, it is fundamentally similar to X- and γ-ray radiation, but the longer wavelength affects its interaction properties. These properties are dependent on the frequency of the nonionizing radiation. Nonionizing radiation is not normally quantified in terms of dose because of the difficulty in calculating energy absorption per unit mass.

Laser radiation is normally quantified in terms of energy or power density per unit area. Radiofrequency and microwave fields are characterized in terms of the electric field strength, the magnetic field strength, their associated power densities, and induced currents.

In this chapter, the most common types of nonionizing radiation (i.e., radiofrequency, microwave, and laser radiation) are reviewed. Laser, radiofrequency, and microwave radiation differ in wavelength and frequency as well as in their inherent characteristics and hazards. Each of these radiation types is reviewed and their hazards assessed. The discussion of nonionizing radiation begins with a discussion of the sources of radiofrequency and microwave radiation.

8.1
Sources of Radiofrequency and Microwave Radiation

The Earth is bathed in a sea of background radiofrequency (RF) and microwave radiation that arise from both natural and man-made sources. Radiofrequency radiation spans the frequency range from 3 kHz to 300 MHz. Microwave radiation occurs at higher frequency from 300 MHz to 300 GHz. Below 3 kHz, radiation is referred to as extremely low frequency.

Natural NIR background radiation arises from both terrestrial and extraterrestrial sources. All matter with a temperature above absolute zero radiates electromagnetic radiation. This radiation is called blackbody radiation and is a function of absolute temperature. The radiation output is characterized in terms of the monochromatic emissive power $I(\lambda, T)$:

$$I(\lambda, T) = \frac{2\,\pi\,c^2\,h}{\lambda^5\left(e^{\frac{c\,h}{k\,T\,\lambda}} - 1\right)} \tag{8.1}$$

where λ is the output wavelength, T is the absolute temperature (K), h is Planck's constant, k is Boltzmann's constant, and I has units of W/m^3. The emissive power occurs over a range of values and is not monochromatic.

Man-made sources of RF and microwave radiation are generated by a number of approaches including power grid tubes; linear beam tubes such as klystrons; cross-field devices including magnetrons and amplitrons; solid state devices; free electron lasers; dipole antennas; and resonating cavities. Once the RF wave is generated, it is transmitted to an antenna through a waveguide or transmission line.

Electromagnetic antennas or emitters transmit their signals in either a continuous wave (CW) or pulsed output format. In a CW system, the output is composed of continuous waves and the rated power output is expressed in terms of average power. Pulsed systems express their power output in terms of peak power. In a pulsed system, the output is in pulses that are repeated at regular intervals.

The duty factor (D_F) or duty cycle is a useful concept to describe the output of a pulsed source:

$$D_F = \frac{P_{avg}}{P_{peak}} = \frac{I_{avg}}{I_{peak}} = P_w\ P_{rf} \tag{8.2}$$

where P_{avg} is the average power for a pulsed system, P_{peak} is the peak power, I_{avg} is the average current, I_{peak} is the peak current, P_w is the pulse width, and P_{rf} is the pulse repetition frequency. The pulse width is the amount of time that the beam is on and the pulse repetition frequency is the number of beam pulses per unit time. During a full cycle, the reciprocal of the pulse repetition frequency is the time between pulses that represents the time the beam is on plus the time the beam is off.

8.2 Characteristics of Electromagnetic Waves

The capability of electromagnetic waves to penetrate a medium depends on the properties of that medium and the frequency of the incident radiation. For plane waves traveling in free space, the electric and magnetic field vectors are in phase and orthogonal to each other and to their direction of propagation. The strengths of the electric and magnetic fields are measured in units of V/m and A/m, respectively. Their direction of propagation is characterized by the Poynting vector $(\vec{S})$ defined in terms of the electric and magnetic fields:

$$\vec{S} = \vec{E} \times \vec{H} = \left|\vec{E}\right|\left|\vec{H}\right| \sin\theta \tag{8.3}$$

where θ is the angle between the electric and magnetic field vectors.

The electromagnetic wave is characterized by an energy (E_{EM})

$$E_{EM} = h\,\nu = \frac{h\,c}{\lambda} \tag{8.4}$$

where ν is the frequency of the radiation (Hz = cycle/s), h is Planck's constant, and the λ is the wavelength of the radiation:

$$\lambda = \frac{c}{\nu} \tag{8.5}$$

8.3 Antennas

An antenna focuses and intensifies the artificially generated electromagnetic radiation. The antenna's radiation and its associated generating equipment must be evaluated for its radiation hazard.

8.3.1 Stationary Antennas

Microwave and electromagnetic radiation can be focused into intense beams because the antenna size is much larger than the wavelength of the emitted radiation. The ability of the antenna to focus the radiated electromagnetic energy into a beam is characterized by the gain (G) of the antenna. Gain is defined by a ratio of power values assuming that the same amount of transmitter power is available in both situations:

$$G = \frac{P}{P_o} \tag{8.6}$$

where

P = power from the antenna in the far field at point X
P_o = power from an isotropic radiator at point X

Gain may also be expressed in terms of decibel units. The decibel (dB) is a dimensionless number used to express a simple ratio. For antennas, the gain in dB (G') is:

$$G' = 10 \log_{10}(G)\ \text{dB} \tag{8.7}$$

For a parabolic antenna, the gain is determined by its area (A) and emitted wavelength:

$$G = \frac{4\,\pi\,\eta\,A}{\lambda^2} \tag{8.8}$$

where η is the antenna's aperture efficiency.

An antenna produces radiation with properties that vary with distance. These distances are historically defined in terms of three hazard assessment regions extending radially from the source: the near field (nf), intermediate field (if), and far field (ff). In a hazard assessment, it is convenient to define these fields in terms of the average power density of the antenna.

The average power density (W_o) emitted from a uniformly illuminated antenna is given by:

$$W_o = \frac{P_{avg}}{A} \tag{8.9}$$

where

W_o = average power density (W/m^2)
P_{avg} = power output of the antenna (W)
A = area of the antenna or radiating device (m^2)

The near-field maximum power density is four times the average power density:

$$W_{nf} = \frac{4\,P_{avg}}{A} = 4\,W_0 \tag{8.10}$$

For simplicity, safety assessments usually assume that the power density is constant throughout the near field having a $4\,W_0$ value. For the case of a circular antenna, this power density is conservatively defined to extend to the distance:

$$R_{nf} = \frac{D^2}{4\,\lambda} \tag{8.11}$$

where

R_{nf} = maximum extent of the near field
D = diameter of the radiator or antenna

The distance defined by R_{nf} is much larger than the physical extent of the near field, but it provides a measure of safety in protecting workers against the harmful effects of the electromagnetic fields.

For safety assessments, an intermediate field is defined to lie between the near field and the far field. The distance between the antenna and the end of the intermediate field and start of the far field is defined to be:

$$R_{ff} = \frac{0.6\ D^2}{\lambda} \tag{8.12}$$

To complete the specification of the hazard assessment methodology, the intermediate and far-field power densities are defined:

$$W_{if} = 4\ W_0 \left(\frac{R_{nf}}{r} \right) \tag{8.13}$$

$$W_{ff} = 2\ W_0 \left(\frac{R_{ff}}{r} \right)^2 \tag{8.14}$$

where r is the distance from the radiating device. The intermediate field power density decreases with distance in a linear manner, and the power density in the far field decreases as $1/r^2$.

8.3.2 Rotating Antennas

At any given location, the output from a rotating system (W_r) is less than that from a stationary radiator (W_s). The power density produced at any point from a rotating antenna is:

$$W_r = F\ W_s \tag{8.15}$$

where F is a rotational reduction factor. This factor is independent of the scanning rate or rotational frequency of the antenna.

For the far field, the reduction factor (F_{ff}) is defined by

$$F_{ff} = \frac{\theta_{1/2}}{\theta_s} \tag{8.16}$$

where

θ_s = emitter scan angle in radians. For full rotation through 360°, this angle is 2 π.

$\theta_{1/2}$ = half-power beam width of the antenna. The half-power beam width is the angle spanning the region where the power density exceeds 50% of its peak value.

8.4 Attenuation by Biological Systems

As electromagnetic radiation traverses a medium, energy is lost through a variety of loss mechanisms including joule heating. The wave induces currents in the medium whose propagation is opposed by the resistance of the medium. The power or energy deposited per unit time in the medium is:

$$P = I^2 R \tag{8.17}$$

where P is the power or heating produced in the medium by the induced current (I) as a result of penetration of the electromagnetic wave, and R is the electrical resistance of the medium being penetrated. The heating occurs from ionic currents induced by the EM field and through vibrations of polar molecules induced by the oscillating EM field.

The wave's energy loss is continuous and the decrease in power density as the EM field traverses the medium is given by:

$$W(t) = W(0)\, e^{-2bt} \tag{8.18}$$

where

$W(t)$ = power density at a depth t into the medium
$W(0)$ = initial power density at the surface of the medium
t = depth of penetration into the medium or absorbing material
b = absorption coefficient for the absorbing medium

The absorption coefficient is a complex function that is dependent on the radiation frequency and the properties of the medium including its conductivity, permittivity, and permeability:

$$b = \omega \left(\frac{\mu\,\varepsilon}{2}\right)^{1/2} \left\{ \left[1 + \left(\frac{\sigma}{\omega\,\varepsilon}\right)^2 \right]^{1/2} - 1 \right\}^{1/2} \tag{8.19}$$

where

ω = angular frequency of the radiation (1/s) = $2\,\pi\,\nu$
σ = conductivity of the medium = $(\text{ohm-m})^{-1}$
ε = permittivity of the medium = $K\,\varepsilon_o$
ε_o = permittivity of free space = 8.85×10^{-12} $C^2/\text{N-m}^2$
K = relative dielectric constant
μ = permeability of the medium
For biological media, μ is essentially equal to the permeability of free space = $4\,\pi \times 10^{-7}$ N/A^2.

8.5
Biological Effects

The primary physical effect of microwave or radiofrequency electromagnetic radiation is heating. Heating mechanisms include joule heating due to ionic currents or polar molecule vibration induced by oscillating electromagnetic fields.

Thermal effects occur for exposures greater than about 10 mW/cm^2 and include hyperthermia or increases in body temperature. Common temperature-induced effects include cataract formation in the lens of the eye and loss of fertility because of temperature changes in testicles.

Non-thermal effects, which may occur at or below 10 mW/cm^2, are not clearly understood, and the data is less definitive. The more common non-thermal effects include increased fatigue, headaches, irritability, sleepiness, and decreases in olfactory sensitivity.

8.6
Protection Standards

Ionizing radiation protection standards are well established and the radiation dose is measured with a variety of techniques. These conditions do not yet exist in the nonionizing realm.

Table 8.1 provides guidance for the maximum permissible exposure from nonionizing radiation for radiation controlled areas. Other guidance is available for the maximum permissible exposure from nonionizing radiation for uncontrolled radiation areas, and for induced and contact currents for both controlled and uncontrolled environments.

Table 8.1 provides a frequency-dependent protection standard. Workers can be subjected to multiple frequencies and an appropriate protection mechanism must be provided. If a worker is exposed simultaneously to a number of different frequencies, the recommended exposure limit is governed by the sum rule:

$$\sum_{i=1}^{n} \frac{W_i}{L_i} \leq 1 \tag{8.20}$$

where

W_i = power density for frequency i
L_i = protection standard for frequency i
n = number of frequencies exposing the individual
i = frequency label

Table 8.1 Maximum Permissible Exposures for Controlled RF Environments.

Frequency Range (MHz)	RMS Electric Field Strength (E)[a] (V/m)	RMS Magnetic Field Strength (H)[a] (A/m)	RMS Power Density (S) E-Field, H-Field (W/m^2)	Averaging Time $\|E\|^2$, $\|H\|^2$, or S (min)
0.1–1.0	1842	$16.3/f_M$	$(9000{,}100\,000/f_M^{\,2})$[b]	6
1.0–30	$1842/f_M$	$16.3/f_M$	$(9000/f_M^{\,2}{,}100\,000/f_M^{\,2})$[b]	6
30–100	61.4	$16.3/f_M$	$(10{,}100\,000/f_M^{\,2})$[b]	6
100–300	61.4	0.163	10	6
300–3000	–	–	$f_M/30$	6
3000–30 000	–	–	100	$19.63/f_G^{\,1.079}$
30 000–300 000	–	–	100	$2.524/f_G^{\,0.476}$

Source: IEEE Standard C95.1-2005.
Note – f_M is the frequency in MHz, f_G is the frequency in GHz, and RMS is the root-mean-square value.

[a] For exposures that are uniform over the dimensions of the body, such as certain far-field plane-wave exposures, the exposure field strengths and power densities are compared with the MPEs in the table. For non-uniform exposures, the mean values of the exposure fields, as obtained by spatially averaging the squares of the field strengths or averaging the power densities over an area equivalent to the vertical cross-section of the human body (projected area), or a smaller area depending on the frequency, are compared with the MPEs in the table.

[b] These plane-wave equivalent power density values are commonly used as a convenient comparison with MPEs at higher frequencies and are displayed on some instruments in use.

8.7 Measurement of Electromagnetic Fields

Protection standards require measurement of the electric and magnetic field strengths and associated power densities. The measurement of electromagnetic fields utilizes basic physical principles that convert the incident field strength into a measurable quantity (e.g., current or voltage). Measurements must be carefully performed since the position of the individual performing the measurement and the instrumentation probe orientation can affect the observed quantity.

Electric fields are measured using parallel plates, dipoles, thermocouples, and diodes. Magnetic field measurements utilize current loops, thermocouples, and diodes.

As an illustration of the physics behind these techniques, consider a loop of a good conducting wire. If this loop moves through a magnetic field, a current is produced. The magnetic field strength is related to the magnitude of the induced

current. In a similar manner, parallel conducting plates submerged in an electric field generate an output voltage that can be calibrated to reflect the electric field strength and power density.

Electromagnetic radiation probes use dipoles and thermocouples to detect the electric field components of the field by measuring the output voltages. Schottky diodes convert the root-mean-square value of the detected field components to a direct current voltage. By using diodes instead of thermocouples, it is possible to handle a much wider field strength range. Thermocouples can also be damaged more easily than diodes at high power levels.

These techniques and physical principles are used in the construction of a survey instrument. For example, a magnetic field probe often consists of three separate current loops oriented at 90° to each other. This configuration provides a spatially averaged magnetic field value.

8.8 Laser Radiation

The term laser is commonly used to describe a device that produces coherent, monochromatic light in a well-defined direction. Laser is an acronym for light amplification by stimulated emission of radiation. It is the process of stimulated emission that leads to the unique characteristics of laser radiation which include: (1) monochromatic wavelength (same frequency); (2) spatial coherence (same phase); and (3) emission in a specific direction.

Laser output spans a wide range of wavelengths. As a matter of notation, laser radiation is classified in terms of the wavelength of the emitted radiation: Ultraviolet C (UV-C) @ 100–280 nm, UV-B @ 280–315 nm, UV-A @ 315–400 nm, visible @ 400–700 nm, Near-infrared (IR-A) @ 700–1400 nm, IR-B @ 1400–3000 nm, and IR-C @ 3,000–1.0×10^6 nm.

8.8.1 Radiometric and Photometric Terms and Units

Two sets of units are commonly used to describe light. These are the photometric and radiometric units. Photometric units are used to describe the response of the eye to light and are not related to physical phenomena. Radiometric units are physical units whose definition is not limited to a narrow range of the electromagnetic spectrum, but are related to the absolute measurement of radiation. This text utilizes radiometric units to describe both laser properties and protection standards.

Radiometric quantities and units that are frequently encountered in laser studies include radiant energy, radiant power, radiant intensity, radiance, radiant exposure, and irradiance. Radiant energy is energy in the form of electromagnetic waves usually expressed in joules (W-s). Radiant power is the time rate of flow of radiant energy, which is expressed in watts. Radiant intensity is radiant power in a given direction or radiant flux emitted from the source per unit solid angle

(steradian), in the direction of propagation which is usually expressed in W/sr. Radiance is radiant flux (radiant power) output per unit solid angle per unit area in units of W/sr-cm^2.

Radiant exposure and irradiance are frequently used in protection standards. Radiant exposure is the total energy per unit area incident upon a given surface. This unit is used to express exposure to pulsed laser radiation and it is expressed in units of J/cm^2. Irradiance is the power per unit area incident on a surface, has units of W/cm^2, and is used to express exposure to continuous wave laser radiation.

8.8.2
Properties of the Laser

Three basic components are common to all laser devices. These components include an amplifying medium, energy source, and optical cavity. An amplifying or active medium is the material in which a population inversion occurs. Mixtures of helium and neon gases or a ruby crystal are examples of amplifying media.

A source of energy is required to drive or pump the lasing medium to produce a population inversion. In a HeNe laser, this involves passing a direct electrical current through a mixture of helium and neon gasses. In a pulsed ruby laser, the pumping system is a flashlamp.

An optical cavity is required to support the amplification and propagation of the laser beam. This cavity contains the active medium, sets up standing waves, and contains reflective surfaces. Laser amplification and stimulated emission occur within the optical cavity.

There are a variety of lasers in use and the selection of the particular laser depends on the desired wavelength and power level. Representative lasers (associated wavelength) include: H_2 (0.16 μm), Xe_2 (0.17 μm), HeCd (0.4416 μm), argon (0.4880 and 0.5143 μm), rhodamine 6G (0.59 μm), HeNe (0.6328 μm), ruby (0.6943 μm), krypton (0.7525 μm), GaAs (0.91 μm), Nd glass (1.06 μm), Nd:YAG (1.064 μm fundamental; 532 μm first harmonic), HF (3.1 μm), CO (5.5 μm), CO_2 (10.6 μm), H_2O (118.6 μm), and HCN (773 μm).

8.9
Biological Effects from Laser Radiation

Mechanisms of biological damage from laser light include both temperature effects caused by energy absorption and photochemical reactions that change the character of the irradiated tissue. Physical damage depends on the energy (wavelength) of the laser radiation and the tissue being irradiated, and the most sensitive tissues are the skin and the eye. Other mechanisms are governed by acoustic transients and the effect of chronic exposure.

These effects vary according to the absorbency of tissue at a particular wavelength. The effect will also be influenced by the energy of the laser beam, the size of the area irradiated, and the exposure duration.

Acoustic transients result from the effect that the mechanical shock wave can have on the tissue structure. An acoustic wave propagates through tissue and deposits its energy with associated tissue damage.

Chronic exposure effects occur over time and result from prolonged exposures. For example, chronic exposure to solar ultraviolet radiation leads to premature aging of the skin and an increased risk of cancer. Chronic infrared exposure is believed to be a contributor to cataracts.

8.9.1 Eye

The eye is the organ most vulnerable to laser-induced injury. In comparison with other organs, lower energy depositions result in damage and that damage is usually more serious. The biological effects of the absorbed laser radiation on the eye are summarized in Table 8.2.

Table 8.2 Biological Effects of Laser Radiation in Selected Eye Tissues.

Eye Tissue	Wavelength	Effect
Cornea	UV-B and UV-C (surface absorption)	Conjunctivitis (inflammation of corneal membranes) with recovery within 48 hours.
	UV-B and UV-C (deeper corneal absorption)	Photochemical reactions (cornea loses transparency and becomes milky in color) with an onset of 6–12 hours. Recovery requires a corneal graft.
	IR-B and IR-C	Flash burns. IR-B and IR-C absorption is very large.
Lens	UV-A	Absorption by the lens results in a generally permanent injury. UV-A is a major contributor to cataractogenesis, and also results in a loss of elasticity.
	IR-A and IR-B	Prolonged exposure to low-level IR-A causes cataract formation. Most injuries result from the temperature rise in absorbing tissue.
Retina	IR-A and visible	Damage is caused by thermal, photochemical, and acoustic effects. Exposure duration is a critical factor in the injury extent.
		The irradiance is much higher at the retina because the lens focuses light to a smaller area.
		The retina is not normally susceptible to UV absorption. Injury from UV radiation can occur if the individual is taking photosensitizing drugs or if the lens has been removed.
Retinal Pigment Epithelium	IR-A and visible	Retinal thermal injury occurs when energy is absorbed faster than removed by the choroid blood supply. A temperature rise in the pigment epithelium and thermal injury to the retina results.

8.9.2 Skin

In general, laser-induced skin injuries are not as significant as eye injuries. In some cases, only the external dead skin layer is irradiated. Most injuries eventually heal. Skin damage mechanisms include thermal and photochemical effects. The threshold for skin injury is comparable to the corneal hazard level.

From a laser irradiation perspective, the skin structure consists of three broad classifications. The main skin layers are the epidermis, dermis, and subcutaneous tissue. The skin reflects most visible and IR-A radiation. Ultraviolet (UV-B and UV-C) and infrared (IR-B and IR-C) radiation are highly absorbed by the epidermis. The epidermis is less sensitive to UV-A. However, with enough laser power and exposure duration, incident radiation of any optical wavelength penetrates the epidermis and produces internal injury.

UV-B and UV-C cause erythema and photochemical reactions, which result in darkening or tanning of the skin. Chronic UV exposure causes premature aging of the skin and increases the risk of cancer.

Laser-induced thermal injury to the skin is most significant at the largest infrared wavelengths. Higher irradiance is required for visible and near-infrared light. However, the pain associated with thermal injury to the skin is normally sufficient to warn the exposed individual.

8.10 Laser Standards

Laser safety standards consider the focusing action of the lens in establishing the maximum permissible exposure of the cornea. The applicable American National Standards Institute (ANSI) standard is ANSI Z136.1-2007, and a portion of this safety standard is summarized in Table 8.3 for point-source ocular exposure to a laser beam.

The values of Table 8.3 for visible light are based on the conservative assumption that the maximum quantity of light is admitted to the eye. This occurs for a pupil diameter of 7 mm, which is the maximum opening of the iris diaphragm.

Within the retinal hazard region (0.4 to 1.4 μm) sources are considered to be either extended or point. Point sources subtend a visual angle less than or equal to 1.5 mrad (α_{min}). Most collimated beams meet the point-source condition. Extended sources subtend an angle larger than α_{min}. Examples of extended sources include laser arrays, multiple diode lasers, and nonuniform diffuse reflections.

Maximum permissible exposure (MPE) recommendations are promulgated by the American National Standards Institute (ANSI) and the American Conference of Government Industrial Hygienists (ACGIH). These recommendations have been incorporated into applicable Federal Regulations.

Table 8.3 Maximum Permissible Exposure for Point-Source Ocular Exposure to a Laser Beam.[a, b]

Wavelength λ (μm)	Exposure Time t (s)	MPE
Visible		
0.400 to 0.700	1.0×10^{-13} to 1.0×10^{-11}	1.5×10^{-8} J/cm^2
0.400 to 0.700	1.0×10^{-11} to 1.0×10^{-9}	$2.7\ t^{0.75}$ J/cm^2
0.400 to 0.700	1.0×10^{-9} to 1.8×10^{-5}	5.0×10^{-7} J/cm^2
0.400 to 0.700	1.8×10^{-5} to 10.0	$1.8\ t^{0.75} \times 10^{-3}$ J/cm^2
0.500 to 0.700	10 to 3.0×10^4	1.0×10^{-3} W/cm^2
Thermal		
0.450 to 0.500	10 to T_1	1.0×10^{-3} W/cm^2
Photochemical		
0.400 to 0.450	10 to 100	1.0×10^{-2} J/cm^2
0.450 to 0.500	T_1 to 100	$C_B \times 10^{-2}$ J/cm^2
0.400 to 0.500	100 to 3.0×10^4	$C_B \times 10^{-4}$ J/cm^2

[a] $C_B = 1$ for λ = 0.400 to 0.450 μm; $C_B = 10^{[20(\lambda - 0.450)]}$ for λ = 0.450 to 0.600 μm.
$T_1 = 10 \times 10^{[20(\lambda - 0.450)]}$ s for λ = 0.450 to 0.500 μm.
[b] *Source:* ANSI Z136.1-2007.

8.10.1 Intrabeam Exposures

Intrabeam exposures to laser radiation are specified in units of radiant exposure or irradiance. Radiant exposure (H) is related to irradiance (E) by the relationship $H = E\,t$, where t is the exposure time. Emergent radiant exposure and emergent irradiance are defined at the exit of the laser aperture. These emergent values are analogous to the saturation activity since they are the largest values that are produced by the laser system. The emergent radiant exposure (H') and emergent irradiance (E') are defined as:

$$E' = \frac{P}{A'} \tag{8.21}$$

$$H' = \frac{e}{A'} \tag{8.22}$$

where
P = power output of the laser (W)
A' = area of the laser aperture (cm^2)
e = energy output of the laser per pulse (J/pulse)

For a Gaussian beam, the laser beam radius is:

$$r_{GB} = (a^2 + d^2\ D^2/4)^{1/2} \tag{8.23}$$

where
a = aperture radius
d = distance from the aperture
D = divergence angle

Equation (8.23) illustrates the physical situation in which the area of a laser beam increases as its distance from the aperture increases. The increase is determined by the divergence angle that is typically on the order of a milliradian (mrad).

8.10.2
Nominal Ocular Hazard Distance

Laser safety assessments frequently required the calculation of the Nominal Ocular Hazard Distance (NOHD) or the distance at which the maximum permissible exposure (MPE) is reached. Assuming that atmospheric attenuation can be neglected, the NOHD is given by:

$$\text{NOHD} = \frac{2}{D}\sqrt{\frac{P}{\pi\,(\text{MPE})} - a^2} \tag{8.24}$$

8.10.3
Diffuse Reflections

Laser standards are also defined in terms of the diffuse reflection of a laser or an extended source laser. When light strikes a surface, it is either reflected from or interacts with that surface. The quantity of reflected light depends on the surface topology and the wavelength and energy of the light. Laser light follows similar effects, but the large energy densities create significant material challenges for reflecting surfaces and beam optical systems. The reflection of laser radiation is characterized as specular, diffuse, or mixed.

Specular reflection occurs when laser light strikes a smooth or polished surface. The angle of incidence equals the angle of reflection for the case of specular reflection. Diffuse reflection occurs when laser radiation strikes a rough surface. If the reflected light appears to be of equal intensity for all observation directions, the reflection is diffuse. Mixed reflection occurs if a portion of the incident radiation undergoes specular reflection and the remainder is diffusely reflected.

8.10.4
Nominal Hazard Zone

The nominal hazard zone (NHZ) is the space within which the level of direct, reflected, or scattered radiation exceeds the applicable MPE. Exposure levels beyond the boundary of the NHZ are below the appropriate MPE. The NHZ is given by the relationship:

$$\mathrm{NHZ} = \sqrt{\frac{\rho_\lambda \; P \cos\theta_v}{\pi\,(\mathrm{MPE})}} \tag{8.25}$$

where

ρ_λ = spectral reflectance of a diffuse or specular object at wavelength λ
θ_v = viewing angle from the normal to a reflecting surface

8.10.5
Skin Exposures

Standards for the maximum permissible skin exposure are provided in ANSI Z136.1-2007. Table 8.4 provides a portion of the ANSI Standard for skin exposure to a laser beam. The skin MPE values are both frequency and time dependent. Applications of these limits are discussed in the Laser Safety Calculation Section.

Table 8.4 Maximum Permissible Exposure for Skin Exposure to a Laser Beam.[a, b]

Wavelength λ (μm)	Exposure Time t (s)	MPE
Photochemical		
0.315 to 0.400	10 to 1.0×10^3	1.0 J/cm^2
0.315 to 0.400	1.0×10^3 to 3.0×10^4	1.0×10^{-3} W/cm^2
Visible and Near Infrared		
0.400 to 1.400	1.0×10^{-9} to 1.0×10^{-7}	$2.0\ C_A \times 10^{-2}$ J/cm^2
0.400 to 1.400	1.0×10^{-7} to 10	$1.1\ C_A\ t^{0.25}$ J/cm^2
0.400 to 1.400	10 to 3.0×10^4	$0.2\ C_A$ W/cm^2
Far Infrared		
1.400 to 1.500	1.0×10^{-9} to 1.0×10^{-3}	0.1 J/cm^2
1.400 to 1.500	1.0×10^{-3} to 10	$0.56\ t^{0.25}$ J/cm^2
1.400 to 1.500	10 to 3.0×10^4	0.1 W/cm^2

[a] $C_A = 1$ for $\lambda = 0.400$ to 0.700 μm; $C_A = 10^{(2[\lambda - 0.700])}$ for $\lambda = 0.700$ to 1.050 μm; $C_A = 5$ for $\lambda = 1.050$ to 1.400 μm.
[b] *Source:* ANSI Z136.1-2007.

8.11 Free Electron Lasers

A free electron laser (FEL) is a device that converts the kinetic energy of an electron into electromagnetic (EM) energy. This conversion occurs in a manner that creates coherent, monochromatic radiation emitted in a specified direction. The two basic components in a FEL are an accelerator that generates energetic electrons and a device that converts the electron's kinetic energy into EM radiation.

Most of the diversity in FEL design is associated with the device for generating energetic electrons. A number of approaches for electron beam generation are utilized including radiofrequency (RF) linear accelerators (LINACs), superconducting accelerators, microtrons, van de Graaff accelerators, induction accelerators, pulse line accelerators, and storage rings.

The conversion of electron kinetic energy into EM radiation requires an energy transfer between the electron beam and the standing wave over an extended distance. In order to achieve the requisite energy transfer, synchronization between the electron velocity and the phase velocity of the output wave must occur. Efficient energy transfer must overcome two challenges. First, in a vacuum an EM wave has a phase velocity equal to the speed of light, but the electrons have a velocity that is less than c. Second, since the beam electrons are uniformly distributed along the direction of motion, bunching must occur for the electrons to be in a phase of the EM field in which the photon wave extracts energy from the electron beam. This condition is necessary to facilitate the sustained interchange of energy from the electron beam to the EM field. The manner of extraction determines the wavelength and energy of the resultant EM field.

In order to transfer energy most efficiently, an oscillating or periodically cycling EM field is used to produce an undulating beam trajectory. This trajectory is readily achieved using insertion device magnets with flux-concentrating poles. The desired EM wave is produced in an optical resonator. This condition is similar to the buildup of energy in the optical cavity of a conventional laser. The insertion device magnet pole faces are used to concentrate the energy flux in the plane perpendicular to the direction of the electron beam.

The rate of change of an electron beam's energy in an electric field ($\mathbf{E}$) is:

$$m\,c^2\,\frac{\mathrm{d}\gamma}{\mathrm{d}t} = e\,\mathbf{v}\cdot\mathbf{E} \tag{8.26}$$

where m is the electron rest mass, γ is the Lorentz factor (Equation (7.5)), e is the electron charge, and $\mathbf{v}$ is the electron velocity. In order to extract energy from the electron beam, the right-hand side of (8.26) must be positive when averaged over an extended time interval:

$$\mathbf{v}\cdot\mathbf{E} > 0 \tag{8.27}$$

For a relativistic beam, it is a slow process to tune the FEL by changing γ because this requires adjusting magnets in the beam transport system. The magnetic field

strength is altered by changing the wiggler period (i.e., the pole spacing in the wiggler magnet), but this is an expensive wiggler modification.

The FEL has a number of unique features. It operates over a wide range of frequencies since no physical resonance is involved. The output wavelength is selected by specifying the electron beam energy and wiggler period. In addition, the FEL is capable of delivering high peak power and average output power because it transfers energy from a high power, relativistic electron beam.

8.12 Federal Regulations and Laser Standards

The Federal agency that promulgates performance standards for laser products is the Food and Drug Administration (FDA). The Occupational Safety and Health Administration (OSHA) regulates the use of lasers and specifies user qualifications, posting and labeling requirements, exposure levels, and safety equipment. The Code of Federal Regulations (CFR) containing the FDA's regulations is 21 CFR. OSHA regulations are promulgated in 29 CFR.

8.12.1 Laser Classes

In the ANSI Z136.1-2007 Standard, lasers are classified into one of seven classifications (1, 1M, 2, 2M, 3R, 3B, and 4) depending upon the relative hazard of the emitted laser radiation. The seven ANSI laser classes and their descriptions are:

1. Class 1 lasers are considered incapable of producing damaging radiation levels during operation, and are exempt from any control measures or other forms of surveillance. The Class 1 accessible emission limit (AEL) is determined from:

$$\text{AEL} = (\text{MPE})\, A_{\text{LA}} \tag{8.28}$$

   where MPE is the maximum permissible exposure for the unaided eye for the limiting exposure duration, and A_{LA} is the area of the limiting aperture for the unaided eye. Selection of the MPE and limiting aperture is discussed in the Laser Safety Calculation Section.

2. Class 1M lasers are considered to be incapable of producing hazardous exposure conditions during normal operation unless the beam is viewed with an optical instrument such as an eye-loupe (diverging beam) or a telescope (collimated beam). This class is exempt from any control measures other than to prevent potentially hazardous optically aided viewing; and is exempt from other forms of surveillance. The AEL for Class 1 is the identical energy for Class 1M, but measurement conditions differ. Class 1M or 2M laser output will be limited to the AEL for Class 3B under aided viewing measurement conditions.

3. Class 2 lasers emit in the visible portion of the spectrum (0.4 to 0.7 μm). Eye protection is normally afforded by the aversion response mechanism. The Class 2 and 2M AELs are determined in the same manner as the Class 1 AEL, except MPEs are based on an exposure duration of 0.25 s (aversion response). For continuous wave sources, the MPE is 2.5 mW/cm^2, and the AEL is 1 mW.

4. Class 2M lasers emit in the visible portion of the spectrum (0.4 to 0.7 μm). Eye protection is normally afforded by an aversion response for unaided viewing. However, Class 2M is potentially hazardous if viewed with certain optical aids. Class 1M or 2M laser output will be limited to the AEL for Class 3B under aided viewing measurement conditions. The Class 2 and 2M AELs are determined in the same manner as the Class 1 AELs, except MPEs are based on an exposure duration of 0.25 s (aversion response). For continuous wave sources, the MPE is 2.5 mW/cm^2, and the AEL is 1 mW.

5. Class 3R lasers are potentially hazardous under some direct and specular reflection viewing conditions if the eye is appropriately focused and stable. The probability of an actual injury is small, and these lasers will not pose a fire hazard or diffuse reflection hazard. Class 3R lasers have an accessible output between 1 and 5 times the Class 1 AEL for wavelengths shorter than 0.4 μm or longer than 0.7 μm, or less than 5 times the Class 2 AEL for wavelengths between 0.4 and 0.7 μm.

6. Class 3B lasers may be hazardous under direct and specular reflection viewing conditions, but are normally not a diffuse reflection or fire hazard. This laser class has an AEL that is defined in terms of wavelength ranges.
   - The first range includes lasers operating outside the retinal hazard region (< 0.4 μm or greater than 1.4 μm), and emits accessible radiant power in excess of the Class 3R AEL. However, these lasers are limited by the following conditions: (a) they cannot emit an average radiant power in excess of 0.5 W for an exposure duration (T) ≥ 0.25 s; or (b) the lasers cannot produce a radiant energy greater than 0.125 J within an exposure duration < 0.25 s.
   - The second wavelength range includes visible (0.4 to 0.7 μm) and near infrared (0.7 to 1.4 μm) lasers, which emit in excess of the AEL of Class 3R. However, these lasers: (a) cannot emit an average radiant power in excess of 0.5 W for $T \geq 0.25$ s; or (b) cannot emit a radiant energy greater than 0.03 C_A J/pulse, where C_A is a correction factor which increases the MPE in the near infrared (IR-A) spectral region. For this limit, pulses separated by t_{min} are to be considered as one pulse. For a pulsed laser, t_{min} is the maximum duration for which the MPE is the same as the MPE for a 1 ns exposure.

7. Class 4 lasers are a hazard to the eye or skin from the direct beam, and may pose a diffuse reflection or fire hazard. These lasers also produce laser-generated air contaminants and hazardous plasma radiation (i.e., UV and blue light [0.18–0.55 μm]). Class 4 lasers are those that emit radiation that exceed the Class 3B AEL.

Table 8.5 Requirements by Laser Classification.[a]

Class[b]	Procedural and Administrative Controls	Training	Medical Surveillance	Laser Safety Officer
1	Not Required	Not Required	Not Required	Not Required
1M	Required	Application Dependent[d]	Application Dependent[d]	Application Dependent[d]
2	Not Required[c]	Not Required[c]	Not Required	Not Required
2M	Required	Application Dependent[d]	Application Dependent[d]	Application Dependent[d]
3R	Not Required[c]	Not Required[c]	Not Required	Not Required[c]
3B	Required	Required	Suggested	Required
4	Required	Required	Suggested	Required

[a] Derived from ANSI-Z136.1-2007.
[b] Equipment Labels and Protective Housing are required for all laser classes.
[c] Not required except for conditions of intentional intrabeam exposure applications.
[d] Certain uses of Class 1M or 2M lasers or laser systems that exceed Class 1 or Class 2 because they do not satisfy Measurement Condition 1 may require hazard evaluation and/or manufacturer's information.

Additional requirements for the 7 ANSI Laser Classes are provided in Table 8.5. These requirements include the need for procedures and administrative controls, training, medical surveillance, a laser safety officer, equipment labels, and a protective housing.

8.12.2 Laser Safety Calculations

Conventional lasers operate in the infrared, visible, and ultraviolet regions, and free electron laser output has the potential to span the infrared, visible, ultraviolet, X-ray, and gamma-ray regions of the electromagnetic spectrum. The ionizing X-ray and gamma-ray regions are addressed using the conventional tools summarized in previous discussion. The nonionizing regions require a different approach.

A common approach for performing a laser safety calculation is to use the methodology of the American National Standards Institute publication ANSI Z136.1-2007. In order to perform a laser safety calculation, a number of parameters must be specified. These include the size of the limiting aperture, exposure time, and maximum permissible exposure (MPE). The selection of laser safety parameters depends on the format of the laser output (pulsed or continuous). The basis for the selection of these parameters is described in the subsequent discussion.

8.12.2.1 Limiting Aperture

ANSI Z136.1-2007 defines the limiting aperture based on the wavelength and pulse duration. If the beam area is larger than the limiting aperture, the standard recommends use of the actual area in the laser safety calculation. However, if the beam area is smaller than the limiting aperture, use the limiting aperture in the calculation. For radiation within the range of 400 to 1400 nm and for pulse durations between 1.0×10^{-13} and 3.0×10^{4} s, the limiting aperture diameter for the eye is 7 mm and the 3.5 mm for the skin.

8.12.2.2 Exposure Time/Maximum Permissible Exposure

For continuous wave (CW) lasers, the MPE is determined from the exposure time and the wavelength of the laser radiation. The exposure time is determined by the laser wavelength. For visible CW lasers (400–700 nm), the exposure duration is the maximum time of anticipated exposure. If purposeful staring into the beam is not intended or anticipated, the aversion (blinking) response time of 0.25 s is used as the exposure duration.

For non-visible CW Lasers (less than 400 nm or greater than 700 nm), the exposure duration is the maximum time of anticipated exposure. For the hazard evaluation of retinal exposures in the near-infrared (700–1400 nm), an exposure duration of 10 s provides an adequate hazard criterion for either incidental viewing or purposeful staring conditions. Eye movements provide a natural exposure-time limitation. For special applications, such as health care or experimental conditions, longer exposure durations may be appropriate.

For pulsed lasers, the MPE is derived from three pulsed laser calculation rules. The selected MPE is the minimum value derived from these rules. Rule 1 addresses the single-pulse MPE and specifies that the exposure time is the pulse width and the MPE is determined from the laser wavelength and the exposure duration.

The second rule protects against the average power MPE for thermal and photochemical hazards:

$$\text{MPE} - 2 = \frac{\text{CW MPE for the same wavelength}}{\text{PRF}} \tag{8.29}$$

$$\text{MPE} - 2 = \left(\frac{\text{W/cm}^2}{\text{pulses/s}}\right)\left(\frac{\text{J}}{\text{W-s}}\right) = \frac{\text{J}}{\text{cm}^2\text{-pulse}} \tag{8.30}$$

where PRF is the pulse repetition frequency.

The third rule addresses a multiple-pulse MPE for thermal hazards. Rule 3 protects against sub-threshold pulse cumulative thermal injury:

$$\text{MPE} - 3 = (\text{individual pulse MPE})\, n^{-1/4}; \quad n^{-1/4} \leq 1 \tag{8.31}$$

where n is the number of pulses during the exposure duration (T):

$$n = \text{PRF} \cdot T \tag{8.32}$$

The exposure duration is determined as noted above for CW lasers. Rule 3 does not apply to skin exposures.

8.13 Controlling Laser Radiation

Methods to control exposure to laser radiation include engineering and administrative controls and the use of personnel protective equipment. The type of control measure is governed by regulatory requirements and the laser technique employed.

Engineering controls are design features or devices that are applied to the laser, laser beam, or environment traversed by the laser beam to restrict personnel exposure. A summary of engineering controls is provided in Table 8.6.

Administrative controls are less preferable than physical or engineering controls and include procedural controls, warning signs, and labels. Standard operating

Table 8.6 Selected Engineering Controls for Laser Radiation.

Engineering Control	Description
Beam Attenuation	Absorbing media or filters. Beam expanders.
Beam Shutter	Terminates the laser beam before it exits the aperture.
Beam Path Control	Removes the beam from the area of sensitive tissues. Eliminates reflective surfaces.
Beam Path Exposure	Eliminates intrabeam viewing.
Beam Stops	Safely absorbs the exit beam without producing scattered radiation.
Door Interlocks	Links access doors to the laser power supply.
Filtered Viewing Devices	Reduces laser exposure to the eye by attenuating the laser radiation.
Key Switch	Controls activation of the power supply to the laser.
Protective Housing	Limits the emitted radiation. Interlocks should be required for removal.
Remote Monitoring/ Viewing Devices	Removes personnel from direct or scattered radiation.
Remote Firing	Firing controls permit personnel to work outside beam areas.
Warning Devices (lights and buzzers)	Alerts personnel to laser operation by providing a pre-firing signal or a time delay before energizing the laser.

procedures, safety rules, access control, education and training, and maintenance manuals are examples of laser administrative controls.

8.14
Personnel Protective Equipment

Personnel protective equipment includes laser eyewear and protective garments. Eyewear is wavelength dependent and the transmission of visible light is an important consideration. The selection of eyewear depends on a variety of factors including the wavelength of the laser radiation, MPE, and user comfort. Protective garments include clothing and gloves, which offer skin protection.

Eyes may be protected from laser radiation using protective eyewear. The attenuation of laser radiation by the eyewear is characterized in terms of its optical density. The required optical density (OD) for protective goggles to reduce the radiant exposure (H) or irradiance (E) to the maximum permissible exposure (MPE) at a distance d from the laser is given by either of the following relationships:

$$\mathrm{OD} = \log_{10}\left[\frac{H}{\mathrm{MPE}}\right] \tag{8.33}$$

$$\mathrm{OD} = \log_{10}\left[\frac{E}{\mathrm{MPE}}\right] \tag{8.34}$$

8.15
Spectral Effectiveness of Ultraviolet Radiation

The permissible exposure limit for occupational exposure to ultraviolet radiation incident upon the skin or eye depends on the wavelength of the incident radiation. The eyes and skin are exposed to ultraviolet radiation from industrial sources including electrical arcs, gas and vapor discharges, and incandescent sources. Since UV radiation often occurs over a broad frequency range, it is useful to weight the UV spectrum to obtain the integrated hazard relative to a predetermined wavelength. Performing this weighting leads to a spectral effectiveness function, which is an effective output relative to a specific UV wavelength.

The effective irradiance (E_{eff}) of a broadband UV source is weighted against the peak of the spectral effectiveness curve (270 nm) and is obtained from the relationship:

$$E_{\mathrm{eff}} = \sum_{\lambda} E_{\lambda}\, S_{\lambda}\, \Delta\lambda \tag{8.35}$$

where

E_{eff} = effective irradiance relative to a monochromatic source at 270 nm ($\mathrm{W/cm^2}$)

λ = wavelength of the UV radiation (nm)
E_λ = spectral irradiance (W/cm^2-nm) at wavelength λ
S_λ = relative spectral effectiveness at wavelength λ
$\Delta\lambda$ = band width (nm)

8.16 Scenarios

Scenario 8.1

You are a senior health physicist at a War Department radar facility at the Bertelson Air Force Base (BAFB). The radar system is a circular-aperture, parabolic dish microwave radar that has a diameter of 10 m, a peak power of 1 MW, a gain of 20 dB, and a duty factor of 0.001. This system operates at a frequency 3000 MHz.

The facility is preparing for its initial operation and you are responsible for ensuring all applicable standards are met. The base commander, Major General Emily Sylvester Post, is concerned with safety and identified the following questions for your attention.

BAFB uses the 1992 ANSI Standard for worker protection. The BAFB standard for power density at 3000 MHz is 10 mW/cm^2 for controlled environments and 2 mW/cm^2 for uncontrolled environments. For hazard-assessment purposes, BAFB procedures define the near-field and far-field distances as $D^2/4\,\lambda$ and $0.6\,D^2/\lambda$, respectively.

8.1 What is the major effect on the human body from exposure to microwave radiation? List four factors affecting the extent of injury.
8.2 List five items of information you need to perform the hazard evaluation.
8.3 What is the average power that the antenna radiates?
8.4 What are the near- and far-field distances?
8.5 What are the relationships of power density and distance in the near and far fields?
8.6 What is the significance of near and far fields with respect to power density measurements?
8.7 What is the maximum power density at the near field?
8.8 Is an individual who works at about 300 m along the beam axis overexposed according to the 1992 ANSI Standard?
8.9 You install a fence at 250 m around the installation. Is it possible for anyone outside that fence to exceed the 1992 ANSI Standard for continuous exposure?
8.10 Following initial operations, a worker complained about being overexposed to microwave radiation. You plan to investigate the allegation. List two steps you will take.

Scenario 8.2

You are the laser safety officer at the Omar D'Tentmaker Research Laboratory at the University of Southeastern Rhode Island. This facility utilizes a number of lasers including a continuous wave (CW) neon laser. The neon laser is operated from the center of a laboratory that is 20-meters square and has the following operating parameters: an output power of 20 W, a 1/e beam diameter of 2 mm, a beam divergence of 1 mrad, and an output wavelength of 540 nm.

The beam is directed towards the center of one wall where it terminates on a diffusely reflecting target that is mounted to the wall. The target's spectral reflectance (ρ_λ) is 0.90.

The applicable safety standard is:

Maximum Permissible Exposure for Direct Ocular Exposure Intra-beam Viewing from a Laser Beam.[a, b]

Wavelength λ (μm)	Exposure Time t (s)	MPE
0.400 to 0.700	1.0×10^{-9} to 1.8×10^{-5}	5.0×10^{-7} J/cm^2
0.400 to 0.700	1.8×10^{-5} to 10.0	$1.8\ t^{3/4} \times 10^{-3}$ J/cm^2
0.400 to 0.550	10 to 1.0×10^4	1.0×10^{-2} J/cm^2
0.550 to 0.700	10 to T_1	$1.8\ t^{3/4} \times 10^{-3}$ J/cm^2
0.550 to 0.700	T_1 to 1.0×10^4	$10\ C_B \times 10^{-3}$ J/cm^2
0.400 to 0.700	10^4 to 3.0×10^4	$C_B \times 10^{-6}$ W/cm^2

[a] $C_B = 1$ for $\lambda = 0.400$ to 0.550 μm; $C_B = 10^{(15[\lambda - 0.550])}$ for $\lambda = 0.550$ to 0.700 μm.
$T_1 = 10 \times 10^{[20(\lambda - 0.550)]}$ s for $\lambda = 0.550$ to 0.700 μm.

[b] Derived from ANSI Z136.1 as presented in the 1999 American Board of Health Physics Examination question.

8.11 Match the following radiometric quantities to their appropriate units.

a. Radiant energy	1. W/cm^2
b. Radiant power	2. W/sr
c. Radiant intensity	3. J
d. Radiance	4. W/sr-cm^2
e. Radiant exposure	5. J/cm^2
	6. W
	7. J/sr-cm^2

8.12 Briefly define and describe each of the following terms:

a. Nominal Hazard Zone (NHZ).

b. Nominal Ocular Hazard Distance (HOHD).

Why is the specification NHZ more appropriate for indoor laser laboratories?

8.13 Estimate and describe the nominal hazard zone (NHZ) for this laser laboratory. Assume the intrabeam MPE applies.

8.14 What is the minimum optical density (OD) of protective eyeware required to reduce the laser irradiance below the intrabeam MPE? Assume a 0.25 s accidental viewing time.

8.15 Assume a minimum OD of 5 is required for protective eyewear. A researcher has obtained specifications from three proposed laser eyewear manufacturers. These have the following optical densities at the indicated wavelengths and luminous transmission values:

Specifications	Brand X	Brand Y	Brand Z
OD @ 400 nm	3	5	5
OD @ 450 nm	4	7	7
OD@ 500 nm	5	7	8
OD @ 540 nm	5	7	8
OD @ 600 nm	3	4	7
OD @ 650 nm	1.5	3	6
Luminous Transmission	35 %	6 %	< 1 %

Which manufacturer's eyewear (Brand X, Y, or Z) would you recommend as best to use in this laser laboratory and why?

8.16 Match the following Food and Drug Administration (FDA) laser class with the appropriate characteristic that best describes the class.

1. Class I	a. can damage skin or eye from diffuse reflection.
2. Class II	b. not an ocular hazard.
3. Class III	c. 0.25 s exposure threshold.
4. Class IV	d. momentary intrabeam viewing is hazardous.

8.17 Briefly describe the three basic components of all lasers.

Scenario 8.3

You are employed by Quantum Laser Associates, a consulting firm that conducts training seminars. Upon the completion of a seminar at Misguided Electronics, the instructor will administer a short quiz to assess student achievement. Develop the answer key to the following quiz.

8.18 As it applies to lasers, what does the term "stimulated emission" mean?

8.19 Name three characteristics of laser radiation that uniquely distinguish it from other radiation sources.

8.20 All lasers and laser systems are classified in accordance with the accessible emission limits (AELs). Name each of the ANSI laser classes and give a brief (general) description of each. Specific numerical AELs do not have to be given.

8.21 For the four laser types specified below, place a Y or N in the boxes indicating whether the tissue may be significantly affected by the laser radiation.

Laser Type	Wavelength (μm)	Tissues Significantly Affected			
		Skin	Cornea	Lens	Retina
CO_2	10.6				
Nd-YAG	1.06				
Ar	0.325				
HeNe	0.633				

8.22 Name the Federal agency, which promulgates manufactures' performance standards for laser products.

8.23 Which title of the Code of Federal Regulations contains this agency's regulations?

8.24 Name six controls that are specified in the ANSI standard for control of the most hazardous class of lasers.

Scenario 8.4

You are a military health physicist assigned as laser safety officer at the Hillary Lumberman Air Force Base in Jackass Flats, Arkansas. The following questions relate to the characteristics of a HeNe laser (633 nm) and ruby laser (694.3 nm). A portion of the applicable protection standard follows:

Maximum Permissible Exposure for Direct Ocular Exposure Intra-Beam Viewing from a Laser Beam.[a, b]

Wavelength λ (μm)	Exposure Time t (s)	MPE
0.400 to 0.700	10^{-9} to 1.8×10^{-5}	5×10^{-7} J/cm^2
0.400 to 0.700	1.8×10^{-5} to 10	$1.8\ t^{3/4} \times 10^{-3}$ J/cm^2
0.400 to 0.550	10 to 10^4	1×10^{-2} J/cm^2
0.550 to 0.700	10 to T_1	$1.8\ t^{3/4} \times 10^{-3}$ J/cm^2
0.550 to 0.700	T_1 to 10^4	$10\ C_B \times 10^{-3}$ J/cm^2
0.400 to 0.700	10^4 to 3×10^4	$C_B \times 10^{-6}$ W/cm^2

[a] $C_B = 1$ for $\lambda = 0.400$ to 0.550 μm; $C_B = 10^{15(\lambda - 0.550)}$ for $\lambda = 0.550$ to 0.700 μm; $T_1 = 10 \times 10^{20(\lambda - 0.550)}$ s for $\lambda = 0.550$ to 0.700 μm.

[b] Derived from ANSI Z136.1 as presented in the 2000 American Board of Health Physics Examination question.

8.25 Match the following ANSI Z136.1-2007 correction factors with their definitions.

Correction Factors	Definitions
1. C_A	a. Correction factor which increases the MPE for ocular exposure, based on pre-retinal absorption of radiant energy in the spectral region between 1.15–1.40 μm.
2. C_B	b. Correction factor which reduces the MPE for repetitive-pulse exposure of the eye.
3. C_C	c. Correction factor which increases the MPE in the near-infrared (IR-A) region (0.7–1.4 μm), based on reduced absorption properties of melanin pigment granules found in the skin and in the retinal pigment epithelium.
4. C_E	d. Correction factor used for calculating the extended source MPE for the eye from the point source MPE, when the source subtends a visual angle exceeding α_{min}.
5. C_P	e. Correction factor which increases the MPE in the red end of the visible spectrum (0.45–0.60 μm), based on greatly reduced photochemical hazards.
	f. Correction factor which increases the MPE for repetitive-pulse exposure of the eye.
	g. Correction factor which decreases the MPE for ocular exposure, based upon pre-retinal absorption of radiant energy in the spectral region between 1.15–1.40 μm.

8.26 What is the fundamental and first harmonic wavelength for a Nd-YAG laser?

a. 10.6 μm and 5.3 μm.
b. 10.6 μm and 21.2 μm.
c. 1.064 μm and 0.532 μm.
d. 1.064 μm and 2.128 μm.
e. 0.633 μm and 0.317 μm.

8.27 A ruby laser has the following characteristics: pulse energy = 20 J, pulse rate = 2/min, pulse duration = 10 μs, beam divergence = 15 mrad, and aperture diameter = 2 mm. What is the required optical density for protective goggles to reduce the radiant exposure to the maximum permissible exposure at one meter from the laser?

8.28 A 50 mW HeNe continuous-emission laser is used to give a light show at an outdoor concert. The laser has an aperture diameter of 3 mm and a beam divergence of 0.3 mrad. Calculate the emergence irradiance.

8.29 For the conditions of question 8.28, calculate the hazardous intra-beam viewing distance (that is the distance where the MPE is not exceeded assuming continuous viewing). Neglect atmospheric attenuation.

Scenario 8.5

You are the designated laser safety officer (LSO) at the Army's Dakota Area Test Range (DATR), an outdoor military battlefield tactics range, which has controlled air space in order to exclude commercial or private aviation from the area. A variety of military aircraft, ground troops, and mechanized equipment utilize the range. In addition to conventional weapons, numerous lasers are actively employed at DATR. The DATR lasers are operated for long exposure durations during field activities. For all potential exposures, purposeful staring into a laser beam is neither intended nor anticipated. A table of maximum permissible exposure values follows.

Maximum Permissible Exposure for Direct Ocular Exposure Intra-Beam Viewing from a Laser Beam.[a, b]

Wavelength (μm)	Exposure Duration (s)	Maximum Permissible Exposure	
		(J/cm^2)	(W/cm^2)
0.4 to 0.7	10^{-9} to 18×10^{-6}	0.5×10^{-6}	–
0.4 to 0.7	18×10^{-6} to 10	$1.8\ t^{3/4} \times 10^{-3}$	–
0.4 to 0.55	10 to 10^4	10×10^{-3}	–
0.55 to 0.7	10 to T_1	$1.8\ t^{3/4} \times 10^{-3}$	–
0.55 to 0.7	T_1 to 10^4	$10\ C_B \times 10^{-3}$	–
0.4 to 0.7	10^4 to 3×10^4	–	$C_B \times 10^{-6}$
0.7 to 1.05	10^{-9} to 18×10^{-6}	$0.5\ C_A \times 10^{-6}$	–
0.7 to 1.05	18×10^{-6} to 10^3	$1.8\ C_A\ t^{3/4} \times 10^{-3}$	–
0.7 to 1.05	10^3 to 3×10^4	–	$320\ C_A \times 10^{-6}$
1.05 to 1.4	10^{-9} to 50×10^{-6}	$5\ C_C \times 10^{-6}$	–
1.05 to 1.4	50×10^{-6} to 10^3	$9.0\ C_C\ t^{3/4} \times 10^{-3}$	–
1.05 to 1.4	10^3 to 3×10^4	–	$1.6\ C_C \times 10^{-3}$

[a] Limiting Aperture Diameter: 0.7 cm
$T_1 = 10 \times 10^{20(\lambda - 0.55)}$ for 0.55 to 0.7 μm; $C_A = 1.0$ for 0.4 to 0.7 μm
$C_B = 1.0$ for 0.4 to 0.55 μm; $C_A = 10^{2(\lambda - 0.7)}$ for 0.7 to 1.05 μm
$C_B = 10^{15(\lambda - 0.55)}$ for 0.55 to 0.7 μm; $C_A = 5.0$ for 1.05 to 1.4 μm
$C_C = 1.0$ for 1.05 to 1.15 μm; $C_A = 10^{18(\lambda - 1.15)}$ for 1.15 to 1.2 μm
$C_C = 8.0$ for 1.2 to 1.4 μm

[b] Derived from ANSI Z136.1 as presented in the 2001 American Board of Health Physics Examination question.

Maximum Permissible Exposure for Skin Exposure to a Laser Beam.[a, b]

Wavelength (μm)	Exposure Duration (s)	Maximum Permissible Exposure	
		(J/cm^2)	(W/cm^2)
0.4 to 1.4	10^{-9} to 10^{-7}	$2\ C_A \times 10^{-2}$	–
0.4 to 1.4	10^{-7} to 10	$1.1\ C_A \times t^{1/4}$	–
0.4 to 1.4	10 to 3×10^4	–	$0.2\ C_A$

[a] Limiting Aperture Diameter: 0.35 cm
C_A is defined above.
[b] Derived from ANSI Z136.1 as presented in the 2001 American Board of Health Physics Examination question.

8.30 For a laser with the following specifications [Wavelength = 1.06 μm, Pulse Repetition Frequency = 0.5 Hz, Beam Width (Laser Exit – 1/*e*) = 0.9 cm, Energy/Pulse = 1.0 J, Pulse Width = 25×10^{-9} s, and Divergence = 1 mrad (1/*e*)], calculate an appropriate MPE. Justify the selection of exposure duration used for the MPE calculation.

8.31 For the laser with specifications given in question 8.30 and an MPE of 7.0×10^{-6} J/cm^2, determine the nominal ocular hazard distance (NOHD).

8.32 For the two continuous-wave (CW) lasers with parameters listed in the following table, determine the organs at risk for biological damage from unprotected exposure at 1) the exit port and 2) 2 kilometers (km). Justify the selection of the exposure duration for MPE calculations.

Laser Type	Power (W)	Beam Diameter (1/*e*) (cm)		λ (μm)	Irradiance (W/cm^2)		Divergence (1/*e*) (radians)
		Exit	@2 km		Exit	@2 km	
Argon	0.1	8	21	0.515	2.0×10^{-3}	2.9×10^{-4}	65×10^{-6}
GaAs	0.02	0.8	200.8	0.905	0.04	6.3×10^{-7}	1.0×10^{-3}

8.33 For the two lasers listed in question 8.32, what additional hazards exist from use of the lasers on this range? Assume that aircraft, equipment, or personnel are within 2 km of the lasers' exit ports and a long exposure duration may be used. List only the two most important additional hazards. If the answer is exclusive to just one of the lasers, note this in your solution.

8.34 For the lasers of emission wavelength listed in the left column of the following table, select the tissue(s) with significant energy absorption from the list on the right column. Select only one answer per laser wavelength. Selections from the right column may be used more than once.

Wavelength (μm)	Tissues with Significant Energy Absorption
1. 1.3 _____	A. Retina and Cornea
2. 5.0 _____	B. Skin and Retina
3. 0.28_____	C. Skin and Cornea
4. 0.5 _____	D. Skin, Retina, Cornea, and Lens
5. 0.36_____	E. Skin and Lens

Scenario 8.6

Gigantic Power and Light (GP&L) is reviewing business opportunities to enhance its profit margin, and is developing a communications system that operates at 2800 MHz. You have been retained by GP&L to assist in its technology review. Assume that accessible power densities are as large as 10 mW/cm^2.

This company has adopted "GP&L-Reference Man" for electromagnetic field calculations. In this model, the body is treated as a slab with a surface area of 0.9 m^2 and a thickness of 2.0 cm. The composition of the slab is water. Model parameters of relevance include an average conductivity of body tissue of 2.2/Ω-m and an average relative dielectric constant of body tissue of 45. Useful constants include: the permittivity of free space = 8.85×10^{-12} $C^2/nt\text{-}m^2$, permeability of body tissue = 1.26×10^{-6} nt/A^2, density of body tissue = 1.0 g/cm^3, and specific heat of body tissue = 4178 J/kg-C°. The following unit conversions should be considered when solving this problem: Ω = $kg\text{-}m^2/s\text{-}C^2$, A = C/s, and nt = $kg\text{-}m/s^2$.

8.35 What is the absorption coefficient for body tissue?
8.36 At what depth in tissue will the power density be reduced to 0.1 mW/cm^2?
8.37 What is the penetration depth in tissue?
8.38 What is the maximum time an individual can reside in the communications field assuming that a limiting tissue temperature rise is 2 °C. Assume the body's cooling mechanisms do not significantly reduce the heat rise caused by the incident radiation. The average power density absorbed in the GP&L phantom is reasonably well approximated by the power density value calculated at 1 cm.

Scenario 8.7

You have been hired as the chief health physicist by Enrichment Technology Group (ETG), an emerging technology company. ETG is developing laser techniques to enrich a variety of isotopes for commercial sale.

A preliminary set of experiments utilizes a dye laser operating at 600 nm in a pulsed mode. The laser has a pulse repetition frequency of 100 pps and a pulse width of 1.0 ms. The initial experiments produce enriched B-10 atoms. A photodiode detector is used for all current measurements.

The average current equivalent power measured with a photodiode detector is 40 μA. This detector has a calibration factor of 2 μW/μA. The measured $1/e^2$ beam diameter at 2.0 (10.0) meters from the laser aperture is 3.0 mm (6.0 mm).

8.39 What is the divergence angle of the dye laser?
8.40 What is the laser's duty cycle?
8.41 What is the average laser power?
8.42 What is the peak power?
8.43 What is the energy per pulse?

Scenario 8.8

You are employed by Quantum Universal Applications Corporation (QUAC), a high-tech manufacturing organization. As a principle health physicist in QUAC's health and safety department, you are reviewing a recently completed hazards assessment for corporate facilities. You are particularly concerned with the nonionizing radiation hazards present in the Green Bay Division's Quantum Gravitational Packing Laboratory.

In the space biology laboratory, a UV light box installation has been completed. It is used to view and photograph electrophoresis gels stained with rubidium bromide and operates at a peak wavelength of 300 nm. The laboratory staff also use a biological safety cabinet (BSC) containing a bulb with a peak wavelength of 280 nm. Gel manipulation requires about 20 minutes each day. Work in the BCS requires about 30 minutes each day. The bulb in the cabinet is illuminated at all times.

Work is also performed in a microwave laboratory that has acquired a surplus microwave transmitter that will be used in microwave health effects experiments. This device operates with a frequency of 10 GHz, a peak power of 2000 W, a pulse width of 1 ms, a pulse repetition rate of 200 pps, an antenna gain of 16 dB, and a maximum horn antenna dimension of 0.2 m.

A table of ultraviolet radiation exposure threshold limit values (TLVs) and spectral weighting functions are provided.

Ultraviolet Radiation Exposure TLVs and Spectral Weighting Functions.

Wavelength (n/m)	TLV (J/m^2)	TLV (mJ/cm^2)	Relative Spectral Effectiveness (S_λ)
250	70	7.0	0.430
260	46	4.6	0.650
270	30	3.0	1.000
280	34	3.4	0.880
290	47	4.7	0.610
300	100	10.0	0.300
310	2000	200.0	0.015

$h = 6.626 \times 10^{-34}$ J-s; $1\ J = 6.24 \times 10^{12}$ MeV

8.44 The manufacturer reports in its operating manual that the irradiance at 60.96 cm from the light box is 1 μW/cm^2 and at 15.24 cm from the BSC the level is 0.6 μW/cm^2. Assuming the entire irradiance is at the peak wavelengths given, is the TLV exceeded?

8.45 Is the assumption of irradiance at the peak wavelength accurate?

8.46 Briefly describe a better means of assessing the exposure to an individual working with this equipment.

8.47 What are two biological tissues at risk from excessive UV light exposure? How would excessive exposure be manifest for each of these tissues, i.e., what symptom would be present in each case?

8.48 List two steps you would recommend to lower the exposure to UV light in this lab.

8.49 List two precautions that could be followed to prevent harm to technicians working in the microwave laboratory.

8.50 The microwave device's horn antenna transmits into the room and the closest human access distance is 3 meters away. Assuming far-field conditions, calculate the maximum plane-wave free space power density at this distance in units of milliwatts per square centimeter.

8.51 The microwave laboratory staff are not permitted in the test room when the device is energized. However, on this day a lab worker ignores a warning sign on the door, enters the room, and remains there for 2 minutes before leaving for the day. Measurements in the room indicate the lab worker was exposed to a free-space power density of 25 mW/cm^2. No levels could be detected outside the room. Was the lab worker exposed in excess of the ACGIH or ANSI microwave recommendations?

8.52 The microwave laboratory has purchased a microwave device to cure a particular form of adhesive. The device operates at a frequency of 2400 MHz. Workers are concerned about possible health effects including cancer induction. Can the radiation emitted by this source cause ionization and subsequent damage to DNA?

8.53 What is the primary effect of the question 8.52 microwave device on tissue? Describe why it causes this effect.

8.54 The ALARA principle is generally applied to radiation exposure controls while most of the microwave laboratory hazards are limited by TLVs. What is the basis for the difference between these two concepts?

Scenario 8.9

You are interviewing for the position of Laser Safety Officer (LSO) at the University of Wisconsin Packertown (UWP). During a normal workweek, an LSO is asked a variety of questions concerning laser hazards and safety. UWP administration wants to ensure a competent selection. The previous LSO was a political appointee and thought the term laser meant, "lunar amplification for the saturated emission of rocks".

To avoid a similar hiring error, the interview process includes questions to test the technical competence of candidate LSOs. The questions and supporting data follow.

Selected Ocular Transmission and Absorption Data.[a]

Wavelength (nm)	Percentage transmission for the human ocular system in the ocular focus region (400–1400 nm)	Percentage absorption in the human pigment epithelium
248	NA	0
337	NA	0
628	95	45
10 600	NA	0

[a] The limiting aperture of the eye is 7 mm.

8.55 Preventing eye injury is a primary concern of laser safety programs. Identify the anatomical structure(s) of the eye that are the most sensitive to damage for each of the following lasers:
   a. Far-infrared, carbon dioxide laser (10.6 μm).
   b. Visible, gold vapor laser (0.628 μm).
   c. UV-A, nitrogen laser (0.337 μm).
   d. UV-C, krypton fluoride excimer laser (0.248 μm).
   Justify the selection of the anatomical structure(s).

8.56 Maximum Permissible Exposure Limits vary considerably within a narrow range of the visible spectrum. For example, MPEs of 2.5×10^{-3} W/cm^2 and 1.7×10^{-5} W/cm^2 apply respectively to quarter-second exposures for wavelengths of 0.647 μm (red) and 0.530 μm (green). Briefly explain the source(s) of these differences.

8.57 Deleterious effects to the skin are associated with exposure to lasers with wavelengths in the ultraviolet region of the electromagnetic spectrum. List the skin effect(s) associated with the following UV bands:
   a. UV-A (0.315–0.400 μm).
   b. UV-B (0.280–0.315 μm).
   c. UV-C (0.100–0.280 μm).

8.58 Lasers have gained wide acceptance for refractive eye surgery. Surgical units, such as LASIK (Laser In Situ Keratomileusis) systems, are typically equipped with laser alignment devices, which operate in the visible region. Assuming that the beam divergence is negligible, calculate the maximum allowable power output (in units of mW) for the following alignment system. The system includes a continuous-wave diode laser operating at a wavelength of 635 nm with an associated MPE of 0.01 J/cm^2. The laser to eye distance is 0.2 m and the beam has a diameter of 3.5 mm. A patient's blinking reflex time is inhibited for surgery and the maximum alignment time is 1.2 s per eye.

8.59 UWP's "Fighting Cheeze Men" football team plans to add a laser show to their pre-game entertainment routine. This show will include figure tracing and direct audience scanning. The proposed laser array includes a Q-Switched 40 W Nd:YAG laser operating at a wavelength of 532 nm (1st harmonic). Laser array design specifications include a 20 μs pulse width, a 25 kHz pulse repetition frequency, a 0.2 milliradian beam divergence, a beam aperture diameter of 2 mm, and a scanning rate of 20 000 cm/s. The minimum laser-to-audience distance is 25 m and the MPE is $1.8\ t^{3/4} \times 10^{-3}$ J/cm^2.

Using the MPE as your guide, is this equipment appropriate for use in such a show? Assume no optically aided viewing.

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Part III
Answers and Solutions

The solution section of this text further develops the theory and introduces additional applications of the principles developed in Parts I and II. The answers and solutions presented in Part III illustrate many of the practical difficulties encountered in real-world applications. The student is encouraged to carefully examine these solutions in order to gain the maximum benefit from this text.

Solutions for Chapter 1

Scenario 1.1

Question 1.1: c

The specific activity of C-14 in living organisms is relatively constant during the organism's lifetime. Following the organism's death, the C-14-specific activity decreases at a rate governed by the C-14 half-life.

Question 1.2

The desired age can be determined by solving the activity equation for t

$$A(t) = A_0 \exp(-\lambda t)$$
$$= \text{activity (Curies, dpm, cpm, Bq, etc.) at time } t$$

where
t = time
λ = disintegration constant (time^{-1}) = $\ln(2)/T_{1/2}$
A_0 = initial activity

The age is given by

$$t = (-1/\lambda) \ln(A/A_0)$$

where
λ = $(0.693)/5715 \text{ years} = 1.21 \times 10^{-4}/\text{year}$
A/A_0 = $(1.5 \times 10^{-1} \text{ Bq/g})/(1.67 \times 10^{-1} \text{ Bq/g}) = 0.898$

$$t = -\ln(0.898)/(1.21 \times 10^{-4}/\text{year}) = 889 \text{ years}$$

Scenario 1.2

Question 1.3

$$a \rightarrow b \rightarrow c$$

where
a = Mo-99
b = Tc-99m
c = Tc-99

$$A_b = C\,Y\,\frac{\lambda_b\,A_{a_0}}{\lambda_b - \lambda_a}\left[\exp(-\lambda_a\,t) - \exp(-\lambda_b\,t)\right]$$

where
C = chemical yield = 0.95
Y = fractional yield = 0.876
λ_b = $(0.693)/T_b = 0.693/6.0$ hr $= 1.155 \times 10^{-1}$/hr for Tc-99m
λ_a = $(0.693)/T_a = 0.693/67.0$ hr $= 1.034 \times 10^{-2}$/hr for Mo-99

$$\begin{aligned} A_b &= (0.95)\,(0.876) \\ &\quad \times [(1000\text{ mCI})\,(1.155 \times 10^{-1}\text{ hr})/(1.55 \times 10^{-1} - 1.034 \times 10^{-2}/\text{hr}] \\ &\quad \times [\exp(-1.034 \times 10^{-2} \times 48) - \exp(1.155 \times 10^{-1} \times 48)] \\ &= 553\text{ mCi} \end{aligned}$$

Question 1.4

The Tc-99m activity obtained by a second milking 24 hr later is given by

$$A_b = C\,Y\,\frac{\lambda_b\,A'_{a_0}}{\lambda_b - \lambda_a}\left[\exp(-\lambda_a\,t) - \exp(-\lambda_b\,t)\right] + C\,A_{b_1}\exp(-\lambda_b\,t)$$

where
t = 24 hr
C = 0.95
Y = 0.876
λ_b = 1.155×10^{-1}/hr
λ_a = 1.034×10^{-2}/hr

The Mo-99 activity remaining after 48 hours is given by

$$A'_{a_0} = A_{a_0}\exp(-\lambda_a\,t')$$

where

t' = 48 hr (assumes generator arrives at the hospital 48 hr after its manufacture)

A'_{a_0} = 1000 mCi exp [-1.034×10^{-2}/hr × 48 hr] = 609 mCi

A_{b_1} = 5% Tc-99m left over from the last milking

= {[553 mCi/(0.95)] – 553 mCi} = 29 mCi

A_b = (0.95) {0.876) (609 mCi)
× (1.155×10^{-1}/hr)/(1.155×10^{-1}/hr – 1.034×10^{-2}/hr)
× {exp [$-(1.034 \times 10^{-2}$/hr × 24 hr)] – exp [$-(1.155 \times 10^{-1}$/hr × 24 hr)]
+ (29 mCi) (0.95) exp [$-(1.155 \times 10^{-1}$/hr × 24 hr)]}

= 401 mCi

Scenario 1.3

Question 1.5: d

Because $\lambda_B > \lambda_A$, the parent and daughter will eventually reach the condition of transient equilibrium.

Question 1.6

$$T_{1/2}(A) = T_A = 10 \text{ hr} = 0.693/\lambda_A$$
$$T_{1/2}(B) = T_B = 1 \text{ hr} = 0.693/\lambda_B$$

The daughter will reach its maximum activity at a time t given by

$$t = \{\ln(\lambda_B/\lambda_A)\}/(\lambda_B - \lambda_A)$$
$$= \{\ln[0.693/1 \text{ hr})/(0.693/10 \text{ hr})]\} \times [(0.693/1 \text{ hr}) - (0.693/10 \text{ hr})]^{-1}$$
$$= 3.69 \text{ hr}$$

Scenario 1.4

Question 1.7: b

The tissue dose from thermal neutrons is determined by the (n, γ) reaction with hydrogen and the (n, p) reaction with nitrogen.

Question 1.8: e

Tissue dose from fast neutrons is due principally to elastic scattering with nuclei.

Question 1.9: c

The most probable process for energy deposition by a 1-MeV photon in tissue is Compton scattering.

Question 1.10: a

Inelastic scattering by atomic electrons is the principal mechanism of dose deposition by a 5-MeV alpha particle that stops in tissue.

Question 1.11: d

Inelastic scattering by atomic electrons is the principal mechanism of dose deposition by a 100-keV beta particle that stops in tissue.

Question 1.12: d

$$100 \text{ keV} \times \frac{1000 \text{ eV}}{\text{keV}} \times \frac{1 \text{ ion pair}}{34 \text{ eV}} = 2941 \text{ ip}$$

Question 1.13: a

$$100 \text{ keV} \times \frac{1000 \text{ eV}}{\text{keV}} \times \frac{1 \text{ ion pair}}{3.4 \text{ eV}} = 29412 \text{ ip}$$

Question 1.14: b

A nuclide that undergoes orbital electron capture emits a neutrino and the characteristic X-rays of the daughter.

Question 1.15: c

The specific gamma-ray emission rate for Cs-137 is approximately $0.33 \text{ R hr}^{-1} \text{ Ci}^{-1} \text{ m}^2$.

Question 1.16: e

The thyroid is an organ for which the ALI is determined by the limit for non-stochastic effects.

Scenario 1.5

M_p and M_d are expressed as mass defects (p = parent and d = daughter).

Question 1.17: c

I-126 (K-capture) → Te-126

For K-capture:

$$M_p = M_d + b + Q$$

or

$$Q = M_p - M_d - b = [-87.90 - (-90.05) - 0.03]\ \text{MeV} = 2.12\ \text{MeV}$$

Question 1.18: c

I-126 (positron decay) → Te-126

For positron decay:

$$M_p = M_d + 2\,M_e + Q$$

or

$$Q = M_p - M_d - 2\,M_e = [-87.90 - (-90.05) - 2\,(0.51)]\ \text{MeV} = 1.13\ \text{MeV}$$

Question 1.19: e

I-126 (beta decay) → Xe-126

For beta day:

$$M_p = M_d + Q$$

or

$$Q = M_p - M_d = [-87.90 - (-89.15)]\ \text{MeV} = 1.25\ \text{MeV}$$

Question 1.20: e

For both internal and external exposures, the antineutrino will be the least significant dose contributor. Neutrino and antineutrino interaction cross-sections are much smaller than those for the other particles.

Question 1.21: d

The 32-keV Te X-rays are released when the electron capture event creates a vacancy in the inner shells and electrons from the outer shells fill the vacancy.

Scenario 1.6

Question 1.22: d

$$\lambda = \text{disintegration constant} = \ln(2)/T_{1/2}$$
$$\overline{T} = 1/\lambda = T_{1/2}/0.693 = 64.2 \text{ hr}/0.693 = 92.6 \text{ hr}$$

Question 1.23: c

$$\text{SA} = \lambda N$$

where
N = number of atoms per gram of Y-90

$$\text{SA} = \frac{0.693}{64.2 \text{ hr}} \frac{1 \text{ hr}}{3600 \text{ s}} \frac{6.02 \times 10^{23} \text{ atoms/mole}}{90 \text{ g/mole}} \frac{1 \text{ dis}}{\text{atom}} \frac{1 \text{ Bq}}{\text{dis/sec}} \frac{1 \times 10^3 \text{ g}}{\text{kg}}$$
$$= 2.01 \times 10^{19} \text{ Bq/kg}$$

Question 1.24: d

Using the serial decay relationship, we obtain

$$A(\text{Y-90}) = \frac{\lambda(\text{Y-90})\, A_0(\text{Sr-90})}{\lambda(\text{Y-90}) - \lambda(\text{Sr-90})} \{\exp[-\lambda(\text{Sr-90})\, t] - \exp[-\lambda(\text{Y-90})\, t]\}$$

At $t = 72.0$ hr,

$$A(\text{Y-90}) = 3.4 \text{ MBq}$$

$$\lambda(\text{Sr-90}) = (0.693/27.7 \text{ year})\,(1 \text{ year}/365 \text{ days})\,(1 \text{ day}/24 \text{ hr})$$
$$= 2.86 \times 10^{-6}/\text{hr}$$
$$\lambda(\text{Y-90}) = (0.693/64.2 \text{ hr}) = 1.08 \times 10^{-2}/\text{hr}$$

The initial activity of Sr-90 is obtained by solving the first equation for this quantity:

$$A_0(\text{Sr-90}) = \frac{A(\text{Y-90};\, t = 72\ \text{hr})\,[\lambda(\text{Y-90}) - \lambda(\text{Sr-90})]}{\lambda(\text{Y-90})} \times \frac{1}{\exp[-\lambda(\text{Sr-90})\,t] - \exp[-\lambda(\text{Y-90})\,t]}$$

$$= \frac{(3.4\ \text{MBq})\,(1.08 \times 10^{-2}/\text{hr} - 2.86 \times 10^{-6}/\text{hr})}{(1.08 \times 10^{-2}/\text{hr})} \times \frac{1}{[\exp(-2.86 \times 10^{-6}/\text{hr} \times 72\ \text{hr}) - \exp(-1.08 \times 10^{-2}/\text{hr} \times 72\ \text{hr})]}$$

$$= 3.4\ \text{MBq}/(1.0 - 0.46) = 6.29\ \text{MBq}$$

Scenario 1.7

Question 1.25

$$\dot{X} = \frac{i}{V\,p} \times \frac{T}{T_{\text{STP}}} \times \frac{P_{\text{STP}}}{P}\,k$$

where

T_{STP} = 0 °C = 273 K
P_{STP} = 760 mm Hg
k = $[2.58 \times 10^{-4}\ \text{C/kg-R}]^{-1}$

$$\dot{X} = \frac{1 \times 10^{-9}\ \text{C/s}}{4\ \text{cm}^3 \times 1.293 \times 10^{-6}\ \text{kg/cm}^3} \times \frac{283\ \text{K}}{273\ \text{K}} \times \frac{760\ \text{mm Hg}}{755\ \text{mm Hg}}\,k$$

$$= \frac{2.02 \times 10^{-4}\ \text{C/kg-s}}{2.58 \times 10^{-4}\ \text{c/kg-R}} = 0.783\ \text{R/s}$$

Question 1.26

The optimum counting time for a fixed counting interval is obtained from the relationship

$$t_{s+b}/t_b = (R_{s+b}/R_b)^{1/2}$$

where

t_{s+b} = count time for the sample
t_b = count time for the background
R_{s+b} = sample count rate = (2400 + 300) cpm
R_b = background count rate = 300 cpm
$t_{s+b} + t_b$ = 100 min

$$
\begin{aligned}
t_{s+b}/t_b &= t_{s+b}/(100\ \text{min} - t_{s+b}) \\
&= [(2400\ \text{cpm} + 300\ \text{cpm})/(300\ \text{cpm})]^{1/2} \\
t_{s+b}/(100\ \text{min} - t_{s+b}) &= 3 \\
t_{s+b} &= 75\ \text{min}
\end{aligned}
$$

Question 1.27

Sample count rate	=	$R_{s+b} = 600$ counts/10 min = 60 cpm
t_{s+b}	=	10 min
Background count rate R_b	=	56 cpm (40-min count)
t_b	=	40 min
Net count rate	=	$R_s = 60\ \text{cpm} - 56\ \text{cpm} = 4\ \text{cpm}$

The standard deviation is

$$
\begin{aligned}
\sigma &= \{R_{s+b}/t_{s+b} + R_b/t_b\}^{1/2} \\
&= (60\ \text{cpm}/10\ \text{min} + 56\ \text{cpm}/40\ \text{min})^{1/2} \\
&= (7.4)^{1/2}\ \text{cpm} = 2.72\ \text{cpm}
\end{aligned}
$$

At the 95% confidence interval (one-tail), the sample count rate is

$$
\begin{aligned}
95\%\ \text{C.I.} &= R_s \pm 1.65\,\sigma \\
&= 4\ \text{cpm} \pm 1.65 \times 2.7\ \text{cpm} \\
&= 4\ \text{cpm} \pm 4.5\ \text{cpm}
\end{aligned}
$$

Because the sample count rate at a 95% confidence interval (one-tail test) includes "zero", it should be concluded that the net sample count rate is not statistically different from the background count rate. Therefore, the sample does not contain any net activity (above background).

Scenario 1.8

Question 1.28

The minimum sample and background counting time, required to ensure a lower limit of detection (LLD) at the 95% confidence level less than or equal to 0.10 DAC for I-131, may be obtained as follows:

$$\text{LLD} = 4.66\,(1/k)\,(R_b/t_b)^{1/2}$$

where

R_b = background count rate = 50 cpm
t_b = background count time (minutes)

k = correction factor (cpm-m^3/Bq)
= (detector efficiency) (sampling efficiency) (sample volume) (60 dpm/Bq)
= (0.2 cpm/dpm) (0.7) (5 liters/min × 10 min) (1 m^3/1000 liters) × (60 dpm/Bq)
= 0.42 cpm – m^3/Bq

LLD = 0.1 DAC = (0.1)(900 Bq/m^3) = 90 Bq/m^3

Solving the basic equation for t_b, we obtain:

$$t_b = \left(\frac{4.66}{k(\text{LLD})}\right)^2 R_b = \left[\frac{4.66}{(0.42 \text{ cpm-m}^3/\text{Bq})\,(90 \text{ Bq/m}^3)}\right]^2 (50 \text{ cpm}) = 0.76 \text{ min}$$

Question 1.29

Methods to reduce the counting time include:

1. Improving the collection efficiency increases the number of counts collected. This increases k and therefore reduces t_b.
2. Decreasing the background levels with shielding reduces R_b and therefore directly reduces t_b.
3. Increasing the counter efficiency by improving the geometry to increase the collected counts or by obtaining a more efficient detector.
4. Increasing the sample flow rate which increases the number of counts collected, which then increases k and therefore reduces t_b.
5. Increasing the sampling time. t_b is inversely proportional to the sample time squared.

Scenario 1.9

Question 1.30

In order to be representative of isokinetic sampling conditions, the linear flow into the sampling nozzle must be equal to the linear flow in the stack. This requires that the ratio of volumetric flow to flow area be a constant:

$$\frac{\text{Volumetric flow (stack)}}{\text{Area (stack)}} = \frac{\text{Volumetric flow (nozzle)}}{\text{Area (nozzle)}}$$

$$\frac{(20 \text{ m}^3/\text{min})}{(\pi)\,(0.25 \text{ m})^2} = \frac{200 \text{ liters/min } (10^{-3} \text{ m}^3/\text{liter})}{(\pi)\,r^2}$$

where
r = internal radius of the isokinetic probe.

$$r^2 = \frac{0.2\ \text{m}^3/\text{min}\ (0.0625\ \text{m}^2)}{20\ \text{m}^3/\text{min}}$$

$$r = 2.5\ \text{cm}$$

Therefore, the sample nozzle has an internal diameter of 5 cm.

Question 1.31

Flow patterns within a stack or duct may be distorted near blends, interferences, or transition regions, as well as at the entrance and exit to the duct. Therefore:

1. Sampling locations should be at least 5–10 stack diameters downstream of bends/duct transitions and as far upstream from the atmospheric exit as practical.
2. For large-diameter ducting, radial and axial variations in the velocity and particle composition may occur. If these differences are significant, multiple axial and radial sampling locations may be needed for a representative composite sample.

Question 1.32

Filter type	Advantages	Disadvantages
Cellulose	Low ash content	High (burial) losses for alpha counting
	Easily dissolved or decomposed	Not suitable for high-temperature applications
	Low airflow resistance	
	Good efficiency for respirable particles (0.3–10 μm)	
	Strong and not easily damaged	
Glass fibers	High collection efficiency for respirable particles	Not easily dissolved
	Low airflow resistance	Fragile/careful handling
	Can be used at higher temperatures than cellulose	
	Low burial loss for alpha counting	
Membrane	Wide range of pore sizes (0.01–10 μm)	Fragile to handle
	Easily dissolved in many solvents	High airflow resistance
	High collection efficiency for respirable particles	Not suitable for high temperatures
	Larger pore sizes (1–10 μm) are well-suited for sampling alpha emitters because lower collection efficiency is offset by minimal burial loss	
	Small-pore-size varieties have very high collection efficiencies and minimal alpha burial losses	
	Low ash content	

Scenario 1.10

Question 1.33

The larger initial count rate is due to the combination of natural products and long-lived alpha activity (Pu-239) collected on the filter. After 48 hr, it is safe to assume that all the radon decay products have decayed away because the longest half-life in the series is only 27 min.

Question 1.34

Pu-239 represents the long-lived alpha contribution. The concentration of Pu-239 can be determined from the relationship

$$C = \frac{(R_S)\,(\mathrm{FA})}{(E_d)\,(E_f)\,(\mathrm{SA}_f)\,(F)\,(T_s)\,(\mathrm{DA})}$$

where

R_S = net count rate (cpm)
FA = total filter area = 500 cm^2
E_d = detector efficiency = 0.3 cpm/dpm
E_f = filtration efficiency = 0.8
SA_f = filter paper self-absorption factor = 0.4
F = filter sampling rate = 55 ft^3/min
T_s = filter sampling time = 60 min
DA = detector area = 60 cm^2

$$C = \frac{(220 \text{ counts}/100 \text{ min} - 20 \text{ counts}/100 \text{ min})\,(500 \text{ cm}^2)}{(0.3 \text{ cpm/dpm})\,(0.8)\,(0.4)\,(55 \text{ ft}^3/\text{min})\,(60 \text{ min})} \times \frac{1}{(0.0283 \text{ m}^3/\text{ft}^3)\,(60 \text{ cm}^2)}$$

$$= 1.86 \text{ dpm/m}^3$$

The standard deviation is

$$\sigma^2 = \sigma_b^2 + \sigma_{s+b}^2$$

$$\sigma = (R_b/t_b + R_{s+b}/t_{s+b})^{1/2}$$

where

R_b = (20 counts/100 min) = 0.2 cpm
R_{s+b} = (200 counts/100 min) = 2.2 cpm

$$\sigma = (2.2 \text{ cpm}/100 \text{ min} + 0.2 \text{ cpm}/100 \text{ min})^{1/2} = 0.155 \text{ cpm}$$

σ may be related to concentration:

$$\sigma = \frac{(0.155 \text{ cpm})\,(500 \text{ cm}^2)}{(0.3 \text{ cpm/dpm})\,(0.8)\,(0.4)\,(55 \text{ ft}^3/\text{min})} \times \frac{1}{(60 \text{ min})\,(0.0283 \text{ m}^3/\text{ft}^3)\,(60 \text{ cm}^2)}$$

$$= 0.14 \text{ dpm/m}^3$$

$$C \pm \sigma = (1.86 \pm 0.14) \text{ dpm/m}^3$$

Question 1.35

$$\text{LLD} = 4.66\,(R_b/t_b)^{1/2} = 4.66\,(0.2 \text{ cpm}/100 \text{ min})^{1/2} = 0.21 \text{ cpm}$$

This LLD may be related to concentration:

$$C_{\text{LLD}} = \frac{(0.21 \text{ cpm})\,(500 \text{ cm}^2)}{(0.3 \text{ cpm/dpm})\,(0.8)\,(0.4)\,(55 \text{ ft}^3/\text{min})} \times \frac{1}{(60 \text{ min})\,(0.0283 \text{ m}^3/\text{ft}^3)\,(60 \text{ cm}^2)}$$

$$= 0.20 \text{ dpm/m}^3$$

Scenario 1.11

Question 1.36

$$\overline{T} = \text{Mean lifetime} = 1.44\,T_{1/2}$$

The total number of disintegrations (N) is given in terms of the half-life and activity (A) of the isotope

$$N = 1.44\,T_{1/2}\,A$$

The total alpha energy is the sum of the products of the total number of disintegrations and the alpha energy available for each daughter. By definition, Rn-222 is not considered in the calculation. The total alpha energy is given by

$$E_{\text{total}} = \sum_i E^i_{\text{total}}\,N_i$$

where

i = sum over the Po-218, Pb-214, Bi-214, and Po-214

An activity of 100 pCi/liter for radon and its daughters is assumed.

For Po-218,

$$\begin{aligned}
\overline{T} &= 1.44\ (3.05\ \text{min}) = 4.4\ \text{min} \\
N &= 1.44\ (3.05\ \text{min})\ (100 \times 10^{-12}\ \text{Ci})\ (3.7 \times 10^{10}\ \text{dis/s-Ci}) \times (60\ \text{s/min}) \\
&= 976\ \text{dis} \\
E_{\text{alpha}} &= (6.00 + 7.68)\ \text{MeV/dis} = 13.68\ \text{MeV/dis} \\
E_{\text{total}} &= 13.68\ \text{MeV/dis} \times 976\ \text{dis} = 1.34 \times 10^{4}\ \text{MeV}
\end{aligned}$$

For Pb-214,

$$\begin{aligned}
\overline{T} &= 1.44\ (26.8\ \text{min}) = 38.6\ \text{min} \\
N &= 1.44\ (26.8\ \text{min})\ (100 \times 10^{-12}\ \text{Ci})\ (3.7 \times 10^{10}\ \text{dis/s-Ci}) \times (60\ \text{s/min}) \\
&= 8567\ \text{dis} \\
E_{\text{alpha}} &= 7.68\ \text{MeV/dis} \\
E_{\text{total}} &= 7.68\ \text{MeV/dis} \times 8567\ \text{dis} = 6.58 \times 10^{4}\ \text{MeV}
\end{aligned}$$

For Bi-214,

$$\begin{aligned}
\overline{T} &= 1.44\ (19.7\ \text{min}) = 28.4\ \text{min} \\
N &= 1.44\ (19.7\ \text{min})\ (100 \times 10^{-12}\ \text{Ci})\ (3.7 \times 10^{10}\ \text{dis/s-Ci}) \times (60\ \text{s/min}) \\
&= 6298\ \text{dis} \\
E_{\text{alpha}} &= 7.68\ \text{MeV/dis} \\
E_{\text{total}} &= 7.68\ \text{MeV/dis} \times 6298\ \text{dis} = 4.84 \times 10^{4}\ \text{MeV}
\end{aligned}$$

For Po-214,

$$\begin{aligned}
\overline{T} &= 1.44\ (1.0 \times 10^{-6}\ \text{min}) = 1.4 \times 10^{-6}\ \text{min} \\
N &= 1.44\ (1.0 \times 10^{-6}\ \text{min})\ (100 \times 10^{-12}\ \text{Ci})\ (3.7 \times 10^{10}\ \text{dis/s-Ci}) \times (60\ \text{s/min}) \\
&= 3.2 \times 10^{-4}\ \text{dis} \\
E_{\text{alpha}} &= 7.68\ \text{MeV/dis} \\
E_{\text{total}} &= 7.68\ \text{MeV/dis} \times 3.2 \times 10^{-4}\ \text{dis} = 2.46 \times 10^{-3}\ \text{MeV}
\end{aligned}$$

In summary, we have the following data:

Nuclide	Mean Lifetime	Alpha Energy (MeV)	Disintegrations per 100 pCi	Total Energy per 100 pCi
Po-218	4.4 min	6.00 + 7.68	976	1.34×10^4 MeV
Pb-214	38.6 min	7.68	8567	6.58×10^4 MeV
Bi-214	28.4 min	7.68	6298	4.84×10^4 MeV
Po-214	1.4×10^{-6} min	7.68	3×10^{-4}	0 MeV
Total				1.3×10^5 MeV

The value 1.3×10^5 MeV of alpha energy completes the WL description.

Question 1.37

$$C = (k_2\ N)/(k_1\ e\ f\ t_s) = \text{radon-222 concentration}$$

where
k_1 = 150 dpm alpha/liter-WL
k_2 = (100 pCi/liter)/0.5 = 200 pCi/liter @ 50% equilibrium
N = total alpha counts per minute = 230 counts/minute = 230 cpm
e = counting efficiency = 0.3 cpm/dpm
f = sample pump flow rate = 10 liters/min
t_s = sample collection time = 5 min

$$C = \frac{(230\ \text{cpm})\ (200\ \text{pCi/liter-WL})}{(0.3\ \text{cpm/dpm})\ (10\ \text{liters/min})\ (5\ \text{min})\ (150\ \text{dpm/WL-liter})}$$
$$= 20.4\ \text{pCi/liter}$$

Question 1.38

Measurement for radon and its daughters is performed by using the following methods:

1. Track etch detectors
2. Charcoal absorption
3. Filter paper collection and analysis
4. Continuous working level monitors
5. Passive environmental radon monitors (PERMs)
6. Electrostatic or ELECTRET dosimeters

Scenario 1.12

Question 1.39

The "memory effect" is a potential source of error associated with measuring beta and gamma dose rates while moving in and out of a noble gas environment. While in the cloud, noble gas seeps inside the ion chamber. After leaving the noble gas environment, the instrument continues to respond to the gas within its chamber, which causes the meter to respond and yield a false-positive reading.

Question 1.40

Environmental conditions having an adverse effect on the accuracy of the instrument response include temperature, pressure, radiofrequency signals, and humidity. The temperature and pressure affect the density of the air within the chamber and therefore directly affect the number of ion pairs produced. Radiofrequency signals can interfere with the instrument's signal processing capability. Humidity can disrupt the flow of electricity, causing shorting of the anode/cathode which will lead to erratic readings.

Question 1.41

The geometry factor or the geometric relationship between the source and receptor is the most significant source of error associated with measuring true beta and gamma surface dose rates from contact measurements of small sources. The instrument response is related to the number of ions collected at the center of the detector, and not the dose rate at the surface of the source. This is also affected by the volume of the chamber exposed to the particle's radiation. The smaller the source, the smaller the volume irradiated and the larger the geometry effect.

Question 1.42

A source of error associated with measuring beta dose rates from large-area sources, with each source being a different beta emitter, is the energy of the emitted beta particle. Because beta particles are attenuated across the chamber depending upon their energy, higher-energy beta particles respond closer to the true beta dose rate. Lower-energy beta particles underrespond.

Question 1.43

For high-energy beta sources, beta penetration through the closed window is a source of error implicit in the application of the open and closed window readings. Normally, the open window (OW) measures both beta and gamma radiation, and the closed window (CW) measurement yields only gamma information:

$$OW = \text{beta} + \text{gamma}$$
$$CW = \text{gamma}$$

Therefore

$$OW - CW = \text{beta}$$

With a high-energy beta source, some of the beta particles penetrate the closed window. This results in an overestimate of the gamma contribution which causes the beta dose rate to be underestimated.

Scenario 1.13

Question 1.44: a

ANSI N13.11 does not apply to pocket dosimeters and extremity dosimeters.

Question 1.45

Lens of the eye.

Question 1.46: a

The standard least adequately tests for low-energy beta particles and the spectrum of neutrons escaping from the reactor vessel in a power reactor.

Question 1.47: b

Passing the beta radiation performance standard does not guarantee accurate results for all beta sources.

Question 1.48: a

ANSI N13.11 forms the basis for the National Voluntary Laboratory Accreditation Program for dosimetry processors.

Scenario 1.14

Question 1.49

5 – Proton recoil badge.

High-energy neutron detection is best accomplished with the proton recoil film badge.

Li-6-based dosimeters are preferred for lower-energy neutrons. Other dosimeters do not have good neutron detection capability.

Question 1.50

2 – A TLD albedo containing both Li-6 and Li-7 elements.

The TLD albedo system has the best capacity for measuring both neutrons and gammas, but without the fogging which is often encountered with film systems.

A proton recoil film badge will also work, but not without interference.

Question 1.51

1 – A common film badge.

The film badge has energy-compensating filters which provide satisfactory energy discrimination at the lower X-ray range.

Question 1.52

9 – A calcium sulfate, dysprosium-activated TLD element in a tissue equivalent holder.

This system has a good response to high-energy gammas, low fading, and the tissue equivalent holder approximates conditions of electronic equilibrium.

Item 4, the manganese-activated TLD element, would also work, but its high-energy response is not as good.

Question 1.53

7 – The four-element TLD with lithium borate phosphors, a thin mylar filter over one element, plastic filters over two elements, and an aluminum filter over the fourth element.

Lithium borate is a good photon and electron detector, and the multiple filters allow good energy determination.

Scenario 1.15

Question 1.54

Gamma calibration factor = 6000 TL units/500 mrem
GCF = 12 TL units/mrem

Beta calibration factor = 750 TL units/1000 mrem
BCF = 0.75 TL units/mrem

Gamma dose = (Chip 2 – Control Chip 2)/GCF
= (11 520 – 120) TL units/12 TL units/mrem
= 950 mrem

Beta dose = (Chip 1 – Chip 2)/BCF
= (12 270 – 11 520) TL units/0.75 TL units/mrem
= 1000 mrem

The beta dose varies as a function of depth into tissue. This variation was provided in the problem statement as the fraction (f) of the beta dose equivalent as a function of depth:

Whole-body dose = beta dose (1000 mg/cm^2) + gamma dose
= f(1000 mg/cm^2) × beta dose + gamma dose
= 0.01 × 1000 mrem + 950 mrem = 960 mrem

Skin dose = beta dose (7 mg/cm^2) + gamma dose
= f(7 mg/cm^2) × beta dose + gamma dose
= 1.0 × 1000 mrem + 950 mrem = 1950 mrem

Question 1.55

The correct depth to use for the lens of the eye is 300 mg/cm^2. The beta dose equivalent at 300 mg/cm^2 is 25% of that at 7 mg/cm^2.

Eye dose = beta dose (300 mg/cm^2) + gamma dose
= f(300 mg/cm^2) × beta dose + gamma dose
= 0.25 × 1000 mrem + 950 mrem
= 1200 mrem

Question 1.56

No regulatory limits (e.g., 10 CFR 20) were exceeded. The 10 CFR 20 regulation establishes a 15-rem/year limit for the eye dose equivalent.

Scenario 1.16

Question 1.57: b

The neutron quality factor is not constant between 0 and 20 MeV.

Question 1.58: a

The gamma dose measured on a phantom is greater than the gamma dose measured in air due to the H(n, γ) D reaction in the phantom.

Question 1.59: e

TLD albedo dosimeters calibrated with a bare Cf-252 source will overrespond to soft neutron spectra. Corrections are needed for accurate results.

Question 1.60: c

Neutron bubble detectors are affected by temperature.

Question 1.61: a

For power reactor containment entries, a TLD albedo dosimetry system calibrated to D_2O-moderated Cf-252 will accurately measure the neutron dose equivalent.

Scenario 1.17

Question 1.62: Departments A, B, and C

The nuclear medicine, X-ray and radiation therapy departments will require personnel monitoring for photons because they utilize photon or X-ray sources. The research department will not require photon monitoring because it only uses H-3 and C-14, which are low-energy beta emitters.

Question 1.63: None

Because no department utilizes neutron sources, neutron monitoring is not required.

Question 1.64: Department B

The X-ray department will benefit from the dual monitors because the apron will provide attenuation of the X-ray source.

Question 1.65: Department A

The nuclear medicine department should utilize ring badges because of the handling and preparation of Tc-99m.

Question 1.66: None

However, a misapplication could lead to significant skin doses that would require assessment. The dose factor (rem/hr per μCi/cm^2) for a 7-mg/cm^2 skin depth should be considered. Factors for applicable nuclei are:

Nuclide	Dose factor
H-3	0.0
C-14	1.09
Co-60	4.13
Tc-99m	3.49
Cs-137	6.46

Question 1.67: Department D

The use of H-3 and C-14 suggests that the research department should consider a routine bioassay program.

Question 1.68

Positive characteristics of film dosimeters:

1. Provides a permanent record.
2. Rugged and durable.
3. Responds to a wide exposure range.
4. Provides a stable latent record.
5. Reasonable energy discrimination.
6. Relatively inexpensive.

Question 1.69

Negative characteristics of film dosimeters:

1. No beta sensitivity for energies below about 200 keV.
2. Sensitive to heat.
3. Cannot be reused.
4. Processing time causes delays in information retrieval.

5. Not tissue-equivalent.
6. Fades with time.
7. Energy-dependent.

Question 1.70

Positive characteristics of TLDs:

1. Rugged in construction.
2. Tissue-equivalent.
3. Energy-independent.
4. Applicable over a wide exposure range.
5. Sensitive to low doses.
6. Can be reused.
7. Can be immediately processed on site.
8. Reasonable beta sensitivity.

Question 1.71

Negative characteristics of TLDs:

1. No permanent record.
2. Fades with heat.
3. Sensitive to oil and water.
4. Angular dependence.
5. Supralinearity.
6. Expensive.
7. Maintenance costs are higher than those for film.

Scenario 1.18

Question 1.72

For cylindrical gas ionization chambers, the output current (I) is directly proportional to the volume of the detector:

$$I = k\,V$$

where

k = proportionality constant (A/cm^3)

For a cylinder detector, the volume (V) is given by

$$V = \pi\, r^2\, h$$

where
r = radius of the detector
h = detector height

For the two detectors, we have

$$r_A = 0.25\ \text{cm}, \quad r_B = 0.50\ \text{cm}$$
$$h_A = 5.0\ \text{cm}, \quad h_B = 5.0\ \text{cm}$$

The current outputs of detectors A and B are related by

$$\frac{I_B}{I_A} = \frac{\pi\, r_B^2\, h_B\, k}{\pi\, r_A^2\, h_A\, k}$$

Because $h_A = h_B$ and $r_B = 2\, r_A$, we obtain

$$I_B = I_A\,(2)^2 = 4\, I_A = 4\,(1.0 \times 10^{-10}\ \text{A}) = 4.0 \times 10^{-10}\ \text{A}$$

The corrects answer is b.

Question 1.73

The detector sensitivity (S) is directly proportional to the gas pressure (P) within the detector

$$S = k\, p$$

where
k = constant

The sensitivity obtained by changing the gas pressure may be obtained from the relationship

$$S_1 / S_2 = P_1 / P_2$$

or

$$S_2 = S_1\,(P_2 / P_1) = (1.2 \times 10^{-10}\ \text{A-hr/R}) \cdot \frac{11\,400\ \text{torr}}{7600\ \text{torr}} = 1.8 \times 10^{-10}\ \text{A-hr/R}$$

Question 1.74

The detector current (I) may be written in terms of the conditions at STP, the detector volume (V), the detector gas density (ρ), ambient temperature (T), pressure (P), and the dose equivalent rate ($\dot{H}$):

$$I = k\ p\ V \frac{T_{STP}}{T} \frac{P}{P_{STP}} \dot{H}$$

where

T_{STP} = 273 K
T = 20 °C
P_{STP} = 760 torr
P = 7600 torr
p = 1.29 kg/m^3
V = 100 cm^3
I = 9.0×10^{-14} A
k = (2.58×10^{-4} coulomb/kg-R) (1 hr/3600 s) (1 A s/coulomb)
= 7.17×10^{-8} A-hr/R-kg

$\dot{H}$ may be obtained from the current equation:

$$\dot{H} = I \Big/ \left(k\ p\ V \frac{T_{STP}}{T} \frac{P}{P_{STP}} \right)$$

$$= \frac{9.0 \times 10^{-14}\ \text{A}}{(7.17 \cdot 10^{-8}\ \text{A-hr/R-kg})\,(1.29\ \text{kg/m}^3)\,(100\ \text{cm}^3)\,(1\ \text{m}/100\ \text{cm})^3} \times \frac{1}{[273\ \text{K}/(273\ \text{K} + 20\ \text{K})]\,[7600\ \text{torr}/760\ \text{torr}]}$$

$$= \frac{9.0 \times 10^{-14}\ \text{A}}{8.62 \times 10^{-11}\ \text{A-hr/R}}$$

$$= (1.04 \times 10^{-3}\ \text{R/hr})\,(1000\ \text{mR/R})\,(1\ \text{mrem}/1\ \text{mR})\,(1\ \text{mSv}/100\ \text{mrem})$$

$$= 0.0104\ \text{mSv/hr}$$

Question 1.75

The detector was calibrated under the following conditions:

$$T_C = (273 + 20)\ \text{K} = 293\ \text{K}$$
$$P_C = 591.6\ \text{torr}$$

Measurements were made at sea level:

$$T = (273 + 36)\ \text{K} = 309\ \text{K}$$
$$P = 760\ \text{torr}$$
$$\dot{H} = 1\ \text{mSv/hr}$$

From the first equation of this scenario, we have

$$I = K\ \dot{H} \frac{P}{T}$$

where
K = constant for a given detector.

Because $I = I_C$, the dose equivalent rate $(\dot{H}_C)$ that will be read in a 1 mSv/hr field is

$$\dot{H}_C = \dot{H}\,(P/P_C)\,(T_C/T)$$
$$= (1\text{ mSv/hr})\,(760\text{ torr}/591.6\text{ torr})\,(293\text{ K}/309\text{ K}) = 1.22\text{ mSv/hr}$$

The correct answer is d.

Scenario 1.19

Question 1.76

The total gamma dose rate at 1 ft is

$$\dot{H} = (S/4\,\pi\,r^2)\,(u_{en}/p)\,E_{gamma}$$

where
S = gamma source strength = 2.2×10^6 gammas/s
r = distance from the source = 1 ft = 30.48 cm
E_{gamma} = gamma-ray energy of source = 6.1 MeV
u_{en}/p = mass-energy absorption coefficient
= 0.0178 cm^2/g for muscle

$$\dot{H} = [(2.2 \times 10^6\text{ gammas/s})/(4\,\pi)\,(30.48\text{ cm})^2] \times (0.0178\text{ cm}^2/\text{g})$$
$$\times (6.1\text{ MeV/gamma})\,(1.6 \times 10^{-6}\text{ erg/MeV})\,(1\text{ rad}/100\text{ erg/g})$$
$$\times (3600\text{ s/hr})\,(1\text{ rem/rad})\,(1000\text{ mrem/rem})$$
$$= 1.18\text{ mrem/hr}$$

Question 1.77

The total neutron dose rate at 1 ft is given by

$$\dot{H}_n = (S_n/4\,\pi\,r^2)\,k$$

where
S_n = neutron source strength = 2.0×10^5 n/s
k = flux-to-dose conversion factor
= (2.5 mrem/hr)/(20 n s^{-1} cm^{-2})

$$\dot{H}_n = \frac{(2.5\text{ mrem/hr})\,(2.0 \times 10^5\text{ n s}^{-1})}{(20\text{ n/s-cm}^2)\,(4\,\pi)\,(30.48\text{ cm})^2} = 2.14\text{ mrem/hr}$$

Question 1.78: a

The best shielding arrangement is polyethylene followed by lead. The polyethylene thermalizes the neutrons, and lead attenuates both the capture gamma rays and the source's 6.1-MeV gammas.

Question 1.79

The required gamma shielding thickness is obtained as follows:

$$\dot{H}(x) = \dot{H}(0)\, B \exp(-u\, x)$$

where

$\dot{H}(0)$ = unshielded exposure rate

T = transmission factor

= $\dot{H}(x)/\dot{H}(0) = 1/5 = 0.2 = B \exp(-u\, x)$

From the table of buildup factors, a table of transmission factors can be assembled:

i	$u\, x_i$	B_i	T_i
1	1.0	1.18	0.434
2	2.0	1.40	0.189

When it is assumed that $\ln(T)$ versus $u\, x$ is linear in this region of shield thickness, interpolation leads to the required $u\, x$ value that corresponds to $T = 0.2$:

$$u\, x = u\, x_1 + (u\, x_2 - u\, x_1)\frac{\ln(T/T_1)}{\ln(T_2/T_1)}$$

$$= 1.0 + (2.0 - 1.0)\frac{\ln(0.2/0.434)}{\ln(0.189/0.434)} = 1.0 + \frac{-0.7747}{-0.8313} = 1.93$$

$$x = 1.93/u$$

u_a/p = mass attenuation coefficient for lead = 0.0435 cm^2/g

$p = 11.35\ \text{g/cm}^3$

$u = (u_a/p)\, p$

$$x = \frac{1.93}{(0.0435\ \text{cm}^2\ \text{g}^{-1})\,(11.35\ \text{g cm}^{-3})} = 3.91\ \text{cm}$$

Although logarithmic interpolation is justified by the problem data, a linear interpolation solution is provided for illustrative purposes. If linear interpolation is assumed for the T versus $u\, x$ relationship, the following result is obtained:

$$u\,x = u\,x_1 + (u\,x_2 - u\,x_1)\left(\frac{T - T_1}{T_2 - T_1}\right)$$

$$= 1.0 + (2.0 - 1.0)\left(\frac{0.2 - 0.434}{0.189 - 0.434}\right) = 1.0 + \frac{0.234}{0.245} = 1.96$$

$$x = \frac{1.96}{(0.0435\ \text{cm}^2/\text{g})\,(11.35\ \text{g/cm}^3)} = 3.97\ \text{cm}$$

The best approach depends on the shape of the data set being interpolated. This shape must be ascertained before interpolation.

Scenario 1.20

Question 1.80

Method	Advantages	Disadvantages
Time for decay	By waiting, there will be reduced exposures during decontamination and decommissioning (D&D).	Longer time will require controls throughout the storage period.
	Costs for D&D can be deferred to a future date.	There is a potential for continuing releases to the environment.
	Waiting will permit new technology to emerge to limit dose and cost.	Costs for future waste disposal are uncertain and will likely increase.
	More time is allowed for a D&D fund to accumulate.	Waiting will prohibit use of land, buildings, and facility resources.
Immediate removal	Costs are known.	During removal, the potential for releases to the environment are increased.
	Problem is solved, and resources may be used for other purposes.	Transportation risks will be increased during removal operations.
	Waste will be removed to a storage facility designed for this purpose.	Higher occupational exposures will be received compared to a delayed disposal.
Shielding	May be the lowest cost option.	A possession licence will be required.
	Some occupational exposure, but not as high as removal.	Future liability may include eventual removal.

Question 1.81

Based upon the data provided, the depth of excavation required to allow free release can be obtained from the relationship

$$I = I_0 \exp(-u\,x) \text{ (no buildup)}$$

Based upon the available information, excluding buildup is appropriate. The required excavation depth can be obtained by rearranging this equation and by inserting the given data into this equation:

$$x = -(1/u) \ln(I/I_0)$$

where

Relaxation length = $1/u = 15$ cm
I_0 = 20 μR/hr
I = 5 μR/hr

$$x = -(15 \text{ cm}) \ln \frac{5\mu\text{R/hr}}{20\ \mu\text{R/hr}} = 20.8 \text{ cm}$$

Question 1.82

$$A = A_0 \exp(-\lambda\, t)$$
$$t = (-1/\lambda) \ln(A/A_0)$$

For Co-60, we have

$$\lambda = 0.693/T_{1/2} = 0.693/5.27 \text{ year} = 0.131 \text{ year}^{-1}$$
$$t = (-1/0.131 \text{ year}^{-1}) \ln(5/20) = 10.54 \text{ year}$$

Question 1.83

The mass attenuation coefficient is 0.06 cm^2/g and the density of concrete is 2.5 g/cm^3. Buildup is assumed to be a constant factor of 2.

To obtain the shield thickness, the following relationship is utilized:

$$I = I_0\, B \exp(-u\, X) \text{ (buildup included)}$$

where

B = buildup factor
u = (u/p) (p)
u/p = 0.06 cm^2/g
p = 2.5 g/cm^3

Solving for X, we obtain

$$X = (-1/u)\ln(I/I_0\ B)$$
$$= (-1/0.06\ \text{cm}^2/\text{g} \times 2.5\ \text{g/cm}^3) \times \ln(5\ \mu\text{R/hr}/2 \times 20\ \mu\text{R/hr})$$
$$= 13.9\ \text{cm}$$

Scenario 1.21

Question 1.84

The reaction described in this problem is

$$\text{Incident particle} + \text{Al-27} \rightarrow \text{Na-24} + \text{ejectile}$$

with a production cross-section of 20 mb.

The general equation for the buildup of Na-24 activity is

$$A = N\ \sigma\ \phi\ [1 - \exp(-\lambda\ t_{\text{irrad}})] \times \exp(-\lambda\ t_{\text{decay}})$$

For this problem, the saturation activity is assumed and the activating flux is to be determined:

$$A_{\text{sat}} = N\ \sigma\ \phi = \text{saturation activity of Na-24}$$

where

N = number of atoms of Al-27 in the target
σ = cross-section for producing Na-24 from Al-27 = 20 mb
ϕ = activating flux

Solving for the activating flux, we obtain

$$\phi = A_{\text{sat}}/N\ \sigma$$

where

A_{sat} = 4.0×10^7 dis/s
N = $(6.02 \times 10^{23}$ atoms/mole) (1 g)/(27 g/mole) = 2.23×10^{22} atoms of Al-27
σ = (20 mb) (1 b/1000 mb) (1×10^{-24} cm^2/barn) = 2×10^{-26} cm^2

$$\phi = \frac{(4.0 \cdot 10^7\ \text{dis/s})\ (1\ \text{particle/dis})}{(2.23 \times 10^{22}\ \text{atoms})\ (2.0 \times 10^{-26}\ \text{cm}^2/\text{atom})}$$
$$= 8.97 \times 10^{10}\ \text{particles/cm}^2\text{-s}$$

Question 1.85

The general activation relationship can be written as

$$A = A_{sat}\,[1 - \exp(-\lambda\, t_{irrad})]\exp(-\lambda\, t_{decay})$$

Because we are interested in the time immediately after shutdown, $t_{decay} = 0$ and the activity relationship becomes

$$A = A_{sat}\,[1 - \exp(-\lambda\, t_{irrad})]$$

The activity after the 30-hr irradiation is

$$A = (4.0 \times 10^7\ \text{Bq})\,\{1 - \exp[-(0.693/15\ \text{hr})\,(30\ \text{hr})]\} = 3.0 \times 10^7\ \text{Bq}$$

Question 1.86

The dose-equivalent rate to a person standing a distance r from an unshielded point activation source is given by

$$\dot{D} = A\,E\,u/(4\,\pi\,r^2)$$

where

u = energy absorption coefficient (cm^2/g) = u_{en}/p
= $(2.3 \times 10^{-5}\ \text{cm}^{-1})/(0.00129\ \text{g/cm}^3) = 0.018\ \text{cm}^2/\text{g}$
$E = \sum_i E_i\,Y_i = [1.4\,(1.0) + 2.8\,(1.0)]\ \text{MeV} = 4.2\ \text{MeV}$

A point source approximation is reasonable because the size of the 1-gram mass is small with respect to the 1-meter distance. With this approximation the dose rate is

$$\begin{aligned}\dot{D} &= (4.0 \times 10^7\ \text{dis/s})\,(4.2\ \text{MeV})\,(1.6 \times 10^{-6}\ \text{erg/MeV}) \\ &\quad \times (1\ \text{rad}/100\ \text{erg/g})\,(0.018\ \text{cm}^2/\text{g})/(4)\,(3.14)\,(100\ \text{cm})^2 \\ &= 3.85 \times 10^{-7}\ \text{rad/s} \times 1000\ \text{mrem/rem} \times 3600\ \text{s/hr} \times 1\ \text{rem/rad} \\ &= 1.39\ \text{mrem/hr}\end{aligned}$$

Scenario 1.22

Question 1.87

$$\dot{X} = A\,G/r^2$$

where

A = source activity = 1.0×10^7 Ci

G = gamma constant for Co-60
r = distance from the point source

$$\dot{X} = (1.0 \times 10^7 \text{ Ci})\,(1.32 \text{ R-m}^2/\text{hr-Ci})/(3.0 \text{ m})^2 = 1.47 \times 10^6 \text{ R/hr}$$

Question 1.88

The maximum photon fluence $I(r)$ at the exterior wall surface is given by

$$I(r) = I_0(r) \sum_i B_i \; f_i \; \exp(-u_i \; x)$$

where
B_i = buildup factor for energy E_i
u_i = attenuation factor for 1-m-thick concrete wall for energy E_i
u_1 = 0.140/cm for 1.17 MeV
u_2 = 0.130/cm for 1.33 MeV
x = thickness of the concrete wall = 1 m
E_1 = energy of Co-60 gamma ray #1 = 1.17 MeV
E_2 = energy of Co-60 gamma ray #2 = 1.33 MeV
f_i = fraction of I_0 contributed by E_i or the partial yield of the i th gamma ray
$f_1 = f_2 = 0.5$

Each term contributing to $I(r)$ will be calculated separately. The unattenuated flux is given by

$$I_0 = A \; k \; Y/(4 \; \pi \; r^2) = \text{unattenuated flux @ 3.5 m (gammas cm}^{-2} \text{ s}^{-1})$$

where
S = source strength (gammas/s) = $A \; k \; Y$
A = source activity (Ci) = 1×10^7 Ci
k = conversion factor (3.7×10^{10} dis/s-Ci)
Y = yield (gammas/dis) = 2 gammas/dis for Co-60
r = distance from the source (cm) = 350 cm

$$I_0 = (1 \times 10^7 \text{ Ci})\,(3.7 \times 10^{10} \text{ dis/s-Ci})\,(2 \text{ gammas/dis})/(4\,\pi)\,(350 \text{ cm})^2$$
$$= 4.81 \times 10^{11} \text{ gammas cm}^{-2} \text{ s}^{-1}$$

Determine the attenuation provided by the concrete wall for each gamma ray:

1.17 MeV: $i = 1$
1.33 MeV: $i = 2$

$$\exp[-(u_1 \; x)] = \exp(-0.140/\text{cm} \times 100 \text{ cm}) = 8.3 \times 10^{-7}$$
$$\exp[-(u_2 \; x)] = \exp(-0.130/\text{cm} \times 100 \text{ cm}) = 2.3 \times 10^{-6}$$

Determine the buildup factors for each energy by interpolation:

$$u_1\ x = (0.140/\text{cm})\,(100\ \text{cm}) = 14.0$$
$$u_2\ x = (0.130/\text{cm})\,(100\ \text{cm}) = 13.0$$

i	B	
	$u\,x = 10$	$u\,x = 15$
1	17.5	30.6
2	16.3	28.2

$$B_1(14) = 17.5 + \frac{14 - 10}{15 - 10} \times (30.6 - 17.5) = 28.0$$
$$B_2(13) = 16.3 + \frac{13 - 10}{15 - 10} \times (28.2 - 16.3) = 23.4$$

Finally, apply these buildup factors and sum to obtain the total fluence rate:

$$\begin{aligned} I(r) &= I_0(r) \sum_i B_i\ f_i\ \exp(-u_i\ x) \\ &= (4.8 \times 10^{11}\ \text{gammas cm}^{-2}\ \text{s}^{-1}) \\ &\quad \times [(0.5)\,(8.3 \times 10^{-7})\,(28) + (0.5)\,(2.3 \times 10^{-6})\,(23.4)] \\ &= 1.9 \times 10^7\ \text{gammas/cm}^2\text{-s} \end{aligned}$$

Question 1.89

Buildup factors apply to broad-beam or poor geometry conditions. For these cases, scattering becomes important and the buildup factor is needed to correct for this scattering. The correct answer is d.

Scenario 1.23

Question 1.90

The linear attenuation coefficient (u) is defined by

$$I = I_0\ \exp(-u\ x)$$

where
I = shielded radiation exposure rate (R/hr)
I_0 = unshielded radiation exposure rate (R/hr)
x = shield thickness (cm)

u may be obtained from its defining equation:

$$u = -\frac{1}{x}\ln(I/I_0)$$

Using the values in the data tables leads to the following values of $u(x)$ for the narrow beam:

$$u(1.0\text{ cm}) = -\frac{1}{1.0\text{ cm}}[\ln(29.5\text{ mR/hr}/127.0\text{ mR/hr})] = 1.46\text{ cm}^{-1}$$
$$u(2.0\text{ cm}) = -\frac{1}{2.0\text{ cm}}[\ln(7.7\text{ mR/hr}/127.0\text{ mR/hr})] = 1.40\text{ cm}^{-1}$$
$$u(3.0\text{ cm}) = -\frac{1}{3.0\text{ cm}}[\ln(1.9\text{ mR/hr}/127.0\text{ mR/hr})] = 1.40\text{ cm}^{-1}$$

The linear attenuation coefficient for the given data is in the range of 1.40–1.46/cm. An average of these values yields 1.42/cm.

Question 1.91

The buildup factor (B) is defined for broad-beam conditions by the equation

$$I = I_0\ B(x)\exp(-u\ x)$$

Solving for $B(x)$ leads to the result

$$B(x) = \frac{I}{I_0\ \exp(-u\ x)}$$

B (2.5 cm) is requested. It can be obtained from the B (2.0 cm) and B (3.0 cm) values:

$$B(2.0\text{ cm}) = \frac{I}{I_0\ \exp(-u\ x)} = \frac{13.0\text{ mR/hr}}{(127\text{ mR/hr})\exp(-2.0/\text{cm}\times 2.0\text{ cm})} = 5.59$$
$$B(3.0\text{ cm}) = \frac{I}{I_0\ \exp(-u\ x)} = \frac{4.0\text{ mR/hr}}{(127\text{ mR/hr})\exp(-2.0/\text{cm}\times 3.0\text{ cm})} = 12.71$$

The value at 2.5-cm thickness is obtained by linear interpolation. Simple linear interpolation leads to the following value:

$$B(2.5\text{ cm}) = (5.59 + 12.71)/2 = 9.15$$

The use of linear or logarithmic interpolation will be governed by the available data and the manner by which $B(x)$ scales with x.

Question 1.92

The mass-attenuation coefficient (u/p) is

$$u/p = \frac{2.0/\text{cm}}{18.9\ \text{g/cm}^3} = 0.11\ \text{cm}^2/\text{g}$$

Scenario 1.24

Question 1.93

The dose from the K-40 distributed throughout the body can be estimated by determining its activity in the body:

$$A = \lambda N = (0.693/T_{1/2})\, N$$

where

$T_{1/2}$ = half-life (years) = 1.2×10^9 years
N = number of atoms of K-40 in reference man
= $m A_0/M$
m = mass of K-40 in the body = 140 g × 0.00012
M = gram atomic weight of K-40 = 40 g/GAW
A_0 = Avogadro's number = 6.02×10^{23} atoms/GAW
N = (140 g × 0.00012) (6.02×10^{23} atoms/GAW)/(40 g/GAW)
= 2.53×10^{20} atoms

$$A = (0.693/1.2 \times 10^9\ \text{years})\,(2.53 \times 10^{20}\ \text{atoms})\,(1\ \text{dis/atom})$$
$$= 1.46 \times 10^{11}\ \text{dis/year}$$

$\overline{e}$ = average energy deposited into the tissue by the K-40 decay
= $E_{\max} f/3$
$E_{\max}$ = maximum beta energy of the K-40 decay = 1.3 MeV
f = probability that a beta particle is emitted when K-40 decays
= 0.9
$\overline{e}$ = (1.3 MeV) (0.9)/3 = 0.39 MeV

The dose rate delivered to the body from the K-40 $(\dot{D})$ is given by

$$\dot{D} = \frac{A\,\overline{e}}{m}\,k$$

where

m = mass of the whole body = 70 000 g
k = conversion factor = (1.6×10^{-6} erg/MeV)/(100 erg/g-rad)

$$\dot{D} = \frac{(1.46 \times 10^{11} \text{ dis/year})\,(0.39 \text{ MeV/dis})\,(1.6 \times 10^{-6} \text{ erg/MeV})}{(70\,000 \text{ g})\,(100 \text{ erg/g-rad})}$$
$$= 0.013 \text{ rad/year} \times 1 \text{ year}/52 \text{ weeks}$$
$$= 2.5 \times 10^{-4} \text{ rad/week}$$

Question 1.94: c

K-40 has no regulatory significance in whole-body counting, but it provides an important qualitative system check.

Question 1.95: e

ICRP-26 focuses upon minimizing the total effective dose equivalent, which is the sum of the external (deep dose equivalent) and internal (committed effective dose equivalent) doses. As such, the use of air samples and stay-time calculations instead of respirator usage is acceptable if the total dose is minimized.

Question 1.96: a

Electronic equilibrium is least likely at the surface of the skin.

Scenario 1.25

The reader should note that medical exposure is not included as part of the occupational exposure. This scenario reflects a radiation protection program that takes the diagnostic exposure into account in order to determine a worker's readiness for duty.

Question 1.97

Initial activity in the thyroid:

q = administered activity (μCi) = 1.0 μCi
$A(0)$ = initial activity in the thyroid = $q f_2$ = 0.3 × 1.0 μCi = 0.3 μCi

Effective half-life:

T_{phy} = 8.08 days
T_{bio} = 74 days
T_{eff} = (8.08 days) (74 days)/(8.08 days + 74 days) = (7.28 days) (24 hr/day)
= 174.8 hr

Cumulated activity:

$\tilde{A}$ = $A(0)$ (1.44) (T_{eff}) = (0.3 μCi) (1.44) (174.8 hr) = 75.51 μCi-hr

Thyroid dose:

$$D = \tilde{A}\, S(T \leftarrow S) = 1.44\, A(0)\, T_{\text{eff}}\, S(T \leftarrow S)$$
$$= (0.3\ \mu\text{Ci})\,(1.44)\,(174.8\ \text{hr})\,(2.2 \times 10^{-2}\ \text{rad}/\mu\text{Ci-hr}) = 1.66\ \text{rad}$$

Question 1.98

Because the ICRP-10 investigation level is 300 nCi, the HP program places a restriction on the worker at 10% of this value, or 30 nCi.

The worker reaches unrestricted status at 30 nCi. Therefore, you must find the time for the 300-nCi intake to decay to 30 nCi. This is given by a simple exponential decay using the effective half-life. Because the diagnostic procedure investigates a potential thyroid abnormality, this technique is only an estimate of the dose. The intake retention function would be required to provide a retrospective assessment of the actual dose. This can be accomplished by either *in vitro* or *in vivo* techniques.

$$A(t) = A(0)\exp[(-0.693/T_{\text{eff}})\,t]$$
$$A(0) = 300\ \text{nCi}$$
$$A(t) = 30\ \text{nCi}$$

Solving for t, we obtain

$$t = -(T_{\text{eff}}/0.693)\ln[A(t)/A(0)]$$
$$= -(1.44)\,(7.28\ \text{days})\ln(30/300) = 24.2\ \text{days}$$

Question 1.99: e

Only statement e is incorrect per the ICRP-10 investigation level specification.

Scenario 1.26

Question 1.100

Assuming an instantaneous distribution of the 10 Ci and no dilution by the ventilation system, the dose to the individual in the room is given by

$$D = C(\text{BR})\,t(\text{DCF})$$

where

C = tritium air concentration
= $(10\ \text{Ci})/(10\ \text{ft} \times 20\ \text{ft} \times 30\ \text{ft} \times 0.02832\ \text{m}^3/\text{ft}^3) = 5.89 \times 10^{-2}\ \text{Ci/m}^3$
BR = breathing rate = $3.5 \times 10^{-4}\ \text{m}^3/\text{s}$
t = exposure time = 30 min
DCF = dose conversion factor = 158 rem/Ci

$$D = C(\text{BR})\, t(\text{DCF})$$
$$= (5.89 \times 10^{-2}\ \text{Ci/m}^3)\,(3.5 \times 10^{-4}\ \text{m}^3/\text{s})\,(30\ \text{min} \times 60\ \text{s/min}) \times (158\ \text{rem/Ci})$$
$$= 5.86\ \text{rem}$$

Question 1.101

To calculate the dose to the individual at the site boundary, the following assumptions are made:

1. All H-3 is released to the environment in 30 min.
2. The person is stationed at the plume centerline during the entire accident.
3. The entire plume passes the individual in 30 min.

$$D = A\,(X/Q)\,(\text{BR})\,(\text{DCF})$$

where
A = activity released = 10 Ci
X/Q = atmospheric diffusion factor = 1.0×10^{-4} s/m^3

$$D = (10\ \text{Ci})\,(1.0 \times 10^{-4}\ \text{s/m}^3)\,(3.5 \times 10^{-4}\ \text{m}^3/\text{s})\,(158\ \text{rem/Ci})$$
$$= 5.53 \times 10^{-5}\ \text{rem}$$

Question 1.102

Assuming uniform distribution of the tritium and exponential removal by the ventilation system, the concentration of tritium activity $C(t)$ is given by

$$C(t) = C_0\ \exp(-r\ t)$$

where
r = ventilation removal rate = 3/hr × 1 hr/60 min = 0.05/min

The average concentration $(\overline{C})$ in the room during the time of the exposure (t = 30 min) is obtained by integrating $C(t)$ from $t = 0$ to t:

$$\overline{C} = \int_0^t C(t)\,\mathrm{d}t \Big/ \int_0^t \mathrm{d}t = C_0\,[1 - \exp(-r\ t)]/(r\ t)$$

For t = 30 min,

$$\overline{C} = \frac{(5.89 \times 10^{-2}\ \text{Ci/m}^3)\,[1.0 - \exp(-0.05/\text{min} \times 30\ \text{min})]}{(0.05/\text{min})\,(30\ \text{min})}$$
$$= 3.05 \times 10^{-2}\ \text{Ci/m}^3$$

The dose is given by

$$\begin{aligned} D &= \overline{C}(\mathrm{BR})\, t(\mathrm{DCF}) \\ &= (3.05 \times 10^{-2}\ \mathrm{Ci/m^3})\,(3.5 \times 10^{-4}\ \mathrm{m^3/s})\,(30\ \mathrm{min} \times 60\ \mathrm{s/min}) \times (158\ \mathrm{rem/Ci}) \\ &= 3.04\ \mathrm{rem} \end{aligned}$$

Question 1.103

The assumptions are the same as noted in question 1.101. The same quantity of tritium is released, and the individual is present during the entire release. Therefore, the dose is the same and independent of the room ventilation characteristics:

$$D = 5.53 \times 10^{-5}\ \mathrm{rem}$$

Question 1.104

The assumptions are the same as noted in question 1.102. To solve this problem, the average concentration to give a dose of 0.5 rem should be found.

$$\begin{aligned} D &= \overline{C}(\mathrm{BR})\, t(\mathrm{DCF}) \\ \overline{C} &= D/[(\mathrm{BR})\, t(\mathrm{DCF})] \\ &= \frac{0.5\ \mathrm{rem}}{(3.5 \times 10^{-4}\ \mathrm{m^3/s})\,(30\ \mathrm{min} \times 60\ \mathrm{s/min})\,(158\ \mathrm{rem/Ci})} \\ &= 5.02 \times 10^{-3}\ \mathrm{Ci/m^3} \end{aligned}$$

From question 1.102:

$$\overline{C} = C_0\,[1 - \exp(-r\,t)]/r\,t$$

The results of question 1.102 suggest that $r > 3$/hr, which corresponded to a dose of 3 rem. Because

$$1 - \exp(-r\,t) \rightarrow 1$$

as r increases, we can simplify the previous equations:

$$\begin{aligned} \overline{C} &= C_0/r\,t \\ r &= C_0/\overline{C}\,t \\ &= (5.89 \times 10^{-2}\ \mathrm{Ci/m^3})/[(5.02 \times 10^{-3}\ \mathrm{Ci/m^3})\,(0.5\ \mathrm{hr})] = 23.5/\mathrm{hr} \end{aligned}$$

Scenario 1.27

Question 1.105

To calculate the CDE and weighted CDE or committed effective dose equivalent (CEDE), the data for Cs-137 and Ba-137m must be organized:

Cs-137

Targets (T)	SEE ($T \leftarrow$ Lungs)	U_S	$U_S \times$ SEE
Gonads	0.0	1.9×10^{4}	0.0
Breast	0.0	1.9×10^{4}	0.0
Red marrow	0.0	1.9×10^{4}	0.0
Lungs	1.9×10^{-4}	1.9×10^{4}	3.6
Thyroid	0.0	1.9×10^{4}	0.0
Bone surfaces	0.0	1.9×10^{4}	0.0
SI wall	0.0	1.9×10^{4}	0.0
ULI wall	0.0	1.9×10^{4}	0.0
LLI wall	0.0	1.9×10^{4}	0.0
Uterus	0.0	1.9×10^{4}	0.0
Adrenals	0.0	1.9×10^{4}	0.0

Cs-137

Targets (T)	SEE ($T \leftarrow$ Total Body)	U_S	$U_S \times$ SEE
Gonads	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Breast	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Red marrow	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Lungs	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Thyroid	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Bone surfaces	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
SI wall	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
ULI wall	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
LLI wall	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Uterus	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}
Adrenals	2.7×10^{-6}	7.7×10^{6}	2.1×10^{1}

Ba-137m

Targets (T)	SEE ($T \leftarrow$ Lungs)	U_S	$U_S \times$ SEE
Gonads	5.7×10^{-8}	1.8×10^{4}	1.0×10^{-3}
Breast	2.7×10^{-6}	1.8×10^{4}	4.9×10^{-2}
Red marrow	2.5×10^{-6}	1.8×10^{4}	4.5×10^{-2}
Lungs	9.5×10^{-5}	1.8×10^{4}	1.7
Thyroid	2.6×10^{-6}	1.8×10^{4}	4.7×10^{-2}
Bone surfaces	2.0×10^{-6}	1.8×10^{4}	3.6×10^{-2}
SI wall	5.9×10^{-7}	1.8×10^{4}	1.1×10^{-2}
ULI wall	8.0×10^{-7}	1.8×10^{4}	1.4×10^{-2}
LLI wall	1.7×10^{-7}	1.8×10^{4}	3.1×10^{-3}
Uterus	2.1×10^{-7}	1.8×10^{4}	3.8×10^{-3}
Adrenals	4.9×10^{-6}	1.8×10^{4}	8.8×10^{-2}

Ba-137m

Targets (T)	SEE ($T \leftarrow$ Total Body)	U_S	$U_S \times$ SEE
Gonads	4.7×10^{-6}	7.3×10^{6}	3.4×10^{1}
Breast	3.9×10^{-6}	7.3×10^{6}	2.8×10^{1}
Red marrow	4.3×10^{-6}	7.3×10^{6}	3.1×10^{1}
Lungs	4.0×10^{-6}	7.3×10^{6}	2.9×10^{1}
Thyroid	3.9×10^{-6}	7.3×10^{6}	2.8×10^{1}
Bone surfaces	4.0×10^{-6}	7.3×10^{6}	2.9×10^{1}
SI wall	4.9×10^{-6}	7.3×10^{6}	3.6×10^{1}
ULI wall	4.8×10^{-6}	7.3×10^{6}	3.5×10^{1}
LLI wall	4.9×10^{-6}	7.3×10^{6}	3.6×10^{1}
Uterus	4.9×10^{-6}	7.3×10^{6}	3.6×10^{1}
Adrenals	5.2×10^{-6}	7.3×10^{6}	3.8×10^{1}

The following table summarizes the intermediate result:

$$\text{Sum} = H_{50,T}/1.6 \times 10^{-10} = \sum_S \sum_j \left[U_S \sum_i \text{SEE}(T \leftarrow S_i) \right]_j$$

$$= \left[U_S \sum_i \text{SEE}(T \leftarrow S_i) \right]_{\text{Cs-137}} + \left[U_S \sum_i \text{SEE}(T \leftarrow S_i) \right]_{\text{Ba-137m}}$$

$U_S \times$ SEE Summary and Combination of Parent (Cs-137) and Daughter (Ba-137m) Results.

Targets (T)	Cs-137		Ba-137m		
	($T \leftarrow L$)[a] +	($T \leftarrow$ TB)[b] +	($T \leftarrow L$) +	($T \leftarrow$ TB) =	Sum
Gonads	0.0	2.1×10^{1}	1.0×10^{-3}	3.4×10^{1}	5.5×10^{1}
Breast	0.0	2.1×10^{1}	4.9×10^{-2}	2.8×10^{1}	4.9×10^{1}
Red marrow	0.0	2.1×10^{1}	4.5×10^{-2}	3.1×10^{1}	5.2×10^{1}
Lungs	3.6	2.1×10^{1}	1.7	2.9×10^{1}	5.5×10^{1}
Thyroid	0.0	2.1×10^{1}	4.7×10^{-2}	2.8×10^{1}	4.9×10^{1}
Bone surfaces	0.0	2.1×10^{1}	3.6×10^{-2}	2.9×10^{1}	5.0×10^{1}
SI wall	0.0	2.1×10^{1}	1.1×10^{-2}	3.6×10^{1}	5.7×10^{1}
ULI wall	0.0	2.1×10^{1}	1.4×10^{-2}	3.5×10^{1}	5.6×10^{1}
LLI wall	0.0	2.1×10^{1}	3.1×10^{-3}	3.6×10^{1}	5.7×10^{1}
Uterus	0.0	2.1×10^{1}	3.8×10^{-3}	3.6×10^{1}	5.7×10^{1}
Adrenals	0.0	2.1×10^{1}	8.8×10^{-2}	3.8×10^{1}	5.9×10^{1}

[a] L = Lungs.
[b] TB = total body.

$H_{50,T}$ may be obtained from the previous table by multiplying the final column (Sum) by 1.6×10^{-10}.

$$\text{CDE} = H_{50,T} = 1.6 \times 10^{-10} \times \sum_S \sum_j \left[U_S \sum_i \text{SEE}(T \leftarrow S_i) \right]_j$$

Targets (T)	$H_{50,T}$ (Sv/Bq)	w_T	$w_T H_{50,T}$ (Sv/Bq)
Gonads	8.8×10^{-9}	0.25	2.2×10^{-9}
Breast	7.8×10^{-9}	0.15	1.2×10^{-9}
Red marrow	8.3×10^{-9}	0.12	1.0×10^{-9}
Lungs	8.8×10^{-9}	0.12	1.1×10^{-9}
Thyroid	7.8×10^{-9}	0.03	2.3×10^{-10}
Bone surfaces	8.0×10^{-9}	0.03	2.4×10^{-10}
SI wall	9.1×10^{-9}	0.06	5.5×10^{-10}
ULI wall	9.0×10^{-9}	0.06	5.4×10^{-10}
LLI wall	9.1×10^{-9}	0.06	5.5×10^{-10}
Uterus	9.1×10^{-9}	–[a]	–
Adrenals	9.4×10^{-9}	–[a]	–
Remainder	9.4×10^{-9}	0.12	1.1×10^{-9}

[a] The ICRP chooses to not assign a weight of 0.06 to both the uterus and the adrenals. Instead, the largest $H_{50,T}$ value is selected (adrenals), and this value is assigned a weighting factor of 0.12. The organ with this weight is defined to be the remainder.

$$\text{CEDE} = \sum_{\text{T}} w_T \, H_{50,T} = 8.7 \times 10^{-9} \text{ Sv/Bq (by summing the last column).}$$

The ALI may be determined from calculation of the stochastic (S) and nonstochastic (NS) results. The ALI is taken to be the smaller of the two values:

$$\text{ALI}_\text{S} \leq \frac{0.05 \text{ Sv}}{\sum_T \text{w}_\text{T} \, H_{50,T}} = \frac{0.05 \text{ Sv}}{8.7 \times 10^{-9} \text{ Sv/Bq}} = 5.7 \times 10^6 \text{ Bq}$$

$$\text{ALI}_\text{NS} \leq \frac{0.5 \text{ Sv}}{H_{50,T}} = \frac{0.5 \text{ Sv}}{9.4 \times 10^{-9} \text{ Sv/Bq}} = 5.3 \times 10^7 \text{ Bq}$$

where the largest CDE value (remainder) is used to provide the smallest nonstochastic ALI value.

Comparing these two values yields an ALI value of 5.7×10^6 Bq. This is the value that satisfies both the stochastic and nonstochastic inequalities.

The value of the DAC follows from its definition:

$$\text{DAC} = \text{ALI}/2400 \text{ m}^3 = 5.7 \times 10^6 \text{ Bq}/2400 \text{ m}^3 = 2.4 \times 10^3 \text{ Bq/m}^3$$

By convention, the ICRP rounds these values to one significant figure. Therefore,

$$\text{ALI} = 6 \times 10^6 \text{ Bq (inhalation)}$$

$$\text{DAC} = 2 \times 10^3 \text{ Bq/m}^3$$

Scenario 1.28

Question 1.106

The total transfer rate constant $k^{(j)}$ for insoluble material in a given segment of the ICRP-30 gastrointestinal (GI) tract is given by

$$k^{(j)} = \lambda + 1/T^{(j)}$$

where

$T^{(j)}$ = mean residence time for the segment
j = 1, 2, 3, and 4 for the stomach (S), small intestine (SI), upper large intestine (ULI), and lower larger intestine (LLI), respectively
λ = disintegration constant

Of the total quantity of material entering a given segment of the GI tract, the fraction decaying in that segment will be given by $\lambda/k^{(j)}$. The fraction that will be transported to the next segment is given by

$$\frac{1/T^{(j)}}{k^{(j)}} = \frac{1}{k^{(j)}\,T^{(j)}}$$

Using this information, the activity entering and decaying in each segment based on an intake q into the stomach is given by:

Segment	Activity Entering	Activity Decaying
S	q	$\frac{\lambda}{k^{(1)}}\,q$
SI	$\frac{q}{k^{(1)}\,T^{(1)}}$	$\frac{\lambda}{k^{(2)}}\,\frac{1}{k^{(1)}\,T^{(1)}}\,q$
ULI	$\frac{q}{k^{(1)}\,T^{(1)}\,k^{(2)}\,T^{(2)}}$	$\frac{\lambda}{k^{(3)}}\,\frac{q}{k^{(1)}\,T^{(1)}\,k^{(2)}\,T^{(2)}}$
LLI	$\frac{q}{k^{(1)}\,T^{(1)}\,k^{(2)}\,T^{(2)}}\,\frac{1}{k^{(3)}\,T^{(3)}}$	$\frac{\lambda}{k^{(4)}}\,\frac{q}{k^{(1)}\,T^{(1)}\,k^{(2)}\,T^{(2)}}\,\frac{1}{k^{(3)}\,T^{(3)}}$

The cumulated activity $\tilde{A}(j)$ for each segment is obtained by multiplying the activity decaying in each segment by the radiological mean lifetime ($1/\lambda$) of Tc-99m.

$$\tilde{A}(1) = \frac{q}{k^{(1)}}$$

$$\tilde{A}(2) = \frac{q}{k^{(2)}\, k^{(1)}\, T^{(1)}}$$

$$\tilde{A}(3) = \frac{q}{k^{(3)}\, k^{(1)}\, T^{(1)}\, k^{(2)}\, T^{(2)}}$$

$$\tilde{A}(4) = \frac{q}{k^{(4)}\, k^{(1)}\, T^{(1)}\, k^{(2)}\, T^{(2)}\, k^{(3)}\, T^{(3)}}$$

where

$k^{(j)} = \lambda + \dfrac{1}{T^{(j)}}$

$\lambda = 0.693/6.03 \text{ hr} = 0.1149/\text{hr}$

$q = 1\ \mu\text{Ci}$

$T^{(1)} = 1 \text{ hr}$

$T^{(2)} = 4 \text{ hr}$

$T^{(3)} = 13 \text{ hr}$

$T^{(4)} = 24 \text{ hr}$

Using these values and the model's mean residence times lead to the cumulated activity in each GI tract segment:

$$\tilde{A}(1) = \frac{1\ \mu\text{Ci}}{(0.1149 + 1/1)/\text{hr}} = 0.897\ \mu\text{Ci-hr} \tag{S}$$

$$\begin{aligned}\tilde{A}(2) &= \frac{1\ \mu\text{Ci}}{(1 \text{ hr})\,[(0.1149 + 1/1)/\text{hr}]\,[(0.1149 + 1/4)/\text{hr}]} \\ &= 2.458\ \mu\text{Ci-hr}\end{aligned} \tag{SI}$$

$$\begin{aligned}\tilde{A}(3) &= \frac{1\ \mu\text{Ci}}{(1 \text{ hr})\,(4 \text{ hr})\,[(0.1149 + 1/1)/\text{hr}]\,[(0.1149 + 1/4)/\text{hr}]} \\ &\quad \times \frac{1}{(0.1149 + 1/13)/\text{hr}} \\ &= 3.204\ \mu\text{Ci-hr}\end{aligned} \tag{ULI}$$

$$\begin{aligned}\tilde{A}(4) &= \frac{1\ \mu\text{Ci}}{(1 \text{ hr})\,(4 \text{ hr})\,(13 \text{ hr})\,[(0.1149 + 1/1)/\text{hr}]\,[(0.1149 + 1/4)/\text{hr}]} \\ &\quad \times \frac{1}{[(0.1149 + 1/13)/\text{hr}]\,[(0.1149 + 1/24)/\text{hr}]} \\ &= 1.574\ \mu\text{Ci-hr}\end{aligned} \tag{LLI}$$

Question 1.107

The absorbed doses for each segment may be obtained by utilizing the tabulated S values (rad/μCi-hr) and the cumulated activities calculated in question 1.106. The dose to the target organ is obtained by summing over the four source organs comprising the GI tract model's organs:

$$D(T) = \sum_S S(T \leftarrow S)\, \tilde{A}(S)$$

Explicitly writing these sums permits the dose equivalent to the walls of each segment to be determined:

$$\begin{aligned} D(1) &= S(1 \leftarrow 1)\, \tilde{A}(1) + S(1 \leftarrow 2)\, \tilde{A}(2) \\ &\quad + S(1 \leftarrow 3)\, \tilde{A}(3) + S(1 \leftarrow 4)\, \tilde{A}(4) \\ &= (1.3 \times 10^{-4} \text{ rad/μCi-hr})\,(0.897 \text{ μCi-hr}) \\ &\quad + (3.7 \times 10^{-6} \text{ rad/μCi-hr})\,(2.458 \text{ μCi-hr}) \\ &\quad + (3.8 \times 10^{-6} \text{ rad/μCi-hr})\,(3.204 \text{ μCi-hr}) \\ &\quad + (1.8 \times 10^{-6} \text{ rad/μCi-hr})\,(1.574 \text{ μCi-hr}) \\ &= 1.407 \times 10^{-4} \text{ rad} \times 1 \text{ rem/rad} = 1.407 \times 10^{-4} \text{ rem} \end{aligned}$$

The last line is based upon the quality factor of the gamma radiation emitted from Tc-99m.

$$\begin{aligned} D(2) &= S(2 \leftarrow 1)\, \tilde{A}(1) + S(2 \leftarrow 2)\, \tilde{A}(2) \\ &\quad + S(2 \leftarrow 3)\, \tilde{A}(3) + S(2 \leftarrow 4)\, \tilde{A}(4) \\ &= (2.7 \times 10^{-6} \text{ rad/μCi-hr})\,(0.897 \text{ μCi-hr}) \\ &\quad + (7.8 \times 10^{-5} \text{ rad/μCi-hr})\,(2.458 \text{ μCi-hr}) \\ &\quad + (1.7 \times 10^{-5} \text{ rad/μCi-hr})\,(3.204 \text{ μCi-hr}) \\ &\quad + (9.4 \times 10^{-6} \text{ rad/μCi-hr})\,(1.574 \text{ μCi-hr}) \\ &= 2.634 \times 10^{-4} \text{ rad} \times 1 \text{ rem/rad} = 2.634 \times 10^{-4} \text{ rem} \end{aligned}$$

$$\begin{aligned} D(3) &= S(3 \leftarrow 1)\, \tilde{A}(1) + S(3 \leftarrow 2)\, \tilde{A}(2) \\ &\quad + S(3 \leftarrow 3)\, \tilde{A}(3) + S(3 \leftarrow 4)\, \tilde{A}(4) \\ &= (3.5 \times 10^{-6} \text{ rad/μCi-hr})\,(0.897 \text{ μCi-hr}) \\ &\quad + (2.4 \times 10^{-5} \text{ rad/μCi-hr})\,(2.458 \text{ μCi-hr}) \\ &\quad + (1.3 \times 10^{-4} \text{ rad/μCi-hr})\,(3.204 \text{ μCi-hr}) \\ &\quad + (4.2 \times 10^{-6} \text{ rad/μCi-hr})\,(1.574 \text{ μCi-hr}) \\ &= 4.853 \times 10^{-4} \text{ rad} \times 1 \text{ rem/rad} = 4.853 \times 10^{-4} \text{ rem} \end{aligned}$$

$$
\begin{aligned}
D(4) &= S(4 \leftarrow 1)\,\tilde{A}(1) + S(4 \leftarrow 2)\,\tilde{A}(2) \\
&\quad + S(4 \leftarrow 3)\,\tilde{A}(3) + S(4 \leftarrow 4)\,\tilde{A}(4) \\
&= (1.2 \times 10^{-6}\ \text{rad}/\mu\text{Ci-hr})\,(0.897\ \mu\text{Ci-hr}) \\
&\quad + (7.3 \times 10^{-6}\ \text{rad}/\mu\text{Ci-hr})\,(2.458\ \mu\text{Ci-hr}) \\
&\quad + (3.2 \times 10^{-6}\ \text{rad}/\mu\text{Ci-hr})\,(3.204\ \mu\text{Ci-hr}) \\
&\quad + (1.9 \times 10^{-4}\ \text{rad}/\mu\text{Ci-hr})\,(1.574\ \mu\text{Ci-hr}) \\
&= 3.283 \times 10^{-4}\ \text{rad} \times 1\ \text{rem/rad} = 3.283 \times 10^{-4}\ \text{rem}
\end{aligned}
$$

The ULI receives the largest exposure from the postulated 1-μCi stomach intake.

Question 1.108

Considering that the facility administrative organ limit is 15 rem, the ULI will be limited to this value. For insoluble Tc-99m, this dose equivalent would result from an intake of Q microcuries:

$$
Q = \frac{15\ \text{rem/year}}{4.853 \times 10^{-4}\ \text{rem}/\mu\text{Ci}} = 30\,900\ \mu\text{Ci/year}
$$

Because the residence time in the GI tract is relatively short, the intake time period is somewhat arbitrary. Because we are interested in an annual limit, the annual average intake rate for occupational exposure is

$$
P = \frac{30\,900\ \mu\text{Ci/year}}{(40\ \text{hr/week})\,(50\ \text{weeks/year})} = 15.45\ \mu\text{Ci/hr}
$$

This value can be converted into the maximum allowed occupational concentration limit (MAOC). Because the intake rate (1100 ml/day) is known, we obtain

$$
\text{MAOC} = \frac{15.45\ \mu\text{Ci/hr}}{(1100\ \text{ml/day})\,(1\ \text{day}/8\ \text{hr})} = 0.11\ \mu\text{Ci/ml}
$$

Scenario 1.29

Question 1.109

The ICRP-26 risk coefficient is 2×10^{-4} excess cancer deaths/rem.

$$
\begin{aligned}
\text{Risk} &= (\text{Risk Coefficient})\,(\text{Dose}) \\
&= (2.0 \times 10^{-4}\ \text{excess cancer deaths/rem})\,(100\,000\ \text{mrem}) \\
&\quad \times (1\ \text{rem}/1000\ \text{mrem}) \\
&= 0.02
\end{aligned}
$$

There is about a 1 in 50 probability of developing a radiation-induced fatal cancer over the worker's lifetime.

Question 1.110

Normal:	0.2×1000	=	200
Radiation:	3 in 100	=	3
			203

203 in 1000 is the total probability of developing a fatal cancer.

Question 1.111: a

The ICRP-26 cancer risk model is based on an absolute risk model.

Question 1.112: b

The PC tables are based on a relative risk model.

Question 1.113: b

Smoking history is considered when using the PC tables to estimate risk. Items a, c, d, and e are true.

Scenario 1.30

Question 1.114: c

Tritiated thymidine will cause a greater biological effect because it is incorporated into the cell's nucleus.

Question 1.115

The correct matches are:

Strontium-90 (soluble)	Bone
Cesium-137	Total body
Plutonium-239 (soluble)	Bone
Uranium-238 (insoluble)	Lung/kidney
Radon-222	Lung

Question 1.116

The listed order (a, b, c, d, e) is from most to least radiosensitive.

Question 1.117: d

Occupational exposure to X-rays does not account for 1% of the cataracts in X-ray technicians.

Question 1.118: c

Statements a, b, d, and e are correct.

Scenario 1.31

Question 1.119: e

Most soluble radioactive material is removed from the body through excretion via urine.

Question 1.120: d

Prior to 1 January 1993, the requirements of 10CFR20 were based upon ICRP-2. Thereafter, ICRP-26/30 forms the basis of the revised 10CFR20.

Question 1.121: d

The ICRP-26 annual dose equivalent limit to prevent nonstochastic effects in tissue is 0.5 Sv (50 rem).

Question 1.122: a

The stochastic limit is based on uniform whole-body radiation. On the principle that the risk of a stochastic effect should be equal whether the whole body is uniformly irradiated or the radiation dose is distributed in a nonuniform manner, the ICRP recommended that stochastic effects for occupational exposures be limited to 50 mSv (5 rem) effective dose equivalent in 1 year.

Question 1.123: d

ICRP-26 replaced the critical organ with the concept of tissue region or target tissue.

Question 1.124: c

Large sets of occupational radiation exposure data are fit by a log-normal distribution.

Question 1.125: a

The annual dose limit for the eye was changed from 0.3 Sv (30 rem) to 0.15 Sv (15 rem).

Question 1.126: a

NCRP-116 recommends that the occupational cumulative effective dose limit be 10 mSv × age (y). For example, a 30-year-old worker is limited to 300 mSv cumulative effective dose.

Question 1.127: c

Note that when this question was asked on the 1989 ABHP Comprehensive Examination, it was based on the ICRP-26 risk coefficient of 2×10^{-2} radiation induced effects (rie)/Sv:

$$r = (2 \times 10^{-2} \text{ rie/Sv})\,(0.005 \text{ Sv}) = 1 \times 10^{-4} \text{ rie}$$

BEIR V and its 8×10^{-2} rie/Sv risk coefficient yields 4×10^{-4} rie. BEIR VII does not provide specific risk coefficients. However, a value of 5×10^{-2} excess cancer deaths/Sv can be extracted from the BEIR VII data (see Appendix V).

Question 1.128: b

Safe industries in the United States have an annual average fatal accident rate of 1×10^{-4}.

Scenario 1.32

Question 1.129: e

The skin response to acute radiation exposure in correct chronological order is erythema, dry desquamation, moist desquamation, and recovery.

Question 1.130: c

Factor	Affects Severity of the Reaction
Skin pigmentation	Yes
Fractionation of dose	Yes
Charged particle equilibrium at the basal cell layer	No
Dose rate	Yes
LET	Yes

Question 1.131: b

The basal cell layer is the most radiosensitive.

Question 1.132

$T_{physical}$ = 8 days
$T_{biological}$ = 5 days
T_{eff} = (8 days) (5 days)/(8 days + 5 days) = 3.08 days
$A(t)$ = $A(0)$ exp(–0.693 t/T_{eff})
$A(t)/A(0)$ = 0.1
t = ln $[A(t)/A(0)]$ $(-T_{eff}/0.693)$ = ln(0.1) (–3.08 days/0.693) = 10.2 days

Question 1.133: a

The weighting factor is used to relate the effective dose equivalent to skin dose. Effective dose equivalent = 0.01 × skin dose. The radiogenic skin cancer risk is low.

Scenario 1.33

Question 1.134

The types of information from animal studies that can be useful in human risk estimation include:

1. Predicting the effects of high-LET radiations including neutrons, heavy ions, and high-energy particles.
2. Estimating the effects at low dose rates.
3. Gauging effects for dose fractionation sequences.
4. Predicting the effects of radionuclide intakes.
5. Predicting radiation-induced mutations.
6. Estimating the relative effects of doses as a function of the radiation type.
7. Gauging the effects of the presence of mitigating, protective, or synergistic agents.

Question 1.135

The radiobiological factors which affect either the onset or the development of malignant tumors in experimental animals include:

Radiation type	Radiation dose
Radiation dose rate	Dose fractionation
Linear energy transfer	Sex
Age at exposure	General health
Diet	Genetic predisposition toward tumor formation
Presence of other agents	

Question 1.136: b

In this question, you are to assume that all radiation-induced effects lead to death.

$$\begin{aligned} \text{ed} &= \text{excess deaths} \\ &= (\text{BEIR V total risk coefficient})\,(\text{acute dose equivalent}) \\ &= (8 \times 10^{-2}\ \text{ed/Sv})\,(0.1\ \text{Sv}) = 8 \times 10^{-3}\ \text{ed} \end{aligned}$$

Question 1.137: d

The additive risk model has been dropped in the BEIR V Committee Report in favor of the multiplicative risk model.

Question 1.138: a

The constant (1) does not ensure positive values of the excess risk estimate. All other items – b, c, d, and e – are correct.

Scenario 1.34

Question 1.139

In this question, you are asked to describe and discuss the optimal bioassay approach for the I-131 intake. Advantages and disadvantages for *in vivo* and for *in vitro* methods of analysis as related to this case are also requested.

Optimum Approach:
Whole body or thyroid counting would be the preferred approach. The radiation types emitted by I-131 are readily detected using whole body counters or thyroid counters. These *in vivo* count methods require minimum participation by the worker and are quickly repeated. The methods of analyzing the data are well established and accepted.

In vivo:

Advantages:
1. The method is quick and requires little effort by the subject.
2. An immediate assessment of intake is provided from the thyroid activity and intake retention function.
3. Detectors can be used to isolate and count the thyroid.
4. The method is well established and defensible.
5. The thyroid measurements are directly related to the accumulated thyroid activity and thyroid dose.

Disadvantages:

1. The location of the radioactive material within the body may not be clearly defined unless a shielded detector is used – i.e., a whole body count will detect the contamination but not necessarily its exact location.
2. Quantification of the intake is complicated by differences between the subject and Reference Man in terms of body organ sizes, chest wall thicknesses, organ shape and weight, and the absorption and scattering characteristics of the individual.
3. Inaccuracies are introduced if the individual has external contamination.
4. I-131 may be simulated by other radionuclides, including radon daughters.
5. The method requires the presence of the worker during each thyroid count.
6. Internal contamination in the gastrointestinal tract or respiratory system, particularly within a few days of intake, leads to inaccuracies if these depositions are interpreted as a thyroid uptake.

In vitro:

Advantages:

1. The method is well established and defensible.
2. The input parameters (intakes and intake retention functions) are tabulated and readily available.
3. Urine is a medium that is readily obtained and analyzed.
4. The exposed individual need not be present during the analysis of the 24 h urine samples.

Disadvantages:

1. Samples require time for collection and additional time for preparation and analysis.
2. The method depends on the worker to properly implement the required sampling protocol.
3. The method does not provide an immediate quantification of the intake. Analysis of the sample with a subsequent model-dependent calculation is required.
4. The use of standard models will lead to inaccurate results unless the subject behaves like Reference Man.
5. Excreta must be collected over time.
6. The collected sample only represents the excretion and not necessarily the thyroid burden. This is particularly important if a thyroid-blocking agent (e.g., KI) was used.

Question 1.140

In this question, you are requested to describe how your approach to bioassay might change as time goes by, given the metabolic model for iodine. Within the scope of this model:

Initial Phase:
Approximately 75% of the I-131 as NaI is excreted from the body in the urine in 1–2 days with an effective halftime of about 6 hours. In the early phase, whole body/thyroid counting is the easiest approach. However, most of the activity is excreted via urine. Therefore, urinalysis during the first few days would be helpful in verifying the intake. However, limited data will be obtained since a 24-hour void is the usual protocol.

Later Phase:
The remaining 25% of I-131 is trapped in the subject's thyroid, reaching a maximum about 24 hours post-intake, and is excreted with an effective halftime of about 7 days. After the I-131 localizes in the thyroid, whole body or thyroid counting will be performed. Periodic urine sampling should also be performed to verify the intake models.

In vivo counting should be performed throughout the uptake evaluation period since it is easy and requires a minimum of effort. As noted above, periodic urine samples also establish the individual's retention function and subsequent intake.

Question 1.141

Based on the thyroid-counting data, the best estimate of the subject's intake is provided by the unweighted least-squares methodology of NUREG/CR-4884. This methodology and the following data permit determination of the intake:

i	Time Post Intake (d)	$A(i)$ Thyroid Activity (kBq)	IRF
1	1	250	0.133
2	7	230	0.0995
3	10	130	0.0751

$$I = \frac{\sum_{i=1}^{3} A(i)\,\mathrm{IRF}(i)}{\sum_{i=1}^{3} [\mathrm{IRF}(i)]^2}$$

$$= \frac{(0.133)\,(250\ \mathrm{kBq}) + (0.0995)\,(230\ \mathrm{kBq}) + (0.0751)\,(130\ \mathrm{kBq})}{(0.133)^2 + (0.0995)^2 + (0.0751)^2}$$

$$= \frac{65.9\ \mathrm{kBq}}{0.0332} = 1.98 \times 10^3\ \mathrm{kBq} = 1.98\ \mathrm{MBq}$$

The reader should note that the aforementioned approach is the one advocated in NUREG/CR-4884 (1987). More recent work (K. Skrable *et. al., Elements of*

Exposure, Bioassay, and Internal Dose Assessment Programs, in Proceedings of the Health Physics Society 2002 Summer School, Medical Physics Publishing, Madison, WI 2002) suggests that a weighted least-squares modification of this intake equation is warranted. The suggested intake equation is:

$$I = \frac{\sum_{i=1}^{3} A(i)}{\sum_{i=1}^{3} [\mathrm{IRF}(i)]}$$
$$= \frac{(250\ \mathrm{kBq} + 230\ \mathrm{kBq} + 130\ \mathrm{kBq})}{(0.133 + 0.0995 + 0.0751)}$$
$$= \frac{610\ \mathrm{kBq}}{0.308} = 1980\ \mathrm{kBq} = 1.98\ \mathrm{MBq}$$

This is the same result obtained from the NUREG/CR-4884 methodology.

Question 1.142

CDE:
The CDE is determine from the intake:

$$\mathrm{CDE} = H_{50,T} = I(\mathrm{DCF})$$

where
I = intake of I-131 = 5 MBq
DCF = Dose conversion factor = 2.9×10^{-7} Sv/Bq intake

$$\mathrm{CDE} = (2.9 \times 10^{-7}\ \mathrm{Sv/Bq})\,(5.0 \times 10^{6}\ \mathrm{Bq}) = 1.45\ \mathrm{Sv}$$

CEDE:

$$\mathrm{CEDE} = w_T\ H_{50,T}$$

where
w_T = organ weighting factor (ICRP 26/30) = 0.03
$H_{50,T}$ = committed dose equivalent = 1.45 Sv

$$\mathrm{CEDE} = (0.03)\,(1.45\ \mathrm{Sv}) = 0.0435\ \mathrm{Sv}$$

Regulatory Limits:
The Regulatory Limits (for both 10CFR20 [NRC Licensees] and 10CFR835 [DOE Licensees]) are based on ICRP 26/30. The organ dose limit (including the thyroid) is 0.5 Sv CDE. The organ dose of 1.45 Sv exceeds the regulatory limit for the thyroid. The CEDE limit is 0.05 Sv and it was not exceeded.

Scenario 1.35

Question 1.143

The ICRP equivalent dose (H_T) is:

$$H_T = \sum_R w_R \, D_{T,R}$$

This sum may be illustrated via the following table:

Radiation Type (R)	w_R (μSv/μGy) × $D_{T,R}$ (μGy)	$H_T = w_R \, D_{T,R}$ (μSv)
Beta	(1 μSv/μGy) (30 μGy)	= 30 μSv
Gamma	(1 μSv/μGy) (70 μGy)	= 70 μSv
Thermal Neutrons	(5 μSv/μGy) (90 μGy)	= 450 μSv
10 MeV Neutrons	(10 μSv/μGy) (25 μGy)	= 250 μSv
Sum		= 800 μSv

Question 1.144

A 26 year old radiation worker had a lifetime effective dose of 0.32 Sv. NCRP-116 recommends that the cumulative occupational exposure be limited to $10 \times N$ mSv, where N is the worker's age in years. Since the worker is 26 years old, he exceeds the NCRP-116 recommendation by 0.32 Sv – 0.26 Sv = 0.06 Sv. Based on NCRP recommendations, the annual dose should be limited to 10 mSv instead of 50 mSv for workers who do not exceed their lifetime limit.

Question 1.145

The worker has a cumulative effective dose in years 1–4 of

$$E = (10 + 30 + 40 + 20)\ \text{mSv} = 100\ \text{mSv}$$

Since ICRP-60 recommends a dose of 100 mSv averaged over 5 years, the worker should receive no exposure during the fifth year.

Question 1.146

A laboratory worker using a Pu/Be neutron source is exposed to a source having a half-life of 2.41×10^4 y. The source emits neutrons with an average energy of

about 4.5 MeV with a maximum energy of about 11 MeV. The source's neutrons are generated from an (alpha, n) reaction.

The dosimeter must monitor neutrons as well as capture gamma rays generated from neutron reactions. The ^{6}Li phosphors respond to thermal neutrons via the ^{6}Li(n, α)^{3}H reaction. The ^{6}LiF material responds to neutrons and photons while ^{7}LiF responds only to photons. A number of dosimeter configurations are possible. Two options are noted.

Option 1:

Chip	Material	Thickness (cm)	Cover
2[a]	^{7}LiF	0.38	1000 mg/cm^2 plastic
3[b]	^{7}LiF	0.15	7 mg/cm^2 mylar
4[c]	^{6}LiF	0.38	300 mg/cm^2 plastic
6[d]	^{6}LiF	0.38	300 mg/cm^2 plastic and Cd filter

[a] Provides whole body photon dose (deep dose).
[b] Provides the shallow dose equivalent from photons.
[c] Yields thermal neutron dose and photon eye dose.
[d] Yields fast neutron dose and photon eye dose.
The Cd filter eliminates the thermal neutron contribution.

Option 2:

Chip	Material	Thickness (cm)	Cover
5[a]	^{7}LiF	0.38	300 mg/cm^2 plastic
6[b]	^{6}LiF	0.38	300 mg/cm^2 plastic and Cd filter

[a] An additional Cd filter is added to Chip 5.
The ^{7}LiF chip under the Cd filter responds to external gamma radiation and gamma radiation produced by thermal neutrons absorbed in the Cd filter.
Its response should be subtracted from the response of the ^{6}LiF chip to obtain the net neutron response.
[b] The ^{6}LiF chip under the Cd filter responds to fast neutrons, external gamma radiation, and gamma radiation produced by thermal neutrons captured in the Cd filter.

Question 1.147

The X-ray technologist will be exposed to photons, but no neutron dose is expected. Therefore, the Li-6 chips are not required. A number of possible dosimeter configurations are possible. Two options are noted.

Option 1:

Chip	Material[a]	Thickness (cm)	Cover
2[b]	^{7}LiF	0.38	1000 mg/cm^2 plastic
3[c]	^{7}LiF	0.15	7 mg/cm^2 mylar
5[d]	^{7}LiF	0.38	300 mg/cm^2 plastic

[a] For the X-ray application only ^{7}LiF chips are needed.
[b] Provides the deep dose.
[c] Provides the shallow dose.
[d] Provides the eye dose.

Option 2:
For measuring the effective dose, the single dosimeter with a 1000 mg/cm^2 cover (Chip 2) is adequate.

Question 1.148: d

A portable meter (i.e., BF_3) could be used to determine the neutron effective dose to an individual with knowledge of how the instrument responds to the spectrum as compared to the neutron calibration source and with knowledge of the individual's stay-time.

A BF_3 portable neutron survey meter responds to neutron energies between thermal and about 10 MeV, and has reasonable gamma rejection up to several Sv/h. Once the spectral response and stay time are known, the neutron effective dose is determined.

Scenario 1.36

Question 1.149

The following ICRP-60 definitions are provided:

a. Equivalent dose (H_T)
The average absorbed dose in the tissue or organ (T), due to all radiations (R), weighted by a radiation weighting factor w_R is defined to be the equivalent dose (H_T):

$$H_T = \sum_R w_R \, D_{T,R}$$

where
$D_{T,R}$ = average absorbed dose in tissue T due to radiation of type R

The radiation weighting factor replaced the ICRP-26 quality factor which is related to the physical effects of the radiation at a point.

b. Effective dose (E)

The effective dose is analogous to the ICRP-26/30 effective dose equivalent. Effective dose expresses the relative detriment associated with each irradiated tissue or organ, and its response is expressed as if the whole body were irradiated. The tissue weighting factor (w_T) is obtained by expressing the detriment of each tissue-specific cancer or hereditary disease relative to the total aggregated detriment. With these definitions, the effective dose is

$$E = \sum_T w_T \; H_T$$

The effective dose can also be written in terms of the double sum of the average absorbed dose in tissue T due to radiation R:

$$E = \sum_R w_R \sum_T w_T \; D_{T,R} = \sum_T w_T \sum_R w_R \; D_{T,R}$$

c. Committed effective dose [$E(50)$]

The committed effective dose $E(50)$ is the effective dose expected over the 50 year period following the intake (I) of radioactive material:

$$E(50) = \frac{I}{\text{ALI}} 20 \text{ mSv} = \sum_{T=1}^{12} w_T \; H_T(50) + w_{\text{remainder}} \frac{\sum_{T=13}^{22} m_T \; H_T(50)}{\sum_{T=13}^{22} m_T}$$

where

$H_T(50)$ = 50 year committed equivalent dose to target organ or tissue T
m_T = mass of the remainder tissue
$w_{\text{remainder}}$ = 0.05
ALI = ICRP-60 annual limit on intake

As noted in Appendix IV, ICRP-60 uses only one ALI.

Question 1.150

The ICRP-60 recommended occupational limits on effective dose, from both internal and external sources, are as follows:

a) An effective dose limit of 20 mSv/year (2 rem/year) to be averaged over a period of 5 years, and
b) No more than 50 mSv (5 rem) should be received in a single year.

Question 1.151

For a declared pregnant worker, the ICRP-60 recommended limits for the protection of the embryo/fetus are as follows:

a. External exposure
A dose equivalent limit of 2 mSv (200 mrem) to the surface of the mother's abdomen.

b. Intake of radioactive material
For the remainder of the pregnancy, the mother's internal exposure is limited to 1/20 of an ALI,

$$\sum_i \frac{I_i}{\text{ALI}_i} \leq \frac{1}{20}$$

where i labels the intake of the i^{th} radionuclide.

Question 1.152

The committed effective dose to the worker is obtained from a relationship, which assumes that only I-131 is present in the room:

$$E(50) = \frac{I(\text{I-131})}{\text{ALI}(\text{I-131})} 20 \text{ mSv}$$

where the ICRP-60 ALI is based on the 20 mSv annual dose limit. In order to utilize this relationship, the intake I must be determined. This can be accomplished from the relationship:

$$I = k\,\overline{C}(\text{BR})\,t$$

where

I = inhalation intake of I-131 (Bq)
k = conversion factor = (1000 mL/L) (1 cm^3/mL) (1 $\text{m}^3/1.0 \times 10^6$ cm^3)
$\overline{C}$ = average room concentration = 8.3×10^4 Bq/m^3
BR = worker's breathing rate = 20 L/min
t = exposure time(min) = 30 min

Using these values, leads to the desired intake of I-131:

$$\begin{aligned} I &= (8.3 \times 10^4 \text{ Bq/m}^3)\,(20 \text{ L/min})\,(1000 \text{ mL/L})\,(1 \text{ cm}^3\text{/mL}) \\ &\quad \times (1 \text{ m}^3/1.0 \times 10^6 \text{ cm}^3) \times (30 \text{ min}) \\ &= 4.98 \times 10^4 \text{ Bq} \end{aligned}$$

Knowledge of the intake permits the committed effective dose to be determined from the I-131 inhalation ALI (1.0×10^6 Bq):

$$E(50) = \frac{4.98 \times 10^4 \text{ Bq}}{1.00 \times 10^6 \text{ Bq}} 20 \text{ mSv} = 0.996 \text{ mSv}$$

Question 1.153

The ICRP-60 formulation is based on the assumption that a worker's body structure and metabolic function is accurately represented by Reference Man. A worker who has a thyroid disease, malfunction, or no thyroid is not be accurately described by the Reference Man model. Therefore, worker-specific information is needed to more accurately determine the committed effective dose. These specific data include:

1. Actual breathing rate of the worker.
2. Thyroid mass of the worker.
3. Worker's retention or elimination function for the actual I-131 uptake. This could be obtained by periodic thyroid counting, urine samples, or whole body counting.

Scenario 1.37

Question 1.154

This question requests that the primary mode of interaction for the following radiation types in tissue be specified:

a. Fast neutrons. Elastic scattering with nuclei is the dominant mode of interaction for fast neutrons with tissue.
b. Thermal neutrons. Tissue dose with thermal neutrons arises principally from (n, γ) reactions with hydrogen and (n, p) reactions with nitrogen.
c. Co-60 gamma rays. The most probable process for energy deposition by a Co-60 photon in tissue is Compton scattering.

Question 1.155

The linear energy transfer (LET) and relative biological effectiveness (RBE) explain the shapes of the dose response curves. Fission neutrons have a high LET value, higher quality factor (Q) and higher relative biological effectiveness while the corresponding gamma radiation values are smaller.

Question 1.156

The neutron relative biological effectiveness (RBE_n) is defined as:

$$RBE_n = \frac{D_\gamma}{D_n}$$

where
D_γ = gamma absorbed dose to produce an effect
D_n = neutron absorbed dose to produce the same effect

For this question, the effect is 0.5 chromosome aberrations per cell.

For neutrons:

$$e_n = 0.6\, D_n$$
$$D_n = \frac{e_n}{0.6/\text{Gy}} = \frac{0.5}{0.6/\text{Gy}} = 0.833\ \text{Gy}$$

For gamma rays, the relationship is linear quadratic having the form ($a\,x^2 + b\,x + c = 0$), which has the solution:

$$x = \frac{-b \pm \sqrt{b^2 - 4\,a\,c}}{2\,a}$$

The gamma-ray relationship:

$$e_\gamma = 0.0157\, D_\gamma + 0.05\, D_\gamma^2 = 0.5$$

can be rearranged into the standard linear quadratic form with $a = 0.05/\text{Gy}^2$, $b = 0.0157/\text{Gy}$, and $c = -0.5$:

$$0.05\, D_\gamma^2 + 0.0157\, D_\gamma - 0.5 = 0$$

$$D_\gamma = \frac{\frac{-0.0157}{\text{Gy}} \pm \sqrt{\left(\frac{0.0157}{\text{Gy}}\right)^2 - 4\left(\frac{0.05}{\text{Gy}^2}\right)(-0.5)}}{2\left(\frac{0.05}{\text{Gy}^2}\right)} = \frac{-0.0157 \pm 0.317}{0.1}\ \text{Gy}$$
$$= 3.01\ \text{Gy}, -3.33\ \text{Gy}$$

Only the positive root has a physical meaning because dose is by definition a positive definite quantity. With the determination of the doses for neutrons and

gamma rays to cause the effect of 0.5 chromosome aberrations/cell, the neutron RBE is determined:

$$\mathrm{RBE}_n = \left(\frac{D_\gamma}{D_n}\right) = \frac{3.01\ \mathrm{Gy}}{0.833\ \mathrm{Gy}} = 3.61$$

These results can also be obtained by plotting the neutron and gamma-ray dose response curves (i.e, e vs D). A graphical solution is obtained by plotting the horizontal line $e = 0.5$ chromosome aberrations/cell. The absorbed dose values used in the RBE_n equation are the values corresponding to the intersection of the neutron and gamma-ray dose response curves with the $e = 0.5$ chromosome aberrations/cell line.

Question 1.157

The maximum RBE occurs at low doses as $D \rightarrow 0$. Therefore, as the neutron and gamma-ray doses approaches zero, the chromosome aberration relationships become:

$$e_n = 0.6\ D_n$$
$$e_\gamma = 0.0157\ D_\gamma + 0.05\ D_\gamma^2 \rightarrow 0.0157\ D_\gamma$$

In the last equation, the quadratic term becomes negligible as the dose approaches zero.

Using these relationships, the maximum RBE is:

$$\mathrm{RBE}_{n,\max} = \lim_{\substack{D_\gamma \rightarrow 0 \\ D_n \rightarrow 0}} \left(\frac{D_\gamma}{D_n}\right) = \frac{\dfrac{e}{0.0157}}{\dfrac{e}{0.6}} = 38.2$$

where

e = number of effects occurring at threshold

Question 1.158

The Code of Federal Regulations, Title 10, Part 20 provides neutron quality factor information. In this regulation, Table 1004(b). 1, *Quality Factors and Absorbed Dose Equivalencies*, notes that the quality factor for neutrons of unknown energy is 10. This value is consistent with the mean quality factor values provided in Table 1004(b). 2, *Mean Quality Factors, Q, and Fluence per Unit Dose Equivalent for Monoenergetic Neutrons:*

Neutron Energy (MeV)	Mean Quality Factor (Q)[a]
2.5×10^{-8} (thermal) – 1.0×10^{-3}	2
1.0×10^{-2}	2.5
1.0×10^{-1}	7.5
5.0×10^{-1} – 1.0	11
2.5	9
5.0	8
7.0	7
10.0	6.5
14.0	7.5
20.0	8
40.0	7
60.0	5.5
100	4
200 – 400	3.5

[a] Value of the quality factor (Q) at the point where the dose equivalent is maximum in a 30-cm diameter cylindrical tissue-equivalent phantom.

Scenario 1.38

Question 1.159

The Co-60 activity as measured on day 0 was 7.77×10^5 Bq [$A(0)$], and this value corresponds to an intake retention function [$IRF(t)$] of 0.63 for 1 μm particles. The intake is determined from these $t = 0$ values:

$$
\begin{aligned}
I(t) &= A(t)/\mathrm{IRF}(t) \\
I(0) &= A(0)/\mathrm{IRF}(0) \\
&= 7.77 \times 10^5 \text{ Bq}/0.63 = 1.23 \times 10^6 \text{ Bq} \times (1 \text{ ALI}/1 \times 10^6 \text{ Bq}) \\
&= 1.23 \text{ ALI} \times 100\% = 123\% \text{ ALI}
\end{aligned}
$$

Question 1.160

The committed effective dose equivalent (CEDE) for an inhalation intake of 9.25×10^5 Bq of 1 μm AMAD Class Y Co-60 is given by the relationship:

$$\text{CEDE} = \left(\frac{I}{\text{ALI}_{\text{stochastic}}}\right)\left(\frac{50\ \text{mSv}}{\text{ALI}}\right)$$
$$= \left(\frac{9.25 \times 10^5\ \text{Bq}}{1 \times 10^6\ \text{Bq/ALI}}\right)\left(\frac{50\ \text{mSv}}{\text{ALI}}\right) = 46.3\ \text{mSv}$$

Question 1.161

The total effective dose equivalent may be written in terms of the quantities given in this problem:

$$H_{\text{TEDE}} = H_{\text{external}} + \text{CEDE(Co-60)} + w_T\ H_T\,(\text{I-131})$$

where

H_{TEDE} = Total effective dose equivalent (TEDE)
H_{external} = External radiation exposure from Co-60 = 2.5 mSv
CEDE(Co-60) = Committed Effective Dose Equivalent from the Co-60 intake
= 0.5 mSv
H_T = Committed Dose Equivalent to the Thyroid from the I-131 intake
= 6.0 mSv
w_T = ICRP 26 thyroid weighting factor = 0.03

With these values, the TEDE is determined:

$$H_{\text{TEDE}} = 2.5\ \text{mSv} + 0.5\ \text{mSv} + 0.03 \times 6.0\ \text{mSv} = 3.0\ \text{mSv} + 0.18\ \text{mSv}$$
$$= 3.18\ \text{mSv}$$

Question 1.162

The CDE to the lungs from the Co-60 inhalation may be determined by correcting the 1 μm CDE for particle size. The CDE to the lungs for a 1.11×10^6 Bq intake of Class Y Co-60 (1 μm) is given by the relationship:

$$H_{50}\,(1\ \mu\text{m}) = I(1\ \mu\text{m})\ k_{\text{lung}}\,(1\ \mu\text{m})$$

where

H_{50} (1 μm) = Committed dose equivalent to the lung from 1 μm particles
I = Intake of Co-60 (Class Y, 1 μm) = 1.11×10^6 Bq
k_{lung} = 3.4×10^{-7} Sv/Bq (Class Y Co-60, 1 μm)

With these values the dose from 1 μm particles is determined:

$$H_{50}\,(1\ \mu\text{m}) = (3.4 \times 10^{-7}\ \text{Sv/Bq})\,(1.11 \times 10^6\ \text{Bq}) = 0.377\ \text{Sv}$$

The dose corresponding to 10 μm particles is given by the relationship:

$$\frac{H_{50}\,(10\ \mu\text{m})}{H_{50}\,(1\ \mu\text{m})} = f_{\text{N-P}}\,\frac{D_{\text{N-P}}\,(10\ \mu\text{m})}{D_{\text{N-P}}\,(1\ \mu\text{m})} + f_{\text{T-B}}\,\frac{D_{\text{T-B}}\,(10\ \mu\text{m})}{D_{\text{T-B}}\,(1\ \mu\text{m})} + f_{\text{P}}\,\frac{D_{\text{P}}\,(10\ \mu\text{m})}{D_{\text{P}}\,(1\ \mu\text{m})}$$

where

H_{50} (10 μm)	=	CDE from 10 μm particles
H_{50} (1 μm)	=	CDE from 1 μm particles = 0.377 Sv
f_j	=	Fractional contributions of the CDE to the reference tissue from initial deposition in the *j*-th ICRP-26 lung region
j	=	Lung compartment label (nasopharyngeal passages (N-P), tracheobronchial region (T-B), and pulmonary region (P))
$f_{\text{N-P}}$ (10 μm)	=	0.0
$f_{\text{T-B}}$ (10 μm)	=	0.0
f_{P} (10 μm)	=	1.00
D_j (AMAD)	=	Fraction of the intake deposited in the *j*-th ICRP lung compartment for a given particle size (AMAD)
$D_{\text{N-P}}$ (1 μm)	=	0.30
$D_{\text{T-B}}$ (1 μm)	=	0.08
D_{P} (1 μm)	=	0.25
$D_{\text{N-P}}$ (10 μm)	=	0.87
$D_{\text{T-B}}$ (10 μm)	=	0.08
D_{P} (10 μm)	=	0.05

With these values, the dose from the 10 μm particles is determined:

$$H_{50}\,(10\ \mu\text{m}) = (0.377\ \text{Sv})\left[0\,\frac{0.87}{0.30} + 0\,\frac{0.08}{0.08} + 1.0\frac{0.05}{0.25}\right]$$
$$= 0.377\ \text{Sv}(0.2) = 0.0754\ \text{Sv}$$

Solutions for Chapter 2

Scenario 2.1

Question 2.1

NCRP-155 recommends that members of the public are limited to 1 mSv/year. For radiation exposure recommendations, members of the public include other patients, visitors to the medical facility, and facility staff not specifically trained in radiation safety.

For the photon-shielding calculation in the next question, it is reasonable to approximate 1 mSv = 100 mrem = 100 mR. Since the adjacent patient is in the hospital for one week and the therapy patient stays for at least one week, the exposure rate delivered to the adjacent patient is:

$$\dot{X}_{\text{NCRP-155}} = \frac{100 \text{ mR}}{(1 \text{ wk})\,(7 \text{ d/wk})\,(24 \text{ h/d})} = 0.6 \text{ mR/hr}$$

Question 2.2

Exposure Rate to Adjacent Patient:

$$\dot{X}_0 = m_{\text{eqRa}}\, G_{\text{Ra}}\, (1 - P)/r^2$$

where

$$G_{\text{Ra}} = 8.25 \text{ mR-ft}^2\text{/mg-hr}$$

For this application, the exposure rate to the adjacent patient, assuming no shielding in wall A, is

$$\dot{X}_0 = (70 \text{ mg})\,(8.25 \text{ mR-ft}^2\text{/hr-mg})\,(1 - 0.3)\,(1/4 \text{ ft})^2 = 25.3 \text{ mR/hr}$$

Shielding Required to Meet NCRP-155 Recommendations: The number of half-value layers (N) required to reduce $\dot{X}_0$ to the NCRP-155 value is

$$(1/2)^N = \dot{X}_{\text{NCRP-155}} / \dot{X}_0$$

where

$\dot{X}_{\text{NCRP-155}}$ = 0.6 mR/hr
$(1/2)^N$ = (0.6 mR/hr/25.3 mR/hr) = 0.024
$N \ln(1/2)$ = ln(0.024) HVL
N = 5.4 HVL

The required thickness is just N times the half-value thickness:

$$t = N\, t_{\text{HVL}}$$

Because one HVL = 0.65 cm (for Cs-137 in lead), the thickness of lead required to reduce the dose rate to the NCRP-155 value is

$$t = 5.4 \text{ HVL} \times 0.65 \text{ cm/HVL} = 3.5 \text{ cm Pb}$$

This type of problem could also be solved from barrier transmission versus shield thickness tables or curves.

Question 2.3

The linear attenuation coefficient (u) for concrete is

$$u = (0.06 \text{ cm}^2/\text{g})\,(2.4 \text{ g/cm}^3) = 0.144 \text{ cm}^{-1}$$

The linear attenuation coefficient can be related to the HVL as follows:

$$I = I_0 \exp(-u\, x)$$

If $x = 1$ HVL, then $I/I_0 = 1/2$. Solving for the HVL leads to the relationship

$$\text{HVL} = 0.693/u$$

For the values given in this problem,

$$\text{HVL} = 0.693/0.144 \text{ cm}^{-1} = 4.81 \text{ cm concrete}$$

The exposure rate for transmission through the concrete floor is solved using the exposure rate equation:

$$\dot{X}_0 = m_{\text{eqRa}}\, G_{\text{Ra}}\, (1 - P)/r^2$$
$$= (70 \text{ mg})\,(8.25 \text{ mR-ft}^2/\text{hr-mg})\,(1 - 0.3)/(7 \text{ ft})^2 = 8.25 \text{ mR/hr}$$

Using the same NCRP dose guideline for the nursery as for the adjacent patient and the half-value layer equation permits the ceiling concrete thickness to be determined:

$$
\begin{aligned}
(1/2)^N &= (0.6\ \text{mR/hr})/(8.25\ \text{mR/hr}) = 0.073 \\
N \ln(1/2) &= \ln(0.073)\ \text{HVL} \\
N &= 3.78\ \text{HVL} \times 4.81\ \text{cm concrete/HVL} = 18.2\ \text{cm} \\
&= 18.2\ \text{cm concrete (1 in./2.54 cm)} \\
&= 7.2\ \text{in. concrete}
\end{aligned}
$$

Question 2.4

Radiation protection differences between Cs-137 and I-125 brachytherapies include:

Iodine-125
1. Permanent implant.
2. Seeds inserted directly into the tissue.
3. Lower external exposure rate at the surfaced of the patient.
4. The window or thickness of the radiation instrument is critical to the measured radiation level.

Cesium-137
1. Temporary implant.
2. Afterloading devices are utilized to load the seeds.
3. Higher external exposure rate at the surface of the patient.
4. The dose rate can be detected by most gamma instrumentation.

Question 2.5

The dose equivalent (D) is given in terms of the relationship

$$D = B\,t\,S\,w_T = D_T\,w_T$$

where
w_T = thyroid weighting factor

The effective half-life is needed in order to calculate the dose equivalent to the thyroid from the uptake:

$$
\begin{aligned}
T_{\text{eff}} &= \frac{(60\ \text{days})\,(138\ \text{days})}{(60\ \text{days} + 138\ \text{days})} = 41.8\ \text{days} \times 24\ \text{hr/day} \\
&= 1004\ \text{hr}
\end{aligned}
$$

The dose to the thyroid is given by

$$D_T = B\, t\, S$$

where

B = organ burden = 300 nCi
t = mean lifetime = $T_{eff}/\ln(2)$
S = I-125 S-factor = 3×10^{-3} rad/(μCi-hr)
D_T = (1004 hr/0.693) (300 nCi × 1 μCi/1000 nCi) (3×10^{-3} rad/μCi-hr)
= 1.30 rad × (rem/rad) = 1.30 rem

The contribution to the effective dose equivalent is obtained from the thyroid weighting factor:

$$D = 0.03 \times 1.30 \text{ rad} = 39.1 \text{ mrad} \times 1 \text{ mrem/mrad} = 39.1 \text{ mrem}$$

Question 2.6

The sealed source is slowly leaking iodine, and the thyroid burden will continue to increase. This is not a statistical variation in the thyroid counting.

Scenario 2.2

Question 2.7

Because C-14 is long-lived, the contribution of the glucose and exhaled CO_2 can be handled individually. The requisite relationship for the dose to tissue T due to an activity deposition (A) is

$$D_T = (A/\lambda)\, S$$

where

λ = $0.693/T_{1/2}$

Glucose:
The half-life for glucose being metabolized into CO_2 is

$$T_{1/2} = 67 \text{ hr}$$

The dose to the liver from the glucose injection is written as

$$D_{liver} = [200\ \mu\text{Ci}/(0.693/67\ \text{hr})]\,(5.8 \times 10^{-5}\ \text{rad}/\mu\text{Ci-hr})$$
$$= 1.12 \text{ rad}$$

The contribution to the glucose to the whole-body dose equivalent is

$$D_{\text{whole body}}\,(\text{glucose}) = (0.06)\,(1.12\ \text{rad}) = 0.067\ \text{rad}$$

CO_2:
CO_2 is released from the whole body via the lungs with a half-life of

$$T_{1/2} = 1.2\ \text{hr}$$

The whole-body dose from the exhaled CO_2 is obtained by using the S value:

$$D_{\text{whole body}} = [200\ \mu\text{Ci}/(0.693/1.2\ \text{hr})]\,(1.5 \times 10^{-5}\ \text{rad}/\mu\text{Ci-hr}) = 0.005\ \text{rad}$$

Total Committed Effective Dose Equivalent:

$$\begin{aligned} D &= \text{total committed effective dose (rad)} \\ &= D_{\text{whole body}}\,(CO_2) + w_{\text{liver}}\ D_{\text{liever}}\,(\text{glucose}) \\ &= 0.005\ \text{rad} + 0.06 \times 1.12\ \text{rad} = 0.072\ \text{rad} \\ H &= 0.072\ \text{rad} \times 1\ \text{rem/rad} = 0.072\ \text{rem} \end{aligned}$$

Question 2.8

$$\dot{V} = V\,r\,k$$

where
k = conversion factor = (24 hr/day) (365 day/year) $(2.832 \times 10^4\ \text{ml/ft}^3)$
V = room volume

$$\begin{aligned} \dot{V} &= (18\ \text{ft} \times 20\ \text{ft} \times 8\ \text{ft})\,(3/\text{hr})\,(24\ \text{hr/day})\,(365\ \text{day/year}) \\ &\quad \times (2.832 \times 10^4\ \text{ml/ft}^3) \\ &= 2.14 \times 10^{12}\ \text{ml/year} \end{aligned}$$

$$\dot{A} = A\,N\,f$$

where
f = loss factor = 0.2

$$\dot{A} = (200\ \mu\text{Ci/subject})\,(20\ \text{subjects/year})\,(0.2) = 800\ \mu\text{Ci/year}$$

The average room concentration may be determined from the information derived above:

$$\overline{C} = \dot{A}/\dot{V} = (800\ \mu\text{Ci/year})/(2.14 \times 10^{12}\ \text{ml/year}) = 3.73 \times 10^{-10}\ \mu\text{Ci/ml}$$

Because this is much less than the hospital's administrative limit, no action is warranted.

Question 2.9

Based upon the volume, one could reasonably conclude that the administrative limit (AL) will not present a problem.

$$V_a = V\ N\ k = \text{air volume (ml) in the hospital}$$

where

N = number of room volumes in the total volume = 50

k = 2.832×10^4 ml/ft^3

$$V_a = (18\ \text{ft} \times 20\ \text{ft} \times 8\ \text{ft})\ (50)\ (2.832 \times 10^4\ \text{ml/ft}^3)$$
$$= 4.08 \times 10^9\ \text{ml}$$

The average room concentration may be determined from the information derived above:

$$\overline{C} = A/V = (800\ \mu\text{Ci})/(4.08 \times 10^9\ \text{ml})$$
$$= 1.96 \times 10^{-7}\ \mu\text{Ci/ml}$$

Any amount of ventilation would reduce the activity; and because it is 1/10 the AL, there is no problem. You could also conclude from the previous calculation that if the air in the room is much less than the AL, then any recirculated air would further reduce the concentration.

Question 2.10

$$A = \text{quantity of } {}^{14}CO_2 \text{ released into the room per patient}$$
$$= \int_0^t \frac{dN_{CO_2}}{dt}\,dt = \int_0^t \lambda_{CO_2}\ N_{CO_2}(t)\,dt$$

Let C replace CO_2 for simplicity and let G denote glucose:

$$\lambda_G = 0.693/67\ \text{hr} = 0.0103/\text{hr}$$
$$\lambda_C = 0.693/1.2\ \text{hr} = 0.5775/\text{hr}$$

The serial decay relationship is used to write

$$A = \int_0^t \frac{\lambda_C\ \lambda_G}{\lambda_C - \lambda_G}\ N_G(0)\ [\exp(-\lambda_G\ t) - \exp(-\lambda_C\ t)]\,dt$$

Performing the integration from $t = 0$ to $t = 6$ hr (the release period) yields

$$A = \frac{\lambda_C \lambda_G}{\lambda_C - \lambda_G} N_G(0) \times \left[\frac{1 - \exp(-\lambda_G t)}{\lambda_G} - \frac{1 - \exp(-\lambda_C t)}{\lambda_C} \right]$$

$$= \frac{(0.0103/\text{hr})\,(0.5775/\text{hr})}{0.5775/\text{hr} - 0.0103/\text{hr}} (200\ \mu\text{Ci})$$

$$\times \left\{ \frac{[1 - \exp(-0.0103/\text{hr} \times 6\ \text{hr})]}{0.0103/\text{hr}} - \frac{[1 - \exp(-0.5775/\text{hr} \times 6\ \text{hr})]}{0.5775/\text{hr}} \right\}$$

$$= 8.68\ \mu\text{Ci}$$

Scenario 2.3

Question 2.11

$$W = E\, N_v\, N_p\, k$$

where

E = 120 mA-s/view
N_v = 4 views/patient
N_p = 40 patients/week
k = conversion factor (1 min/60 s)

$$W = (120\ \text{mA-s/view})\,(4\ \text{views/patient})\,(40\ \text{patients/week})\,(1\ \text{min}/60\ \text{s})$$

$$= 320\ \text{mA-min/week}$$

Question 2.12

The correct answer is d. NCRP-49 recommends 10 mR/week to be conservative.

Question 2.13

Orientation	Number of Views per Patient	Number of Images Directed Toward Control Booth
Cephalocaudal	2	0
Mediolateral	2	1
Total	4	1

Only the mediolateral images on one side would be directed at this wall.

$$U_{CB} = V_{CB}/N_v = 1/4 = 0.25$$

The correct answer is b. Only 1 of 4 images is directed at the control booth.

Question 2.14

The correct answer is e. Based on NCRP-49, the occupancy of the control booth would be 1.00.

Question 2.15

$$\dot{X}_{CB}^{\text{unshielded}}(r) = O_{PB}\,(T_{CB})\,(U_{CB})\,(W)\,k\,(r_0/r)^2$$

where

O_{PB} = 1.0 mR/mA-s @ 100 cm (primary beam output)
T_{CB} = 0.5
U_{CB} = 0.5
W = 160 mA-min/week
k = conversion factor (60 s/min)
r = distance from X-ray source to point of interest (mediolateral position)
= 200 cm
r_0 = location of measured primary beam output (cm)

$$\begin{aligned}\dot{X}_{CB}^{\text{unshielded}}(r) &= (1.0\text{ mR/mA-s})\,(100\text{ cm}/200\text{ cm})^2\,(0.5)\,(0.5)\\ &\quad\times(160\text{ mA-min/week})\,(60\text{ s/min})\\ &= 600\text{ mR/week}\end{aligned}$$

The 600 mR/week value must be reduced to 5 mR/week. These values permit the calculation of the transmission factor (T):

$$T = \frac{\dot{X}(\text{NCRP-49})}{\dot{X}_{CB}^{\text{unshielded}}} = \frac{5\text{ mR/week}}{600\text{ mR/week}} = 8.33\times10^{-3}$$

The required shielding is obtained from the gypsum transmission curve which yields a thickness of 1.1 cm. The 30° angle reduces the required shielding thickness as illustrated in Figure S2.1.

$$\cos 30^\circ = t/1.1\text{ cm}$$
$$t = 0.866\times1.1\text{ cm} = 0.95\text{ cm}$$

which is the required thickness.

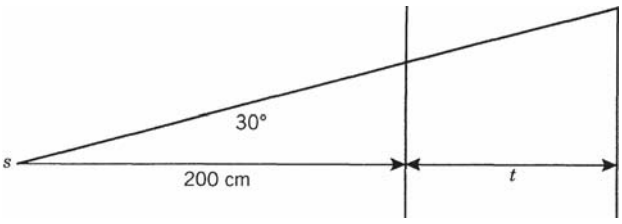


Figure S2.1 Relative location of X-ray source (*s*) and the gypsum wall of thickness *t*.

Question 2.16

$$\dot{X}^{\text{scatt}}_{\text{DO, unshielded}}(r) = O_{\text{PB}}(W)\, k\, f_{\text{scatter}}(\text{OF}_{\text{DO}})\,(r_0/r)^2$$

= exposure rate in the doctor's office without gypsum board addition (mR/week) due to the scattered radiation

where

O_{PB} = 1.0 mR/mA-s @ 100 cm
W = 160 mA-min/week
k = (60 s/min)
r = distance from X-ray source to point of interest (doctor's office) = 150 cm
r_0 = 100 cm = location of the primary beam measurement
f_{scatter} = 0.0003
OF_{DO} = 0.6

$$\begin{aligned}\dot{X}^{\text{scatt}}_{\text{DO, unshielded}} &= (1.0\ \text{mR/mA-s})\,(100\ \text{cm}/150\ \text{cm})^2 \\ &\quad \times (0.0003)\,(160\ \text{mA-min/week})\,(60\ \text{s/min}) \\ &\quad \times (0.6)\,(1.0\ \text{m}/0.5\ \text{m})^2 \\ &= 3.07\ \text{mR/week}\end{aligned}$$

Because this is less than the 5-mR/week criterion, no shielding is needed.

Question 2.17

$$\dot{X}^{\text{unshielded}}_{\text{DO, leakage}}(r) = (\dot{X}_{\text{L}}(r_0)/i)\,(W)\,k\,(T_{\text{DO}})\,(r_0/r)^2$$

where

$\dot{X}_{\text{L}}(r_0)$ = leakage dose rate = 100 mR/hr at 100 cm (r_0)
r_0 = location of leakage measurement
i = 7 mA
W = workload = 640 mA-min/week
k = conversion factor (1 hr/60 min)
r = distance from X-ray source to point of interest (doctor's office) is 200 cm
T_{DO} = occupancy factor for the doctor's office = 0.5

$$\begin{aligned}\dot{X}^{\text{unshielded}}_{\text{DO, leakage}} &= (100\ \text{mR}/7\ \text{mA-hr})\,(100\ \text{cm}/200\ \text{cm})^2 \\ &\quad \times (640\ \text{mA-min/week})\,(1\ \text{hr}/60\ \text{min})\,(0.5) \\ &= 19.0\ \text{mR/week}\end{aligned}$$

Because this is more than the 5-mR/week design criterion, gypsum board shielding is needed. The required thickness can be calculated from the following relationship:

$$\dot{X}^{\text{shielded}}_{\text{DO, leakage}} = \dot{X}^{\text{unshielded}}_{\text{DO, leakage}} \exp(-0.693\, t/\text{HVL})$$
$$= \text{design exposure rate (mR/week)} = 5 \text{ mR/week}$$

where

HVL = half-value layer = 0.5 cm for 30 kVp
t = shield thickness required to reduce the unshielded exposure rate to the criterion (5 mR/week)
5 mR/week = (19.0 mR/week) exp (–0.693 t/0.5 cm)

Solving for t yields

$$\ln(5/19.0) = -0.693\, t/0.5 \text{ cm}$$
$$t = -(0.5 \text{ cm}/0.693) \ln(5/19.0)$$
$$= 0.96 \text{ cm of gypsum board}$$

Scenario 2.4

Question 2.18

General Formulation:

a = Mo-99, $T_{1/2}$ = 66 hr
b = Tc-99m, $T_{1/2}$ = 6 hr

$$a \xrightarrow{\lambda_a} b \xrightarrow{\lambda_b} c$$

$$N_b = \frac{\lambda_a\, N_{a_0}}{\lambda_b - \lambda_a} [\exp(-\lambda_a\, t) - \exp(-\lambda_b\, t)]$$
$$A_b = \frac{\lambda_b\, A_{a_0}}{\lambda_b - \lambda_a} [\exp(-\lambda_a\, t) - \exp(-\lambda_b\, t)]$$

where

N_i = number of atoms of isotope i
A_i = activity of isotope i

Using the relationship, $\lambda_i = \ln(2)/T^i_{1/2}$, the activity equation can be written in terms of the half-lives:

$$A(\text{Tc-99m}) = A(\text{Mo-99})\,[T(\text{Mo-99})/(T(\text{Mo-99}) - T(\text{Tc-99m}))]$$
$$\times [\exp(-0.693\, t/T(\text{Mo-99})) - \exp(-0.693\, t/T(\text{Tc-99m}))]$$
$$= 1000 \text{ mCi}\,[66/(66-6)]$$
$$\times [\exp(-0.693 \times 24/66) - \exp(-0.693 \times 24/6)]$$
$$= 1000\,(1.1)\,(0.7772 - 0.0625) \text{ mCi} = 786 \text{ mCi}$$

Question 2.19

The specific activity C ($t = 8$ hr) at 1600 hours (4:00 p.m.) is

$$C(t_2) = C(t_1)\,[\exp - 0.693\, t/T\,(\text{Tc-99m})]$$
$$C(1600 \text{ hours}) = 80 \text{ mCi/cm}^3\ [\exp(-0.693 \times 8/6)] = 31.75 \text{ mCi/cm}^3$$

V_a = volume of Tc-99m to be added (cm^3)
V_l = volume of sulfur colloid liquid with the kit = 5 cm^3
(*Note:* This is not radioactive material.)
C = concentration at 1600 (mCi/cm^3)
C_d = desired concentration = 10 mCi/cm^3

The volume of added material is obtained from an activity balance prior to and following the addition of the sulfur colloid liquid:

$$A = V_a\, C = (V_a + V_l)\, C_d$$

which may be solved for the added volume:

$$V_a\, C = V_a\, C_d + V_l\, C_d$$
$$V_a\,(C - C_d) = V_l\, C_d$$
$$V_a = \frac{V_l\, C_d}{C - C_d} = \frac{(5 \text{ cm}^3)\,(10 \text{ mCi/cm}^3)}{31.75 \text{ mCi/cm}^3 - 10 \text{ mCi/cm}^3} = 2.3 \text{ cm}^3$$

Question 2.20

$$\dot{X} = A\, G \exp(-u\, x)/r^2$$
$$= \text{exposure rate (mR/hr)}$$

where
G = gamma constant = 0.56 R-cm^2/mCi-hr
u = attenuation coefficient = 3.25/cm
A = 50 mCi
r = 50 cm
x = 0.5 cm

Buildup is ignored; that is, assume $B = 1$:

$$\dot{X} = \frac{(50 \text{ mCi})\,(0.56 \text{ R-cm}^2/\text{mCi-hr})}{(50 \text{ cm})^2} \times \exp(-3.25/\text{cm} \times 0.5 \text{ cm})\,(1000 \text{ mR/R})$$
$$= 2.2 \text{ mR/hr}$$

Question 2.21

The remaining activity of Mo-99 is determined from

$$\begin{aligned} A(t) &= A_0 \exp[-0.693 \times t/T_{1/2}\,(\text{Mo-99})] \\ &= (1000\ \text{mCi}) \exp\{-[0.693/66\ \text{hr})\ (90\ \text{days} \times 24\ \text{hr/day})]\} \\ &= 1.41 \times 10^{-7}\ \text{mCi} \times 1\ \text{Ci}/10^3\ \text{mCi} \times 10^{12}\ \text{pCi/Ci} = 141\ \text{pCi} \end{aligned}$$

This activity may also be expressed in dpm:

$$\begin{aligned} A(t) &= 1.41 \times 10^{-7}\ \text{mCi} \times 1\ \text{Ci}/10^3\ \text{mCi} \times 3.7 \times 10^{10}\ \text{dis/s-Ci} \\ &\quad \times 60\ \text{s/min} \\ &= 314\ \text{dpm} \end{aligned}$$

Question 2.22

Because 3700 dpm is much larger than the remaining activity from Mo-99 (314 dpm), there must be another source. This could be from long-lived radiochemical impurities from the original Mo-99 production. Tc-99 is probably not the source of the additional counts unless the generator column is fractured.

Assuming that Ru-106 ($T_{1/2}$ = 372 days) has been seen in old generator columns at a fraction of microcurie levels, it is a likely candidate for explaining the 3700-dpm value. One should also evaluate that Tc-99 is the source of the contamination.

Tc-99 ($T_{1/2}$ = 213 000 years) is also present in the column. Its weak beta (293 keV) should not penetrate the glass wall of the column. It is also unlikely that there would be enough Tc-99 contamination on the outside of the column to account for the 3700-dpm, or enough bremsstrahlung from the inside to explain the 3700-dpm value.

Scenario 2.5

Question 2.23

The beamline exposure rate $(\dot{X})$ is given by

$$\dot{X} = k \frac{S\,E}{A\,W} (u_{\text{en}}/p)_{\text{air}}$$

where

S = photon source strength = 4.0×10^{11} photons/s
E = photon energy = 30 keV/photon
A = beam area = 0.05 cm × 12.3 cm
W = 34 eV/ion pair

u_{en}/p = energy absorption coefficient = 0.15 cm^2/g-air
k = conversion factor

$$\begin{aligned}\dot{X} &= \frac{(4.0 \times 10^{11} \text{ photons/s})\,(30 \text{ keV/photon})}{(0.05 \text{ cm} \times 12.3 \text{ cm})\,(34.0 \text{ eV/ip})} \\ &\quad \times (0.15 \text{ cm}^2\text{/g-air})\,(1000 \text{ eV/keV})\,(1.6 \times 10^{-19} \text{ coulomb/ip}) \\ &\quad \times (1000 \text{ g/kg})\,(1 \text{ R-kg-air}/2.58 \times 10^{-4} \text{ coulomb}) \\ &= 53.4 \text{ R/s}\end{aligned}$$

Question 2.24

For simplicity assume no scattered radiation enters the beamline. This assumption permits the attenuation relationship to be utilized:

$$\dot{X}(t) = \dot{X}_0 \exp(-u\,t)$$

where
$\dot{X}(t)$ = 2.0 mR/hr (by design)
$\dot{X}_0$ = 53.4 R/s (exposure rate in beamline)
u = $(u/p)\,(p)$ = (1.12 cm^2/g) (2.7 g/cm^3) = 3.02/cm

Solving for the desired thickness (t) yields

$$\begin{aligned}t &= -(1/u)\ln[\dot{X}(t)/\dot{X}_0] \\ &= -(1/3.02/\text{cm})\ln[(2.0 \text{ mR/hr} \times 1 \text{ hr}/3600 \text{ s})/53.4 \text{ R/s} \times 1000 \text{ mR/R}] \\ &= 6.08 \text{ cm}\end{aligned}$$

Question 2.25

For simplicity assume that the exposure rate is constant across the area of the beam.

The time required for the beam to move past any point is

$$t = \frac{\text{beam width}}{\text{traveling rate}} = \frac{0.5 \text{ mm}}{60 \text{ mm/s}} = 8.33 \times 10^{-3} \text{ s}$$

The exposure is just the exposure rate times the time:

$$X = \dot{X}\,t = (53.4 \text{ R/s})\,(8.33 \times 10^{-3} \text{ s}) = 0.445 \text{ R}$$

Question 2.26

The transmission ionization chamber will not provide a good estimate of the surface dose. This ion chamber would measure the exposure rate over the area

of the beam. Because the patient is moving relative to the beam, the exposure is delivered over a larger area than the beam area. Therefore, the average exposure rate delivered over the entire area irradiated is less than the beamline exposure rate. The two rates are the same only if the patient is stationary.

Question 2.27

The beamline exposure rate and the vertical velocity of the patient must be monitored to ensure that the patient exposure is maintained below the 1.0-R criterion.

Scenario 2.6

Question 2.28

X = exposure per view at 40 in.
S = 4 mR/mA-s for 70 kVp at 40 in.
I = time/current = 25 mA-s/view

$$X = S\,I = (4\ \text{mR/mA-s})\,(25\ \text{mA-s/view}) = 100\ \text{mR/view}$$

Assuming that the thickness of the child is about 4 in, the exposure at the surface of the child (X_0) is

$$X_0 = X\,(d/d_0)^2$$

where
d = 40 in. = source image distance
d_0 = distance to surface of child = 40 in. – 4 in. = 36 in.

$$X_0 = 100\ \text{mR/view}\,\frac{(40\ \text{in.})^2}{(40\ \text{in.} - 4\ \text{in.})^2} = 123\ \text{mR/view}$$

In order to estimate the abdomen dose, the distance between the primary beam and the woman's abdomen (d_1) must be determined. From the problem statement, this distance is 18 in. The abdomen skin dose (X_1) may be written in terms of these parameters:

$$X_1 = X_0\ f_{\text{scatter}}\ N\,(d_0/d_1)^2$$

where
f_{scatter} = scattered radiation intensity from the primary beam at 1 m normalized to the primary beam intensity
= 0.001

N = number of views = 5 views
X_1 = (123 mR/view) (0.001) (5 views) (39.37 in./18 in.)^2
= 2.94 mR

Question 2.29

In order to estimate the fetal exposure, recall that the woman was wearing a lead apron. Assuming the lead apron is 0.5 mm equivalent lead, it would lead to a transmission factor (TF) of 10%. A depth dose factor (DDF) of about 30% would also be a reasonable assignment.

The fetal exposure (X_F) can be written in terms of these factors

$$X_F = X_1 \text{ (DDF) (TF)} = (2.46 \text{ mR}) (0.3) (0.1) = 0.074 \text{ mR}$$

Question 2.30

NRC and NCRP guidance is to limit fetal exposure to 500 mR or less during its gestation.

Question 2.31

The exposure calculations, assumptions, and relevant regulatory guidance should be presented to the attending physician. Dose effects from fetal exposures should also be discussed with the physician. The final responsibility to provide consultation with the patient resides with the physician.

Question 2.32

The practice of excluding the mother appears to be overly conservative when the risk to the fetus is compared to the benefit to the child being X-rayed. The use of mechanical restraints is an option that should also be considered.

Scenario 2.7

Question 2.33

I-131 poses both an internal and external radiation hazard due to its associated beta and gamma radiation. From an internal viewpoint, each microcurie taken up by the thyroid leads to a dose commitment of about 6.6 rem. The gamma constant for I-131 is 2.2 R hr^{-1} mCi^{-1} cm^2, which indicates that gamma exposure must be evaluated. The short radiological half-life of 8 days suggests that the hazards will be relatively short-lived, but still long enough that good radiological controls practices are warranted.

Question 2.34

The exposure rate a distance r from a point source of activity A is

$$\dot{X}(r) = A\,G/r^2$$

$$\dot{X}(\text{contact}) = (200\text{ mCi})\left(2.2\,\frac{\text{R-cm}^2}{\text{hr-mCi}}\right)\Big/(1.0\text{ cm})^2 = 440\text{ R/hr}$$

The contact exposure rate suggests that source shielding and good exposure control practices are warranted.

$$\dot{X}(1.0\text{ m}) = (200\text{ mCi})\left(2.2\,\frac{\text{R-cm}^2}{\text{hr-mCi}}\right)\Big/(100\text{ cm})^2$$

$$= 0.044\text{ R/hr} \times 1000\text{ mR/R} = 44\text{ mR/hr}$$

Exposure control for hospital personnel and other patients should be considered following this administration because fields around the patient will be on the order of 10–100 mR/hr.

Question 2.35

The following health physics practices are recommended:

1. *Minimize Airborne Radioactivity.* Administration of the I-131 would be in a basic pH solution to minimize the evolution of airborne I-131. Good room ventilation is recommended to minimize the airborne concentration of any I-131 that volatilizes.
2. *Minimize Radiation Exposures.* It is recommended that the 200-mCi solution be contained in a shielded vial to minimize radiation exposures to the technician administering the I-131. The use of bedside shields should also be considered.

   Family members and visitors should be instructed to maintain their distance from the patient (10 feet) during their visit to minimize the radiation exposure. Similar guidance will be provided to the hospital staff.

   Following the oral administration of I-131 to the patient, fluids are to be administered to rinse the mouth and esophagus. This action minimizes exposure to these organs. Intravenous injection would eliminate the exposures to the mouth and esophagus.
3. *Implement Contamination Control Practices.* The technician wears protective clothing and gloves during the administration process and during the subsequent cleanup. The room setup prior to the administration should minimize the contamination of the patient's clothing, bedding, and room, and any patient excretion including vomiting should be easily controlled. The collection of bedding should also include monitoring for contamination resulting from patient sweating.

4. *Posting of Radiological Areas*. The room must be properly posted and controlled to ensure that exposures are minimized and all applicable standards are met.
5. *Periodic Surveys*. Radiation and contamination surveys should be performed to properly assess the radiological conditions in the room. This information will also impact access to the patient.
6. *Post-administration Cleanup and Waste Disposal*. Post-administration follow-up should properly control any waste material. Good contamination control practices and dose control methods are to be utilized.
   All urine will be collected and stored in waste containers in a waste storage area to permit decay prior to disposal in the sanitary waste. This procedure also minimizes the exposure to downstream populations after the waste water is recycled for public use. *Note:* There is no requirement to store urine, and it may be directly discharged.

Scenario 2.8

Question 2.36

Internal dose can be calculated using either the ICRP or MIRD methodologies. The basic equations for internal dose within the MIRD and ICRP methodologies are:

$$\bar{D} = \tilde{A}\, S = (I/\lambda_{\text{eff}})\, S$$

$$H_{50,T} = 1.6 \times 10^{-10}\, \frac{\text{Sv-g}}{\text{MeV}}\, U_S\ \text{SEE}$$

$$H_E = \sum_T w_T\, H_{50,T}$$

where

$\bar{D}$	=	mean absorbed dose
$\tilde{A}$	=	total cumulated activity
S	=	mean dose per unit cumulated activity
$H_{50,T}$	=	50 year committed dose equivalent
U_S	=	number of transformations in the source organ over 50 years
SEE	=	specific effective energy
H_E	=	committed effective dose equivalent
T	=	tissue/organ label
w_T	=	organ weighting factor

The constant 1.6×10^{-10} is the product of conversion factors 1.6×10^{-13} J/MeV and 1000 g/kg and its units include the definition that a Sv is equivalent to a J/kg.

Three major factors are used to determine the committed effective dose equivalent (CEDE). These factors are: (1) the activity (I) injected into the patient; (2) the total effective removal rate constant (λ_{eff}) in the body, which is used to calculate U_S or $\tilde{A}$; and (3) the specific effective energy or mean dose per unit cumulated

activity, which is the energy in MeV absorbed per gram of the applicable tissue or organ per transformation within the source organ.

Question 2.37

The relative contribution (R) is calculated for each of the three factors (F) listed in Question 2.36:

$$R_i = \frac{F_i\,(\text{Tl-201})}{F_i\,(\text{Tc-99m})}$$

Factor 1 (Intake or Quantity Injected):

$$R_1 = \frac{I\,(\text{Tl-201, Injected})}{I\,(\text{Tc-99m, Injected})} = \frac{148\ \text{MBq}}{370\ \text{MBq} + 740\ \text{MBq}} = \frac{148\ \text{MBq}}{1110\ \text{MBq}} = 0.133$$

Factor 2 (Total Effective Removal Rate Constant):
The total effective removal rate constant (λ_{eff}) is defined as:

$$\lambda_{\text{eff}} = \lambda + \lambda_{\text{b}}$$

where
λ = physical disintegration constant
λ_{b} = biological removal rate

$$\lambda_{\text{b}} = \frac{\ln 2}{T_{\text{b}}} = \left(\frac{0.693}{10\ \text{d}}\right) = 0.0693/\text{d}$$

$$\lambda\,(\text{Tl-201}) = \frac{\ln 2}{T_{1/2}\,(\text{Tl-201})} = \left(\frac{0.693}{73\ \text{h}}\right)\left(\frac{24\ \text{h}}{\text{d}}\right) = 0.228/\text{d}$$

$$\lambda\,(\text{Tc-99m}) = \frac{\ln 2}{T_{1/2}\,(\text{Tc-99m})} = \left(\frac{0.693}{6\ \text{h}}\right)\left(\frac{24\ \text{h}}{\text{d}}\right) = 2.77/\text{d}$$

$$R_2 = \frac{\lambda_{\text{eff}}\,(\text{Tl-201})}{\lambda_{\text{eff}}\,(\text{Tc-99m})} = \frac{(0.0693/\text{d} + 0.228/\text{d})}{(0.0693/\text{d} + 2.77/\text{d})} = \frac{0.297/\text{d}}{2.84/\text{d}} = 0.105$$

Factor 3 (SEE):
The specific effective energy is the energy absorbed per unit mass in the affected organ/tissue per transformation:

$$\text{SEE} = \frac{\sum_i E_i\ Y_i\ \text{AF}_i}{m}$$

where

E_i = energy of the i^{th} photon emitted in the heart profusion study (given in the problem statement)

Y_i = yield of the i^{th} photon emitted in the heart profusion study (given in the problem statement)

AF_i = absorbed fraction of the i^{th} photon emitted in the heart profusion study (given in the problem statement)

m = mass of the affected organ = whole body = 70 000 g

Using these values, the SEE values are determined:

$$\text{SEE(Tl-201)} = \frac{(68.9\text{ keV})(0.27)(0.47) + (70.8\text{ keV})(0.465)(0.47) + (80.3\text{ keV})(0.205)(0.47)}{7.0 \times 10^4}$$

$$= \frac{8.74 + 15.5 + 7.74}{7.0 \times 10^4\text{ g}}\text{ keV/g} = 4.57 \times 10^{-4}\text{ keV/g}$$

$$\text{SEE(Tc-99m)} = \frac{(140.5\text{ keV})(0.891)(0.36)}{7.0 \times 10^4\text{ g}} = 6.44 \times 10^{-4}\text{ keV/g}$$

$$R_3 = \frac{\text{SEE(Tl-201)}}{\text{SEE(Tc-99m)}} = \frac{4.57 \times 10^{-4}\text{ keV/g}}{6.44 \times 10^{-4}\text{ keV/g}} = 0.710$$

The dose relationship (D) composed of the individual R values should be approximately unity to verify the premise that ten times more Tetrolite can be administered and still yield the same dose as a unit thallium injection. In terms of dose, the three factors have the relationship:

$$D \approx \frac{I}{\lambda_{\text{eff}}}\text{SEE}$$

The ratio (ξ) of Tl-201 and Tc-99m doses is:

$$\xi = \frac{D\text{(Tl-201)}}{D\text{(Tc-99m)}} = \frac{\dfrac{I\text{(Tl-201)}}{\lambda_{\text{eff}}\text{(Tl-201)}}\text{SEE(Tl-201)}}{\dfrac{I\text{(Tc-99m)}}{\lambda_{\text{eff}}\text{(Tc-99m)}}\text{SEE(Tc-99m)}}$$

$$= \frac{\dfrac{I\text{(Tl-201)}}{I\text{(Tc-99m)}}\dfrac{\text{SEE(Tl-201)}}{\text{SEE(Tc-99m)}}}{\dfrac{\lambda_{\text{eff}}\text{(Tl-201)}}{\lambda_{\text{eff}}\text{(Tc-99m)}}} = \frac{R_1\,R_3}{R_2}$$

$$= \frac{R_1\,R_3}{R_2} = \frac{(0.133)(0.710)}{(0.105)} = 0.899 \approx 1$$

Question 2.38

If the biological half-life (T_b) of Tc-99m-Tetrolite in the gall bladder is 7 days and the physical half-life (T_p) is 6 hours, the effective half-life (T_e) for Tc-99m in the gall bladder is:

$$T_e = \frac{T_p\ T_b}{T_p + T_b} = \frac{(6\text{ h})\,(1\text{ d}/24\text{ h})\,(7\text{ d})}{(6\text{ h})\,(1\text{ d}/24\text{ h}) + (7\text{ d})} = 0.241\text{ d}$$

Question 2.39

The variability (V) is defined as the ratio of the standard deviation (σ) and the number of counts (N):

$$V = \sigma/N \quad \text{and} \quad \sigma = \sqrt{N}$$
$$= \sqrt{N}/N = 1/\sqrt{N}$$

The ratio of count rates (CR) is:

$$\text{CR(Tc-99m)}/\text{CR(Tl-201)} = 20$$

Using this ratio, the relative variability is determined from the individual variability values:

$$V\text{(Tl-201)} = 1/\sqrt{N}$$
$$V\text{(Tc-99m)} = 1/\sqrt{20\,N}$$

Therefore, the relative variability (r) in the number of counts is:

$$r = \frac{V\text{(Tl-201)}}{V\text{(Tc-99m)}} = \frac{1/\sqrt{N}}{1/\sqrt{20N}} = \sqrt{20} = 4.47$$

Question 2.40

Assuming that the thallium and Tetrolite is administered in identical syringes and that the source geometries are identical, the extremity doses are given by the relationship:

$$D = \left[\frac{S}{4\,\pi\,r^2}\sum_i E_i\ Y_i \left(\frac{\mu_{en}}{\rho}\right)_i\right] t$$

where

S = source strength or activity

r = distance from the source
μ_{en}/ρ = energy absorption coefficient
t = exposure time

The distance is the same in both cases. Assuming that the tissue energy absorption coefficients are proportional to the air absorption coefficients provided in the problem statement, the ratio of doses is:

$$\frac{D(\text{Tl-201})}{D(\text{Tc-99m})} = \frac{\left[\sum_{i=1}^{3} E_i^{\text{Tl-201}}\ Y_i^{\text{Tl-201}} \left(\frac{\mu_{en}}{\rho}\right)_i^{\text{Tl-201}}\right] S_{\text{Tl-201}}\ t_{\text{Tl-201}}}{E^{\text{Tc-99m}}\ Y^{\text{Tc-99m}} \left(\frac{\mu_{en}}{\rho}\right)^{\text{Tc-99m}} \sum_{i=1}^{2} S_i^{\text{Tc-99m}}\ t_i^{\text{Tc-99m}}}$$

$$= \frac{\left[\begin{array}{l}(68.9\ \text{keV})\ (0.27)\ (0.0262\ \text{cm}^2/\text{g}) \\ +\ (70.8\ \text{keV})\ (0.465)\ (0.0262\ \text{cm}^2/\text{g}) \\ +\ (80.3\ \text{keV})\ (0.205)\ (0.0236\ \text{cm}^2/\text{g})\end{array}\right]}{(140.5\ \text{keV})\ (0.891)\ (0.0245\ \text{cm}^2/\text{g})}$$

$$\times \frac{(148\ \text{MBq})\ (1\ \text{min})}{(370\ \text{MBq})\ (1\ \text{min}) + (740\ \text{MBq})\ (1\ \text{min})}$$

$$= \left(\frac{0.487 + 0.863 + 0.388}{3.07}\right)\left(\frac{148}{1110}\right) = 0.0755$$

Scenario 2.9

Question 2.41: d

The NCRP-147 shielding design goal for controlled areas is 0.1 mGy/wk.

Question 2.42: b

The NCRP-147 shielding design goal for uncontrolled areas is 0.02 mGy/wk.

Question 2.43

The unshielded primary air kerma [$K_P(0)$] is determined from the relationship:

$$K_P(0) = \frac{K_P^1\ U\ N}{d_P^2}$$

where

K_P^1 = unshielded primary air kerma per patient from Table 3
= 1.2 mGy-m^2/patient
U = use factor for the primary barrier = 1
N = number of patients examined per week = 300 patients/wk
d_P = distance from the source to the maximally exposed individual beyond the primary barrier = 3 m

$$K_P(0) = \frac{(1.2 \text{ mGy-m}^2/\text{patients})\,(1)\,(300 \text{ patients/wk})}{(3 \text{ m})^2} = 40 \text{ mGy/wk}$$

Question 2.44

The transmission required for the primary barrier to meet the shielding design goal for uncontrolled areas is obtained from the result of question 2.43 and the following relationship:

$$B_P(x_{barrier} + x_{pre}) = \left(\frac{P}{T}\right)\frac{d_P^2}{K_P^1\,U\,N} = \frac{\left(\frac{P}{T}\right)}{K_P(0)}$$

where

P = NCRP-147 shielding design goal for uncontrolled areas = 0.02 mGy/wk
T = occupancy factor = 1.0

Using these values, the transmission is:

$$B_P(x_{barrier} + x_{pre}) = \frac{\frac{0.02 \text{ mGy/wk}}{1.0}}{40 \text{ mGy/wk}} = 5 \times 10^{-4}$$

Question 2.45

The thickness of lead in the primary barrier ($X_{barrier}$) required to achieve the transmission value of question 2.44 is determined from the relationship:

$$x_{barrier} = \frac{1}{\alpha\,\gamma}\ln\left[\frac{\left(\frac{N\,T\,U\,K_P^1}{P\,d_P^2}\right)^{\gamma} + \frac{\beta}{\alpha}}{1 + \frac{\beta}{\alpha}}\right] - x_{pre}$$

where

α = fitting parameter for the chest room from problem Table 1
= 2.283 mm^{-1}
β = fitting parameter for the chest room from the problem's first data table
= 10.74 mm^{-1}

γ = fitting parameter for the chest room from the problem's first data table
= 0.637
N = number of patients examined per week = 300 patients/wk
U = use factor for the primary barrier = 1
T = occupancy factor for the primary barrier = 1
P = shielding design goal for uncontrolled areas = 0.02 mGy/wk
K_P^1 = unshielded primary air kerma per patient from the problem's third data table
= 1.2 mGy-m^2/patient
x_{pre} = Primary beam preshielding from the problem's fourth data table
= 0.85 mm

Using these values permits the primary barrier thickness ($x_{barrier}$) to be determined:

$$x_{barrier} = \frac{1}{(2.283/\text{mm})\,(0.637)} \times \ln\left\{\frac{\left[\frac{(300\ \text{patients/wk})\,(1)\,(1)\,(1.2\ \text{m}^2\text{-mGy/patient})}{(0.02\ \text{mGy/wk})\,(3\ \text{m})^2}\right]^{0.637} + \frac{10.74/\text{mm}}{2.283/\text{mm}}}{1 + \frac{10.74/\text{mm}}{2.283/\text{mm}}}\right\} - 0.85\ \text{mm}$$

$$= 2.16\ \text{mm} - 0.85\ \text{mm} = 1.31\ \text{mm}$$

Question 2.46

The weekly, unshielded secondary air kerma from secondary (leakage and scatter) radiation [$K_{sec}(0)$] at a distance d_{sec} for N patients is:

$$K_{sec}(0) = \frac{K_{sec}^1\ N}{d_{sec}^2}$$

where
K_{sec}^1 = unshielded secondary air kerma which includes the leakage and scatter contributions from the problem's second data table
= 2.7×10^{-3} mGy-m^2/patient
d_{sec} = distance from the source to the maximally exposed individual beyond the secondary barrier (m) = 2.1 m
N = number of patients examined per week = 300 patients/wk

Using these values, leads to the desired unshielded secondary air kerma:

$$K_{sec}(0) = \frac{(2.7 \times 10^{-3}\ \text{mGy-m}^2/\text{patients})\,(300\ \text{patients/wk})}{(2.1\ \text{m})^2} = 0.18\ \text{mGy/wk}$$

Question 2.47

The transmission required for the secondary barrier to meet the shielding design goal for uncontrolled areas is obtained from the result of question 2.46 and the following relationship:

$$B_{\text{sec}}(x_{\text{barrier}}) = \left(\frac{P}{T}\right)\frac{d_{\text{sec}}^2}{K_{\text{sec}}^1\ N} = \frac{\left(\frac{P}{T}\right)}{K_{\text{sec}}(0)}$$

where

P = NCRP-147 shielding design goal for uncontrolled areas = 0.02 mGy/wk
T = occupancy factor = 1.0

Using these values, the transmission is:

$$B_{\text{sec}}(x_{\text{barrier}}) = \frac{0.02\ \text{mGy/wk}}{0.18\ \text{mGy/wk}} = 0.11$$

Question 2.48: d

A qualified expert is an individual certified by one of the following organizations: American Board of Radiology, American Board of Medical Physics, American Board of Health Physics, or Canadian College of Physicists in Medicine. A qualified expert is a medical physicist or medical health physicist who is competent to design radiation shielding in medical X-ray imaging facilities.

Scenario 2.10

Question 2.49

Since the patient's thyroid behaves in an abnormal manner, the absorbed dose is obtained by integrating the dose rate equation:

$$\dot{D}(t) = \frac{q(0)\ e}{m}\ e^{-\lambda_e t}$$

$$D = \int_0^T \dot{D}(t)\ \mathrm{d}t = \int_0^T \frac{q(0)\ e}{m}\ e^{-\lambda_e t}\ \mathrm{d}t = \frac{q(0)\ e}{m\ \lambda_e}\left(1 - e^{-\lambda_e T}\right)$$

where

D = dose over first year
T = period of dose evaluation = 1 y

$q(0)$ = initial activity in the thyroid = $f\,I$
f = fraction of the intake in the thyroid = 0.30
I = Intake of I-131 = 3.7×10^3 MBq
$q(0)$ = (0.3) (3.7×10^3 MBq) = 1.11×10^3 MBq
e = average beta energy deposited in the thyroid = 0.190 MeV/dis
m = thyroid mass = 20 g
λ_e = effective disintegration constant = $0.693/T$
T_e = effective half-life = $\dfrac{T_p\,T_b}{T_p + T_b}$
T_p = physical half-life = 8.05 d
T_b = biological half-life = 90 d

$$T_e = \frac{T_p\,T_b}{T_p + T_b} = \frac{(8.05\text{ d})\,(90\text{ d})}{(8.05\text{ d} + 90\text{ d})} = 7.39\text{ d}$$

$$\lambda_e = \frac{\ln 2}{T_e} = \frac{0.693}{7.39\text{ d}} = 0.0938/\text{d}$$

These values uniquely determine the absorbed dose:

$$D = \frac{q(0)\,e}{m\,\lambda_e}\left(1 - e^{-\lambda_e T}\right)$$

$$= \frac{\left(\dfrac{3600\text{ s}}{\text{hr}}\right)\left(\dfrac{24\text{ hr}}{\text{d}}\right)(1.11 \times 10^3\text{ MBq})\,(10^6\text{ Bq/MBq})\,(\text{dis/Bq-s})\,(0.190\text{ MeV/dis})}{(20g)(0.0938/d)(1kg/1000g)}$$

$$\times (1.6 \times 10^{-13}\text{ J/MeV})\left(\frac{\text{Gy-kg}}{\text{J}}\right)\left[1 - e^{-(0.0938/\text{d})\,(1\text{ y})\,(365\text{ d/y})}\right]$$

$$= 1.55 \times 10^3\text{ Gy}\,[1 - 0] = 1.55 \times 10^3\text{ Gy}$$

Question 2.50

Radiation protection concerns associated with patient room preparation by the hospital radiation protection staff include:

1. Contamination concerns:
   - Control of patient sweat. Plastic bedding and protective clothing for hospital personnel are warranted.
   - Control of patient vomit. The room should be prepared to contain any vomit and the staff should be prepared for this possibility.
   - Control of patient's urine. Patients with bladder control problems require additional precautions. Again, plastic bedding is appropriate.

2. Direct dose concerns:
   – Exposure to adjacent patients. Minimum distances or bedside shields should be considered.
   – Exposure to hospital staff. Bed shields should be utilized. The I-131 source should be shielded to minimize dose to the medical technicians.
   – Exposure to hospital visitors. Strict room access or distance controls are appropriate. Postings will also be required.

3. Patient dose concerns:
   – Oral administration leads to stray dose to the mouth, throat, and esophagus. This dose should be minimized by rinsing these organs with water to remove the I-131 source.
   – An alternative administration method using an iodine capsule would save the dose associated with a liquid administration.

4. Internal Dose concerns:
   – Internal intakes to visitors, family, and friends.
   – Internal intakes by hospital staff.

Question 2.51

Regarding patient room preparation by the hospital radiation protection staff, specific measures that could be taken in room preparation include:

1. Using plastic bedding to control contamination.
2. Using plastic or washable coverings to control contamination within the room.
3. Using bed shields to minimize exposure to staff and visitors.
4. Using a shielded vial/transfer container to minimize exposures prior to administering the I-131.
5. Posting the room following hospital and regulatory guidelines. Posting a sign on the treatment room door informs staff and visitors that the room contains a patient receiving radiation therapy and that radioactive materials are present.
6. Ensuring all room ventilation is properly functioning.
7. Staging protective clothing within the room for the staff and patient.
8. Staging appropriate waste containers within the room. These containers should be properly labeled with instructions for the collection and processing of all patient excreta.
9. Staging clothing collection containers within the room with instructions for the collection and processing of bedding and clothing worn by the patient.
10. Installing barriers (e.g., yellow-magenta rope) at a specified distance from the patient with signs providing appropriate instructions to visitors and staff. Visitor access must be controlled to ensure postings are fully understood and that compliance with posted requirements is achieved.

Question 2.52

Radiation protection measures/controls that should be implemented for the protection of the hospital staff include:

1. Using bed shields to minimize exposure to the hospital staff.
2. Using a shielded vial/transfer container to minimize exposures to the technician and staff.
3. Posting the room following hospital and regulatory guidelines to minimize inadvertent entry and associated exposure to hospital staff.
4. Ensuring all room ventilation is properly functioning to minimize internal intakes.
5. Staging protective clothing within the room for the staff to minimize contamination and internal uptakes.
6. Providing staff training to ensure the proper care of radiation therapy patients. This training should include control of radioactive materials, characteristics of medical isotopes, transport pathways, and ALARA measures to minimize doses.
7. Informing the patient of the nature of their treatment and the characteristics of the radioactive material used in their treatment. Instruct the patient in procedures to be followed to minimize the spread of contamination and the need to minimize the external radiation received by all persons entering their recovery room.
8. Removing contaminated bedding and clothing as they are generated to reduce the potential for intakes and to minimize the source term in the patient's room.
9. Flushing patient excreta through the room's toilet as it is generated. If the patient is ambulatory, he should use the toilet to remove the source term.

Question 2.53

Radiation protection concerns regarding allowing the patient to resume activities as a radiation worker at a nuclear power facility include:

1. The worker's uptake of radioactive material produces a source that irradiates his dosimetry. It will be difficult to distinguish dose from this internal source in the thyroid from external sources at the facility.
2. The worker's body excreta (sweat and urine) lead to cross-contamination problems. The I-131 in the worker's urine presents liquid effluent concerns and sweat will lead to contamination of plant areas.
3. The I-131 administration masks other I-131 internal intakes from the facility by the worker.
4. The I-131 could mask any skin contamination.
5. I-131 is an indicator of fuel defects. If the facility is operating with any fuel defects, the new I-131 intakes will be masked by the medical procedure.

6. The worker will be a source of exposure to coworkers.
7. The worker has the potential to disrupt routine radiation protection activities at the plant. Operations and health physics staff will respond to portal alarms and personal contamination monitor alarms caused by the worker.
8. Cross-contamination of samples (air samples, environmental samples, and survey smears) could also result.

Solutions for Chapter 3

Scenario 3.1

Question 3.1

$$e(E) = N(E)/S \times t$$

As an example of the efficiency calculation, consider the 60- and 1408-keV data:

Gamma Energy (keV)	Calibration Source (gammas/s) Decay Corrected	Ge(Li) Detector Net Counts (1000-second count)
60	2000	83 762
1408	1170	8215

$$e(60\text{ keV}) = \frac{83\,762\text{ counts}/1000\text{ s}}{2000\text{ gammas/s}} = 0.042\text{ counts/gamma}$$

$$e(1408\text{ keV}) = \frac{8215\text{ counts}/1000\text{ s}}{1170\text{ gammas/s}} = 0.007\text{ counts/gamma}$$

Following a similar procedure leads to the efficiency results for the other gamma energies considered in this problem:

***E* (keV)**	60	88	121	344	768	963	1408
***e* (E)**	0.042	0.052	0.050	0.021	0.012	0.009	0.007

Question 3.2

The values of $e(E)$ versus E provided in question 3.1 will lead to the desired curve.

Question 3.3

From the curve or by interpolating the tabular results, we obtain the value e (2000 keV) = 0.004.

The 2000-keV efficiency can be obtained by assuming that the tail of the $e(E)$ versus E curve is linear:

$$\begin{aligned} & e(E) = m\,E + b \\ 963\text{ keV}: \quad & 0.009 = 963\,m + b \qquad [1] \\ 1408\text{ keV}: \quad & 0.007 = 1408\,m + b \qquad [2] \\ [1] - [2]: \quad & 0.002 = (963 - 1408)\,m \\ & m = -4.494 \times 10^{-6} \end{aligned}$$

Inserting this value for m leads to b:

$$\begin{aligned} [1]: \quad & 0.009 = 963\,(-4.494 \times 10^{-6}) + b \\ & b = 0.0133 \end{aligned}$$

Therefore,

$$e(E) = -4.494 \times 10^{-6}\,E + 0.0133$$

Using this equation permits the efficiency at 2000 keV to be determined:

$$E = 2000\text{ keV}: \quad e(2000\text{ keV}) = (-4.494 \times 10^{-6})\,(2000) + 0.0133 = 0.004$$

Question 3.4: a

Between 100 and 300 keV, the counting efficiency decreases because photoelectric absorption decreases.

Question 3.5: a

At low gamma energies (< 80 keV), counting efficiency decreases because detector and housing attenuation become significant.

Question 3.6

The 400.7-keV peak only contains a contribution from Se-75, and the activity of Se-75 can be obtained from its analysis. Once the Se-75 activity is obtained, the Co-57 activity can be derived from the 121.5-keV peak. Finally, the Hg-203 activity is obtained from the 279.2-keV peak once the Se-75 activity is known.

Peak 1: 400.7 keV

$$A_i = C(E)\,k/(t)\,[e(E)]\,[Y(E)]$$
$$e(400\text{ keV}) = 0.018\text{ counts/gamma}$$

$$A(\text{Se-75, 400.7-keV}) = \frac{(629\text{ counts})\,(2.22\times10^6\text{ dpm/}\mu\text{Ci})^{-1}}{(10\text{ min})\,(0.018\text{ counts/gamma})\,(0.116\text{ gammas/dis})}$$
$$= 1.36\times10^{-2}\ \mu\text{Ci}$$

Peak 2: 122.1 keV

Given the activity of Se-75 determined from the 400.7-keV peak, the Se-75 contribution to the 121.1-keV peak can be determined. The Co-57 activity can be determined from the 122.1-keV sample count results once the Se-75 contribution is subtracted.

The Se-75 count rate contributing to the 121.1-keV peak is obtained from the previous relationship:

$$A_i = C(E)\,k/(t)\,[e(E)]\,[Y(E)]$$

Solving for $C(E)$ provides the count rate contribution:

$$C(E) = A_i\,t\,e(E)\,Y(E)/k$$

$$C(\text{Se-75, 121.1 keV}) = (1.36\times10^{-2}\ \mu\text{Ci})\,(10\text{ min})\,(0.05\text{ counts/gamma})$$
$$\times\,(0.173\text{ gammas/dis})\,(2.22\times10^6\text{ dpm/}\mu\text{Ci})$$
$$= 2611.6\text{ counts}$$

$$A(\text{Co-57, 122.1 keV}) = \frac{(7266\text{ counts} - 2612\text{ counts})\,(2.22\times10^6\text{ dpm/}\mu\text{Ci})^{-1}}{(10\text{ min})\,(0.05\text{ counts/gamma})\,(0.859\text{ gammas/dis})}$$
$$= 4.88\times10^{-3}\ \mu\text{Ci}$$

Peak 3: 279.2 keV

A similar approach is utilized to obtain the Hg-203 activity:

$$e(279.5\text{ keV}) = 0.025\text{ counts/gamma}$$

$$C(\text{Se-75, 279.5 keV}) = (1.36\times10^{-2}\ \mu\text{Ci})\,(10\text{ min})\,(0.025\text{ counts/gamma})$$
$$\times\,(0.252\text{ gammas/dis})\,(2.22\times10^6\text{ dpm/}\mu\text{Ci})$$
$$= 1902.1\text{ counts}$$

$$A(\text{Hg-203, 279.2 keV}) = \frac{(2279\text{ counts} - 1902\text{ counts})\,(2.22\times10^6\text{ dpm/}\mu\text{Ci})^{-1}}{(10\text{ min})\,(0.025\text{ counts/gamma})\,(0.815\text{ gammas/dis})}$$
$$= 8.33\times10^{-4}\ \mu\text{Ci}$$

Scenario 3.2

Question 3.7

P-32 is not volatile, even when heated, and can be ignored as an airborne contaminant.
HTO is an airborne hazard.
I-125 is an airborne hazard.

Question 3.8

Air sampling would be performed as follows:

HTO Bubble air through distilled water.
Pass air through desiccant (silica gel).
Cold trap/finger.
Ion chamber.
I-125 Charcoal filters with a vacuum pump impinger.

Question 3.9

Because neither HTO nor sodium iodide resides in dust, anisokinetic sampling can be used.

Question 3.10

An intake (I_i) will be possible for I-125 and H-3. Based upon its properties, P-32 is not volatilized. Because both students are present for the hour, they will each receive intakes of I-125 and H-3.

$$I_i = (\mathrm{BR})\, t\, A_i / V$$

$$V = (20\ \mathrm{ft} \times 20\ \mathrm{ft} \times 10\ \mathrm{ft})\,(28.3\ \mathrm{liters/ft^3}) = 113\,200\ \mathrm{liters}$$

I-125. For I-125, the intake relationship is sufficiently accurate to estimate the activity in the students:

$$I(\text{I-125}) = (20\ \mathrm{liters/min})\,(60\ \mathrm{min})\,(5000\ \mu\mathrm{Ci})/(113\,200\ \mathrm{liters})$$
$$= 53.0\ \mu\mathrm{Ci\ I\text{-}125/student}$$

HTO. Tritium oxide can enter the body via inhalation, ingestion, and skin absorption in approximately equal amounts. Therefore, skin absorption must be included in the HTO intake estimate. Following ICRP-30, a factor of 1.5 is used to account for skin absorption.

$$I(\text{HTO}) = 1.5\,(20\ \mathrm{liters/min})\,(60\ \mathrm{min})\,(10\,000\ \mu\mathrm{Ci})/(113\,200\ \mathrm{liters})$$
$$= 159\ \mu\mathrm{Ci\ HTO/student}$$

Question 3.11

The critical organs for HTO and I-125 are

HTO Whole body
I-125 Thyroid

The amount of HTO in the whole body is given in question 3.10.

The amount of I-125 in the thyroid will be 20–30% of the quantity calculated previously in question 3.10.

Question 3.12

H-3 and I-125 can be counted simultaneously by using liquid scintillation counting (LSC). Samples could be counted in a sodium iodide well counter or other gamma counter, to assess only the I-125. To obtain the tritium activity subtract the absolute activity, obtained from the gamma counter, from the gross absolute activity obtained from the LSC.

P-32 could be ignored because it is not volatile. Its presence or absence could be confirmed by counting in an LSC with a wide window.

Question 3.13

Borotritide chemical reactions lead to the evolution of tritium gas. Therefore, tritiated water collection techniques, which are not efficient collection mechanisms for tritium gas, would not be appropriate. The tritium gas could be oxidized via dry combustion using a palladium catalyst. Alternatively, ion chamber techniques could be employed to detect the tritium gas.

Question 3.14

The intakes noted above were caused by the laboratory exhaust fan motor being turned off for maintenance without the knowledge of the laboratory workers. The following measures would prevent recurrence of this accident:

1. Develop a procedure for maintenance personnel or outside contractors who repair exhaust/ventilation equipment. The procedure should include posting the hood to warn the researcher that the equipment is inoperable.
2. Install a warning light or horn at the hood to notify users when the hood fan is off or not functioning correctly.
3. Improve worker training and include operational checks of hood ventilation or a requirement for workers to contact maintenance prior to operating the hoods.
4. Institute a red tag system for activities that interfere with hood operability.

Question 3.15

Factors that would affect the intake of radioactive material by the workers utilizing the nonoperable hood include:

1. Room air changes or ventilation system characteristics.
2. Uniformity of radionuclide distribution in the air.
3. Location of the workers.
4. Plateout of radioactive material on room or hood structures.
5. Metabolism of the workers.
6. Actual quantity of material volatilized versus the 100% assumption.

Scenario 3.3

Question 3.16: e

Use a 5-in. × 0.06-in. NaI crystal probe to survey all areas suspected of being contaminated. The NaI detector of this configuration will have a good response to the low-energy X-rays. It is also more sensitive than the 1.5-in. × 1-in. NaI probe.

Poorer Answers

a. Take wipes in all suspected areas and count them on a shielded gas-flow proportional counter. You would have to smear every square inch and document the smear location. This would take considerable time, and the wipes would not necessarily remove contamination from all surfaces such as carpeting.
b. Use a pancake GM probe to survey all suspected areas of contamination. This probe will respond, but its efficiency is not as good as the NaI probe.
c. Use a ZnS-coated/photomultiplier-tube-based portable alpha probe to survey all suspected areas. Because the ZnS probe detects only alphas, it will miss significant levels of contamination due to self-shielding by carpet fibers.
d. Use a 1.5-in. × 1.0-in. NaI crystal probe to survey all suspected contaminated areas. The 1.5-in. × 1-in. NaI crystal geometry is not as sensitive as the 5-in. × 0.06-in. NaI probe.

Question 3.17: d

Use a 1.5-in. × 1.0-in. NaI crystal probe to survey all areas suspected of being contaminated. This size and thickness of NaI detector will have the best sensitivity to the 662-keV Cs-137 gammas.

Question 3.18: e

Analyze a 24-hr fecal sample collected on day 2 with a germanium detector/multichannel analyzer. Clearance of insoluble Am-241 particles from the lung will route through the gastrointestinal tract to feces pathway.

Poorer Answers

a. Count nasal swipes on a shielded gas-flow proportional counter to estimate the inhaled activity. Although swipe counting will detect the Am-241, there is no established method to allow a quantitative determination of lung dose based on nasal wipe data.
b. Whole-body counting in a shielded facility via a 3×3 NaI detector/MCA. Whole-body counting for low-energy X-rays would have a very low sensitivity due to their attenuation by the body.
c. Whole-body counting in a shielded facility via a coaxial germanium detector/MCA. Whole-body counting by GeLi for low-energy X-rays has a greater sensitivity than a NaI detector, but it will be less sensitive than fecal counting.
d. Analysis of a 24-hr urine sample collected on day 2 by liquid scintillation counting. A negligible amount of insoluble Am-241 is expected to appear in the urine.

Question 3.19: b

Whole-body counting in a shielded facility via a NaI detector/MCA. This system would easily detect the 662-keV Cs-137 gammas and quantify the Cs-137 uptake.

Poorer Answers

a. Count nasal swipes on a shielded gas-flow proportional counter to estimate the inhaled activity. Although swipe counting will detect the Cs-137, there are too many uncertainties in using this technique to quantify the uptake.
c. Analysis of activity exhaled in the breath. Insignificant activity and calibration uncertainties complicate the use of this approach.
d. Analysis of a 24-hr urine sample collected on day 2 by liquid scintillation counting. The activity going into the urine would vary greatly, and the retention function would be uncertain within the first 2 days.
e. Analyze a 24-hr fecal sample collected on day 2 with a germanium detector/multichannel analyzer. This technique would allow an estimate, but there is more uncertainty than with whole-body counting.

Question 3.20

$\dot{D}_0 = 2.13\ P\ E(\mathrm{QF})/m$
P = activity deposited into the lung (μCi)
E = 5.57 MeV (energy of alphas and recoil atoms deposited into the lung)
m = lung mass (2) = 1000 g

QF = quality factor for alphas = 20 rem/rad

$\dot{D}_0$ = (2.13) [(rad/hr)/(μCi-MeV/g)] (1000 μCi) (5.57 MeV) × (20 rem/rad)/(1000 g)

= 237 rem/hr × 24 hr/d = 5.69×10^3 rem/d

Question 3.21: c

Over a 50-year period, the doses for both cases would be essentially the same. For the 50-year period, details of the intake over the first 100 days are insignificant when compared to the lung retention time.

Question 3.22: b

ICRP-26 specifies a lung weighting factor of 0.12.

Question 3.23: d

Am-241's most common production mode is from the beta decay of Pu-241.

Scenario 3.4

Question 3.24

Dose rate information, a knowledge of the fetal–source distance, and the geometry of the X-ray system can be used to calculate patient exposures including the dose to a fetus.

In order to use the tabular data, the beam quality HVL must be determined. From the available survey data, the measured dose can be plotted, on semilog graph paper, as a function of the filtration (mm Al) to determine an HVL value of 3.0 mm of Al. Interpolation can also be used to obtain the HVL thickness:

4.5-mm Al filtration corresponds to 180 mR
2.5-mm Al filtration corresponds to 280 mR
0.0-mm Al filtration corresponds to 500 mR
HVL of Al filtration corresponds to 250 mR

$$\frac{\log(4.5) - \log(2.5)}{180\ \text{mR} - 280\ \text{mR}} = \frac{\log(4.5) - \log(\text{HVL})}{180\ \text{mR} - 250\ \text{mR}}$$

$$\log(\text{HVL}) = -\frac{180\ \text{mR} - 250\ \text{mR}}{180\ \text{mR} - 280\ \text{mR}}[\log(4.5) - \log(2.5)] + \log(4.5)$$

$$= 0.6532 - 0.7(0.2553) = 0.4745$$

$$\text{HVL} = 2.98\ \text{mm Al}$$

Distances of interest are illustrated in Figure S3.1.

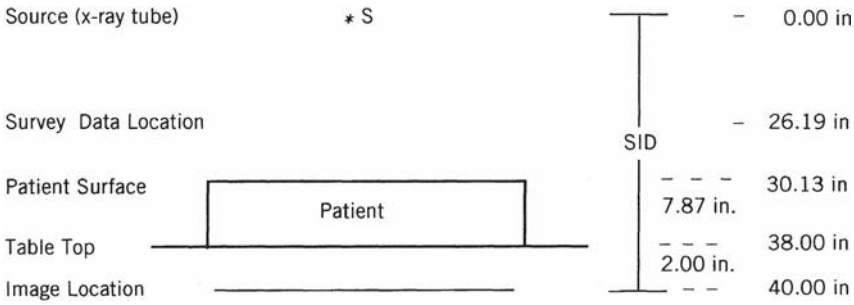


Figure S3.1 Location of the patient, X-ray source (*S*) and image plane for the unanticipated exposure event.

The following dimensions are relevant to the fetal exposure assessment:
- X-ray source location = 0.00 in.
- The patient thickness is 20.0 cm = 7.87 in.
- The location of the survey data is 30 cm above the table top, which is the following distance from the source:
  (40.00 in. – 2.00 in. – 30.00 cm/2.54 cm/in. = 26.19 in.
- The location of the top of the patient relative to the source:
  (40.00 in. – 2.00 in. – 7.87 in.) = 30.13 in.

The information given in the problem statement indicates that the patient's skin is 30.13 in. from the tube. This is illustrated above.

In order to obtain the patient's entrance skin exposure, assume:
1. Linearity with respect to mA-s.
2. The inverse-square law applies.
3. The entrance skin dose can be obtained from the given tabular data.

The embryo/fetus dose can be determined from the tabular data for the entrance skin exposures. To use the tables, we must first determine the absorbed dose for AP and PA films. In the X-ray series, 2 AP and PA films were taken. The total entrance skin exposure (*X*) for both posterior-anterior (PA) and anterior–posterior (AP) films is provided by the following equation, where all distances are measured from the X-ray source:

$$X = \frac{(26.19\text{ in.})^2}{(30.13\text{ in.})^2} \times \frac{35\text{ mA-s}}{30\text{ mA-s}} \times 500\text{ mR} \times 2 = 882\text{ mR} = 0.882\text{ R}$$

In the tables under the 3.0-mm Al column for AP film, an absorbed dose of 330 mrad/R at the skin entrance is obtained. Therefore, for both AP films the embryo absorbed dose is

$$D_{AP} = X_{AP} \times (\text{tabular mrad/R conversion factor})$$
$$= 0.882\text{ R} \times 330\text{ mrad/R} = 291\text{ mrad}$$

For the PA exposure, the 3.0-mm Al column yields 174 mrad/R to the embryo at the skin entrance. Therefore, for both PA films we obtain

$$\begin{aligned} D_{PA} &= X_{PA} \times (\text{tabular mrad/R conversion factor}) \\ &= 0.822\ \text{R} \times 174\ \text{mrad/R} = 153\ \text{mrad} \end{aligned}$$

Therefore, the total embryo absorbed dose (D) is:

$$D = D_{AP} + D_{PA} = 291\ \text{mrad} + 153\ \text{mrad} = 444\ \text{mrad}$$

Question 3.25: b

ICRP-49 indicates that prior to the eighth week following fertilization there is apparently little risk of severe mental retardation for radiation dose *in utero*.

Question 3.26: e

The NCRP-91 recommendations for pregnant radiation workers are to limit the dose equivalent to 500 mrem to the embryo/fetus over the entire pregnancy. More recently, NCRP-116 recommends no more than 0.5 mSV in a month to the embryo-fetus.

Scenario 3.5

Question 3.27

The dose is obtained from the fundamental ICRP-2/10 relationship:

$$\begin{aligned} D(\text{rad}) &= 2.13\ \frac{\text{rad-g-dis}}{\text{hr-MeV-}\mu\text{Ci}}\ \frac{A(0)\ (\mu\text{Ci})\left[\sum_i E_i\ (\text{MeV/dis})\ Y_i\right](\text{AF})}{m\,(\text{g})\ \lambda_{\text{eff}}\ (1/\text{hr})} \\ &= 2.13\ \frac{\text{rad-g-dis}}{\text{hr-MeV-}\mu\text{Ci}}\ \frac{C(\mu\text{Ci/g})\left[\sum_i E_i\ (\text{MeV/dis})\ Y_i\right](\text{AF})\ T_{\text{eff}}\ (\text{d})}{\ln(2)}\ \frac{24\ \text{hr}}{\text{d}} \\ &= 73.8\ \frac{\text{rad-g-dis}}{\text{d-MeV-}\mu\text{Ci}}\ C(\mu\text{Ci/g})\left[\sum_i E_i\ (\text{MeV/dis})\ Y_i\right](\text{AF})\ T_{\text{eff}}\ (\text{d}) \end{aligned}$$

where

D = dose in rads
C = activity concentration in the thyroid (μCi/gram of tissue) = $A(0)/m$
$A(0)$ = initial activity in the thyroid (μCi)
m = mass of thyroid = 30 g
AF = absorbed fraction = 0.7
i = number of photons emitted = 2
E_i = mean photon energy (MeV) of the ith photon

Y_i = yield of the ith photon of energy E_i
E_1 = 0.028 MeV/dis
E_2 = 0.035 MeV/dis
Y_1 = 1.398
Y_2 = 0.067
T_{eff} = effective half-life (days)
= $(T_{\text{phy}} \times T_{\text{bio}})/(T_{\text{phy}} + T_{\text{bio}})$
T_{phy} = 60 days
T_{bio} = 130 days
= (60 days × 130 days)/(60 days + 130 days) = 41.05 days

The activity (2.4 μCi) at 14 days is known. This information and the effective half-life can be used to determine the initial uptake of I-125 in the thyroid:

$$A(t) = A(0)\exp(-0.693\, t/T_{\text{eff}})$$
$$A(0) = 2.4\ \mu\text{Ci}\exp(+0.693 \times 14\ \text{days}/41.05\ \text{days}) = 3.04\ \mu\text{Ci}$$

$$D = 73.8\ \frac{\text{rad-g-dis}}{\text{d-MeV-}\mu\text{Ci}}\ C(\mu\text{Ci/g})\left[\sum_i E_i\,(\text{MeV/dis})\,Y_i\right](\text{AF})\,T_{\text{eff}}\,(\text{d})$$
$$= 73.8\ \frac{\text{rad-g-dis}}{\text{d-MeV-}\mu\text{Ci}}\left(\frac{3.04\ \mu\text{Ci}}{30\ \text{g}}\right)$$
$$\times[(0.028\ \text{MeV/dis})(1.398) + (0.035\ \text{MeV/dis})(0.067)](0.7)(41.05\ \text{d})$$
$$= 8.92\ \text{rad}$$

Question 3.28

$$H = D + w_{\text{T}}\ H_{\text{T}}$$
$$= 1.25\ \text{rem} + 0.03 \times 25\ \text{rem} = 2.0\ \text{rem}$$

Question 3.29

No overexposure occurred. The ICRP-26 organ dose limit is 50 rem (nonstochastic), and the total effective dose equivalent limit is 5 rem (stochastic).

Scenario 3.6

Question 3.30

The total activity (A) of Co-60 in the column may be obtained by assuming that there is no decay during the sampling period. This assumption is valid in view of the Co-60 (5.3 years) half-life.

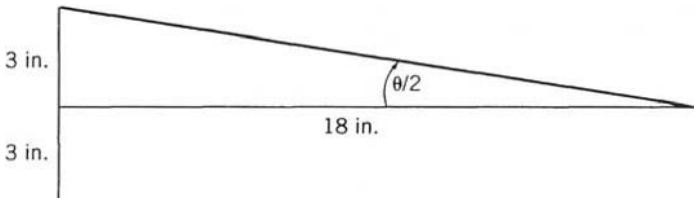


Figure S3.2 Line source geometry for determination of the included angle (θ).

$$
\begin{aligned}
A &= V\,C\,e \\
&= (5000 \text{ liters})\,(1000 \text{ mL/liter})\,(1.92 \times 10^{-2}\ \mu\text{Ci/mL})\,(0.95) \\
&= 9.12 \times 10^{4}\ \mu\text{Ci} = 91.2 \text{ mCi}
\end{aligned}
$$

The exposure rate from the column may be obtained from a line source approximation:

$$\dot{D} = G\,C_{\text{L}}\,\theta/h$$

Determination of the angle θ is illustrated in Figure S3.2.

$$
\begin{aligned}
\tan\theta/2 &= 3/18 = 0.1667 \\
\theta &= \tan^{-1}(0.3333333) = 18.92^{\circ} = 0.33 \text{ rad} \\
C_{\text{L}} &= 91.2 \text{ mCi}/(6.0 \text{ in.} \times 2.54 \text{ cm/in.}) = 5.98 \text{ mCi/cm} \\
\dot{D} &= 13.2\,\frac{\text{R-cm}^2}{\text{mCi-hr}}\,(5.98 \text{ mCi/cm})\,(0.33) \\
&\quad \times (1000 \text{ mR/R})/(18 \text{ in.} \times 2.54 \text{ cm/in.}) \\
&= 570 \text{ mR/hr}
\end{aligned}
$$

A reasonable approximation to this dose rate may be obtained from a point-source approximation:

$$
\begin{aligned}
\dot{D} &= G\,A/r^2 \\
&= 13.2\,\frac{\text{R-cm}^2}{\text{mCi-hr}}\,(91.2 \text{ mCi}) \times 1000 \text{ mR/R}/(18 \text{ in.} \times 2.54 \text{ cm/in.})^2 \\
&= 576 \text{ mR/hr}
\end{aligned}
$$

Question 3.31

Safety concerns regarding the storage of the material are as follows:

1. Radiolytic decomposition of water results in the generation and accumulation of hydrogen and oxygen gases, which present an explosive hazard. To assess this concern, the gas generation must be quantified.

Gas Generation Assumptions

a. Storage time is 1 year.
b. Co-60 activity will be assumed to be constant during the interval (91.2 mCi).
c. Anion resin is used for I-129, and cation resin is used for Co-60.
d. All beta energy emitted by radioactive decay is absorbed for gas generation. One-half is absorbed by the anion resin and one-half by the cation resin. No gamma absorption is assumed.

The energy absorbed per year (E) is given by

$$\begin{aligned} E &= (91.2 \times 10^{-3}\ \text{Ci})\,(3.7 \times 10^{10}\ \text{dis/s-Ci})\,(0.096\ \text{MeV/dis}) \\ &\quad \times (365\ \text{days/year})\,(24\ \text{hr/day})\,(3600\ \text{s/hr}) \\ &= 1.02 \times 10^{16}\ \text{MeV/year} \end{aligned}$$

The quantity of hydrogen generation (V) is obtained from the absorbed energy and radiolytic rates:

$$\begin{aligned} V &= (1.02 \times 10^{16}\ \text{MeV/year})\,(1.0 \times 10^{6}\ \text{eV/MeV})\,(0.5) \\ &\quad \times [(0.6\ \text{H}_2\ \text{molecules}/100\ \text{eV}) + (0.13\ \text{H}_2\ \text{molecules}/100\ \text{eV})] \\ &\quad \times\ (22.4\ \text{liters/g-mole}) \times (1000\ \text{cm}^3/\text{liter})/6.023 \times 10^{23}\ \frac{\text{H}_2\ \text{molecules}}{\text{g-mole}} \\ &= 1.38\ \text{cm}^3/\text{year} \end{aligned}$$

This gas-generation rate is insignificant. Moreover, the careful researcher will have vented his storage column, which further reduces the potential of the buildup of an explosive mixture.

2. Radiation damage to the plexiglass resin column resulting in its fracture or eventual failure with a subsequent release of its contents.

Material Damage Assumptions

a. Co-60 activity will be assumed to be constant during the interval (91.2 mCi).
b. All the beta energy is absorbed in the resin column. This is reasonable based upon the beta energies encountered in this scenario.
c. A minimum amount of gamma energy is absorbed in the resin column. Attenuation coefficients suggest that this is a valid assumption.

In order to assess the dose to the resin column walls, we first calculate the dose to the resin contained within the column. The average dose is the total energy absorbed (E) divided by the volume of the resin column (v) contents:

$$
\begin{aligned}
D_{\text{resin}} &= E/v \\
&= (1.02 \times 10^{16}\ \text{MeV/year})\,(1\ \text{year})\,(1.6 \times 10^{-8}\ \text{g-rad/MeV}) \\
&\quad \times (\text{cm}^3/1\ \text{g}) \times \frac{1}{(3.14)\,(0.5\ \text{in.})^2\,(6\ \text{in.})\,(2.54\ \text{cm/in.})^3} \\
&= 2.11 \times 10^6\ \text{rad}
\end{aligned}
$$

The dose to the wall of the resin column will be a fraction of the dose delivered to the contents of the column. This fraction will vary between 0 and 1, depending on the nature of the contents and the container. A guide to the correct factor is derived from the ICRP-30 ingestion model; this suggests that for water contents in tissue, the dose to the walls is about 0.5 times the dose delivered to the contents.

$$
D_{\text{column}} = (0.5)\,D_{\text{resin}} = (0.5)\,(2.11 \times 10^6\ \text{rad}) = 1.06 \times 10^6\ \text{rad}
$$

Radiation damage to the column walls is possible in this dose range. The damage will involve discoloration of the plexiglass. Some fine cracking is also possible. Rupture of the container is not likely during the first year.

Question 3.32

Type of Control	Control Measure
Administrative	1. Perform a radiological safety evaluation to assess potential hazards resulting from the storage. 2. The operation and usage of the column should be specified in a written procedure. 3. An ALARA (as low as reasonably achievable) evaluation should be performed to limit radiation exposures.
Operational	1. Access controls including high radiation control should be imposed. 2. Temporary shielding should be added to limit radiation exposures. 3. Appropriate radiological postings and warning signs should be installed.

Question 3.33

The definition of the lower limit of detection (LLD) is

$$
\text{LLD} = \frac{4.66\ S_b}{e\ v\ Y\ \exp(-\lambda\ t)}
$$

All factors defining the LLD remain constant except for the sample volume (v). Therefore, the reduction in the LLD is related to the ratio of sample volumes:

$$
\text{LLD} = 3.0 \times 10^{-10}\ \mu\text{Ci/ml}\ (1.0\ \text{liter}/5000\ \text{liters}) = 6.0 \times 10^{-14}\ \mu\text{Ci/ml}
$$

Scenario 3.7

Question 3.34

The transport index of a package is defined as the maximum dose equivalent rate at a distance of 1.0 meter from the surface of the package.

The transport index is specified to be 3. Therefore, the maximum dose equivalent rate at 1 m from the surface of the package is 3 mrem/hr.

Question 3.35

The package dimensions are 60 × 60 × 60 cm. From the transport index definition, it is known that the dose equivalent rate 1.0 m from the surface of the package or 130 cm (100 cm + 30 cm) from the source is 3.0 mrem/hr. Assuming that the source resides at the center of the package and that the additional air and carton offer minimal attenuation and buildup, the point-source approximation can be used to calculate the dose rate at the package surface:

$$\dot{D}(r) = S/r^2$$

where S is the source strength. Using this equation, we can write

$$\dot{D}(a)\, r_a^2 = \dot{D}(b)\, r_b^2$$

Letting a = 130 cm and b = 30 cm (the distance to the package surface), the dose rate at 30 cm is obtained:

$$\dot{D}(30\text{ cm}) = 3.0\text{ mrem/hr}\ (130\text{ cm}/30\text{ cm})^2 = 56.3\text{ mrem/hr}$$

Question 3.36

A medical package is legal for shipment on a passenger aircraft if the following conditions are met:

1. The transport index (TI) does not exceed 3.0.
2. The highest surface dose equivalent rate does not exceed 200 mrem/hr.

For the packages in question, the TI = 3.0 and the maximum surface dose rate is 56.3 mrem/hr. Therefore, this is a legal package.

Question 3.37

The shipment contains four packages as described in the scenario. The TI of four packages is just four times the TI for a single package, or 4 × 3 (mrem/hr) = 12 (mrem/hr).

Question 3.38

The TLD package is 50 cm from the package surface or 80 cm (30 m + 50 cm) from the source assumed to reside at the center of the package. The dose delivered to the TLD package from the four generator packages at the end of a 6-hr exposure is

$$\begin{aligned} D &= (3 \text{ mrem/hr-packages})\,(130 \text{ cm}/80 \text{ cm})^2\,(4 \text{ packages})\,(6 \text{ hr}) \\ &= 190.1 \text{ mrem} \end{aligned}$$

Question 3.39

Provisions made by commercial vendors to compensate for transient exposures include:

1. The package will contain control dosimeters. These dosimeters were exposed to a known exposure and are used to determine the transit dose that is to be subtracted from the TLD shipment:

   Transit dose = Total TLD dose – Known exposure

2. The TLD package will contain a warning label stating that the package is sensitive to X-rays to preclude unnecessary X-ray exposure.

Question 3.40

The package is returned to its manufacturer 2 weeks after its initial labeling. Assuming that the Tc-99m and Mo-99 are in secular equilibrium, the activity decays with an effective half-life of the parent. The TI at the end of the 2-week period is

$$\begin{aligned} \text{TI}(t) &= \text{TI}_0 \exp(-\lambda t) \\ &= (3)\,[(-0.693/67 \text{ hr})\,(14 \text{ days} \times 24 \text{ hr/day})] = 0.1 \end{aligned}$$

Question 3.41

Another package has the following properties:

TI = 2.0
Surface dose equivalent rate = 15.0 mrem/hr

Because the TI exceeds 1, a radioactive type III label is required.

Scenario 3.8

Question 3.42

Actions taken to evaluate and correct the spread of radioactive material are as follows:

1. Secure the area and control access.
2. Contain the leaking source and store it in a radiologically controlled area.
3. Determine the contamination levels on the source.
4. Survey the area to determine the extent and degree of contamination. Appropriately post the area based on the survey results.
5. Determine which and how many workers were contaminated.
6. Survey and decontaminate the affected personnel.
7. Obtain nasal swabs and bioassay on the workers involved in the incident.
8. Contact the source manufacturer to determine if the source was manufactured with biologically inert microspheres or other material to limit their biological impact.
9. Notify university, state, and Federal officials as required by your health physics procedures.

Question 3.43

Pertinent polonium-210 decay and hazard parameters are as follows:

Primary mode of decay:	Alpha 5.3 MeV @ 100%
	Gamma 0.8 MeV @ 0.001%
Half-life:	138 days
Hazard type:	Internal

Question 3.44

Measurement	Method	Locations	Concerns/Precautions
Air	Particulate	Lab involved	Interfering nuclides
	High-volume grab sample	Adjacent labs	Representativeness of the sample
			Measurement sensitivity
Smears	Filters	Lab benches, walls, and floors	All locations not easily accessible
	Sanitary napkins	Remainder of building	Radon/thoron interference
	Mops	Worker residences	

Measurement	Method	Locations	Concerns/Precautions
Surveys	Alpha probe	Campus buildings	Interfering nuclides
	Large-area detectors	Parking Lots	Measurement sensitivity
		Personal vehicles	Surveys on rough surfaces
		Worker residences	Po-210 may be masked by asphalt, soil, vegetation, and carpet
		Personal property	
		Roadways	

Question 3.45

Areas that should receive priority attention for decontamination are as follows:

Area	Cleanup Motivation
University buildings (excluding the affected lab)	Permit the university to function normally
Parking lots and roads	Ensure that uncontrolled areas are returned to a normal status
Worker residences and vehicles	Ensure that uncontrolled areas are returned to a normal status

Question 3.46

Items for inclusion in the incident report are as follows:

1. Root cause and contributing factors for the incident.
2. Quantification of the degree and extent of the contamination.
3. Listing of personnel contaminated, their contamination levels, and the associated internal exposures.
4. Recommendations to preclude recurrence of this type of event.
5. Program modifications suggested by this event (changes to survey frequency, personnel frisking requirements, changes to training program, and source inspection requirements).
6. Corrective actions taken and planned with responsible person and due date identified.
7. Incident recovery cost.

8. Sequence of events of the incident and recovery, including dates, times, and personnel involved.
9. Identification of regulatory consequences and required reporting requirements.
10. Personnel statements provided during the investigation.

Scenario 3.9

Question 3.47

Relative Hazards	P-32	I-125	HTO
Skin dose potential	High	Moderate	Low
Bioassay requirement	Low	High	High
Eye hazard	High	Low	None
Personnel dosimetry requirement	High	High	None
Air sampling requirement	Low	High to moderate	High to moderate

Question 3.48

Low-atomic-number material should be utilized to shield the 1.71-MeV beta particle from P-32. One choice would be Lucite or plexiglass with a density of about 1.18 g/cm^3.

The range of the P-32 beta particle may be approximated by the beta-range relationship:

$$R = 412\,E^{1.265-0.0954\,\ln(E)}$$

where R is the range of the beta particle in mg/cm^2 and E is the maximum beta energy in MeV.

$$R = 412\,(1.71)^{1.265-0.0954\,\ln(1.71)} = 790.2\ \text{mg/cm}^2$$

The thickness of material is given by

$$\begin{aligned} t(\text{cm}) &= R(\text{mg/cm}^2)/p(\text{g/cm}^3) \\ &= (790.2\ \text{mg/cm}^2)\,(1\ \text{g}/1000\ \text{mg})/(1.18\ \text{g/cm}^3) \\ &= 0.670\ \text{cm} \times 1\ \text{in.}/2.54\ \text{cm} \\ &= 0.264\ \text{in. or about 3/8 in. to be conservative} \end{aligned}$$

Question 3.49

The beta shielding thickness does not change. All P-32 betas are stopped by the 3/8-in. thickness that will stop the maximum beta energy particle. However, the increase in P-32 activity leads to enhanced bremsstrahlung production. The fraction (f) of beta energy converted into photons is:

$$f = \frac{3.5 \times 10^{-4}}{\text{MeV}} Z\, E_{\beta}^{\max}$$

where Z is the atomic number of the shielding material attenuating the beta particle (6.56 for Lucite) and $E_{\beta}^{\max}$ is the maximum energy (1.71 MeV for P-32) of the beta particle spectrum. The equivalent photon activity associated with the 1000 mCi P-32 source is:

$$A_{\gamma} = f\, A_{\beta} = \frac{3.5 \times 10^{-4}}{\text{MeV}} Z\, E_{\beta}^{\max}\, A_{\beta}$$
$$= \left(\frac{3.5 \times 10^{-4}}{\text{MeV}}\right)(6.56)\,(1.71\ \text{MeV})\,(1000\ \text{mCi}) = 3.93\ \text{mCi}$$

Given the magnitude of the effective photon activity, additional shielding to reduce the photon dose should be evaluated based on the laboratory configuration.

Question 3.50

P-32 waste material should be stored for 10 half-lives to permit radioactive decay to background levels. After documented surveys, the P-32 waste can be discarded as common trash.

Question 3.51

KI should not be taken before the procedure. The researcher could have an allergic reaction to the KI, and there is no need for this practice if the fume hood is working properly. If there were an incident, KI would still be effective if it were taken within hours of the uptake. It would be advantageous if experimenters were cleared for KI use in advance by a university physician. A listing of those qualified for KI use should be maintained by the RSO.

Question 3.52

A liquid scintillation counter should be utilized. Both beta particles and gamma rays will produce light in a liquid scintillation cocktail. Appropriate calibration standards and laboratory procedures would be required to ensure that sufficient accuracy is maintained.

Scenario 3.10

Question 3.53

The Au-198 activity when the sample was removed from the reactor is determined from the relationship:

$$A(T) = A(0)\, e^{-\lambda T}$$

where

$A(T)$ = activity after the foil leaves the reactor at time T
$A(0)$ = activity of the sample immediately after exiting the reactor
A(T) = $C(T)/e\ Y$
$C(T)$ = count rate at 11 hours = 8.274×10^5 c/min
T = decay time = 11 hr
e = counting efficiency = 0.273 c/d
Y = yield = 0.955

$$A(T) = \frac{8.274 \times 10^5 \text{ c/min}}{(0.273 \text{ c/d})\,(0.955)} = (3.17 \times 10^6 \text{ d/min})\,(1 \text{ min}/60 \text{ s})\,(\text{Bq-s/d})$$
$$= 5.28 \times 10^4 \text{ Bq}$$

The activity present when the foil exited the reactor [A(0)] is:

$$A(0) = A(T)\, e^{+\lambda T}$$

where

λ = disintegration constant = ln 2/$T_{1/2}$ = 0.693/2.695 d = 0.257/d

$$A(0) = (5.28 \times 10^4 \text{ Bq})\, e^{+(0.257/\text{d})\,(11 \text{ hr})\,(1 \text{ d}/24 \text{ hr})} = (5.28 \times 10^4 \text{ Bq})\,(1.13)$$
$$= (5.97 \times 10^4 \text{ Bq})$$

Question 3.54

The gamma dose equivalent rate $(\dot{H})$ may be determined from the point source relationship:

$$\dot{H} = \frac{A\,G}{r^2}$$

where

G = gamma emission constant
A = applicable activity as specified in the problem statement = 130 MBq
r = distance for dose equivalent evaluation = 0.1 m

The gamma emission constant is defined in terms of conventional units. When the energy E is expressed in MeV, G has the following form:

$$G = \frac{1}{2}\sum_i E_i\ Y_i$$

and has the units R-m^2/hr-Ci. The sum is over the number (i) of photons emitted and Y is the photon yield. For Au-198,

$$G = \left(\frac{E\ Y}{2}\frac{\text{R-m}^2}{\text{hr-Ci}}\right)\left(\frac{0.95\ \text{rad}}{\text{R}}\right)\left(\frac{1\ \text{Gy}}{100\ \text{rad}}\right)\left(\frac{\text{Sv}}{\text{Gy}}\right)\left(\frac{1\ \text{Ci}}{3.7\times10^4\ \text{MBq}}\right)$$
$$= 1.3\times10^{-7}\ \frac{\text{Sv-m}^2}{\text{hr-MBq}}\ E\ Y$$

E = Au-198 gamma ray energy (MeV) = 0.412 MeV
Y = Au-198 gamma ray yield or abundance = 0.955

$$G = \left(1.3\times10^{-7}\ \frac{\text{Sv-m}^2}{\text{hr-MBq}}\right)(0.412)\ (0.955) = 5.1\times10^{-8}\ \frac{\text{Sv-m}^2}{\text{hr-MBq}}$$

With these values, the dose equivalent rate is determined:

$$\dot{H} = \frac{(130\ \text{MBq})\left(5.1\times10^{-8}\ \frac{\text{Sv-m}^2}{\text{hr-MBq}}\right)}{(0.1\ \text{m})^2} = (6.6\times10^{-4}\ \text{Sv/hr})\ (1000\ \text{mSv/Sv})$$
$$= 0.66\ \text{mSv/hr}$$

Question 3.55: b

The gold foil target contains 1.0×10^{24} Au-197 atoms (N), the incident thermal neutron flux (ϕ) is 1.0×10^{11} n/cm^2-s, and the thermal neutron activation cross-section (σ) for Au-197 (n, γ) is 98.8 b. With these values, the saturation activity is:

$$A_{sat} = N\ \sigma\ \phi = (1.0\times10^{24}\ \text{atoms})\ (98.8\ \text{b/atom})\ (1.0\times10^{-24}\ \text{cm}^2\text{/b})$$
$$\times\ (1.0\times10^{11}\ \text{n/cm}^2\text{-s})\ (1\ \text{dis/n})\ (1\ \text{Bq-s/dis})$$
$$= 9.88\times10^{12}\ \text{Bq}$$

Scenario 3.11

Question 3.56

The average airborne iodine concentration in each room is determined by considering the buildup of activity on the filter. The rate of change of activity on the filter is given by:

$$\frac{dA}{dt} = \text{activity collected} - \text{activity removed} = F\,C - \lambda\,A = K - \lambda\,A$$

where

F = sampler flow rate
C = air concentration
λ = radioactive decay constant
K = sampling rate
A = activity on the filter

The sampling rate is assumed to be constant and this assumption simplifies the integration:

$$\int_0^{A(T)} \frac{dA}{K - \lambda\,A} = \int_0^T dt$$

$$A(T) = \frac{K}{\lambda}\left(1 - e^{-\lambda T}\right) = \frac{F\,C}{\lambda}\left(1 - e^{-\lambda T}\right)$$

where

$A(T)$ = activity at the end of the sampling interval
C = average concentration during the sampling period

The previous equation can be solved for C:

$$C = \frac{A(T)\,\lambda}{F\left(1 - e^{-\lambda T}\right)}$$

Since the sampling system has a specified efficiency, the previous equation must be modified. In addition, three rooms are involved and the room-dependent parameters are indicated below:

$$C_i = \frac{A_i(T_i)\,\lambda}{e\,F_i\left(1 - e^{-\lambda T_i}\right)}$$

where

C_i = average airborne I-125 concentration Room No. i
i = room label = 1, 2, or 3

A = filter activity of I-125
$A_1 = 0.56$ MBq for Room 1
$A_2 = 0.15$ MBq for Room 2
$A_3 = 1.76$ MBq for Room 3
e = air sampler filter efficiency = 0.95
F_i = sampling rate in the i^{th} room
$F_1 = 20$ L/min
$F_2 = 15$ L/min
$F_3 = 30$ L/min
T_i = sampling time assumed to be the duration of the procedure
$T_1 = 5$ hr
$T_2 = 2$ hr
$T_3 = 6$ hr
λ = decay constant for ^{125}I

$$= \frac{0.693}{(60.14\text{ d})(24\text{ h/d})} = 4.80 \times 10^{-4}/\text{hr}$$

Room No. 1:

$$C_1 = \frac{A_1(T)\,\lambda}{e\,F_1\,(1 - e^{-\lambda T_1})}$$

$$= \frac{(0.56\text{ MBq})\,(4.8 \times 10^{-4}/\text{hr})\,(1000\text{ L/m}^3)}{(0.95)\,(20\text{ L/min})\,(60\text{ min/hr})\,[1 - \exp(-4.8 \times 10^{-4}/\text{hr})\,(5\text{ hr})]}$$

$$= 0.0984\,\frac{\text{MBq}}{\text{m}^3}$$

Room No. 2:

$$C_2 = \frac{A_2(T)\,\lambda}{e\,F_2\,(1 - e^{-\lambda T_2})}$$

$$= \frac{(0.15\text{ MBq})\,(4.8 \times 10^{-4}/\text{hr})\,(1000\text{ L/m}^3)}{(0.95)\,(15\text{ L/min})\,(60\text{ min/hr})\,[1 - \exp(-4.8 \times 10^{-4}/\text{hr})\,(2\text{ hr})]}$$

$$= 0.0878\,\frac{\text{MBq}}{\text{m}^3}$$

Room No. 3:

$$C_3 = \frac{A_3(T)\,k}{e\,F_3\,(1 - e^{-\lambda T_3})}$$

$$= \frac{(1.76\text{ MBq})\,(4.8 \times 10^{-4}/\text{hr})\,(1000\text{ L/m}^3)}{(0.95)\,(30\text{ L/min})\,(60\text{ min/hr})\,[1 - \exp(-4.8 \times 10^{-4}/\text{hr})\,(6\text{ hr})]}$$

$$= 0.172\,\frac{\text{MBq}}{\text{m}^3}$$

As a matter of completeness, the concentration expression could have been simplified since λT is small. By expanding the exponential, a simplified result is obtained:

$$C = \frac{A(T)\,\lambda}{e\,F\,(1 - e^{-\lambda T})} = \frac{A(T)\,\lambda}{e\,F\,[1 - (1 - \lambda\,T)]} = \frac{A(T)\,\lambda}{e\,F\,(1 - 1 + \lambda\,T)} = \frac{A(T)}{e\,F\,T}$$

Question 3.57

One of the technicians was exposed to a room air concentration of 0.192 MBq/m^3 for a three-hour procedure. The CDE and CEDE from this procedure is obtained from the DAC values:

$$\text{CDE} = \frac{C\,t}{\text{DAC-NS}}\,k$$

$$\text{CEDE} = \frac{C\,t}{\text{DAC-S}}\,k'$$

where

CDE = committed dose equivalent
CEDE = committed effective dose equivalent
C = I-125 room air concentration = 0.192 MBq/m^3
t = exposure time = 3 hr
DAC-NS = non-stochastic DAC value = 0.001 MBq/m^3
DAC-S = stochastic DAC value = 0.002 MBq/m^3
k = non-stochastic dose conversion factor

$$= \frac{0.5\ \text{Sv}}{2000\ \text{DAC-hr}}$$

k' = stochastic dose conversion factor

$$= \frac{0.05\ \text{Sv}}{2000\ \text{DAC-hr}}$$

With these values the CDE and CEDE are determined:

$$\text{CDE} = \frac{(0.192\ \text{MBq/m}^3)\,(3\ \text{hr})}{(0.001\ \text{MBq/m}^3\text{-DAC})}\,\frac{0.5\ \text{Sv}}{2000\ \text{DAC-hr}} = 0.144\ \text{Sv}$$

$$\text{CEDE} = \frac{(0.192\ \text{MBq/m}^3)\,(3\ \text{hr})}{(0.002\ \text{MBq/m}^3\text{-DAC})}\,\frac{0.05\ \text{Sv}}{2000\ \text{DAC-hr}} = 0.0072\ \text{Sv}$$

As an alternative solution, the dose equivalents may be calculated from the intake. The intake from this procedure is:

$$I = C\ \text{BR}\ t$$

where

I = intake of I-125
C = I-125 room air concentration = 0.192 MBq/m^3
BR = breathing rate = 1.2 m^3/hr
t = exposure time = 3 hr
I = (0.192 MBq/m^3) (1.2 m^3/hr) (3 hr) = 0.691 MBq

The dose equivalents are obtained from the intake:

$$\text{CDE} = \frac{I}{\text{ALI-NS}}\,0.50\ \text{Sv} = (0.691\ \text{MBq}/2\ \text{MBq})\,0.5\ \text{Sv} = 0.173\ \text{Sv}$$

$$\text{CEDE} = \frac{I}{\text{ALI-S}}\,0.05\ \text{Sv} = (0.691\ \text{MBq}/4\ \text{MBq})\,0.05\ \text{Sv} = 0.00864\ \text{Sv}$$

The differences in the two solutions are attributed to the fact that the ICRP in Publications 26 and 30 retains only one significant digit for the ALI and DAC values. No limits were exceeded. The respective limits are 5 rem for the CEDE and 50 rem for the CDE.

This problem can also be solved by considering the number of DAC-hr (N) received in the 3 hr period. Using the limiting non-stochastic DAC (0.001 MBq/m^3):

$$N = \frac{C\,T}{\text{DAC}_{\text{NS}}} = \frac{(0.192\ \text{MBq/m}^3)\,(3\ \text{hr})}{(0.001\ \text{MBq/m}^3\text{-DAC})} = 576\ \text{DAC-hr}$$

Again, no dose limits are exceeded because the number of DAC-hr is less than the limit of 2000 DAC-hr per year.

Question 3.58

One of the laboratory technicians is a declared pregnant worker. Three weeks following a procedure, she showed a decay-corrected intake (I) of 0.27 MBq, with minimal deep dose equivalent. Using the methodology of the previous problem, the CDE and CEDE are:

$$\text{CDE} = \frac{I}{\text{ALI-NS}}\,0.50\ \text{Sv} = (0.27\ \text{MBq}/2\ \text{MBq})\,0.50\ \text{Sv} = 0.0675\ \text{Sv}$$

$$\text{CEDE} = \frac{I}{\text{ALI-S}}\,0.05\ \text{Sv} = (0.27\ \text{MBq}/4\ \text{MBq})\,0.05\ \text{Sv} = 0.00338\ \text{Sv} = 3.38\ \text{mSv}$$

An alternate solution is available by using the dose conversion factors:

CEDE/unit intake = $(6.53 \times 10^{-9}$ Sv/Bq) $(1 \times 10^6$ Bq/MBq)
= 6.53×10^{-3} Sv/MBq

Thyroid dose/unit intake = $(2.16 \times 10^{-7}$ Sv/Bq) $(1 \times 10^6$ Bq/MBq)
= 0.216 Sv/MBq

$$\begin{aligned} \text{CDE} = I\,(\text{CDC/unit intake}) &= (0.27\ \text{MBq})\,(0.216\ \text{Sv/MBq}) \\ &= 0.0583\ \text{Sv} \end{aligned}$$

$$\begin{aligned} \text{CEDE} = I\,(\text{CEDE/unit intake}) &= (0.27\ \text{MBq})\,(6.53 \times 10^{-3}\ \text{Sv/MBq}) \\ &= 1.76 \times 10^{-3}\ \text{Sv} \end{aligned}$$

The NCRP's fetal dose recommendation for the pregnancy is provided in NCRP Report No. 116. For occupational situations, the NCRP recommends a monthly equivalent dose limit of 0.5 mSv to the embryo/fetus (excluding medical and natural background) once the pregnancy is known (declared).

A dose conversion factor was not provided for a fetus. The fetal dose can be approximated by the gonad dose conversion factor (1.84×10^{-11} Sv/Bq):

$$\begin{aligned} \text{CDE (Fetus)} &= (0.27\ \text{MBq})\,(1.84 \times 10^{-11}\ \text{Sv/Bq})\,(1.0 \times 10^{6}\ \text{Bq/MBq}) \\ &= 4.97 \times 10^{-6}\ \text{Sv}\ (1000\ \text{mSv/Sv}) \\ &= 4.97 \times 10^{-3}\ \text{mSv} \end{aligned}$$

Therefore, the NCRP-116 fetal dose recommendation was not exceeded.

Question 3.59

Radiological controls that could be established to reduce personnel exposures during the procedures using I-125 include:

1. Use shadow shielding to reduce the direct exposure.
2. Add ventilation to increase the removal of airborne I-125.
3. Use the I-125 in a basic solution to minimize its volatilization.
4. Perform the procedures in a confinement or hood to minimize both internal and external doses.
5. Train personnel to perform the task in a more efficient manner.
6. Use tongs or tools to minimize the external dose by increasing the distance to the target tissue.
7. If engineering controls are not feasible, use respiratory protection to minimize the intake if respirator use is consistent with ALARA principles.
8. Use a blocking agent to minimize the I-125 intake.
9. Use protective clothing to minimize skin absorption of I-125.
10. Use face shields or safety glasses to minimize the eye dose.
11. Perform periodic screening of worker's thyroids to investigate and correct workplace conditions that could lead to a significant intake.
12. Review iodination procedures to minimize releases to the local environment.
13. Improve the face velocity in the fume hood to minimize releases.
14. Use alarming air monitors to investigate and correct workplace conditions that could lead to a significant intake.

Question 3.60

An air-conditioning repairman enters Room #3 for the final hour of the chromatography procedure and stays in the room for an additional 2 hours. The repairman is an unmonitored member of the general public. His DAC-hours of exposure (E) are calculated from the relationship:

$$E\ (\text{total}) = E_1\ (\text{hour 1}) + E_2\ (\text{hours 2 and 3})$$

where the repairman is exposed to a constant concentration during hour 1 due to the LPLC procedure and an exponentially decreasing concentration during hours 2 and 3:

$$E_1 = \frac{C(0)\,t}{\text{DAC-S}}$$

where

$C(0)$ = I-125 room air concentration = 0.1 MBq/m^3
t = exposure time (hour 1) = 1 hr
DAC-S = stochastic DAC value = 0.002 MBq/m^3

$$E_1 = \frac{(0.1\ \text{MBq/m}^3)\,(1\ \text{hr})}{(0.002\ \text{MBq/m}^3\text{-DAC})} = 50\ \text{DAC-hr}$$

$$E_2 = \frac{\int_0^T C(0)\,e^{-kt}\,dt}{\text{DAC-S}} = \frac{\frac{C(0)}{k}\left(1 - e^{-kT}\right)}{\text{DAC-S}}$$

where

k = total removal rate
T = 2 hour exposure time (hours 2 and 3):

$$k = \lambda + \frac{F}{V}$$

where

F = Room No. 3 exhaust rate = 15 m^3/min
V = Room No. 3 volume = 60 m^3

$$\begin{aligned} k &= \lambda + \frac{F}{V} \\ &= \frac{0.693}{(60.14\ \text{d})\,(24\ \text{hr/d})} + \left(\frac{15\ \text{m}^3/\text{min}}{60\ \text{m}^3}\right)\left(\frac{60\ \text{min}}{\text{hr}}\right) \\ &= 4.80 \times 10^{-4}/\text{hr} + 15/\text{hr} = 15/\text{hr} \end{aligned}$$

$$E_2 = \frac{\dfrac{0.1\ \text{MBq/m}^3}{15/\text{hr}}\left[1 - e^{-(15/\text{hr})\,(2\ \text{hr})}\right]}{0.002\ \text{MBq/m}^3\text{-DAC}} = 3.33\ \text{DAC-hr}$$

$$E = E_1 + E_2 = 50\ \text{DAC-hr} + 3.33\ \text{DAC-hr} = 53.3\ \text{DAC-hr (Stochastic)}$$

The CEDE is

$$\text{CEDE} = \left[(53.3\ \text{DAC-hr})\left(\frac{0.05\ \text{Sv}}{2000\ \text{DAC-hr}}\right)\right] = 0.00133\ \text{Sv} = 1.33\ \text{mSv}$$

Public exposures are limited to 1 mSv CEDE in a year. Therefore, the exposure exceeds the 1 mSv limit for a member of the public.

Solutions for Chapter 4

Scenario 4.1

Question 4.1

The intake based on the CAM concentration estimate is

$$\begin{aligned} I &= k\,t\,(\mathrm{BR})\,C \\ &= (80\ \mathrm{s})\,(20\ \mathrm{liters/min})\,(1\ \mathrm{min/60\ s})\,(1000\ \mathrm{ml/liter})\,(0.74\ \mathrm{Bq/ml}) \\ &= 1.97 \times 10^{4}\ \mathrm{Bq} \end{aligned}$$

Calculation of organ dose equivalents:

$$H = I \sum_i f_i\,(\mathrm{DCF}_i)$$

$$\begin{aligned} H\,(\mathrm{lung}) = 1.97 \times 10^{4}\ \mathrm{Bq}\,[&(1.2 \times 10^{-2})\,(7.0 \times 10^{-5}\ \mathrm{Sv/Bq}) \\ &+ (1.4 \times 10^{-1})\,(6.7 \times 10^{-5}\ \mathrm{Sv/Bq}) \\ &+ (3.1 \times 10^{-2})\,(6.7 \times 10^{-5}\ \mathrm{Sv/Bq}) \\ &+ (8.1 \times 10^{-1})\,(7.0 \times 10^{-9}\ \mathrm{Sv/Bq}) \\ &+ (1.5 \times 10^{-3})\,(1.8 \times 10^{-5}\ \mathrm{Sv/Bq})] \\ = 0.242\ \mathrm{Sv} \end{aligned}$$

$$\begin{aligned} H\,(\mathrm{liver}) = 1.97 \times 10^{4}\ \mathrm{Bq}\,[&(1.2 \times 10^{-2})\,(9.7 \times 10^{-7}\ \mathrm{Sv/Bq}) \\ &+ (1.4 \times 10^{-1})\,(9.2 \times 10^{-7}\ \mathrm{Sv/Bq}) \\ &+ (3.1 \times 10^{-2})\,(9.2 \times 10^{-7}\ \mathrm{Sv/Bq}) \\ &+ (8.1 \times 10^{-1})\,(1.1 \times 10^{-9}\ \mathrm{Sv/Bq}) \\ &+ (1.5 \times 10^{-3})\,(1.5 \times 10^{-6}\ \mathrm{Sv/Bq})] \\ = 3.4 \times 10^{-3}\ \mathrm{Sv} \end{aligned}$$

$$
\begin{aligned}
H\ (\text{bone surfaces}) = 1.97 \times 10^{4}\ \text{Bq}\ [&(1.2 \times 10^{-2})\,(3.5 \times 10^{-6}\ \text{Sv/Bq}) \\
&+ (1.4 \times 10^{-1})\,(3.5 \times 10^{-6}\ \text{Sv/Bq}) \\
&+ (3.1 \times 10^{-2})\,(3.5 \times 10^{-6}\ \text{Sv/Bq}) \\
&+ (8.1 \times 10^{-1})\,(3.8 \times 10^{-9}\ \text{Sv/Bq}) \\
&+ (1.5 \times 10^{-3})\,(5.4 \times 10^{-5}\ \text{Sv/Bq})] \\
= 0.0142\ &\text{Sv}
\end{aligned}
$$

$$
\begin{aligned}
H\ (\text{red bone marrow}) = 1.97 \times 10^{4}\ \text{Bq}\ [&(1.2 \times 10^{-2})\,(3.0 \times 10^{-7}\ \text{Sv/Bq}) \\
&+ (1.4 \times 10^{-1})\,(2.7 \times 10^{-7}\ \text{Sv/Bq}) \\
&+ (3.1 \times 10^{-2})\,(2.7 \times 10^{-7}\ \text{Sv/Bq}) \\
&+ (8.1 \times 10^{-1})\,(2.7 \times 10^{-10}\ \text{Sv/Bq}) \\
&+ (1.5 \times 10^{-3})\,(4.3 \times 10^{-6}\ \text{Sv/Bq})] \\
= 1.1 \times 10^{-3}\ &\text{Sv}
\end{aligned}
$$

Question 4.2

$$
\begin{aligned}
H = \sum_{T} W_T\ H_T\ &= (0.242\ \text{Sv})\,(0.12)_{\text{lung}} + (3.4 \times 10^{-3}\ \text{Sv})\,(0.06)_{\text{liver}} \\
&+ (0.0142\ \text{Sv})\,(0.03)_{\text{bone surfaces}} + (1.1 \times 10^{-3}\ \text{Sv})\,(0.12)_{\text{RBM}} \\
&= 0.0298\ \text{Sv}
\end{aligned}
$$

Question 4.3

Stochastic Limit: 5 rem/year total effective dose equivalent.

Nonstochastic Limit: 50 rem/year committed dose equivalent for the organs of interest for this intake.

Neither limit was exceeded.

Question 4.4

Extensive bioassay follow-up will be required. This will include urinalysis, fecal analysis, and *in vivo* lung counting. All voids should be collected the first week for Pu and Am analysis. A routine frequency should be established for several months thereafter until elimination functions are established.

No clinical symptoms are expected for these dose levels.

Scenario 4.2

Question 4.5

The intake activity (A) can be defined in terms of the count rate on the filter paper:

$$A = \mathrm{CR}/e\,k$$

Knowing the activity permits the determination of the average air concentration ($\overline{C}$) inhaled by the worker:

$$\overline{C} = A/f\,t$$

The intake I can be estimated using the relationship

$$I = (\mathrm{CR})\,(\mathrm{BR})/e\,k\,f\,(\mathrm{PF})$$

which may be written in terms of the available information given in the problem:

$$I = (\overline{C})\,(\mathrm{BR})\,t/\mathrm{PF}$$

Using these relationships, the intakes for workers "A" and "B" can be determined:

Worker A:

$$A = \frac{30\,000\ \mathrm{cpm}}{(0.3\ \mathrm{cpm/dpm})\,(2.2 \times 10^6\ \mathrm{dpm/\mu Ci})} = 4.55 \times 10^{-2}\ \mu\mathrm{Ci\ on\ the\ filter}$$

$$\overline{C} = \frac{4.55 \times 10^{-2}\ \mu\mathrm{Ci}}{(2\ \mathrm{liters/min})\,(10\ \mathrm{min})}$$

$$= 2.3 \times 10^{-3}\ \mu\mathrm{Ci/liter\ average\ inhaled\ concentration}$$

$$I = (2.3 \times 10^{-3}\ \mu\mathrm{Ci/liter})\,(20\ \mathrm{liters/min})\,(10\ \mathrm{min})/(10)$$

$$= 4.5 \times 10^{-2}\ \mu\mathrm{Ci\ intake}$$

Worker B:

$$A = \frac{20\,000\ \mathrm{cpm}}{(0.3\ \mathrm{cpm/dpm})\,(2.2 \times 10^6\ \mathrm{dpm/\mu Ci})} = 3.03 \times 10^{-2}\ \mu\mathrm{Ci\ on\ the\ filter}$$

$$\overline{C} = \frac{3.03 \times 10^{-2}\ \mu\mathrm{Ci}}{(2\ \mathrm{liters/min})\,(10\ \mathrm{min})}$$

$$= 1.5 \times 10^{-3}\ \mu\mathrm{Ci/liter\ average\ inhaled\ concentration}$$

$$I = (1.5 \times 10^{-3}\ \mu\mathrm{Ci/liter})\,(20\ \mathrm{liters/min})\,(10\ \mathrm{min})/(1.0)$$

$$= 3.0 \times 10^{-3}\ \mu\mathrm{Ci\ inhaled}$$

Question 4.6

The material is insoluble, and therefore class Y values for retention and elimination functions and dose conversion factors are appropriate.

For class Y material, the deposition in the pulmonary lung will result in a long-term dose contribution. The long-term activity in the pulmonary lung will be

$$A_L = I\, D_P\, d_Y = \text{long-term pulmonary deposition (μCi)}$$

where

I = intake (μCi)

D_P = deposition factor which depends on the particle size and the location of the deposition (for pulmonary deposition of 1-μm particles, $D_P = 0.25$)

d_Y = class Y long-term retention function

= sum of long-term compartment F values (i.e., those with $T = 500$ days)

= $F_e + F_g + F_h = 0.05 + 0.4 + 0.15 = 0.60$

From ICRP-30, for the Pulmonary Region

Region	Compartment	Class Y	
		T (day)[a]	F[b]
Pulmonary	e	500	0.05
	f	1.0	0.40
	g	500	0.40
	h	500	0.15

[a] T = compartment removal half-times.
[b] F = compartmental fractions.

The long-term pulmonary dose is given by

$$D = A_L \sum_i f_i\, (\mathrm{DCF}_i)$$

where

D = long-term pulmonary dose (rem)

i = Number of isotopes in the air mixture inhaled by the worker

f_i = isotopic fraction in the mixture

DCF_i = inhalation lung dose equivalent conversion factors (rem/μCi)

For this problem, the following are to be used

Isotope	f_i	DCF_i (rem/μCi) (first year)	DCF_i (rem/μCi) (50 years)
U-234	0.10	230	1100
U-235	0.02	210	1000
U-238	0.88	200	980

Using these relationships, the desired information may be obtained.

Worker A:

$$A_L = (4.55 \times 10^{-2}\ \mu\text{Ci})\,(0.25)\,(0.6) = 6.8 \times 10^{-3}\ \mu\text{Ci}$$
$$D = [(0.1)\,(1100) + (0.02)\,(1000) + (0.88)\,(980)]\ \text{rem}/\mu\text{Ci}\ (6.8 \times 10^{-3}\ \mu\text{Ci})$$
$$= 6.75\ \text{rem committed dose equivalent}$$

Worker B:

$$A_L = (3.03 \times 10^{-1}\ \mu\text{Ci})\,(0.25)\,(0.6) = 4.5 \times 10^{-2}\ \mu\text{Ci}$$
$$D = [(0.1)\,(1100) + (0.02)\,(1000) + (0.88)\,(980)]\ \text{rem}/\mu\text{Ci}\ (4.5 \times 10^{-2}\ \mu\text{Ci})$$
$$= 44.6\ \text{rem committed dose equivalent}$$

Question 4.7

The percentage is represented by a ratio of DCF values:

$$\%\ \text{DCE} = \frac{\sum_i f_i\ \text{DCF}_i\ (\text{first year})}{\sum_i f_i\ \text{DCF}_i\ (50\ \text{years})}$$
$$= \frac{(0.1)\,(230) + (0.02)\,(210) + (0.88)\,(200)}{(0.1)\,(1100) + (0.02)\,(1000) + (0.88)\,(980)} = 20.5\%$$

Question 4.8

$$H_E = \sum_T W_T\ H_T \leq 5\ \text{rem effective dose (stochastic dose limit)}$$

Using this information, worker doses can be determined:

Worker A: $(6.75\ \text{rem})\,(0.12) = 0.81\ \text{rem} < 5\ \text{rem}$
Worker B: $(44.6\ \text{rem})\,(0.12) = 5.35\ \text{rem} > 5\ \text{rem}$

Only Worker B exceeds the stochastic limit for the lung (5 rem/0.12 = 41.6 rem). The nonstochastic limit is 50 rem, which is not exceeded by either worker.

Question 4.9

Extensive bioassay follow-up is required. This will include urinalysis, fecal analysis, and *in vivo* lung counting. All voids should be collected for the first week (isotopic uranium, thorium, and radium analyses), and a routine frequency is established for several months thereafter. *In vivo* lung counting should be performed as soon as possible with repeated analysis every few months until elimination functions are determined.

Scenario 4.3

Question 4.10

Ionizing Radiation Hazards

1. X-rays are produced from the high-voltage (30 and 50 kV) power supplies. Extremity dosimeters and low-energy X-ray sensitive dosimeters would be part of a radiation monitoring program.
2. Airborne alpha and beta hazards exist during maintenance operations.
3. The dust/debris resulting from cleaning the ionization chamber may present disposal problems due to the quantity of TRU present.
4. Criticality concerns from the generation of highly enriched product cause potential gamma and neutron hazards.
5. Alpha/beta hazards exist from the raw feed material and enriched product.

Nonionizing Radiation Hazards

1. High-intensity laser beams can damage the skin and eyes.
2. Unshielded high-strength electromagnetic fields may cause biological injury.
3. Reflected laser light from the photoexcitation process may injure the eyes.

Other Hazards

1. Heat buildup from the interaction of the electron beam with the uranium feed material.
2. Noise hazards due to the high-energy systems transporting significant quantities of matter.

Question 4.11

Engineering design features to minimize ionizing radiation hazards include the following:

1. X-ray hazards can be minimized with proper shielding.
2. Facility design should position high-voltage equipment away from locations where personnel will be routinely located.
3. Particulate alpha and beta airborne concentrations can be minimized with proper airflow and ventilation system design.
4. Air should be HEPA filtered and not recirculated from higher airborne concentration areas to lower concentration areas.
5. Proper sizing and arrangement of transfer lines and storage containers will reduce the probability of a criticality accident.
6. Alarming air monitors should be installed in areas where uranium dust is or may be present.
7. Criticality alarms should be installed with detectors properly located in enriched materials storage areas or near enriched material transport areas.
8. Glove boxes and fume hoods should be designed to facilitate chamber operations, maintenance, and cleaning.

Question 4.12

Engineering design features to minimize nonionizing radiation hazards include the following:

1. Beam tubes and optical transport systems should be designed to keep laser radiation out of occupied areas.
2. Interlocks should be used in all areas of high-intensity laser radiation.
3. Areas with high-strength electromagnetic fields should be contained in separate rooms with strict access controls.

Question 4.13

In order to determine the dose to the technician, the following assumptions are made:

1. The buildup from the polyethylene shield is negligible.
2. The average energy of the photons and neutrons is representative of the spectrum, and therefore monoenergetic energies can be used in the calculation.
3. The use of dose conversion factors is appropriate for the total dose and dose rate for the neutron and gamma energies of this problem.
4. A point-source approximation is applicable due to the small size of the critical mass (source) and the distance of the technician from the source.

Neutron Dose: The unattenuated neutron dose is

$$\begin{aligned} D_{\mathrm{n}}^{0} &= \frac{(1.0 \times 10^{16}\ \text{fissions})\,(3\ \text{neutrons/fission})}{(4)\,(3.14)\,(20\ \text{ft} \times 30.48\ \text{cm/ft})^{2}} \\ &\quad \times [(2.5\ \text{mrad/hr})/(20\ \text{n/cm}^{2}\text{-s})]\,(1\ \text{hr}/3600\ \text{s}) \\ &= 2.23 \times 10^{5}\ \text{mrad} \end{aligned}$$

Because the neutron attenuation is 0.3, for 2.5-MeV neutrons passing through 4 in. of polyethylene, the neutron dose to the technician is

$$D_{\mathrm{n}} = (0.3)\,(2.23 \times 10^{5}\ \text{mrad})\,(1\ \text{rad}/1000\ \text{mrad}) = 66.9\ \text{rad}$$

Gamma Dose: The unattenuated gamma dose is

$$\begin{aligned} D_{\mathrm{g}}^{0} &= \frac{(1.0 \times 10^{16}\ \text{fissions})\,(8\ \text{gammas/fission})}{(4)\,(3.14)\,(20\ \text{ft} \times 30.48\ \text{cm/ft})^{2}} \\ &\quad \times [(1.0\ \text{R/hr})/(5.5 \times 10^{5}\ \text{gammas/cm}^{2}\text{-s})]\,(1\ \text{hr}/3600\ \text{s}) \\ &= 8.65\ \text{R} \end{aligned}$$

The attenuated gamma dose is

$$\begin{aligned} D_g &= D_g^0 \exp(-ut) \\ &= 8.65 \text{ R } \exp[-(0.0727 \text{ cm}^2/\text{g})\,(4 \text{ in.})\,(2.54 \text{ cm/in.})\,(1.4 \text{ g/cm}^3)] \\ &= 3.08 \text{ R} \times 0.95 \text{ rad/R} = 2.93 \text{ rad} \\ D_{\text{total}} &= 66.9 \text{ rad} + 2.9 \text{ rad} = 69.8 \text{ rad} \end{aligned}$$

Scenario 4.4

Question 4.14

Assumptions for questions 4.14 and 4.15:

1. The criticality is adequately represented by a point source.
2. The buildup from the polyethylene shield is small; that is, $B = 1.0$.
3. The neutron and gamma spectra are adequately represented by their mean energies.
4. The neutron and gamma dose conversion factors are representative of their associated spectra.

The general equation for the neutron dose is

$$\begin{aligned} D_n &= N_{\text{fissions}}\, \nu\, k \exp(-ut)\, B/(4\,\pi)\, r^2 \\ k &= \text{conversion factors} \\ &= (2.5 \text{ mrem/hr}/20 \text{ neutrons/cm}^2\text{-s}) \times (1 \text{ hr}/3600 \text{ s}) \end{aligned}$$

Unattenuated Neutron Dose (D_n^0)**:**

$$\begin{aligned} D_n^0 &= \frac{(1.0 \times 10^{16} \text{ fissions})\,(3 \text{ neutrons/fission})}{(4)\,(3.14)\,(10 \text{ ft} \times 12 \text{ in./ft} \times 2.54 \text{ cm/in.})^2} \\ &\quad \times \frac{2.5 \text{ mrem/hr}}{20 \text{ neutrons/cm}^2\text{-s}} \frac{1 \text{ hr}}{3600 \text{ s}} \frac{1 \text{ rem}}{1000 \text{ mrem}} \\ &= 892.7 \text{ rem} \end{aligned}$$

Attenuated Dose (D_n)**:**
Because $B \exp(-ut)$ may be replaced by the neutron attenuation factor, the attenuated dose is written as

$$\begin{aligned} D_n &= D_n^0 \times (\text{neutron attenuation for 2.5 MeV and 12 in. of polyethylene}) \\ &= (892.7 \text{ rem})\,(0.005) = 4.46 \text{ rem} \end{aligned}$$

Question 4.15

The general equation for the gamma dose is

$$D_g = N_{\text{fissions}}\ \nu\ k\ \exp(-ut)\ B/4\ \pi\ r^2$$

Unattenuated Gamma Exposure (X_g^0):

$$X_g^0 = \frac{(1.0 \times 10^{16}\ \text{fissions})\ (8\ \text{gammas/fission})}{(4)\ (3.14)\ (10\ \text{ft} \times 12\ \text{in./ft} \times 2.54\ \text{cm/in.})^2} \times \frac{1\ \text{hr}}{3600\ \text{s}} \frac{1\ \text{R/hr}}{5.5 \times 10^5\ \text{gammas/cm}^2\text{-s}}$$
$$= 34.6\ \text{R}$$

Attenuated Exposure (X_g):

$$X_g = X_g^0\ e^{-ut}$$
$$= (34.6\ \text{R})\ \exp(-0.0727\ \text{cm}^2/\text{g} \times 12\ \text{in.} \times 2.54\ \text{cm/in.} \times 1.4\ \text{g/cm}^3)$$
$$= 1.56\ \text{R}$$

Gamma dose (D_g):

$$D_g = (1.56\ \text{R})\ (0.95\ \text{rad/R})\ (1\ \text{rem/rad}) = 1.48\ \text{rem}$$

Question 4.16

Because the alarm set point is 500 mR/hr and the detector responds to 1/2500 of the actual gamma exposure rate, the actual exposure rate which must exist at the detector to trip an alarm is X_t:

$$X_t/2500 = 500\ \text{mR/hr}$$
$$X_t = 1250\ \text{R/hr}$$

Given that 10^{15} fissions yields a gamma exposure of 2.5 R at 6.0 ft, 1.0×10^{16} fissions will yield an exposure of 25 R at 6.0 ft. Assuming that the 1.0×10^{16} fissions occur uniformly in time during the 1.0-ms transient, the effective exposure rate at 6.0 ft is

$$\dot{X} = 25\ \text{R}/0.001\ \text{s} = 2.5 \times 10^4\ \text{R/s} \times 3600\ \text{s/hr}$$
$$= 9.0 \times 10^7\ \text{R/hr @ 6.0 ft}$$

Assuming that the criticality is adequately represented by a point source at the detector, the inverse-square law is applicable:

$$(\dot{X})\, r^2 = \text{constant}$$
$$(9.0 \times 10^7 \text{ R/hr})\,(6.0 \text{ ft})^2 = (1250 \text{ R/hr})\,(d^2)$$

where d is the maximum distance at which the detector will alarm at the alarm set point. Solving the equation yields $d = 1610$ ft.

Scenario 4.5

Question 4.17

Characteristics of raw oils that make collecting a representative sample a difficult task include:

1. Water contamination.
2. Oils having different density or stratification.
3. Sludge.
4. Various solubilities for different chemical compositions and radionuclides.
5. Contaminants or other liquids discarded in oil collection drums.

Question 4.18

Acceptable methods for obtaining representative samples from the tank are as follows:

1. Recirculation of the tank such that its volume is recirculated at least three times.
2. Collecting a composite of repetitive dip samples taken at various depths.

Question 4.19

Analysis methods for characterizing the waste oil include:

1. Radiochemical analysis to determine the radionuclide content.
2. Chemical toxicity testing to determine the presence of hazardous materials.

Question 4.20

General survey techniques for trash release from the RCA include:

1. Beta and gamma surface contamination monitoring with state-of-the-art thin window detectors.
2. Gamma monitoring of the disposal package with state-of-the-art low-background instrumentation.

Question 4.21

The regulatory accepted limit for release is that no detectable activity above background is present.

Question 4.22

The radiation exposure pathways include:

1. Direct exposure from surface deposition.
2. Consumption of fish living in the lagoon.
3. Well-water consumption.
4. Inhalation of resuspended dust.
5. Consumption of garden vegetables.
6. Consumption of dairy cow's milk from the grass–cow pathway.
7. Consumption of beef from the cattle grazing pathway.

Question 4.23

The exposure to the maximum exposed individual should be less than 5 mrem/year.

Question 4.24

Six general environmental categories that impact approval of alternate disposal methods are as follows:

1. Topographical.
2. Geological.
3. Meteorological.
4. Hydrological.
5. Ground- and surface-water usage.
6. Nature and location of other industrial facilities.

Question 4.25

Other types of information which must be included in the application are as follows:

1. Radiation control procedures used to minimize exposure.
2. Description of licensed radioactive material.
3. Other types of nonlicensed radioactive material.
4. Quantity, type, and chemical form of radioactive material.
5. Levels of radioactivity.
6. Proposed manner and conditions of disposal.
7. Alternative methods of disposal including their economic and radiological impacts.

Scenario 4.6

Question 4.26

The committed dose equivalent to the thyroid from the I-131 intake is determined from the definition of the ALI:

$$H_T = I_{\text{I-131}} \frac{50 \text{ rem}}{\text{ALI (I-131)}}$$

where

$I_{\text{I-131}}$ = intake of I-131 = 19.5 μCi
ALI (I-131) = nonstochastic annual limit on intake for I-131 = 50 μCi

$$H_T = 19.5 \text{ μCi} \frac{50 \text{ rem}}{50 \text{ μCi}} = 19.5 \text{ rem}$$

Question 4.27

The committed effective dose equivalent (CEDE) from the I-131 and Cs-137 intakes is determined from the sum of the dose equivalents from each isotope:

$$H_E = 5 \text{ rem} \left(\frac{I_{\text{I-131}}}{\text{SALI (I-131)}} + \frac{I_{\text{Cs-137}}}{\text{SALI (Cs-137)}} \right)$$
$$= 5 \text{ rem} \left(\frac{19.5 \text{ μCi}}{200 \text{ μCi}} + \frac{80.0 \text{ μCi}}{200 \text{ μCi}} \right) = 2.49 \text{ rem}$$

where the following assumptions are made:

1. The CEDE is evaluated over a 50-year period.
2. The 27-year-old worker has the characteristics of reference man.
3. The 1-μm AMAD particle size is applicable to this uptake.
4. Class D aerosols are representative of the inhaled particles.

The CEDE can also be determined from the relationship:

$$H_E = \sum_T w_T H_T = w_{\text{thyroid}} H_{\text{thyroid}} + w_{\text{wb}} H_{50,\text{wb}}$$

where w_{wb} is the whole-body weighting factor which has a value of 1.0. Using the values in the problem statement,

$$H_E = (0.03) \left(\frac{19.5 \text{ μCi}}{50 \text{ μCi}} 50 \text{ rem} \right) + (1.00) \left(\frac{80.0 \text{ μCi}}{200 \text{ μCi}} 5 \text{ rem} \right)$$
$$= 0.585 \text{ rem} + 2.00 \text{ rem} = 2.59 \text{ rem}$$

The two methods yield slightly different results because ICRP-26 expresses ALI values using only one significant figure.

Question 4.28

To determine if the NCRP-116 occupational dose limits have been exceeded, the limit and the corresponding quantity for this event must be determined.

The NCRP-116 annual dose recommendations include:

1. A total effective dose equivalent (TEDE) less than or equal to 5 rem
2. A 50-year committed dose equivalent (CDE) of 50 rem or less.

TEDE:

$$\text{TEDE} = H_{\text{DDE}} + H_{\text{E}}\ (50\ \text{year})$$

where
H_{DDE} = the deep dose equivalent = (0.7 + 1.2 + 0.8 + 0.3) rem = 3.0 rem
H_{E} (50 year) was obtained in question 4.27.
TEDE = 3.0 rem + 2.49 rem = 5.49 rem
The TEDE exceeds the NCRP-116 recommendation.

CDE:
The dose equivalent (H_{T}) assigned to the thyroid is as follows:

$$H_{\text{T}} = H_{\text{DDE (thyroid)}} + H_{\text{T}}\ (50\ \text{year}) = 3.0\ \text{rem} + 19.5\ \text{rem} = 22.5\ \text{rem}$$

H_{T} does not exceed the NCRP-116 recommendations.

The external dose contribution to the thyroid has not accounted for tissue attenuation prior to radiation reaching the thyroid.

Scenario 4.7

Question 4.29

The worker exposure is most easily determined using the thin disk source approximation:

$$\dot{X}(h) = \pi\, G\, C_{\text{a}}\, \ln \frac{r^2 + h^2}{h^2}$$

where
G = gamma constant = $0.33\ \dfrac{\text{R-m}^2}{\text{hr-Ci}}$
(only Cs-137 contributes significantly to the external exposure)

C_a = gamma emitter isotopic concentration per unit area = A/a
A = total activity (Ci) = C (Cs-137) × V
C (Cs-137) = Cs-137 waste concentration (Ci/liter) = 0.2 Ci/liter
V = spill volume = 100 liters
A = 0.2 Ci/liter × 100 liters = 20 Ci
a = spill area (m^2)
= $\pi r^2 = (3.14)\ (2.5\ m)^2 = 19.625\ m^2$
C_a = 20 Ci/19.625 m^2 = 1.02 Ci/m^2
r = radius of the circular spill = 2.5 m
h = distance above spill central axis = 1.0 m

Using these data permits the determination of the gamma exposure rate:

$$\dot{X} = (3.14)\left(0.33\ \frac{\text{R-m}^2}{\text{hr-Ci}}\right)(1.02\ \text{Ci/m}^2)\ \ln\frac{(2.5\ \text{m})^2 + (1.0\ \text{m})^2}{(1.0\ \text{m})^2}$$
$$= 2.09\ \text{R/hr}$$

Assuming 1 R/hr = 1 rem/hr:

$$\dot{D} = 2.09\ \text{R/hr}\ \frac{\text{rem/hr}}{\text{R/hr}} = 2.09\ \text{rem/hr}$$

The external dose equivalent is easily calculated because the exposure time is known:

$$D = \dot{D}\ t = (2.09\ \text{rem/hr})\ (1.5\ \text{hr}) = 3.14\ \text{rem}$$

Question 4.30

The effective dose equivalents may be obtained from a knowledge of the isotopic air concentration, breathing rate, exposure time, and dose conversion factors.

Isotopic Air Concentrations:

$$C_i = \frac{k\ (\text{CR})_i}{(\text{FR})\ T}$$

where
C_i = isotopic concentration of the i th radionuclide
i = 1, gamma count rate (dpm) for Cs-137
= 2, beta count rate (dpm) for Sr-90
= 3, alpha count rate (dpm) for Pu-238
FR = flow rate of air sampler = 2 liters/min
T = exposure time = 90 min
k = conversion factor = $1/2.2 \times 10^6$ dpm/μCi

$$\text{Cs-137: } C_1 = \frac{1.0 \times 10^5 \text{ dpm}}{(2.2 \times 10^6 \text{ dpm}/\mu\text{Ci})\,(2 \text{ liter/min})\,(90 \text{ min})} = 2.53 \times 10^{-4} \ \mu\text{Ci/liter}$$

$$\text{Sr-90: } C_2 = \frac{(5.0 \times 10^3 \text{ dpm})}{(2.2 \times 10^6 \text{ dpm}/\mu\text{Ci})\,(2 \text{ liter/min})\,(90 \text{ min})} = 1.26 \times 10^{-5} \ \mu\text{Ci/liter}$$

$$\text{Pu-238: } C_3 = \frac{5.0 \times 10^3 \text{ dpm}}{(2.2 \times 10^6 \text{ dpm}/\mu\text{Ci})\,(2 \text{ liter/min})\,(90 \text{ min})} = 1.26 \times 10^{-5} \ \mu\text{Ci/liter}$$

The first-year dose equivalent D_1 may be determined from the following relationship:

$$D_1 = \sum_i C_i \,(\text{BR})\, T\, \text{DCF}_i$$

where

D_1 = first-year dose equivalent
= $(2.53 \times 10^{-4}$ μCi/liter) (20 liters/min) (90 min) $(2.9 \times 10^{-2}$ rem/μCi)
+ $(1.26 \times 10^{-5}$ μCi/liter) (20 liters/min) (90 min) (0.32 rem/μCi)
+ $(1.26 \times 10^{-5}$ μCi/liter) (20 liters/min) (90 min) (30 rem/μCi)
= 0.013 rem + 0.007 rem + 0.680 rem = 0.700 rem

Question 4.31

The 50-year committed effective dose equivalent is obtained in an analogous manner:

H_{50} = 50-year committed effective dose equivalent
= $(2.53 \times 10^{-4}$ μCi/liter) (20 liters/min) (90 min) $(3.2 \times 10^{-2}$ rem/μCi)
+ $(1.26 \times 10^{-5}$ μCi/liter) (20 liters/min) (90 min) (1.30 rem/μCi)
+ $(1.26 \times 10^{-5}$ μCi/liter) (20 liters/min) (90 min) (310 rem/μCi)
= 0.015 rem + 0.029 rem + 7.031 rem = 7.075 rem

Question 4.32

Standards

10CFR20. The committed effective dose equivalent (CEDE) plus annual deep dose equivalent (external dose) (DDE) or total effective dose equivalent (TEDE) is limited to 5.0 rem.

TEDE = CEDE + DDE
= 3.14 rem + 7.08 rem = 10.22 rem

The 10CFR20 TEDE limit is exceeded.

DOE Order 5480.11. The 5-rem DOE limit is the sum of the external dose and the internal dose equivalent for the year of interest.

$$D_{DOE} = 3.14 \text{ rem} + 0.7 \text{ rem} = 3.84 \text{ rem}$$

The DOE 5480.11 criterion is not exceeded.

DOE Radiological Controls Manual. This manual specifies the use of the TEDE limited to 5 rem. The 50-year CEDE is added to the external deep dose equivalent and is compared to the 5-rem limit. The limit was exceeded. See the 10CFR20 discussion.

10CFR835. Limits are the same as stated in the *DOE Radiological Controls Manual*.

Question 4.33

Instructions that should be given to the shift supervisor by the health physicist are as follows:

1. Stop the spill/release in the most dose-effective manner.
2. Once the release is terminated, rope the area and keep unauthorized personnel out of the area.
3. Warn others via plant page to stay out of the area.
4. Minimize exposure in the recovery.
5. Check affected personnel for contamination and decontaminate as necessary.
6. Assess radiological conditions.
7. Notify appropriate personnel (NRC/DOE, plant management, and response teams as warranted).
8. Survey adjacent areas and decontaminate as necessary.
9. Clean up the spill area and minimize radioactive waste generation.
10. Stabilize plant conditions.
11. Establish plant shift schedules for long-term recovery.

Question 4.34

Cs-137 rapidly clears from the body. Most of it is eliminated during the first year post intake.

Pu-238 clears more slowly than Cs-137. A significant portion of the initial intake remains after the first year. Some will still remain after 50 years.

These comments are qualitatively understood by examining the ratio of the DCFs for 1 and 50 years:

Cs-137:

$$\frac{DCF_1}{DCF_{50}} = \frac{2.9 \times 10^{-2}}{3.2 \times 10^{-2}} = 0.91 \qquad \text{(91\% of the dose in the first year)}$$

Pu-238:

$$\frac{DCF_1}{DCF_{50}} = \frac{30}{310} = 0.10 \qquad \text{(10\% of the dose in the first year)}$$

Scenario 4.8

Question 4.35

The release rate (in MBq/s) for U-238 is calculated by assuming the instantaneous release of the contents of the UF_6 cylinder and instantaneous uniform mixing of the UF_6 within the building, followed by release to the stack.

The total number of U atoms released to the building is given by the sum of the contributions from all uranium isotopes:

$$N(\text{U}) = N(\text{U-238}) + N(\text{U-235}) + N(\text{U-234})$$

For 3% U-235 enrichment and a negligible U-234 mass fraction, the number of U-238 [$N(U\text{-}238)$] atoms released into the building is written as:

$$N(\text{U-238}) = f\ M\ \frac{6.02 \times 10^{23}\ \text{atoms/mol}}{A(\text{U-238}) + 6\ A(\text{F-19})}$$

where

f = fraction of the uranium atoms that are U-238 = 1 – 0.03 = 0.97
M = mass of UF_6 in the leaking cylinder = 9071 kg
$A(X)$ = gram atomic mass of species X
A(U-238) = 238 g/mol
A(F-19) = 19 g/mol

Using the given values permits the determination of N(U-238):

$$N(\text{U-238}) = (0.97)\ (9071\ \text{kg})\ (1000\ \text{g/kg})\ \frac{6.02 \times 10^{23}\ \text{atoms/mol}}{(238\ \text{g/mol} + (6)\ (19\ \text{g/mol}))}$$

$$= 1.51 \times 10^{28}\ \text{atoms}$$

The activity (A) of U-238 is given in terms of its definition:

$$A = \lambda\ N = \left(\frac{\ln 2}{4.5 \times 10^9\ \text{y}}\right)(1.51 \times 10^{28}\ \text{atoms})$$

$$\times \left(\frac{1\ \text{dis}}{\text{atom}}\right)\left(\frac{1\ \text{y}}{365\ \text{d}}\right)\left(\frac{1\ \text{d}}{24\ \text{h}}\right)\left(\frac{1\ \text{h}}{3600\ \text{s}}\right)\left(\frac{\text{Bq-s}}{\text{dis}}\right)\left(\frac{1\ \text{MBq}}{10^6\ \text{Bq}}\right)$$

$$= 7.37 \times 10^4\ \text{MBq}$$

The initial activity release rate (Q) from the building volume (V) is given by:

$$Q = A\,F/V$$

where

F = stack flow rate = 2.83×10^2 m^3/min
V = building volume = 2.83×10^4 m^3

With these values, the release rate is determined

$$Q = \frac{(7.37 \times 10^4 \text{ MBq})\,(2.83 \times 10^2 \text{ m}^3/\text{min})\,(1 \text{ min}/60 \text{ s})}{(2.83 \times 10^4 \text{ m}^3)} = 12.3 \text{ MBq/s}$$

Question 4.36

The activity concentration of U-238 at 500 meters downwind at ground level at the plume centerline for class D stability conditions and 5 m/s wind speed is calculated using the Gaussian plume model equation:

$$\chi = \frac{Q}{2\,\pi\,\sigma_y\,\sigma_z\,u}\, e^{-\frac{y^2}{2\,\sigma_y^2}} \left[e^{-\frac{(z-h)^2}{2\,\sigma_z^2}} + e^{-\frac{(z+h)^2}{2\,\sigma_z^2}} \right]$$

where

χ = the ground level concentration (MBq/m^3) = $\chi\,(x, y, z)$
Q = the release rate = 12.3 MBq/s
u = wind speed = 5 m/s
σ_y = horizontal dispersion coefficient = 40 m @ 500 m downwind
σ_z = vertical dispersion coefficient = 20 m @ 500 m downwind
x = down-wind distance = 500 m
y = cross-wind distance from the plume centerline = 0
(calculation is on the plume centerline)
z = height above the ground where the activity is calculated = 0
h = release height (above ground level) = 25 m

With these parameter values and the Gaussian plume model equation, the ground level concentration at 500 m downwind on the plume centerline is determined:

$$\chi(500 \text{ m}, 0, 0) = \frac{(12.3 \text{ MBq/s})}{(2\,\pi)\,(5 \text{ m/s})\,(40 \text{ m})\,(20 \text{ m})}\, e^0 \left[e^{-\frac{(0-25 \text{ m})^2}{2\,(20 \text{ m})^2}} + e^{-\frac{(0+25 \text{ m})^2}{2\,(20 \text{ m})^2}} \right]$$

$$= 4.48 \times 10^{-4} \text{ MBq/m}^3$$

Question 4.37

Assumptions that may contribute to the inaccuracy in the use of the Gaussian plume model in the previous question include:

1. The topography at the site may not be flat and the σ_y and σ_z values may not provide a realistic description of plume dispersion. Hills, valleys, and other structures alter the validity of the Gaussian plume model's assumptions.
2. Meteorological conditions, including inversions and transient effects, introduce additional errors.
3. The presence of lakes or large bodies of water requires the consideration of a sea-breeze effect.
4. The use of only the stack height h does not include the effective release height. The effective release height is given by:

$$H = h + d\left(\frac{v}{u}\right)^{1.4}\left(1 + \frac{\Delta T}{T_o}\right)$$

where
H = effective release height
h = actual stack height
d = stack outlet diameter at the release point
v = exit velocity of the release gas/aerosol from the stack
u = mean wind speed
ΔT = absolute temperature difference between ambient (T_o) and effluent gas/aerosol temperatures (K)
T = absolute temperature of the released gas/aerosol in the stack (K)

5. The model is a "quasi" steady state formulation that requires that average values, applicable over times of at least 15 minutes, be utilized. The model will not yield accurate instantaneous results.
6. Reflection by the ground, partially described by the factor of 2 in the denominator, is only a crude, conservative assumption.
7. The model makes only limited provision for radioactive decay or particulate removal by precipitation or gravitational settling during the transport of the plume.
8. Dispersion is only included in two dimensions (y and z).
9. The release meteorology is assumed to be constant throughout the release.

Question 4.38

The Gaussian dispersion model will tend to overestimate the average ground-level concentration. Question 4.37 items 1, 4, 6, 7, and 8 contribute to this overestimate.

Scenario 4.9

Question 4.39

The Pu-239 activity residing within the glove box before the release occurs is defined by the relationship:

$$A = \lambda N$$

where

A = Pu-239 activity (MBq)

λ = Pu-239 disintegration constant = $0.693/T_{1/2}$

$T_{1/2}$ = Pu-239 half-life = 24 100 y

= $0.693/2.41 \times 10^4$ y = 2.88×10^{-5}/y

N = number of Pu-239 atoms = $\dfrac{m\, A_A}{\text{gam}}$

m = mass of Pu-239 in the glove box prior to the event = 500 g

A_A = Avogadro's number = 6.02×10^{23} atoms/mole

gam = gram atomic mass of Pu-239 = 239 g/mole

N = (500 g) (6.02×10^{23} atoms/mole)/(239 g/mole) = 1.26×10^{24} atoms

With these values, the activity is determined:

$$A = \lambda N = (2.88 \times 10^{-5}/\text{y})\,(1.26 \times 10^{24}\ \text{atoms})\,(1\ \text{dis/atom}) \times (1\ \text{y}/365\ \text{d})\,(1\ \text{d}/24\ \text{h})\,(1\ \text{h}/3600\ \text{s})\,(1\ \text{Bq-s/dis})\,(1\ \text{MBq}/10^6\ \text{Bq}) = 1.15 \times 10^6\ \text{MBq}$$

Question 4.40

Assuming 1.85×10^6 MBq of Pu-239 are initially present in the glove box, the Pu-239 concentration in air in MBq/m^3 in the room 20 minutes after the start of the fire, is obtained using the following assumptions:

1. Complete combustion occurs and the release of the plutonium progresses at a constant rate.
2. All Pu-239 evolves into an airborne form during the 20 minute event. Therefore, the activity release rate (P) is:

   $$P = A/t$$

   where

   A = initial activity in the glove box = 1.85×10^6 MBq

   t = Release time = 20 min

With these values, the activity release rate is:

$$P = 1.85 \times 10^6\ \text{MBq}/20\ \text{min} = 9.25 \times 10^4\ \text{MBq/min}$$

3. All Pu-239 activity remains in an airborne state unless it is removed by the ventilation system or radioactive decay. The total removal constant (k) is the sum of the radioactive decay constant (λ) and the ventilation removal constant (F/V):

$$k = \lambda + F/V$$

where
F = emergency ventilation system flow rate = 7 m^3/min
V = room volume = 6 m × 6 m × 3 m = 108 m^3
$\lambda = 0.693/T_{1/2}$
$= (0.693/2.41 \times 10^4 \text{ y})\ (1 \text{ y}/365 \text{ d})\ (1 \text{ d}/24 \text{ h})\ (1 \text{ h}/60 \text{ min})$
$= 5.47 \times 10^{-11}$/min

Therefore, the total removal rate is:
$k = \lambda + F/V = 5.47 \times 10^{-11}/\text{min} + (7 \text{ m}^3/\text{min})/108 \text{ m}^3 = 6.48 \times 10^{-2}/\text{min}$

The Pu-239 air concentration (C) is simply the ratio of the Pu-239 activity (A) divided by the room volume V. The rate of change of the activity A is given by the production equation:

$$\dot{A} = P\, e^{-k\,t}$$

which is integrated to obtain $A(t)$:

$$A(t) = \frac{P}{k}\left[1 - e^{-k\,T}\right]$$

At $T = 20$ m,

$$A(20 \text{ min}) = \frac{9.25 \times 10^4 \text{ MBq/min}}{0.0648/\text{min}}\left[1 - e^{-(0.0648/\text{min})\,(20 \text{ min})}\right]$$
$$= (1.43 \times 10^6 \text{ MBq})\,(1 - 0.274) = 1.04 \times 10^6 \text{ MBq}$$

With these values, the air concentration of Pu-239 is determined:

$$C = \frac{A}{V} = \frac{1.04 \times 10^6 \text{ MBq}}{108 \text{ m}^3} = 9.63 \times 10^3 \text{ MBq/m}^3$$

Question 4.41

Since the protection factor for a pressure demand SCBA is 1.0×10^4, the time to reach 10^4 DAC is to be determined:

$$C(t) = (1.0 \times 10^4 \text{ DAC})\,(0.08 \text{ Bq/m}^3\text{-DAC})\,(1 \text{ MBq}/10^6 \text{ Bq})$$
$$= 8.0 \times 10^{-4} \text{ MBq/m}^3$$

Following the release, the concentration $C(t)$ will decrease from the initial concentration $C(0)$ according to the relationship:

$$C(t) = C(0)\, e^{-k\,t}$$

The desired time (t) is obtained from the previous relationship:

$$-k\,t = \ln\left[\frac{C(t)}{C(0)}\right]$$

$$t = \left(-\frac{1}{k}\right)\ln\left[\frac{C(t)}{C(0)}\right] = \left(-\frac{1}{0.0648/\text{min}}\right)\ln\left(\frac{8.0\times10^{-4}\ \text{MBq/m}^3}{11.1\ \text{MBq/m}^3}\right)$$
$$= (-15.4)\,(-9.54\ \text{min}) = (147\ \text{min})\left(1\ \text{h}/60\ \text{min}\right) = 2.45\ \text{h}$$

Question 4.42

The air concentration at ground level at the site boundary (on the plume centerline), is obtained from the relationship:

$$\chi = \frac{Q}{2\,\pi\,\sigma_y\,\sigma_z\,u}\exp\left[-\left(\frac{y^2}{2\,\sigma_y^2}\right)\right]\left[\exp\left(-\frac{(z-h)^2}{2\,\sigma_z^2}\right)+\exp\left(-\frac{(z+h)^2}{2\,\sigma_z^2}\right)\right]$$

where

Q = release rate = $p\,C\,F$
- p = HEPA filter penetration = 5.0×10^{-4}
- C = room concentration = 11.1 MBq/m^3
- F = emergency filtration system flow rate = 7 m^3/s

= (5.0×10^{-4}) (11.1 MBq/m^3) (7.0 m^3/min)
= 3.89×10^{-2} MBq/min

χ = concentration @ 1000 m downwind on the plume centerline at ground level

$\overline{u}$ = wind speed = 7 m/s

y = cross-wind distance = 0 (plume centerline)

z = elevation for concentration determination = 0 (ground level)

h = stack height = 10 m

σ_y (1000 m) = horizontal dispersion coefficient (m) = 110 m

σ_z (1000 m) = vertical dispersion coefficient (m) = 65 m

$$\chi = \frac{(3.89\times10^{-2}\ \text{MBq/min})\,(1\ \text{min}/60\ \text{s})}{(2\,\pi)\,(110\ \text{m})\,(65\ \text{m})\,(7\ \text{m/s})}$$
$$\times\exp\left[-\left(\frac{0^2}{2\,(110\ \text{m})^2}\right)\right]\left[\exp\left(-\frac{(0-10\ \text{m})^2}{2\,(65\ \text{m})^2}\right)+\exp\left(-\frac{(0+10\ \text{m})^2}{2\,(65\ \text{m})^2}\right)\right]$$
$$= (2.06\times10^{-9}\ \text{MBq/m}^3)\,(0.988+0.988) = 4.07\times10^{-9}\ \text{MBq/m}^3$$

Solutions for Chapter 5

Scenario 5.1

Question 5.1

$$A = (C_i)\,(F)\,(e)\,[1 - \exp(-\lambda\, t)]/(\lambda)$$

For Co-60 and the demineralizer utilized in this problem, we obtain

$$\lambda = 0.693/(5.26 \text{ year} \times 365 \text{ days/year}) = 3.61 \times 10^{-4}/\text{day}$$
$$F = 350 \text{ liters/min} \times 10^3 \text{ ml/liter} = 3.5 \times 10^5 \text{ ml/min}$$
$$C_i = 6.0 \times 10^{-4} \ \mu\text{Ci/ml}$$
$$t = 200 \text{ days}$$
$$e = 1.0$$

$$A = \frac{(6.0 \times 10^{-4} \ \mu\text{Ci/ml})\,(3.5 \times 10^5 \text{ ml/min})\,(1.0)}{(3.61 \times 10^{-4}/\text{day})\,(1 \text{ day}/1440 \text{ min})} \frac{1.0 \text{ Ci}}{10^6 \ \mu\text{Ci}}$$
$$\times \{1 - \exp[-(3.61 \times 10^{-4}/\text{day} \times 200 \text{ days})]\} = 58.3 \text{ Ci}$$

To compute the gamma dose, the gamma constant (G) is utilized.

$R = 2.0$ ft (the radius of the resin bed)
$h = 1.0$ ft (the distance from disk source to dose location)

The dose rate is to be calculated using the equation provided in the data section:

$$D(h) = 3.14\, G\, C \ln \frac{h^2 + R^2}{h^2}$$

$$\ln (R^2 + h^2)/h^2 = \ln [(4 + 1)/1] = 1.609$$

Area of demineralizer bed = $3.14 \times (2\text{ ft} \times 12\text{ in./ft} \times 1\text{ m}/39.37\text{ in.})^2 = 1.17\text{ m}^2$

$$D(1\text{ ft}) = 3.14 \times 1.609 \times \left[(1.3\text{ R-m}^2/\text{hr-Ci}) \times \frac{58.3\text{ Ci}}{1.17\text{ m}^2}\right]$$
$$= 327.3\text{ R/hr} \times 0.877\text{ rad/R} \times 1\text{ rem/rad} = 287.0\text{ rem/hr}$$

Note: The use of the point source approximation is not applicable.

$$D_{\text{Point}} = G\,A/r^2$$
$$= 1.3\text{ R-m}^2/\text{hr-Ci} \times 58.3\text{ Ci} \times \frac{1}{(1\text{ ft} \times 12\text{ in./ft} \times 1\text{ m}/39.37\text{ in.})^2}$$
$$= 815.8\text{ R/hr}$$

Question 5.2

The shielded-dose calculation requires the decay of demineralizer activity to a time of 6 months post shutdown:

$$A(\text{Co-60}) = 80\text{ Ci} \times \exp\{-[(0.693)\,(0.5\text{ year})/(5.26\text{ years})]\} = 74.9\text{ Ci}$$
$$R = 2.0\text{ ft}$$
$$h = 25.0\text{ ft}$$
$$\ln\frac{R^2 + h^2}{h^2} = \ln\frac{2^2 + 25^2}{25^2} = 6.38 \times 10^{-3}$$

The unshielded dose may be calculated based upon this information:

$$D\text{ (unshielded)} = (3.14)\,(6.38 \times 10^{-3})\,[(74.9\text{ Ci} \times 1.3\text{ R-m}^2\text{ hr}^{-1}\text{ Ci}^{-1}/1.17\text{ m}^2)]$$
$$= 1.67\text{ R/hr}$$

Note: The point source approximation may be used because the field-point distance is at least three times larger than the largest source dimension.

It should also be recognized that the beta-dose component is effectively negated by distance and the demineralizer shell.

$$D_{\text{Point}} = (1.3\text{ R-m}^2/\text{hr-Ci}) \times (74.9\text{ Ci}) \times \frac{1}{(25\text{ ft} \times 12\text{ in./ft} \times 1\text{ m}/39.37\text{ in.})^2}$$
$$= 1.68\text{ R/hr} \quad \text{(essentially the same result)}$$

Co-60 has two gamma rays (1.173 and 1.332 MeV) each with a yield of 100%. Because the average energy of these gammas is about 1.25 MeV, it is sufficient to use the 1.25-MeV attenuation coefficient for lead.

Interpolation of buildup factors could be performed, but is not necessary; that is, the steel shell has been neglected. The use of slightly conservative buildup factors is also justified based upon dose limitations.

$$t = 2.0 \text{ in.} \times 2.54 \text{ cm/in.} = 5.08 \text{ cm} \quad \text{(lead shield thickness)}$$

$$u(\text{Co-60}) = 0.65 \text{ cm}^{-1} \quad \text{(from the 1.25-MeV data)}$$

Calculate the optical thickness (ut) values:

$$\text{Co-60:} \quad ut = 0.65 \text{ cm}^{-1} \times 5.08 \text{ cm} = 3.30$$

The buildup factor is obtained as follows:

$$\text{Co-60:} \quad B\,(3.30) = B\,(ux = 4.0 \text{ @ } 1.0 \text{ MeV}) = 2.26$$

The shielded dose rate may be obtained from the preceding factors and the relationship

$$D = D_0\, B \exp(-ut)$$

$$\begin{aligned} D(2.0 \text{ in. Pb}) = {} & (3.14)\,(6.38 \times 10^{-3}) \\ & \times (1.3 \text{ R-m}^2 \text{ Ci}^{-1} \text{ hr}^{-1})\,(74.9 \text{ Ci})\,(2.26)/1.17 \text{ m}^2 \\ & \times \exp(-3.3) \times (1000 \text{ mR/R}) = 139.0 \text{ mR/hr} \end{aligned}$$

The total dose during the repair operation is

$$\begin{aligned} D_{\text{total}} &= 3 \text{ hr} \times 139 \text{ mR/hr} \\ &= 417 \text{ mR} \times 0.877 \text{ mrad/mR} \times \text{mrem/mrad} = 365.7 \text{ mrem} \end{aligned}$$

This exceeds the 300-mrem limit. Therefore the 2 in. of lead is not acceptable. If no additional shielding were utilized, the worker would exceed his annual limit by (366 – 300) mrem = 66 mrem.

Scenario 5.2

Question 5.3: e

For a hot particle, the beta radiation emitted from the particle is the principal contributor to the skin dose.

Question 5.4: a

The increased use of more sensitive instrumentation is the major reason for the increased detection frequency of hot particles at nuclear power plants.

Question 5.5: d

Prior to 1988, neither the NCRP nor the ICRP provided explicit recommendations for hot-particle skin exposure. NCRP 106 (1989) provided this guidance.

Question 5.6: a

Distributed contamination on the skin is more likely to cause skin cancer when compared to the same isotopic activity residing within a hot particle. Hot particles are more likely to kill cells due to their high localized doses.

Question 5.7: c

For a 2.0-cm Sr-90 disk source, a detector located 1 cm from the source should have a detection efficiency of about 35%.

Question 5.8: c

The beta dose to the worker's finger may be determined from the relationship

$$D = A_i \, DF_i \, t/S$$

where

D = beta dose (rad)
A_i = particle activity (μCi) = 0.39 μCi of Co-60
DF_i = 4.13 rad-cm^2/μCi-hr
t = time particle resides on the skin (hr) = 15.75 hr
S = area over which the dose is averaged (cm^2) = 10 cm^2
D = $(3.9 \times 10^{-1}$ μCi) (4.13 rad-cm^2/μCi-hr) (15.75 hr)/10 cm^2
= 2.54 rad × 1 rem/rad = 2.54 rem

Question 5.9: d

Prior to January 1, 1993, this dose would exceed the 18.75-rem quarterly limit for extremities as outlined in 10 CFR 20.

After January 1, 1993, this dose would exceed the 50-rem annual limit for the skin, hands, and feet dose as outlined in 10 CFR 20.

Question 5.10: e

Thyroid monitoring and urinalysis are not appropriate for hot-particle contamination residing on the skin because no internal hazard is likely. Moreover, hot particles are normally insoluble, and an intake is unlikely.

Question 5.11: a

An energy-compensated GM probe would provide the best sensitivity for detecting hot particles at the exit station.

Question 5.12: b

NCRP 106 established a 75 μCi-hr limit for hot-particle exposures.

Scenario 5.3

Question 5.13: a

Choice "a" results in higher doses. The other choices have been identified in various NRC and EPRI publications as reasons for the downward trend. See NUREG/CR-5158, for example.

Question 5.14: c

Co-60, a neutron activation product, has repeatedly been identified as the radionuclide responsible for at least 80% of these exposures.

Question 5.15: d

Co-60 is the major dose contributor to shutdown radiation fields.

Question 5.16: a

Hydrogen water chemistry increases doses by increasing N-16 in BWR turbines. The other choices are ways of reducing the BWR radiation field.

Question 5.17: g

Lack of knowledge of NRC Regulations by the radiological protection technician has not been a common error in unplanned exposures at nuclear power plants.

Question 5.18: a

The number of certified health physicists at a power reactor is not a useful indicator of radiological performance.

Question 5.19: a

The average annual exposure at nuclear power plants is in the 400 to 800-mrem range.

Question 5.20: b

Choice "b" is the only choice that is exactly the same as the previous requirements. Therefore, it cannot result in program changes. The other choices are changed requirements and should result in some program changes.

Question 5.21: c

The NRC dose limit for protection of the embryo/fetus applies once the female worker has voluntarily made her pregnancy known to her employer.

Question 5.22: b

ICRP-60 and ICRP-103 recommend that a 2-rem/year average or 10 rem over 5 years be adopted. NCRP-116 recommendations suggest that 1 rem/year may be appropriate.

Scenario 5.4

Question 5.23

The activity derived from the irradiation is given by

$$
\begin{aligned}
A &= N\,\sigma\,\phi\,[1 - \exp(-\lambda\, t_{\text{irrad}})]\exp(-\lambda\, t_{\text{decay}}) \\
N &= (10.0\text{ g})\,(6.02 \times 10^{23}\text{ atoms/mole})/(59\text{ g/mole}) = 1.02 \times 10^{23}\text{ atoms} \\
\sigma &= 37\,b \times 1.0 \times 10^{-24}\text{ cm}^2\text{/b-atom} = 3.7 \times 10^{-23}\text{ cm}^2\text{/atom} \\
\phi &= 1.0 \times 10^{10}\text{ n/cm}^2\text{-s} \quad \text{(activation is by the thermal flux only)} \\
t_{\text{irrad}} &= 10\text{ years} \\
t_{\text{decay}} &= 0.5\text{ years} \\
\lambda &= 0.693/5.27\text{ years} = 0.131\text{ year}^{-1} \\
A &= (1.02 \times 10^{23}\text{ atoms})\,(3.7 \times 10^{-23}\text{ cm}^2\text{/atom}) \\
&\quad \times (1.0 \times 10^{10}\text{ n/cm}^2\text{-s})\,(1\text{ dis/n}) \\
&\quad \times [1.0 - \exp(-0.131\text{ year}^{-1} \times 10\,\text{year})] \\
&\quad \times \exp(-0.131\text{ year}^{-1} \times 0.5\text{ year}) \\
&= 2.58 \times 10^{10}\text{ dis/s} \times 1\text{ Ci}/(3.7 \times 10^{10}\text{ dis/s}) = 0.70\text{ Ci}
\end{aligned}
$$

The exposure rate at a point 2.0 m from the small-particle source can be obtained from the point-source approximation. Assuming no attenuation due to air, the exposure rate is

$$
\dot{X}(N) = A\,G/r^2 = \frac{(0.70\text{ Ci}) \times 1.3\,\dfrac{\text{R-m}^2}{\text{hr-Ci}}}{(2.0\text{ m})^2} = 0.23\text{ R/hr}
$$

Question 5.24

The exposure rate at point P a distance of 2.0 m from the small-particle source can be obtained from the point source approximation:

$$\dot{X}(P) = A\,G/r^2 = \frac{(3.0\text{ Ci}) \times 1.3\,\dfrac{\text{R-m}^2}{\text{hr-Ci}}}{(2.0\text{ m})^2} = 0.98\text{ R/hr}$$

Question 5.25

The general line source equation may be utilized for this question:

$$\dot{X}(Q) = G\,C_L\,\theta/h$$

The exposure rate at point *Q*, a distance of 2.0 m from the end of the sample line source, can be obtained by using a line source approximation as illustrated in Figure S5.1.

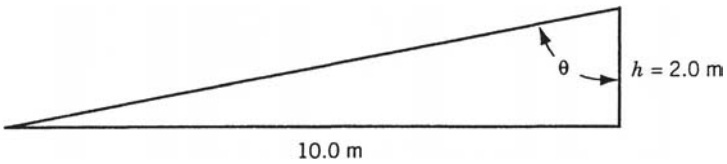


Figure S5.1 Determination of the included angle (θ) for line source geometry.

$$\dot{X} = \frac{G\,C_L\,\theta}{h}$$

$$G = 1.3\,\frac{\text{R-m}^2}{\text{hr-Ci}}$$

$$\tan\theta = 10.0\text{ m}/2.0\text{ m} = 5.0$$

$$\theta = \tan^{-1}(5.0) = 78.69° = 78.69° \times 3.14\text{ rad}/180° = 1.37\text{ rad}$$

$$C_L = 3\text{ Ci}/10\text{ m} = 0.3\text{ Ci/m}$$

$$h = 2.0\text{ m}$$

$$\dot{X} = \frac{1.3\,\dfrac{\text{R-m}^2}{\text{hr-Ci}} \times \dfrac{0.3\text{ Ci}}{\text{m}} \times 1.37}{2.0\text{ m}} = 0.27\text{ R/hr}$$

Question 5.26

The exposure rate at point *X*, a distance of 2.0 m from the thin disk source, may be determined from the relationship:

$$\dot{X}(X) = \pi\,G\,C_a\,\ln\frac{r^2 + h^2}{h^2}$$

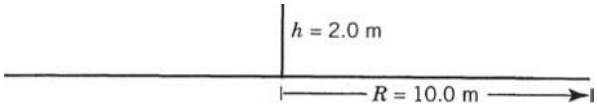


Figure S5.2 Disk source geometry illustrating the 10.0-m-radius spill and the point of interest lying 2.0 m above the centerline of the spill.

The geometric relationships utilized in the application of the disk-source relationship are illustrated in Figure S5.2.

$$G = 1.3\,\frac{\text{R-m}^2}{\text{hr-Ci}}$$

$$C_a = 3\ \text{Ci}/(3.14)\,(10\ \text{m})^2 = 9.55 \times 10^{-3}\ \text{Ci/m}^2$$

$$R = 10.0\ \text{m}$$

$$h = 2.0\ \text{m}$$

$$\dot{X} = 1.3\,\frac{\text{R-m}^2}{\text{hr-Ci}} \times \frac{9.55 \times 10^{-3}\ \text{Ci}}{\text{m}^2} \times 3.14 \times \ln\frac{(10.0\ \text{m})^2 + (2.0\ \text{m})^2}{(2.0\ \text{m})^2} = 0.13\ \text{R/hr}$$

Scenario 5.5

Question 5.27

The hollow pole can produce a radiation streaming path and could result in high dose rates in a narrow beam. The pole needs to have free-flooding holes or shield plugs inserted into the pole.

The installation of holes permits the possibility of trapping debris inside the pole and is the source of additional radioactive waste. Another option would be the use of a solid pole that could easily be decontaminated. This would eliminate both the radiation streaming and radioactive waste generating concerns.

The surfaces of the pole should be polished to a smooth finish. Surfaces that could trap particles or debris must be avoided.

Question 5.28

The cutting head should be designed as a single disposable unit with quick disconnects to hydraulic or power lines. If this has not been part of the vendor supplied tool, then the replacement of cutter jaws will likely require that the tool be removed from the pool. The problem of trapped fragments, debris, or pieces of irradiated hardware in the tool must be considered as part of the radiological evaluation. The tool should be designed to minimize the possibility of trapping irradiated hardware fragments or debris. Large clearances or gaps are desirable. Ease of underwater inspection and cleanout should be available. Appropriate flush connections are to be provided to facilitate cleanout.

Question 5.29

The possibility of lifting an irradiated component out of the water or near the surface of the water, must be addressed. The design of the crane must incorporate physical means to limit the travel of the crane to preclude the loss of the shielding provided by the water in the pool. Limit switches, special rigging, or lanyards should be used to limit the upward vertical travel of the crane.

Alarming radiation monitors or limit switch alarms should be provided to warn the operator of abnormal radiological conditions.

Question 5.30

The ARM detector is at a height of 6 ft above the pool surface and 20 ft from the edge of the pool railing.

Using this height and distance permits the associated angle to be determined as illustrated in Figure S5.3.

$$\tan a = \frac{6 \text{ ft}}{20 \text{ ft}} = 0.3$$

Because the surface of the pool is parallel to the surface of the refueling floor, angle a may be used to determine the distance from the edge of the pool (d). This is illustrated in Figure S5.4.

The distance between the pool and the refueling floor is 18 in.

$$\tan a = 18 \text{ in.}/d = 0.3$$
$$d = 18 \text{ in.}/0.3 = 60 \text{ in.} = 5 \text{ ft}$$

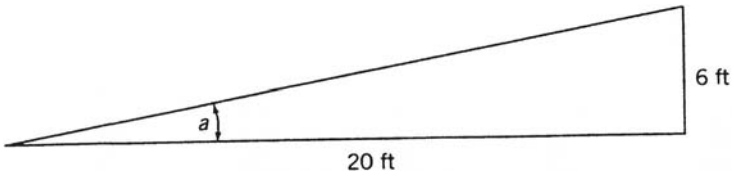


Figure S5.3 Determination of the angle (a) describing the location of the radiation monitor relative to the edge of the spent-fuel pool.

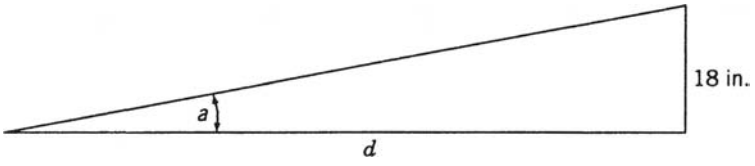


Figure S5.4 Geometry for determining the distance (d) from the edge of the spent-fuel pool. The distance between the surface of the spent-fuel pool and the refueling deck is 18 in.

Question 5.31

The distance (d_1) from the center of the pool to the ARM is

$$d_1 = [(15 \text{ ft} + 20 \text{ ft})^2 + (6 \text{ ft} + 1.5 \text{ ft})^2]^{1/2} = 35.8 \text{ ft}$$

The distance (d_2) from the center of the pool to the edge of the railing is

$$d_2 = [(15 \text{ ft})^2 + (1.5 \text{ ft} + 3.5 \text{ ft})^2]^{1/2} = 15.8 \text{ ft}$$

The alarm setpoint of the installed ARM is adjusted so that it will alarm if a point source at the center of the pool yields an exposure rate of 1 R/hr at the pool railing ($\dot{X}_2$). Using these results, the correct ARM setpoint ($\dot{X}_1$) is given by the inverse-square law:

$$\dot{X}_1 \, d_1^2 = \dot{X}_2 \, d_2^2$$

$$\dot{X}_1 = \dot{X}_2 \, (d_2 / d_1)^2 = 100 \text{ mR/hr} \, (15.8 \text{ ft}/35.8 \text{ ft})^2 = 194.8 \text{ mR/hr}$$

Question 5.32

Characterization Technique	Advantages	Disadvantages
Physical sampling	• Allows for the most precise determination of radionuclide composition. • Calculational conservatism is minimized.	• Sample handling requires strict contamination control practices. • Sampling techniques may produce hot particles that increase the risk of contamination events or unanticipated exposures. • Sample must be shown to be representative of the material. • Irradiation may not have been uniform, which requires multiple samples to be taken.
Underwater surveys	• Direct handling of activated material is not required.	• Material composition uncertainties may require calculations to be overly conservative. • Uncertainties in the measurement technique, such as the probe position, will cause significant errors.

Question 5.33

Characterization Technique	Sources of Error
Physical sampling	• Assumptions regarding the uniform irradiation of the component. • A sufficient number of samples are required to ensure that they are representative. • Only surface conditions are included in the sample. Interior samples are not feasible with the proposed method.
Underwater surveys	• The chemical composition of the material may be incomplete. This includes the fractional quantity of cobalt impurities in the material. • The irradiation history of the material will be incomplete. Core power levels and the activating neutron fluence levels may be incomplete. This is complicated by not knowing the exact core location during the irradiation period. • The position of the probe will be uncertain. This may be complicated by water clarity in the pool and difficulties in accurately positioning the probe relative to the equipment being measured. • These uncertainties complicate and reduce the accuracy of the subsequent model calculations.

Question 5.34

Because of their composition and age, the filters are expected to be in a degraded condition and not have their initial structural integrity. Disintegration of the filters during movement is possible. The disintegration of these filters could lead to the following radiological controls concerns:

1. Release of numerous hot particles.
2. Generation of debris that will float on the surface of the spent fuel pool. Without the water shielding, this debris presents a significant external radiation hazard.

Question 5.35

Appropriate radiation control measures to deal with the hazards outlined in question 5.34 include:

1. Minimize the distance that the filters are transported. An underwater transfer device, supplying structural integrity, could be employed.
2. Utilize tooling and transfer equipment that minimize the possibility of damaging the filters during transport.

Scenario 5.6

Question 5.36

Primary responsibilities of the RCM are as follows:

1. Perform timely calculations of the projected doses from facility releases.
2. Ensure the radiation safety of the personnel at the EOF.
3. Formulate/update Protective Action Recommendations (PARs).
4. Coordinate the movement of off-site sampling teams in order to assess the release.
5. Direct the overall radiological response to the event.
6. Determine if the use of thyroid blocking agents is warranted.
7. Discuss plant radiological conditions, dose projections, and PARs with senior State and Federal Officials.
8. Authorize exposure extensions, emergency exposure requests, and thyroid blocking agents.

Question 5.37

The three fission product barriers are:

1. Fuel element cladding.
2. Reactor coolant system.
3. Containment building.

Question 5.38

Fission Product Barrier Status

1. *Intact*. The fission product barrier is capable of preventing the release of fission products to the environment. This is the normal condition.
2. *In Jeopardy*. The barrier is currently intact, but plant conditions are such that the barrier will likely be breached.
3. *Breached*. The fission product barrier is not able to prevent the release of fission products to the environment.

Question 5.39

Based on the data, we present the status of the three fission product barriers:

Fission Product Barrier	Status
Fuel cladding	Breached as noted by the high activity in the letdown (primary) system.
Reactor coolant system	Breached as evidenced by the high activity in the steam generator blowdown (secondary) sample.
Containment	In jeopardy due to the primary to secondary leakage and the possibility of a release if a relief valve lifts due to increases in secondary system pressure.

Question 5.40

Factors affecting the calculation of off-site doses include:

1. Release magnitude and isotopic composition.
2. Release rate.
3. Meteorological conditions (atmospheric stability class, wind speed, and precipitation).
4. Release height.
5. Release duration.
6. Changing plant conditions due to repairs or equipment failures.

Question 5.41

The projected thyroid dose rate ($\dot{H}$) at 2 miles will be due to the I-131 released, and this value must consider any partitioning from the secondary side of the plant to the environment. The release concentration (C) of I-131 is determined from the I-131 concentration in the steam generator blowdown sample and the iodine partitioning factor:

$$C = (0.015)\,(6.3 \times 10^{-1}\ \mu\text{Ci/cm}^3) = 9.45 \times 10^{-3}\ \mu\text{Ci/cm}^3$$

The release rate (Q) can be obtained from the release concentration by assuming that the release is through the atmospheric relief valve having a flow rate of 1.4×10^7 cm^3/s:

$$\begin{aligned} Q &= (1.4 \times 10^7\ \text{cm}^3/\text{s})\,(9.45 \times 10^{-3}\ \mu\text{Ci/cm}^3) \\ &= 1.32 \times 10^5\ \mu\text{Ci/s} = 0.132\ \text{Ci/s} \end{aligned}$$

$$\dot{H} = \frac{X\,u}{Q}\,Q\,\frac{1}{u}\,(\text{DCF}) \quad @\ 2 \text{ miles}$$

$$= \frac{2.69 \times 10^{-3}}{\text{m}^2}\,(0.132 \text{ Ci/s})\,(3600 \text{ s/hr})$$

$$\times \frac{1}{(15 \text{ mi/hr})\,(1609.36 \text{ m/mi})\,(1 \text{ hr}/3600 \text{ s})}$$

$$\times 77.2\,\frac{\text{rem/s}}{\mu\text{Ci/cm}^3} \times \frac{1\ \text{m}^3}{1 \times 10^6\ \text{cm}^3} \times \frac{1 \times 10^6\ \mu\text{Ci}}{\text{Ci}}$$

$$= 14.7 \text{ rem/hr}$$

Question 5.42

Sheltering should be recommended at this time. Because a long-term release is not expected, evacuation is not warranted.

Question 5.43

A conservative estimate of the dose in the town can be performed by using the dose information calculated at the 2.0-mile distance:

$$D = 14.7 \text{ rem/hr} \times 0.25 \text{ hr} = 3.7 \text{ rem}$$

Again, evacuation is not warranted. The population could not reasonably be expected to be evacuated in a 15-min period. Sheltering is the recommended protective action.

Scenario 5.7

Question 5.44

In order to determine the Rb-88 activity at the beginning of the counting interval, one must know the net filter counts, disintegration constant, counting time interval, and counting efficiency.

The activity at the beginning of the counting interval is

$$A = \lambda N$$

The number of atoms N is given by

$$N = \frac{C_S}{[1 - \exp(-\lambda\, T_{\text{count}})]\, e}$$

where C_S is the number of filter counts in the counting interval (30 min), e is the efficiency, and $1 - \exp(-\lambda T_{count})$ is the fraction of the atoms expected to decay during the counting interval (T_{count}). Therefore, the activity of Rb-88 is written as

$$A_{Rb\text{-}88} = \frac{\lambda C_S}{(1 - \exp[-\lambda T_{count}])\, e}$$

where

λ = $\ln(2)/T_{1/2} = 0.693/17.7$ min = 0.0392/min
C_S = net counts = 1.5×10^5 counts
T_{count} = counting time = 30 min
e = counting efficiency = 0.1 count/disintegration

$$A_{Rb\text{-}88} = \frac{(0.0392/\text{min})\,(1.5 \times 10^5 \text{ counts})}{[1 - \exp(-0.0392/\text{min} \times 30 \text{ min})]\,(0.1 \text{ count/disintegration})}$$

$$= \frac{8.503 \times 10^4 \text{ disintegration/min}}{(3.7 \times 10^{10} \text{ disintegration/s-Ci})\,(60 \text{ s/min})\,(1 \text{ Ci}/10^6 \text{ } \mu\text{Ci})}$$

$$= 0.0383 \text{ } \mu\text{Ci}$$

Question 5.45

To calculate the activity concentration $U_{Rb\text{-}88}$ a knowledge of the counting and sampling times is needed:

T_S (sampling time) = 30 min
T_T (transit time) = 10 min
T_C (counting time) = 30 min

The net counts on the filter paper C_S is a function of the total number of Rb-88 atoms (N) in the sampled air volume:

$$C_S = N\,(\text{FR})\,\frac{1 - \exp(-\lambda T_S)}{\lambda T_S}\exp(-\lambda T_T)\,[1 - \exp(-\lambda T_C)]\,e$$

where

N = number Rb-88 atoms in the sampled air volume (V)

$$= \frac{U_{Rb\text{-}88}}{\lambda} V = \frac{U_{Rb\text{-}88}}{\lambda} F T_S$$

F = sampling flow rate = 30 liters/min
FR = filter retention = 1.0

Utilizing these relationships, the activity concentration is determined to be

$$C_S = \frac{U_{Rb\text{-}88}}{\lambda} F T_S\,(\text{FR})\,\frac{1 - \exp(-\lambda T_S)}{\lambda T_S}\exp(-\lambda T_T) \times [1 - \exp(-\lambda T_C)]\,e$$

$$U_{\text{Rb-88}} = \frac{\lambda^2 C_S}{e\,(\text{FR})\,F\,[1-\exp(-\lambda T_S)]\exp(-\lambda T_T)\,[1-\exp(-\lambda T_C)]}$$

$$= \frac{(0.0392/\text{min})^2\,(1.5\times10^5\ \text{counts})}{(0.1\ \text{counts/dis})\,(1)\,(30\ \text{liters/min}\times1000\ \text{cm}^3/\text{liter})}$$

$$\times\frac{1}{[1-\exp(-0.0392/\text{min}\times30\ \text{min})]\exp(-0.0392/\text{min}\times10\ \text{min})}$$

$$\times\frac{1}{[1-\exp(-0.0392/\text{min}\times30\ \text{min})]}$$

$$= (0.238\ \text{dis/min-cm}^3)\,(1\ \text{min}/60\ \text{s})$$

$$\times(1.0\ \mu\text{Ci}/3.7\times10^4\ \text{dis/s})$$

$$= 1.07\times10^{-7}\ \mu\text{Ci/cm}^3$$

Question 5.46

The airborne activity concentration of Kr-88 is the same as the Rb-88 activity concentration. This assumes that the physical removal rate of Rb-88 is negligible compared with its decay rate. An examination of the relative half-lives of Kr-88 and Rb-88 suggests that secular equilibrium conditions are approximately, which also supports the contention.

Kr-88: $T_{\text{parent}} = 2.84$ hr

Rb-88: $T_{\text{daughter}} = 17.7$ min $\times$ (1 hr/60 min) = 0.295 hr

$T_{\text{parent}} > T_{\text{daughter}}$
(which supports the approach to secular equilibrium contention).

Scenario 5.8

Question 5.47

The protective clothing for the hands and face is based on the Radiological Work Permit requirements. These requirements are as follows:

Hands:

2 pairs of rubber gloves:	2×39.9 mg/cm^2	= 79.8 mg/cm^2
1 pair of glove liners:	$1\times\ 8.6$ mg/cm^2	= 8.6 mg/cm^2
Total		= 88.4 mg/cm^2

Face:

1 bubble hood:	1×63.2 mg/cm^2	$= \underline{63.2 \text{ mg/cm}^2}$
Total		$= 63.2$ mg/cm^2

Based on the available protection noted above, the face will receive the higher shallow-dose equivalent.

Question 5.48

The dose equivalent rate $(\dot{D})$ to the skin of the face evaluated at 7 mg/cm^2 is given by the sum of beta and gamma dose contributions:

$$\dot{D} = \dot{D}_{\text{beta}} + \dot{D}_{\text{gamma}} = f_{\text{beta}}\ \dot{D}_{\text{o,beta}} + f_{\text{gamma}}\ \dot{D}_{\text{o,gamma}}$$

where

f_{beta}	=	fraction of the beta shallow-dose equivalent delivered to the face after attenuation by the bubble hood = 0.08 @ 63 mg/cm^2
$\dot{D}_{\text{o,beta}}$	=	measured beta dose equivalent rate = 0.60 Sv/h @ 7 mg/cm^2
f_{gamma}	=	fraction of the gamma shallow-dose equivalent delivered to the face after attenuation by the bubble hood = 1.00
$\dot{D}_{\text{o,gamma}}$	=	measured gamma-dose equivalent rate = 0.10 Sv/h @ 7 mg/cm^2

The measured beta and gamma dose rates are appropriate to the evaluation depth of 7 mg/cm^2, and the *f* factors are presented as a function of added absorber density thickness. With these values, the dose equivalent rate to the skin of the face is:

$$\dot{D} = (0.08)\,(0.60 \text{ Sv/h}) + (1.00)\,(0.10 \text{ Sv/h}) = 0.148 \text{ Sv/h}$$

Question 5.49

Deterministic effects following irradiation of the skin include:

1. Erythema.
2. Dry desquamation.
3. Moist desquamation.
4. Epilation.
5. Blistering.
6. Necrosis.
7. Ulceration.

Scenario 5.9

Question 5.50

The average airborne concentration C is defined in terms of the relationship:

$$C = \frac{A}{V} = \frac{\dfrac{N}{e\, t_c}}{f\, t_s} = \frac{N}{e\, f\, t_c\, t_s}$$

where

A = activity on the filter
V = volume of air passing through the filter
f = air sampler flow rate = 0.85 m^3/min
t_s = sample collection time = 10 min
N = total filter counts = 4500 c (beta)
e = counter efficiency = 0.32 c/d
t_c = count time = 1 min

Inserting these parameters into the air concentration equation yields:

$$\begin{aligned} C &= \frac{N}{e\, f\, t_c\, t_s} \\ &= \frac{(4500\text{ c})}{(0.32\text{ c/d})\,(0.85\text{ m}^3\text{/min})\,(1\text{ min})\,(10\text{ min})}\left(\frac{1\text{ min}}{60\text{ s}}\right)\left(\frac{\text{Bq-s}}{\text{d}}\right) \\ &= 27.6\text{ Bq/m}^3 \end{aligned}$$

Assuming the most conservative DAC (Co-60 whose DAC is 5.0×10^2 Bq/m^3), the air concentration result is written as the fraction of a DAC (f_{DAC}):

$$f_{DAC} = \frac{C}{\text{DAC}} = \frac{27.6\text{ Bq/m}^3}{500\text{ Bq/m}^3} = 0.0552$$

Question 5.51: a

After soap and water, detergent will be an effective decontamination agent. However, it must be administered carefully to avoid eroding the skin. The detergent is normally applied in the form of a paste and is somewhat more effective than washing with soap. The main disadvantage is that it dissolves fat and erodes skin and must be used with care.

Question 5.52

The intake is obtained from the relationship:

$$I = \frac{A(t)}{K_{IRF}(t)}$$

where
I = intake of Co-60 due to this incident
$A(t)$ = activity derived from the whole body count at time $t = 4.07 \times 10^3$ Bq
t = time of whole body count = 3.0 h after the event
$K_{IRF}(t)$ = intake retention fraction = 0.63 @ 3 hours post-event

With these values, the intake is determined:

$$I = \frac{4.07 \times 10^3 \text{ Bq}}{0.63} = 6.46 \times 10^3 \text{ Bq}$$

The dose (D) from the intake is specified by the relationship:

$$D = \frac{I}{\text{ALI}} 50 \text{ mSv} = \frac{6.46 \times 10^3 \text{ Bq}}{1.0 \times 10^6 \text{ Bq}} 50 \text{ mSv} = 0.323 \text{ mSv}$$

where
D = dose derived from the intake I
ALI = Annual Limit of Intake for Class Y Co-60 = 1.0 MBq

The dose delivered by the intake of an ALI depends upon the internal dosimetry formulation. The relationship noted above represents the 10CFR20 formulation, which governs US power reactors. The existing 10CFR20 (January 1, 2008 revision) is based upon ICRP-26/30 methodology. If ICRP-60 or -103 methodologies were employed, the limit of 50 mSv would be changed to 20 mSv and the corresponding ALI would be used.

Question 5.53

Seventy-two hours later, a follow-up body count is performed on the worker in question 5.52. The activity remaining is only 296 Bq. The reduction between 3 and 75 hours is due to short-term clearance mechanisms before the cobalt is distributed to body tissues. For soluble cobalt, the GI tract, total body, pancreas, liver, spleen, and kidney are the organs most affected by the uptake. If the Co material is insoluble, the tissues at risk are the lung and lower large intestine.

Question 5.54

The intake (I) in DAC-hr is obtained from the relationship:

$$I\,(\text{DAC-hr}) = \frac{t}{\text{PF}} \sum_{i=1}^{N} \frac{C_i}{\text{DAC}_i}$$

where

t = exposure time = 3 hr
PF = protection factor of the respirator = 50
C_i = airborne concentration of isotope i
i = labels isotopes in the sample
($i = 1$ labels Co-60 and $i = 2$ labels Cr-51)
C_i = 3.63×10^3 Bq/m^3 and $C_2 = 2.66$ MBq/m^3
DAC_i = Derived air concentration
($\text{DAC}_i = 5.0 \times 10^2$ Bq/m^3 and $\text{DAC}_2 = 3.0 \times 10^5$ Bq/m^3)

Using these values, the intake is determined:

$$I = \left(\frac{3\text{ h}}{50}\right)\left(\frac{3.63 \times 10^3\ \text{Bq/m}^3}{5.0 \times 10^2\ \text{Bq/m}^3 - \text{DAC}} + \frac{2.66 \times 10^6\ \text{Bq/m}^3}{3.0 \times 10^5\ \text{Bq/m}^3 - \text{DAC}}\right)$$
$$= (0.06\text{ h})\,(7.26\ \text{DAC} + 8.87\ \text{DAC}) = 0.968\ \text{DAC-hr}$$

Solutions for Chapter 6

Scenario 6.1

Question 6.1

Instantaneous Grab Samples. These are on-time or momentary grab samples that provide the instantaneous concentration during the sampling time. Examples of techniques employing this method include: Kusnetz Method, Tsivoglou Method, sealed ionization chamber, scintillation (Lucas) cells, Thomas Method, Rolle Method, two-filter method, and alpha spectroscopy.

Integrated Samples. This is a cumulative method yielding a time-averaged concentration over the duration of the measurement. Examples include: track etch detectors, charcoal canisters, and electrets (E-perms).

Continuous Samples. These are a series of grab measurements performed at prescribed time intervals that yield a chronological record of the instantaneous radon concentration. Examples include: continuous working level meters, flow-through scintillation cells, and flow-through pulse ionization chambers.

Question 6.2

Alpha Track Detectors. Alpha or nuclear track detectors are composed of cellulose nitrate or polycarbonate strips set inside a filtered chamber. Rn-222 diffuses into the chamber via the filter and exposes the strip to the radon gas. After exposure, the strip is chemically etched to enlarge the alpha tracks. The tracks are optically counted, and the track density is calibrated to the average radon concentration over the exposure period.

Activated Charcoal. Radon gas is adsorbed onto an activated charcoal bed. The bed is counted using an appropriate counting method such as NaI gamma ray counting of the photons emitted by progeny ingrowth in the activated charcoal. The Bi-214 photons are the dominant photon contributor.

Electrets (E-Perms). An electret is a dielectric disk. Charged ions formed by the decay of radon and its progeny are collected inside a fixed chamber volume by the quasi-permanently charged electret. The voltage discharge of the disk is linearly proportional to the average radon concentration over the exposure period.

Question 6.3: c

The short-lived Rn-222 decay products include: Bi-214, Pb-214, Po-218, and Po-214. The radon series is summarized in Table 6.1.

Question 6.4: b

Indoor unattached radon decay products are not usually measured using cyclone precollectors.

Question 6.5: c

The fraction of unattached radon decay products in the air does not depend on the radon concentration.

Question 6.6

$$\text{NWLM} = \frac{C_{\text{WL}}\, F_{\text{OCC}}}{k}$$

$$= \frac{(0.09\ \text{WL})\,(24\ \text{hr/day})\,(365\ \text{day/year})\,(0.75)}{(170\ \text{WL-hr/WLM})} = 3.5\ \text{WLM/year}$$

Question 6.7

1. *Breathing Rate (BR)*
   a. NCRP-78: The curve is a relatively flat straight line because the increase in deposition caused by the larger breathing rate is slightly offset by the decreased tidal volume.
   b. BEIR IV: As the breathing rate increases, the total quantity of daughter deposition will increase, but only slightly. The fractional deposition in the upper bronchial region decreases and the total deposition increases less rapidly than the increased alpha energy inhaled. These factors result in a dose versus breathing rate curve that resembles a square-root function because of the square-root variation of mean bronchial dose with minute volume. Minute volume is the product of the tidal volume and the breathing rate.
2. *Equilibrium Factor (EF)*. As the equilibrium factor increases from zero to unity, the dose will increase linearly with a slope of one.
3. *Unattached Factor (UF)*. As the unattached fraction increases, the tracheobronchial dose increases linearly from some constant positive y-intercept value.

The qualitative variation in the tracheobronchial dose as a function of the breathing rate, equilibrium factor, and unattached fraction is summarized in Figure S6.1.

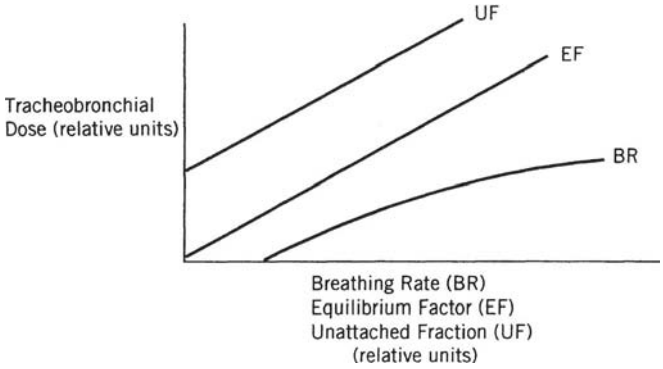


Figure S6.1 Tracheobronchial dose from radon and its daughters as a function of breathing rate, equilibrium factor, and the unattached fraction.

Scenario 6.2

Question 6.8

Mechanisms by which airborne concentrations of radioactive materials are reduced during atmospheric transport include:

1. Radioactive decay.
2. Diffusion or dispersion.
3. Precipitation removal (rainout or washout).
4. Gravitational settling.
5. Ground contact.

Question 6.9

The dose rate for a constant intake of radioactivity, as the activity is building up, is given by

$$\dot{D}(t) = \dot{D}_{\max}\,[1 - \exp(-0.693\,t/T_{1/2})] = \dot{D}_{\max}\,F_{\mathrm{BU}}$$

Once the intake stops, the dose rate decreases as expected from the following relationship:

$$\dot{D}(t') = \dot{D}_{\max}\,F_{\mathrm{BU}}\exp(-0.693\,t'/T_{1/2}) = \dot{D}_{\max}\,F_{\mathrm{D}}\,F_{\mathrm{BU}}$$
$$T_{1/2} = 1\ \mathrm{day}$$

The graph is plotted using the factors

$$F = F_{\mathrm{BU}}\,F_{\mathrm{D}}$$

For this problem, intake occurs only during the first 6 days. During the subsequent 6 days, decay occurs. Using this information, the points defining the desired graph can be determined:

Day (t)	Buildup F_{BU}	Decay F_D	F
1	0.50	1.0	0.50
2	0.75	1.0	0.75
3	0.87	1.0	0.87
4	0.94	1.0	0.94
5	0.97	1.0	0.97
6	0.98	1.0	0.98
7[a]	0.98	0.5	0.49
8	0.98	0.25	0.25
9	0.98	0.13	0.13
10	0.98	0.06	0.06
11	0.98	0.03	0.03
12	0.98	0.02	0.02

[a] The intake stops after day 6. No further buildup occurs in days 7–12.

Question 6.10

The dose rate for a constant intake of radioactive material, as the activity is building up, is given by

$$\dot{D}(t) = \dot{D}_{\max}\,[1 - \exp(-0.693\,t/T_{1/2})] = \dot{D}_{\max}\,F_{\mathrm{BU}}$$

Curve 1 assumes an effective half-life of 1 day, and curve 2 assumes an effective half-life of 2 days.

Day (t)	Curve 1 Buildup F_{BU}	Curve 2 Buildup F_{BU}
1	0.50	0.29
2	0.75	0.50
3	0.87	0.65
4	0.94	0.75
5	0.97	0.82
6	0.98	0.87
7	0.99	0.91
8	1.00	0.94
9	1.00	0.96
10	1.00	0.97
11	1.00	0.98
12	1.00	0.98

Question 6.11

The input rate to the pond's surface is given by

$$\begin{aligned} r_d &= (V)\,(Q)\,(X/Q) \\ &= (1.0 \times 10^{-2}\ \text{m/s})\,(1.0 \times 10^{8}\ \text{Bq/s})\,(1.8 \times 10^{-7}\ \text{s/m}^3) \\ &= 0.18\ \text{Bq/m}^2\text{-s} \end{aligned}$$

Question 6.12

The maximum steady-state concentration in the pond is

$$C_{eq} = (r_d)\,(S)/(\lambda_e)\,(V)$$

$$\lambda_e = \frac{0.693}{8\ \text{days}} + \frac{0.693}{15\ \text{days}} = 0.13/\text{day}$$

$$C_{eq} = \frac{(0.5\ \text{Bq/m}^2\text{-day})\,(100\ \text{m} \times 10\ \text{m})}{(0.13/\text{day})\,(1\ \text{m} \times 10\ \text{m} \times 100\ \text{m})} = 3.76\ \text{Bq/m}^3$$

Question 6.13

$$C_f = (C_{eq})\,(I)/(\lambda_e)$$

$$\lambda_e = \frac{0.693}{8\ \text{days}} + \frac{0.693}{21\ \text{days}} = 0.12/\text{day}$$

$$C_f = \frac{(2\ \text{Bq/m}^3)\,(8 \times 10^{-5}\ \text{m}^3/\text{kg-day})}{(0.12/\text{day})} = 1.34 \times 10^{-3}\ \text{Bq/kg}$$

Question 6.14

The question 6.13 model inaccuracies include:

1. Water evaporation which increases the I-131 concentration.
2. Decreased concentration due to rain and snow.
3. Decreased concentration due to settling.
4. The previous model assumes that all I-131 incorporated into the fish stays there. There will be an excretion function for the fish that reduces the activity.
5. The I-131 removal by other food-chain members is ignored.
6. Physical and chemical removal of I-131 by the pond.
7. Bottom feeding by fish may concentrate I-131.
8. Changes in pond pH may increase the evolution of I-131 from the pond.

Scenario 6.3

Question 6.15

Type of Sample	Preoperational Environmental Monitoring		Sample Analysis
	Location	Method/Frequency	
Air			
Particu-lates	Site boundary	Continuous/weekly filter change or High volume/24 hr every 7 days	1
	Nearest residence	Continuous/weekly filter change or High volume/24 hr every 7 days	1
	Control location (background)	Continuous/weekly filter change or High volume/24 hr every 7 days	1
Water			
Ground-water	Off-site existing wells (within 5 miles)	Grab sample/quarterly	2
	Site boundary (located hydrologically down-gradient)	Grab sample/quarterly	2
	Control location (located hydrologically up-gradient/ background)	Grab sample/quarterly	2
Surface water	Permanent on-site or off-site sources subject to direct drainage from contaminated site areas	Grab sample/quarterly	2
	Surface waters passing through the site or off-site sources subject to direct drainage from contaminated site areas	Grab sample/quarterly	2

Type of Sample	Preoperational Environmental Monitoring		Sample Analysis
	Location	**Method/Frequency**	
Vegetation, Food, and Fish			
Vegetation	Composite grab samples at the site boundary	Sample/spring and fall	3
Food	Composite grab samples within site boundary	Sample/spring and fall	4
	Crops within a few miles of the site boundary	Grab sample/time of harvest	2
Fish	Collection of fish in the site environs that may be subject to site runoff or seepage from contaminated areas	Grab sample/spring and fall	4
Soil and Sediment			
Soil	Immediate disposal area (1 sample per acre)	Grab sample/once	5
	Within site boundary (1 sample per 10 acres)	Grab sample/once	5
Sediment	Upstream and downstream of surface waters	Grab sample/once	5
	On-site surface waters	Grab sample/once	5
Direct Gamma			
	Representative locations across the site	Environmental TLDs/once	6

Analysis:
1 Quarterly composite, gamma scan, gross alpha, and gross beta
2 Gamma scan, gross alpha, gross beta, tritium, and carbon-14
3 Gamma scan, gross alpha, gross beta, tritium, and carbon-14 (species composition based on biomass surrounding plant site)
4 Gamma scan, gross alpha, gross beta, tritium, and carbon-14 (composite of edible portions)
5 Gamma scan, gross alpha, and gross beta
6 Gamma exposure rates from TLD, pressurized ion chamber, or properly calibrated survey instruments

Question 6.16

Major exposure pathways expected during the institutional care period are as follows:

1. Direct gamma from ground contamination and/or dust clouds.
2. Airborne contamination from dust or radioactive gas inhalation.
3. Ground water from contamination migration to a well or stream.
4. Contamination of surface waters through erosion or trench overflow.

Question 6.17

The risks for the various facility options are given by the product of the risk coefficient and the population doses:

Below Ground (BG):

$$\begin{aligned} \text{ecd} &= \text{excess cancer deaths} \\ r_{\text{BG}} &= (1.00 \times 10^{-4}\ \text{ecd/person-rem})\,(10\ \text{person-rem/year})\,(400\ \text{years}) \\ &= 0.4\ \text{ecd} \end{aligned}$$

Above/Below Ground (ABG):

$$\begin{aligned} r_{\text{ABG}} &= (1.00 \times 10^{-4}\ \text{ecd/person-rem})\,(9\ \text{person-rem/year})\,(400\ \text{years}) \\ &= 0.4\ \text{ecd} \end{aligned}$$

Above Ground (AG):

$$\begin{aligned} r_{\text{AG}} &= (1.00 \times 10^{-4}\ \text{ecd/person-rem})\,(30\ \text{person-rem/year})\,(400\ \text{years}) \\ &= 1.2\ \text{ecd} \end{aligned}$$

Question 6.18

The below-ground technology is the preferred technology. Because the site already meets the technical requirements for stability, an enhanced technology is not necessary to ensure the confinement of the radioactivity. Both above/below- and above-ground options would compromise the confinement that the site itself offers.

Worker doses and surface dose rates are higher with the above-ground disposal. Above-ground disposal does not take advantage of the site stability, and the radioactivity has more direct pathways to the environment.

Scenario 6.4

Question 6.19

$$dC/dt = S/V - \lambda C - I C$$
$$C = 25 \text{ pCi/liter}$$
$$V = 100 \text{ m}^3$$

For steady-state conditions, $dC/dt = 0$ and

$$S = V C (\lambda + I)$$
$$= (25 \text{ pCi/liter}) (100 \text{ m}^3 \times 10^6 \text{ cm}^3/\text{m}^3 \times 1 \text{ liter}/1000 \text{ cm}^3)$$
$$\times (0.00755/\text{hr} + 0.2/\text{hr})$$
$$= (5.19 \times 10^5 \text{ pCi/hr}) (1 \mu\text{Ci}/1.0 \times 10^6 \text{ pCi})$$
$$= 0.519 \mu\text{Ci/hr}$$

Question 6.20

$$S_w = N U C_w f_w$$
$$N = 5 \text{ members}$$
$$U = 200 \text{ liters/member-day}$$
$$C_w = 2000 \text{ pCi/liter}$$
$$f_w = 0.70$$
$$S = (5 \text{ members}) (200 \text{ liters/member-day}) (2000 \text{ pCi/liter}) (0.7)$$
$$\times (1 \mu\text{Ci}/1.0 \times 10^6 \text{ pCi})$$
$$= 1.4 \mu\text{Ci/day} \times 1 \text{ day}/24 \text{ hr} = 0.058 \mu\text{Ci/hr}$$

The source term from the water supply is approximately 10% of the total estimated radon source term for the home. Therefore, radon in the water supply alone cannot explain the source of the elevated radon level for the home.

Question 6.21

$$C_{WL} = C (EF) k$$
$$C = 25 \text{ pCi/liter}$$
$$EF = 0.3$$
$$k = \text{conversion factor (1 WL/100 pCi/liter)}$$
$$C_{WL} = (25.0 \text{ pCi/liter}) (0.3) (1 \text{ WL}/100 \text{ pCi/liter}) = 0.075 \text{ WL}$$

Question 6.22

To calculate the effective dose equivalent, the number of working level months per year should be calculated:

$$\begin{aligned}
\text{AWLM} &= (\text{OF})\,(C_{\text{WL}})\,k_1/k_2 \\
C_{\text{WL}} &= 0.1\ \text{WL} \\
\text{OF} &= 0.7 \\
k_1 &= \text{time conversion factor (365 days/year} \times \text{24 hours/day)} \\
k_2 &= 170\ \text{hr/month for WLM estimates} \\
\text{AWLM} &= (365\ \text{days/year})\,(24\ \text{hr/day})\,(0.70)\,(0.10\ \text{WL})/170\ \text{hr/month} \\
&= 3.61\ \text{WLM/year}
\end{aligned}$$

The dose equivalent rate to the tracheobronchial (TB) region of the lung is obtained from the AWLM:

$$\begin{aligned}
D_{\text{L}} &= (\text{DCF}_{\text{TB}})\,(\text{AWLM})\,(\text{QF}) \\
&= \text{committed dose equivalent rate to the TB region of the lung (rem/year)} \\
\text{DCF}_{\text{TB}} &= \text{tracheobronchial dose conversion factor} = 0.7\ \text{rad/WLM} \\
\text{QF} &= \text{quality factor for alpha particles} = 20\ \text{rem/rad} \\
D_{\text{L}} &= (3.61\ \text{WLM/year})\,(0.7\ \text{rad/WLM})\,(20\ \text{rem/rad}) \\
&= 50.5\ \text{rem/year delivered to the TB region of the lung} \\
H &= w_{\text{L}}\ D_{\text{L}} \\
&= \text{committed effective dose equivalent rate (rem/year)} \\
w_{\text{L}} &= \text{weighting factor for the lung (0.06)} \\
H &= 50.5\ \text{rem/year} \times 0.06 = 3.03\ \text{rem/year}
\end{aligned}$$

Question 6.23

Source. Because the water supply and building materials have been eliminated as the principal sources of radon, soil gas entry must be responsible for the home's radon levels.

Mitigation Methods

1. Ventilate the soil under the basement slab.
2. Seal cracks, sumps, and surfaces in the basement foundation.
3. Increase the home's ventilation with outside air.
4. Remove soil around the home's foundation and replace it with low-uranium-concentration material with low porosity.

Scenario 6.5

Question 6.24

K-40 and cosmic radiation are not significant factors in air monitoring. Both radon and thoron and their daughters can have a dramatic impact on air monitoring. When counting for alpha particles, it will be necessary to ensure that the natural products are properly taken into account.

Question 6.25

Cosmic radiation is a significant contributor to sample counting. It is a significant contributor to the background counting rate for low-level beta and gamma counting equipment, even if the equipment is well-shielded from terrestrial radiation sources.

Question 6.26

In vivo counting must account for natural radiation sources. The human subjects contain significant quantities of K-40 and may also have radon/thoron daughter activity on their bodies. K-40 has a positive impact because it provides a qualitative check on the proper operation of the counting equipment.

The equipment in the counting chamber may contain K-40, uranium, and thorium. The air within the chamber may contain radon and thoron and their associated daughter products. Some cosmic radiation will also enter the counting chamber. All of these sources must be properly determined in order to obtain accurate *in vivo* results.

Question 6.27

Cosmic radiation, K-40, the thorium series, and the uranium series will all contribute to background measurements. These sources and their variation make it difficult to detect small man-made contributions to the radiation background.

Question 6.28

Cosmic radiation, K-40, the thorium series, and the uranium series will all contribute to background radiation levels. For this reason, one cannot obtain a zero background for calibrating low-level instruments.

Questions 6.29

Nearly all construction materials, soil, and masonry contain K-40 and trace uranium. Some also contain thorium. Fallout from atomic bomb tests also provide residual radiation levels when they are contained in the construction material.

Shielding materials are often contaminated with natural background or fallout sources. For this reason, care must be exercised in choosing shield materials. Pre-World War II steel is preferred if it is available.

Question 6.30

Radon and thoron daughters are often contaminants in low-level counting laboratories. K-40 and the various isotopes of the uranium and thorium series may be found in laboratory items such as glassware and ceramic materials. Moreover, some of the chemical reagents will contain K-40 and some of the isotopes of the uranium and thorium series.

Scenario 6.6

Question 6.31

Net count rate on the gas channel monitor is

$$\begin{aligned} R &= R_{s+b} - R_b \\ &= 1.0 \times 10^7 \text{ cpm} - 50 \text{ cpm} \\ &= (1.0 \times 10^7 \text{ cpm})/(0.3 \text{ cpm/dpm}) \\ &= 3.33 \times 10^7 \text{ dpm} \end{aligned}$$

The air concentration may be obtained from knowledge of the net count rate and the detector volume:

$$\begin{aligned} C &= R/V \\ &= \frac{(3.33 \times 10^7 \text{ dpm})}{50 \text{ cm}^3} \times \frac{1 \,\mu\text{Ci}}{2.22 \times 10^6 \text{ cpm}} \\ &= 0.3 \,\mu\text{Ci/cm}^3 \end{aligned}$$

The stack release source term (Q) in Ci/min may be obtained from this information and the stack flow rate (F):

$$\begin{aligned} Q &= F\,C \\ &= (1.0 \times 10^5 \text{ ft}^3\text{/min})\,(0.3 \,\mu\text{Ci/cm}^3)\,(28.32 \text{ liters/ft}^3) \\ &\quad \times (1000 \text{ cm}^3\text{/liter})\,(1.0 \text{ Ci/l} \times 10^6 \,\mu\text{Ci}) \\ &= 850 \text{ Ci/min} \end{aligned}$$

Because the release is expected to last for 4 hr (t), the total quantity of Kr-85 released (q) is

$$q = Q\,t = (850\ \text{Ci/min})\,(4\ \text{hr})\,(60\ \text{min/hr})$$
$$= 2.04 \times 10^5\ \text{Ci}$$

The dose is related to these quantities by the relationship

$$D_i = \frac{X\,\overline{u}}{Q}\,(1/\overline{u})\,q\,\text{DCF}_i$$

Total Body Dose:

$$D_i = \frac{(5.0 \times 10^{-4}\ \text{m}^{-2})}{(2.0\ \text{m/s})}\,(2.04 \times 10^5\ \text{Ci})\,4.7 \times 10^{-4}\ \frac{\text{rem-m}^3}{\text{Ci-s}}$$
$$= 2.40 \times 10^{-2}\ \text{rem}$$

Skin Dose:

$$D_i = \frac{(5.0 \times 10^{-4}\ \text{m}^{-2})}{(2.0\ \text{m/s})}\,(2.04 \times 10^5\ \text{Ci})\,6.0 \times 10^{-2}\ \frac{\text{rem-m}^3}{\text{Ci-s}}$$
$$= 3.06\ \text{rem}$$

Question 6.32

Specific notifications would be specified in the facility's emergency plan and its associated implementing procedures. As a minimum, notifications to the host county emergency management coordinator, the state radiation protection branch, and Federal regulators (U.S. Nuclear Regulatory Commission or U.S. Department of Energy) would be recommended.

Because the whole-body exposures are low and the event is occurring and will end during the night, people will probably be indoors. This fact and the low doses suggest that no evacuation is warranted.

Question 6.33

Both on-site and off-site monitoring teams should be deployed. During the event, they should measure air concentrations of Kr-85 in order to verify the accuracy of the dose projections. Following the event, their measurements will confirm that the event has been terminated. In addition, ground deposition samples should be obtained for use by state and local officials to verify that none of the Kr-85 (10.7-year half-life) remains.

Question 6.34

Following the termination of the release, environmental monitoring TLDs should be collected in order to verify the population dose assessments.

Scenario 6.7

Question 6.35

Plant Source	Pathway
Gaseous effluents	• Submersion exposure • Inhalation • Deposition on crops and subsequent ingestion by man • Deposition on animal feed, ingestion of livestock meat • Deposition on animal feed, ingestion of livestock milk • Ground deposition, water contamination, and water immersion • Ground deposition, water contamination, and water ingestion • Ground deposition, water contamination, and aquatic food consumption • Ground deposition to direct exposure from the ground
Liquid effluents	• Immersion exposure • Shoreline exposure • Ingestion of water • Uptake by aquatic food, ingestion of aquatic food • Ground-water deposition, ingestion of livestock meat • Ground-water deposition, ingestion of livestock milk
Direct radiation and shine from N-16 sources or other radionuclides in the turbine hall steam piping systems	• Direct exposure

Question 6.36

The three potentially dominant pathways include:

1. Direct exposure from N-16. Any failed fuel would add to this direct exposure.
2. The submersion dose could be dominant, depending on the fuel performance and how well the turbine hall shields the N-16 gamma rays.
3. Without charcoal absorbers, the gaseous release of radionuclides could be significant.

The dominant pathway would be determined following the assessment of the plant's operating parameters and design characteristics. Calculations, including the information from question 6.37, would be needed to determine the dominant pathway.

Another significant pathway could be the cow's milk pathway. I-131 releases, subsequent ground deposition, and consumption by cattle would also require a more detailed assessment.

Question 6.37

To refine the pathway exposure estimates, additional factors including the following should be evaluated:

1. Population data by sector surrounding the plant.
2. Land use census.
3. Characterization of the population's behavior including their living habits, food preferences, and recreational preferences.
4. Meteorological data including seasonal averages and historical patterns.
5. Plant operating characteristics including the time between release and population exposure.
6. Biological census characterizing biota and their concentration factors for the nuclides that could be released from the facility.
7. Concentration factors for the various pathways.
8. Stream and aquifer volumes and associated flow rates. These water bodies will dilute any liquid effluent releases.
9. Facility release rates and source terms classified by their magnitude, particle size, and chemical form.

The results of a more detailed pathway analysis may suggest design changes to reduce the population dose from the higher-dose pathways. These changes could include a heavier (thicker) turbine casing to shield the N-16 gamma rays or the addition of charcoal absorbers to the gaseous discharge system.

Scenario 6.8

Question 6.38

The raccoon consumes both water and mice. The Ru-106 input rates of each are:

$$P_{\text{water}} = 8 \text{ kg-racoon} \times 0.08 \text{ L/kg racoon-d} \times 9.25 \text{ Bq/L} = 5.92 \text{ Bq/d}$$
$$P_{\text{mouse}} = 8 \text{ kg racoon} \times 0.002 \text{ kg mouse/kg racoon-d} \times 0.74 \text{ Bq/mouse} \times 1 \text{ mouse}/0.02 \text{ kg mouse} = 0.592 \text{ Bq/d}$$

The total input rate of Ru-106 into the raccoon (P) is:

$$P = P_{\text{water}} + P_{\text{mouse}} = 5.92 \text{ Bq/d} + 0.592 \text{ Bq/d} = 6.51 \text{ Bq/d}$$

Assuming a constant input rate P, the rate of change of Ru-106 activity in the raccoon ($\dot{A}$) is:

$$\dot{A}(t) = P\, e^{-k\,t}$$

The Ru-106 activity in the raccoon (A) after a time T is:

$$A(t) = \int_0^T \dot{A}(t)\,\mathrm{d}t = \int_0^T P\,e^{-k\,t}\,\mathrm{d}t = \frac{P}{k}\,(1 - e^{-k\,T})$$

where

k = effective removal rate of Ru-106 from the raccoon = 0.069/d
T = intake period = 1 y = 365 d

$$A(t) = \left(\frac{6.51\ \mathrm{Bq/d}}{0.069/\mathrm{d}}\right)[1 - e^{-(0.069/\mathrm{d})\,(365\ \mathrm{d})}] = 94.3\ \mathrm{Bq}\ (1 - 0) = 94.3\ \mathrm{Bq}$$

Assuming the Ru-106 is equilibrated in the body tissues, the concentration at the end of the first year C_{1y}:

$$C_{1y} = \frac{A}{m}$$

where m is the raccoon mass which is 8 kg. With these values,

$$C_{1y} = \frac{A}{m} = \frac{94.3\ \mathrm{Bq}}{(8\ \mathrm{kg})} = 11.8\ \mathrm{Bq/kg}$$

Question 6.39

The steady state activity is obtained when $k\,T \to \infty$. The previous result demonstrated that saturation was achieved during the first year. Therefore, the steady state activity is:

$$A_{ss} = C_{ss}\,m = (11.8\mathrm{Bq/kg})\,(8\ \mathrm{kg}) = 94.3\ \mathrm{Bq}$$

Question 6.40

The committed effective dose equivalent (CEDE) is defined as:

$$\mathrm{CEDE} = \frac{I}{\mathrm{ALI_S}}\,50\ \mathrm{mSv}$$

where I is the intake and the ingestion ALI is defined in the problem (7×10^6 Bq). The intake is obtained from the relationship:

$$I = \dot{V}\,C\,t$$

where

$\dot{V}$ = volume of water consumed per day = 2.2 liter/d
C = concentration of Ru-106 in the water = 9.25 Bq/liter
t = ingestion period = 1 y = 365 d

Therefore,

$$I = (2.2 \text{ liter/d})\,(9.25 \text{ Bq/liter})\,(365 \text{ d}) = (7.43 \times 10^3 \text{ Bq})$$

With these values, the CEDE is determined:

$$\text{CEDE} = \frac{I}{\text{ALI}_\text{S}}\,50 \text{ mSv} = \frac{7.43 \times 10^3 \text{ Bq}}{7 \times 10^6 \text{ Bq}}\,50 \text{ mSv} = 0.0531 \text{ mSv}$$

This is a small fraction of the dose from background radiation, which is quoted to be about 3 mSv in NCRP-93 or 2 mSv in ICRP-64.

Question 6.41

The effective half-life (T_e) is written in terms of the biological (T_b) and physical (T_p) half-lives:

$$T_e = \frac{T_p\,T_b}{T_p + T_b}$$

The biological half-life is obtained from this relationship:

$$T_e\,(T_p + T_b) = T_e\,T_p + T_e\,T_b = T_p\,T_b$$

$$(T_p - T_e)\,T_b = T_e\,T_p$$

$$T_b = \frac{T_e\,T_p}{(T_p - T_e)}$$

where

T_p = 1.02 y × 365 d/y = 372 d
T_e = 0.693/(0.069/d) = 10.0 d

Using these values, the biological half-life is:

$$T_b = \frac{T_e\,T_p}{(T_p - T_e)} = \frac{(10.0 \text{ d})\,(372 \text{ d})}{(372 \text{ d} - 10.0 \text{ d})} = 10.3 \text{ d}$$

Scenario 6.9

Question 6.42

This question involves a gaseous release described by a Gaussian Plume model.

a. The effective release height is increased.

The effective release height (H) for a gaseous release is given by the relationship:

$$H = h + d\left(\frac{v}{u}\right)^{1.4}\left(1 + \frac{\Delta T}{T}\right)$$

where

h = physical stack height or release elevation
d = stack exit diameter
u = mean wind speed
v = release velocity of the gas
T = absolute temperature of the released gas
ΔT = $T - T(0)$
$T(0)$ = absolute ambient temperature

If the effective release height increases, the exponential terms in the Gaussian Plume Equation lead to a decreasing ground-level concentration:

$$\chi = \frac{Q}{2\,\pi\,\sigma_y\,\sigma_z\,u}\exp\left[-\left(\frac{y^2}{2\,\sigma_y^2}\right)\right]\left[\exp\left(-\frac{(z-H)^2}{2\,\sigma_z^2}\right) + \exp\left(-\frac{(z+H)^2}{2\,\sigma_z^2}\right)\right]$$

where

χ = concentration on the plume centerline at the site boundary at (x, y, z)
Q = release rate
x = down-wind distance
y = cross-wind distance
z = elevation for concentration determination = 0 (ground level)
H = effective release height
σ_y = horizontal dispersion coefficient (m)
σ_z = vertical dispersion coefficient (m)

For a plume centerline ($y = 0$) and ground level ($z = 0$) concentration,

$$\chi = \frac{Q}{2\,\pi\,\sigma_y\,\sigma_z\,u}\left[\exp\left(-\frac{(H)^2}{2\,\sigma_z^2}\right) + \exp\left(-\frac{(H)^2}{2\,\sigma_z^2}\right)\right]$$

$$= \frac{Q}{\pi\,\sigma_y\,\sigma_z\,u}\left[\exp\left(-\frac{(H)^2}{2\,\sigma_z^2}\right)\right]$$

As indicated earlier, increases in H lead to a decrease in the ground-level concentration.

b. The wind speed increases.
As the wind speed (u) increases, the atmospheric stability decreases and the dispersion coefficients σ_y and σ_z increase. An examination of the Gaussian Plume equation verifies that increases in u and the dispersion coefficients, lead to smaller ground-level concentrations.

c. The ground temperature, which initially exceeds the temperature at 100 meters above the ground at the beginning of the release, decreases below the temperature at 100 meters.
In this question, the temperature gradient (Δ) is:

$$\Delta = \frac{T(100\text{ m}) - T(0\text{ m})}{100\text{ m}}$$

Initially, T (0 m) > T (100 m) and the gradient is negative. Finally, T (100 m) > T (0 m) and the gradient is positive.

A negative gradient is indicative of unstable meteorological conditions. For example, Stability Class A has the largest negative gradient and the magnitude of this gradient decreases as the atmospheric stability progresses through Stability Classes B, C, D, and E. Stability Class E includes the zero temperature gradient value. As the temperature gradient becomes positive and increases, the atmospheric stability class moves through increasing stability associated with Stability Classes E, F, and G.

In this question, the atmosphere proceeds from unstable to stable conditions, which decreases the magnitude of the dispersion coefficients (σ_y and σ_z). Decreasing σ_y and σ_z values increases the leading term of the Gaussian Plume Equation $\left(\frac{Q}{2\,\pi\,\sigma_y\,\sigma_z\,u}\right)$. However, the effect on the ground-level concentration depends on the release height H and magnitude of σ_z. Therefore, a definitive answer cannot be given unless H is known and values for σ_y and σ_z are specified.

The problem could also be cast in terms of the lapse rate (L) or negative temperature gradient:

$$L = -\Delta = -\frac{T(100\text{ m}) - T(0\text{ m})}{100\text{ m}}$$

Consideration of the gradient or lapse rate leads to the same result.

d. The gaseous effluent is heated above the ambient temperature and ejected from a vertical stack with a velocity exceeding that of the existing wind speed.
As v/u increases and T increases, the effective stack height (H) increases:

$$H = h + d\left(\frac{v}{u}\right)^{1.4}\left(1 + \frac{\Delta T}{T}\right)$$

An increasing effective release height, leads to a smaller ground-level concentration.

Question 6.43

This question involves a release through a pipe to the bottom and in the middle of a wide, shallow, straight river.

a. The effect on liquid radioactivity concentration if the temperature of the discharge is increased.

Assume that mixing occurs in a volume V of the river and that the mixing volume remains constant. The concentration (C) in the river is:

$$C = \frac{C_0\ V_0}{(V + V_0)}$$

where

C_0 = concentration of radioactivity in the release
V_0 = release volume

Under typical conditions, $V \gg V_0$ and

$$C = \frac{C_0\ V_0}{V}$$

Since water expands on being heated,

$$C_0^{\text{hot}} < C_0^{\text{cold}}$$

The release concentration decreases as the effluent temperature increases. Since the release rate to the river decreases, the concentration in the river also decreases.

The problem can also be solved by generalizing the Gaussian Plume equation for a fluid (i.e., liquid) release:

$$\chi(x, y, z) \rightarrow \chi(x, 0, 0) = \frac{Q}{\pi\ \sigma_y\ \sigma_z\ u}$$

Since the river is shallow, the exponential terms can be ignored. A temperature increase of the release fluid decreases the radioactive material concentration. Since the concentration decreases, the release rate decreases and the river concentration decreases.

b. The effect on liquid radioactivity concentration from an increased velocity of the discharge.

An increased discharge velocity leads to a larger release volume in a given time. The flow rate (F) is related to the release velocity (v) and release area (A):

$$F = v\ A$$

The release volume is rewritten in terms of the flow rate and release time (t):

$$V_0 = F\,t$$

Without additional information, it is reasonable to assume that the release concentration and release time are constant. As V_0 increases, the river concentration increases because more activity is released to the river per unit time:

$$C = \frac{C_0\,V_0}{V} = \frac{C_0\,F\,t}{V}$$

In terms of the generalized Gaussian fluid model, an increase in the discharge velocity increases the release rate (Q). An increase in the release rate, increases the river concentration.

c. The effect of the liquid radioactivity concentration from an increase in the river current.

The river current is characterized in terms of a flow rate (f):

$$C = \frac{C_0\,V_0}{V} = \frac{C_0\,V_0}{f\,t}$$

If all other factors are constant, an increase in river flow rate leads to a decrease in the river concentration.

Considering the generalized Gaussian fluid model, an increase in the river velocity is analogous to an increase in wind speed, which also increases σ_y and σ_z. If all other parameters are constant, an increase in the magnitude of the dispersion parameters decreases the river concentration.

d. The effect time has on the concentration of radioactive material in the sediment on the river bottom.

During the release period, the sediment concentration tends to increase. Assuming that the sediment has a removal efficiency (e), a river flow rate (F), and river concentration (C), the activity in the sediment during the release period (T), and post release time (t) is:

$$A = \frac{C\,F\,e}{k}\left(1 - e^{-k\,T}\right) e^{-k\,t}$$

where k is the total removal constant. Assuming the releases have ceased, normal water erosion and radioactive decay will decrease the sediment concentration as a function of the post-release time.

In addition, as the time increases, the activity concentration in sediment on the river bottom decreases with depth in the sediment. This is a valid conclusion if the activity release rate and sedimentation rate do not change with time. The reason for this conclusion is that radioactivity in sediment at greater depths has

a longer period for decay and the upper layers of sediment remove radioactive material before it reaches the river bottom.

A loose analogy with the generalized fluid model is also possible. The deposition of particulate material on a permeable surface leads to a decreasing concentration with depth. As the time following the release increases, the concentration decreases as the radioactive material decays and other removal mechanisms (e.g., radioactive material leaching) occur.

Question 6.44

A counting laboratory needs to establish the counting time of sediment samples for a given procedure and stated MDA of 3.7×10^{-3} Bq/g. If the counting time for both the background and the samples is the same, a count time of 4 minutes leads to the following MDA:

$$\text{MDA} = \frac{4.65\,\sigma_b}{K\,T} + \frac{3}{K\,T}$$

where

σ_b = standard deviation of the background count = $\sqrt{N_b\,T}$
N_b = background count rate = 85 cpm
T = sample and background count time = 4 min
σ_b = $\sqrt{N_b\,T} = \sqrt{(85\text{ c/min})\,(4\text{min})} = 18.4$ c
K = counting efficiency = 0.2 c/d

These values complete the specification of the MDA:

$$\text{MDA} = \frac{4.65\,\sigma_b}{K\,T} + \frac{3}{K\,T} = \left(\frac{1}{500\text{ g}}\right)\left(\frac{(4.65)\,(18.4\text{ c}) + 3\text{ c}}{(0.2\text{ c/d})\,(4\text{ min})}\right)$$
$$= (0.221\text{ d/min-g})\,(1\text{ min}/60\text{ s})\left(\frac{1\text{ Bq-s}}{\text{d}}\right) = 3.68 \times 10^{-3}\text{ Bq/g}$$

The calculated value for the 500 g sample meets the desired MDA (3.7×10^{-3} Bq/g).

Question 6.45

The rate of change of activity $(\dot{A})$ in the stream is written in terms of a release rate P and removal constant k:

$$\dot{A} = P\,e^{-kt}$$
$$P = \frac{C\,v}{T}$$

where

C = concentration of Cs-137 in the tank that is released to the stream
= 3.26×10^5 Bq/liter having a flowrate of 5.66 m^3/s
v = volume of the tank that is to be released to the stream
$v = 5.68 \times 10^4$ liters
T = time for the release of the tank = 2 d
k = removal rate = F/V
(Cs-137 radioactive decay is insignificant during the release period)
F = tank release rate = 5.68×10^4 liters/2 d = 2.84×10^4 liters/d
V = volume of stream that dilutes the release + release volume
(The release volume (v) $\ll$ stream volume (V))
= (5.66 m^3/s) (3600 s/h) (24 h/d) (2 d) = 9.78×10^5 m^3

Assuming that P is constant over the release period,

$$P = \frac{C\,v}{T} = \frac{(3.26 \times 10^5 \text{ Bq/liter})\,(5.68 \times 10^4 \text{ liters})}{(2 \text{ d})\,(24 \text{ hr/d})\,(3600 \text{ s/hr})} = 1.07 \times 10^5 \text{ Bq/s}$$

With these values, the removal rate is determined:

$$k = \frac{(2.84 \times 10^4 \text{ liters/d})\,(1 \text{ m}^3/1000 \text{ liters})\,(1 \text{ d}/24 \text{ hr})\,(1 \text{ hr}/3600 \text{ s})}{9.78 \times 10^5 \text{ m}^3}$$

$$= 3.36 \times 10^{-10}/\text{s}$$

The activity buildup as a function of time is obtained by integration:

$$A(T) = \int_0^T \dot{A}\,dt = \int_0^T P\,e^{-kt}\,dt = \frac{P}{k}\left(1 - e^{-kT}\right)$$

The concentration of Cs-137 in the water (C) is written in terms of A and the volume of water in the stream (V) and the release water volume (v):

$$C = \frac{A}{V + v}$$

Since $V \gg v$,

$$C = \frac{A}{V} = \frac{P}{k}\frac{\left(1 - e^{-kT}\right)}{V}$$

$$= \left(\frac{1.07 \times 10^5 \text{ Bq/s}}{3.36 \times 10^{-10}/\text{s}}\right)\left[\frac{\left(1 - e^{-(3.36 \times 10^{-10}/\text{s})\,(2 \text{ d})\,(24 \text{ hr/d})\,(3600 \text{ s/hr})}\right)}{(9.78 \times 10^5 \text{ m}^3)\,(1000 \text{ mL/liter})\,(1000 \text{ liters/m}^3)}\right]$$

$$= \frac{(3.18 \times 10^{14} \text{ Bq})\,(1 - 0.99994194)}{9.78 \times 10^{11} \text{ mL}} = 0.0189 \text{ Bq/mL}$$

The intake (I) due to the consumption of 250 g of fish is:

$$I = b\,C\,F$$

where

b = bioaccumulation factor = 2000

C = concentration in the water which is assumed to be in equilibrium with the consumable fish tissue

= 0.0189 Bq/mL

F = consumption of fish = 250 g

Using these values, the intake is determined:

$$I = bCF = (2000)\,(0.0189\ \text{Bq/mL})\,(250\ \text{g})\,(1\ \text{cm}^3/\text{g})\,(1\ \text{mL/cm}^3)$$
$$= 9.45 \times 10^3\ \text{Bq}$$

Using this intake, the dose (D) to the consuming individual is determined:

$$D = \left(\frac{I}{\text{ALI}}\right) 50\ \text{mSv}$$

where

ALI = ingestion ALI = 4 MBq

Using the input values, the dose is obtained:

$$D = \left(\frac{I}{\text{ALI}}\right) 50\ \text{mSv} = \frac{9.45 \times 10^3\ \text{Bq}}{4.0 \times 10^6\ \text{Bq}}\ 50\ \text{mSv} = 0.118\ \text{mSv}$$

Solutions for Chapter 7

Scenario 7.1

Question 7.1: d

The capture of a negative pion at the end of its range results in the deposition of about 40 MeV to the tumor. This energy is transferred via scattering or nuclear interactions to the atoms, molecules, neutrons, or protons at the tumor site. The resulting particles have high LET, and the charged particles have short ranges which selectively deposit the pion's energy at the tumor site. By comparison, Co-60 will irradiate all tissues including the tumor and healthy tissue.

Question 7.2: a

The dominant energy-loss mechanism for charged particles in matter is the collision of these particles with atomic electrons. These collisions do not appreciably deflect the path in tissue of a heavy charged particle like a pion, which is much more massive than the electron.

With an electron or positron beam, the beam particles have the same mass as the atomic electrons, and thus large-angle deflections and beam-spreading occur.

Elastic nuclear collisions through the Coulomb force have a much smaller cross-section, but these collisions involve multiple small-scattering angles which spread the beam. For heavy charged particles, multiple scattering is a significant effect.

For negative pions, multiple scattering by atomic nuclei is the dominant effect (response "a"). Response "e" would apply for electron or positron beams, but it is not the best general answer.

Question 7.3: c

Neutrons, photons, and muons are the penetrating radiations generated with the accelerator running. Residual photons are present from induced radioactivity and can also penetrate thick shielding.

Question 7.4: a

Induced beta and gamma activity will be present in the treatment area.

Question 7.5: b

This energy difference is required to overcome the nuclear binding energy in order to fragment the oxygen, carbon, or nitrogen nucleus.

Question 7.6: a

A negative pion decays into a negative muon, which then decays into an electron. Some neutron contamination is also expected.

Question 7.7

The distance traveled by the pion is expressed in terms of the standard particle range relationship:

R = range in g/cm^2

t = physical distance traveled (cm)

p = density of the material in g/cm^3

$R = t\,p$

$t = R/p = (9.1\ \text{g/cm}^2)/(0.95\ \text{g/cm}^3) = 9.6\ \text{cm}$

Question 7.8

Assumptions

1. Protons and heavier fragments have ranges less than 1 cm and stop in the sphere.
2. Neutrons and gamma photons lose negligible energy in the sphere when compared to protons and heavy fragments.

Based upon these assumptions, the energy deposited into the 1.0-cm sphere per stopped negative pion is:

Particle	Energy Deposited (MeV)
Protons	20
Heavier fragments	17
	37

The absorbed dose is defined in terms of energy deposited per unit mass. The mass of material contained within the 1.0-cm sphere is given by

$$M = \frac{4}{3}\pi r^3 \, p$$

where

M = mass of material (g) contained within a sphere of radius r
r = radius of sphere (cm) = 1.0 cm
p = material density (g/cm^3) $\rightarrow$ 1.0 g/cm^3 is a reasonable approximation for water
M = (4/3) (3.14) (1.0 cm)^3 (1.0 g/cm^3) = 4.19 g

The dose D is obtained from its definition, energy deposited per unit mass, and appropriate conversion factors. It is given by

$$D = \frac{(37 \text{ MeV})(1.6 \times 10^{-13} \text{ J/MeV})(1000 \text{ g/kg})}{(4.19 \text{ g})}\left(\frac{1 \text{ Gy}}{\text{J/kg}}\right) = 1.4 \times 10^{-9} \text{ Gy}$$

Question 7.9

The assumption that charged particles stop in the 1.0-cm sphere is accurate. A 20-MeV proton has a range of about 0.4 cm. The following additional data would be needed to refine the dose estimate:

1. A more accurate neutron and gamma-ray energy spectrum is needed. In addition, the double differential cross-section in terms of energy loss and angle is needed for hydrogen and oxygen.
2. Mean free paths for neutrons and gamma rays in water at the energies of interest.
3. Stopping power for low-energy negative pions in water (see Equations (2.1) and (2.3)).
4. Negative pion–water macroscopic cross-section for low energies (see Equation (2.2)).
5. The incident fluence for low-energy negative pions incident on the water sphere (see Equation (2.2)).

Scenario 7.2

Question 7.10: d

The most likely source of the Al-28 decay is the neutron activation of Al-27 resulting from a gamma–neutron reaction with deuterium in the beam stop. The neutrons may also be derived from electron-induced reactions such as (e, e' n).

Question 7.11: e

Equation (e) describes the Compton scattering of the electron beam.

Question 7.12: c

No radiation hazard would prevent the researcher from attending to his experiment.

Question 7.13

The following radiological controls would be appropriate:

1. Initial neutron surveys at low power.
2. Activation surveys at high power.
3. Determination of airborne contamination and the need for respiratory protection.

Question 7.14

The following data would be needed to properly evaluate the shielding design:

1. Physical configuration of the beam and specification of occupied areas.
2. Gamma-ray attenuation coefficients.
3. Neutron cross-sections.
4. Probability and consequences of gamma–neutron reactions including relevant isotopes produced and their radiological characteristics.

Question 7.15: e

The shielding requirements are essentially the same because the broad-beam TVL for lead for the two accelerators are nearly equal.

Question 7.16: a

To produce neutrons, sufficient energy to overcome the nuclear binding energy must be provided. Because the binding energy per nucleon exceeds 4-MeV, the 40-MeV machine produces significantly more neutrons.

Question 7.17: a

The bremsstrahlung production is sufficiently high for both 4- and 40-MeV electron LINACs to control the shielding design.

Scenario 7.3

Question 7.18

The following gaseous products are expected:

Radioactive	Toxic
C-11	NO_x
O-15	O_3
N-13	

Question 7.19

Target Z. High-*Z* targets increase bremsstrahlung production, which increases photonuclear and ozone production.

Beam Current. Both photonuclear and toxic gas production are directly related to beam current, so increasing current will increase their production rate.

Question 7.20

Essentially all neutrons produced between 35 and 100 MeV are produced within the giant resonance region. This region is sharply peaked in an energy region between the reaction threshold and about 35 MeV. The cross-section is significantly reduced and relatively constant between 35 and 100 MeV. Other mechanisms are present but account for only a small percentage of the neutrons produced.

Question 7.21

If an electron accelerator is properly shielded for bremsstrahlung, it is automatically shielded sufficiently for neutrons produced if the following conditions are meet:

1. The electron energy is less than the photo-pion threshold.
2. The shielding is concrete or another low-*Z* material.

Note: Mazes/labyrinths are separate considerations.

Question 7.22: a

The reaction (production) rate R equals $N \sigma \phi$ and ϕ is proportional to the beam current. Therefore, the production rate is a linear function of the beam current.

Scenario 7.4

Question 7.23

The 10-min nuclide will require about two half-lives to reach the 2-Bq/cm^3 level. During this period, the shorter-lived nuclide will decay through about 10 half-lives. Therefore, the time can be estimated sufficiently accurately using only the longer-lived radionuclide.

$$C(t) = C(0) \exp\,[-(\nu/V + \lambda)\, t]$$

$$2\ \text{Bq/cm}^3 = (6.3 \times 10^4\ \text{Bq/cm}^3) \exp\,\{-[(4.0\ \text{m}^3\text{/s}/560\ \text{m}^3) + 0.693/10\ \text{min} \times 1\ \text{min}(60\ \text{s})]\, t\}$$

$$t = -\ln(2/6.3 \times 10^4)/[(4/560) + (0.693/600)]\ \text{s}$$
$$= 1248.2\ \text{s} = 20.8\ \text{min}$$

Question 7.24

The concentration of the toxic gas varies after shutdown.

$$Z(t) = Z(0) \exp\,[-(\nu/V + 1/T)\, t]$$

$$0.1\ \text{ppm} = 3.5\ \text{ppm} \exp\,\{-[(4\ \text{m}^3\text{/s} \times 60\ \text{s/min}/560\ \text{m}^3) + (1/25\ \text{min})]\, t\}$$

$$t = -\ln(0.1/3.5)/(4 \times 60/560 + 1/25)\ \text{min}$$
$$= 7.59\ \text{min}$$

Scenario 7.5

Question 7.25

Assuming that the distribution of neutrons is isotropic, the dose equivalent rate can be written as

$$\dot{H} = I\, K\, k_1\, P\, (\text{DCF})/(4\, \pi\, r^2)$$

where

I = proton beam current = 25.0×10^{-6} A
k = charge/proton = 1.602×10^{-19} coulomb/proton
K = $1/k = 6.24 \times 10^{18}$ protons/A-s
k_1 = time conversion factor (3600 s/hr)

P = neutron production rate = 1.8×10^{-6} neutrons/proton
DCF = dose conversion factor = 3.5×10^{-8} rem-cm^2/neutron
r = distance from the target = 40 cm

$$\dot{H} = (25 \times 10^{-6}\ \text{A})\,(6.24 \times 10^{18}\ \text{protons/A-s})\,(3600\ \text{s/hr}) \times (1.8 \times 10^{-6}\ \text{neutrons/proton}) \times (3.5 \times 10^{-8}\ \text{rem-cm}^2\text{/neutron})/(4\,\pi)\,(40\ \text{cm})^2 = 1.8\ \text{rem/hr}$$

Question 7.26: b

Assuming a point source approximation, exponential attenuation, and a buildup factor of unity, we obtain the following attenuation relationship:

$$\text{AF} = \exp(-u\,t_b)\,(r_a/r_b)^2$$

where
AF = attenuation factor
r_a = 40 cm = distance from question 7.25
r_b = 400 cm = distance from question 7.26
t_b = 50 cm = shield thickness
u = neutron removal cross-section for concrete 0.08 cm^{-1}

$$\text{AF} = \exp(-0.08/\text{cm} \times 50\ \text{cm})\,(40\ \text{cm}/400\ \text{cm})^2 = 1.8 \times 10^{-4}$$

Question 7.27: a

A BF_3 proportional counter in a polyethylene moderator would have good sensitivity to neutrons while providing the best gamma discrimination.

Poorer Answers

b. GM tube at greater than 2 atmospheres in a polyethylene moderator. The bare GM tube has low neutron sensitivity and no gamma discrimination in a mixed field.
c. Silver-wrapped GM tube inserted in a polyethylene moderator. This detector is neutron-sensitive but has no gamma discrimination.
d. LiI(Eu) scintillator inserted in a polyethylene moderator. The LiI(Eu) system has some sensitivity to photons.
e. Cadmium-wrapped LiI(Eu) scintillator. There is no such detector.

Question 7.28: c

In the laboratory reference frame, both energy and fluence rate are peaked in the forward direction.

Scenario 7.6

Question 7.29

The current can be determined from the relationship

$$\begin{aligned} P &= I\,V \\ &= \text{average beam power (watts)} \\ &= P_{\text{peak}}\,(\text{DF}) \end{aligned}$$

where

P_{peak} = peak beam power = 5 MW
DF = duty factor = 0.01
P = $(5 \times 10^6\ \text{W})\ (0.01) = 5 \times 10^4\ \text{W}$
V = LINAC terminal voltage
= 10 MeV = 1×10^7 eV = 1×10^7 volts × e for an electron beam of charge e
I = average beam current
= $P/V = (5 \times 10^4\ \text{W})/(1 \times 10^7\ \text{V}) = 5 \times 10^{-3}\ \text{A}$

The desired production rate can be found from the empirical ozone production relationship:

$$\begin{aligned} \text{Production rate} &= (600\ \text{eV cm}^{-4}\ \text{A}^{-1}\ \text{s}^{-1})\ G\ I\ d \\ &= (600\ \text{eV cm}^{-4}\ \text{A}^{-1}\ \text{s}^{-1})\ (10.3\ \text{molecules}/100\ \text{eV}) \\ &\quad \times (5 \times 10^{-3}\ \text{A})\ (200\ \text{cm}) \\ &= 61.8\ \text{molecules cm}^{-3}\ \text{s}^{-1} \end{aligned}$$

Question 7.30

The steady-state concentration can be obtained by integrating the concentration rate expression:

$$\dot{C}(t) = \dot{C}(0)\ \exp\left[-(\nu/V + 1/T)\,t\right]$$

from $t = 0$ to time t. Because the production rate $\dot{C}(0)$ is a constant, the concentration at time t, $C(t)$, can be determined:

$$C(t) = \frac{\dot{C}(0)}{(\nu/V + 1/T)}\left\{1 - \exp\left[-(\nu/V + 1/T)\,t\right]\right\}$$

The steady-state concentration $C(\infty)$ occurs at large times – that is, $t \to \infty$:

$$C(\infty) = \frac{\dot{C}(0)}{(\nu/V + 1/T)}$$
$$= \frac{100 \text{ molecules cm}^{-3} \text{ s}^{-1}}{[(5 \text{ m}^3 \text{ s}^{-1}/75 \text{ m}^3) + (1/1800 \text{ s})]} = 1488 \text{ molecules/cm}^3$$

Question 7.31

After equilibrium was established, the cell concentration $C(0)$ was 10 ppm. After beam shutdown, this concentration $C(t)$ will decrease as a function of time:

$$C(t) = C(0) \exp[-(\nu/V + 1/T)\, t]$$

Solving for t yields the time to reach the desired concentration:

$$t = \ln[C(t)/C(0)]/[-(\nu/V + 1/T)]$$
$$= \ln(0.1 \text{ ppm}/10 \text{ ppm})/\{-[(5 \text{ m}^3 \text{ s}^{-1}/75 \text{ m}^3) + (1/2000 \text{ s})]\}$$
$$= 68.6 \text{ s}$$

Scenario 7.7

Question 7.32

The reaction of interest is p + O-16 → C-11 + Li-6. C-11, like most proton accelerator products, is a positron emitter. Consequently, the detected photons have an energy of 0.511 MeV with a yield of 2 gammas per disintegration of C-11.

The activity (A) of C-11 present 1 hour after the accidental exposure is given by the production equation for activation:

$$A = N \sigma \phi (1 - e^{-\lambda t_{\text{irr}}})\, e^{-\lambda t_{\text{d}}}$$

where

N = the number of O-16 atoms in the target tissue
= (1 cm^2) (10 cm) (1.0 g/cm^3) (6.02 × 10^{23} molecules/mole) × (1 atom O-16/molecule water)/(18 g/mole)
= 3.34 × 10^{23} atoms of O-16

σ = C-11 production cross-section = 20 mb/atom of O-16

ϕ = proton fluence rate
$$= \frac{(1.0 \times 10^{12} \text{ protons/pulse})\,(10 \text{ pulses/min})}{(1 \text{ cm}^2)}$$
= 1.0 × 10^{13} protons/cm^2-min

$T_{1/2}$ = C-11 half-life = 20.4 min
λ = C-11 disintegration constant

$$= \frac{\ln 2}{T_{1/2}} = \frac{0.693}{20.4 \text{ min}} = 0.034/\text{min}$$

t_{irr} = irradiation time = 1 min
t_d = decay time = 60 min

The expected count rate (R) in cpm is related to the activity by

$$R = e\,A$$

where

e = efficiency of detector = 0.1 counts/dis
$R = e\,N\,\sigma\,\phi\,(1 - e^{-\lambda t_{irr}})\,e^{-\lambda t_d}$
$= (0.1 \text{ c/dis})\,(3.34 \times 10^{23} \text{ atoms O-16})\,(20 \times 20^{-27} \text{ cm}^2/\text{atoms O-16})$
$\times (1 \text{ dis/p})\,(1.0 \times 10^{13} \text{ p/cm}^2\text{-min})\,(1 - e^{-(0.034/\text{min})\,(1 \text{ min})})\,e^{-(0.034/\text{min})\,(60 \text{ min})}$
$= (6.68 \times 10^9 \text{ cpm})\,(1 - 0.967)\,(0.13) = 2.87 \times 10^7 \text{ cpm}$

Question 7.33

For simplicity, assume that the detector efficiency for the accelerator background gamma radiation is similar to that for the Ra-226 gamma rays.

The NaI detector yields 400 cps in an exposure rate of 10 μR/hr due to Ra-226 gamma rays. A 1-mR/hr gamma background would correspond to a count rate R of

$$R_B = (1 \text{ mR/hr})\,(400 \text{ cps}/[10 \text{ μR/hr}])\,(1000 \text{ μR/mR})\,(60 \text{ s/min})$$
$$= 2.4 \times 10^6 \text{ cpm}$$

From the previous question, the expected arm count rate R equals 2.87×10^7 cpm. The background is a fraction of this value:

$$R_B/R = 2.4 \times 10^6 \text{ cpm}/2.87 \times 10^7 \text{ cpm} = 0.08$$

Because the background is only about 8% of the expected count rate, the activation of the arm should be easily detected.

Question 7.34

It is unlikely that the physicist's recollection of the event will be accurate, and a careful evaluation is warranted to reconstruct the sequence of events. A video-taped simulation with the worker re-enacting the event should be performed. The position of his arm and body relative to the proton beam needs to be accurately determined. Other actions to quantify the physicist's dose include: whole-body counting to assess the arm's isotopic activity, analysis of hair and blood for

induced activity, recreating the event with a phantom and appropriate dosimetry, and medical monitoring of the physicist for acute radiation syndrome symptoms and for indications of tissue damage in the arm. The calculated dose will be an important parameter in determining future medical decisions.

As a bounding case, consider that the 6-GeV proton beam is totally absorbed by the arm. For a 1-cm^2 beam (A) and 10-cm-thick arm (h), this would correspond to a volume (V) of

$$V = A\,h = (1.0\ \text{cm}^2)\,(10.0\ \text{cm}) = 10.0\ \text{cm}^3$$

The dose (D) delivered to the arm is

$$D = \frac{k\,I\,(\text{PRF})\,E\,t}{V\,p}$$

where

I = proton intensity = 1.0×10^{12} protons/pulse
PRF = pulse repetition rate = 10 pulses/min
p = arm density = 1.0 g/cm^3
E = proton beam energy = 6 GeV/proton
k = conversion factor
= $(1.6 \times 10^{-12}$ erg/eV) (1 rad/100 erg g^{-1}) (1×10^9 eV/GeV)
t = exposure time = 1 min

$$D = \frac{(1.0 \times 10^{12}\ \text{protons/pulse})\,(10\ \text{pulses/min})}{(10.0\ \text{cm}^3)\,(1.0\ \text{g/cm}^3)}\,(6\ \text{GeV/proton})$$
$$\times\,(1000\ \text{MeV/GeV})\,(1.6 \times 10^{-13}\ \text{J/MeV})\,(1000\ \text{g/kg})\,(1\ \text{min})\left(\frac{\text{Gy}}{\text{J/kg}}\right)$$
$$= 9.6 \times 10^5\ \text{Gy}$$

This calculation is conservative for the following reasons:

1. The 6-GeV beam will penetrate the arm.
2. The physicist probably did not hold his arm steady for the duration of the exposure.
3. An absorbed dose of 10^6 Gy would lead to a significant temperature rise (ΔT) that would cause the physicist to react and jerk his arm from the beam. The temperature rise is derived from the relationship:

   $$Q = m\,c\,\Delta T$$

   where
   Q/m = absorbed dose delivered to the arm = 10^6 Gy
   c = specific heat of water = 1 cal/g-°C

$$\Delta T = \frac{Q}{m\,c} = \left(\frac{10^6\ \text{Gy}}{1\ \text{cal/g-°C}}\right)\left(\frac{\text{J}}{\text{kg-Gy}}\right)\left(\frac{\text{kg}}{1000\ \text{g}}\right)\left(\frac{1\ \text{cal}}{4.186\ \text{J}}\right) = 239\ °\text{C}$$

A somewhat improved dose estimate is obtained by considering the dynamics of an accelerator beam striking a fixed target. Typically, only about 1% of the beam energy is deposited in the target. Assuming this situation is applicable to the unanticipated exposure situation leads to a somewhat improved estimate of (0.01) (9.6×10^5 Gy) or 9.6×10^3 Gy.

However, the dose is sufficiently large to be of medical concern. The extent of the tissue damage will probably be significant and medical attention is warranted. The affected area should be frequently monitored for changes that may occur.

In addition to the dose to the arm, the whole-body dose should be assessed. The whole-body dose is due to secondary neutron and gamma radiation resulting from reactions within the exposed area. The video may suggest that other areas of the body were subjected to the direct beam. These will also require evaluation.

Scenario 7.8

Question 7.35

An interlock is a device, which automatically shuts off or reduces the radiation emission rate from an accelerator to acceptable levels when an entry portal (e.g., a door) is opened to an elevated radiation area. In certain applications, an interlock can be used to prevent entry into a radiation area.

Personnel entrances into any high-radiation area or exclusion area shall be provided with a structure (door or equivalent device) with shielding equivalent to that required of the surrounding walls or a physical barrier (door, gate, chain, or suitable physical obstruction) at the entrance to a radiation protective labyrinth. The radiation levels outside the structure or barrier shall be no greater than that from the adjacent portions of the shield.

These personnel-access structures shall be equipped with interlock switches that cause the production of radiation from the accelerator to be terminated if the access barrier is opened or breached. Interlocks are emergency devices whose purpose is personnel protection and not as a means for routinely terminating accelerator operation.

Question 7.36

Beam dumps are incorporated into the design of experimental areas for high-energy, high-power accelerators. The purpose of the beam dump is to accept the radiation beam into a shielded cavity. The beam dump's dimensions and materials are designed to significantly reduce the reflection of radiation.

The beam dump is located on the axis of the accelerated beam. In some designs, the lining of the beam dump cavity is selected for its low radiation reflection characteristics. The entrance aperture into the cavity should only be large enough to accept the major components of the beam. The depth of the cavity should be greater than the diameter of the aperture to reduce the solid angle of back

reflection. Radiation shielding surrounding the cavity should provide adequate personnel protection in the area beyond and to the side of the beam dump. Based upon these considerations, the beam dump configuration described in the problem statement is acceptable.

Question 7.37

The radiation types and associated production mechanisms following an interaction between a particle beam and the material it strikes in an accelerator include:

1. X-rays (bremsstrahlung) are produced from the impingement of electrons on the target or neighboring materials.
2. Characteristic X-rays are produced from the impingement of either electrons or ions on the target or neighboring materials.
3. Prompt gamma radiation is generated from the impingement of ions or neutrons on the target or neighboring materials.
4. Neutron radiation is produced by the impingement of electrons, photons, or ions on the target or surrounding materials.
5. Delayed gamma and beta radiation are produced from induced activity in the target or neighboring materials or structures.
6. Pions and kaons, and muons from pion and kaon decays, occur from interactions with nuclei once the threshold energy for their production is reached. Production of these particles would not occur under the operating limitation of 100 MeV noted in the problem statement.

Question 7.38

The peak beam current is obtained from the shielding transmission equation and from the forward direction dose rate current ratio:

$$B_x \leq 1.67 \times 10^{-5} \frac{\text{Gy-hr}}{\text{mSv-min}} \left(\frac{\dot{H}_m \, d^2}{\dot{D}_o \, T} \right)$$

where

B_x = shielding transmission = 0.01 for the 75-cm shield thickness
$\dot{H}_m$ = maximum permissible dose rate limit = 0.025 mSv/hr
d = distance between the source and the reference point = 10 m
$\dot{D}_o$ = maximum permissible dose equivalent rate
= $(4.7 \times 10^2$ Gy-m^2/mA-min$)\ I$
I = peak beam current (mA)
T = Area occupancy factor = 1.0

The desired peak beam current is obtained by substituting the dose-rate equation into the shielding transmission equation:

$$B_x \leq 1.67 \times 10^{-5} \frac{\text{Gy-hr}}{\text{mSv-min}} \left[\frac{\dot{H}_m \, d^2}{\left(4.7 \times 10^2 \frac{\text{Gy-m}^2}{\text{mA-min}} \right) I \, T} \right]$$

This equation can be solved for the peak beam current:

$$I \leq 1.67 \times 10^{-5} \frac{\text{hr}}{\text{mSv}} \left[\frac{\dot{H}_m \, d^2}{\left(4.7 \times 10^2 \frac{\text{m}^2}{\text{mA}} \right) B_x \, T} \right]$$

$$\leq 1.67 \times 10^{-5} \frac{\text{hr}}{\text{mSv}} \left[\frac{(0.025 \text{ mSv/hr}) (10 \text{ m})^2}{\left(4.7 \times 10^2 \frac{\text{m}^2}{\text{mA}} \right) (0.01) (1)} \right] = 8.88 \times 10^{-6} \text{ mA}$$

The maximum beam current is 8.88×10^{-6} mA.

Question 7.39

The ozone concentration is defined as a function of time by the relationship:

$$C(O_3) = k \frac{S_{coll} \, I \, \chi \, t}{V}$$

where

k = conversion factor = $\dfrac{C(O_2) \, G \, e}{A_o}$

$C(O_2)$ = fractional concentration of oxygen in air
= $0.232 \times 1.0 \times 10^6$ ppm = 2.32×10^5 ppm

G = value for ozone production by electron irradiation of oxygen
= 6 molecules/100 eV

e = number of electronic charges per mA-s of electron beam current
= 6.28×10^{15}/mA-s

A_o = adjusted Avogadro's number
= 6.02×10^{23} molecules/22.4 liters of gas at STP

k = $\dfrac{(2.32 \times 10^5 \text{ ppm}) (6 \text{ molecules/100 eV}) (6.28 \times 10^{15}/\text{mA-s})}{(6.02 \times 10^{23} \text{ molecules/22.4 liters}) (1 \text{ keV/1000 eV})}$
= $3.25 \dfrac{\text{L-ppm}}{\text{keV-mA-s}}$

$C(O_3)$ = ozone concentration in ppm

S_{coll} = collision stopping power of electrons in keV/cm
= (1.98 MeV cm^2/g) (1000 keV/MeV) (0.001205 g/cm^3) = 2.39 keV/cm

I = Electron beam current in mA = 0.05 mA
χ = distance traversed in air by the electron beam in cm
= 2.0 m = 200 cm
t = irradiation time = 1 minute
V = volume of the irradiation chamber in liters
= 6 m × 8 m × 2.5 m × (100 cm/m)^3 (1 L/1000 cm^3) = 1.2×10^5 L

The reader should note that the ozone production equation given in the problem statement:

$$C(O_3) = 3.25 \frac{\text{L-ppm}}{\text{keV-mA-s}} \frac{S_{\text{coll}}\, I\, \chi\, t}{V}$$

is a special case of the general production equation. The production term (P) is:

$$P = 3.25 \frac{\text{L-ppm}}{\text{keV-mA-s}} \frac{S_{\text{coll}}\, I\, \chi}{V}$$

The buildup of ozone concentration $(\dot{C})$ in ppm/s is defined by a standard production relationship:

$$\dot{C} = P\, e^{-k\,t}$$

where
k = effective disintegration constant = $1/T + v/V$
T = mean lifetime of ozone = 50 min
v = irradiation cell ventilation rate = 2.0 m^3/min
V = irradiation cell volume = 120 m^3
k = 1/50 min + (2.0 m^3/min)/(120 m^3) = 0.02/min + 0.0167/min
= 0.037/min

The production relationship is integrated to obtain the production equation for ozone concentration as a function of time:

$$C(t) = \frac{P}{k}(1 - e^{-k\,t})$$

The equation provided in the problem statement represents a special case of the production equation when $k\,t$ is small such that the following approximation is valid:

$$e^{-k\,t} \approx 1 - k\,t$$

This approximation results in the simplification:

$$C(t) = \frac{P}{k}[1 - (1 - k\,t)] = \frac{P}{k}\,k\,t = P\,t = 3.25 \frac{\text{L-ppm}}{\text{keV-mA-s}} \frac{S_{\text{coll}}\, I\, \chi}{V}\, t$$

With these parameters, the ozone concentration after 1 minute of continuous operation is:

$$C(O_3) = \left(3.25 \frac{\text{L-ppm}}{\text{keV-mA-s}}\right) \frac{S_{\text{coll}}\, I\, \chi\, t}{V}$$

$$= \left(3.25 \frac{\text{L-ppm}}{\text{keV-mA-s}}\right) \times \frac{(2.39 \text{ keV/cm})\,(0.05 \text{ mA})\,(2.0 \text{ m})\,(100 \text{ cm/m})\,(1 \text{ min})\,(60 \text{ s/min})}{1.2 \times 10^5 \text{ L}}$$

$$= 0.0388 \text{ ppm}$$

Scenario 7.9

Question 7.40

The shield thickness is determined once the transmission (T) is determined:

$$T = \frac{\dot{D}(x)}{\dot{D}(0)}$$

where
$\dot{D}(x)$ = allowable dose equivalent rate in the office = 0.005 mSv/hr
$\dot{D}(0)$ = unshielded dose equivalent rate

Transmission values are obtained from the problem data. For an electron beam kinetic energy of 20 MeV and the 90 degree data (Sideward Direction), the following relationship was provided:

$$\frac{\dot{D}_{IO}}{I} = 20 \text{ Gy-m}^2\text{/mA-min}$$

where
I = average beam current (mA)
$\dot{D}_{IO}$ = X-ray emission rate from a high Z target

Using this relationship, the unshielded absorbed dose rate $\dot{D}(0)$ as a function of distance may be written as:

$$\dot{D}(0) = \left(\frac{\dot{D}_{IO}}{I}\right)\left(\frac{I}{r^2}\right)$$

where a point-source approximation of the target is assumed and

r = perpendicular distance from the target = 5 m
I = average beam current (mA) = $W_p\ P_{rf}\ I_p$
W_p = pulse width = 1.0 μs = 1.0×10^{-6} s
P_{rf} = pulse repetition frequency = 10 Hz = 10/s
I_p = peak beam current = 1 A
I = $(1.0 \times 10^{-6}$ s) (10/s) (1 A) = 1.0×10^{-5} A

Using these values, the unshielded absorbed dose rate is:

$$\dot{D}(0) = \left(\frac{\dot{D}_{IO}}{I}\right)\left(\frac{I}{r^2}\right)$$

$$= \left(20\ \frac{\text{Gy-m}^2}{\text{mA-min}}\right)\left(\frac{1.0 \times 10^{-5}\ \text{A}}{(5\ \text{m})^2}\right)(1000\ \text{mA/A})\ (60\ \text{min/hr})$$

$$= 0.48\ \text{Gy/hr}$$

Since this problem involves photon/bremsstrahlung radiation, a radiation weighting factor or quality factor of unity is appropriate. Therefore, the absorbed dose (Gy) is equivalent to the dose equivalent (Sv). With these values, the transmission (T) is determined:

$$T = \frac{\dot{D}(x)}{\dot{D}(0)} = \frac{(0.005\ \text{mSv/hr})\ (1\ \text{mGy/mSv})\ (1\ \text{Gy}/1000\ \text{mGy})}{0.48\ \text{Gy/hr}}$$

$$= 1.04 \times 10^{-5}$$

In applying this transmission value, the equivalent electron energy is obtained from the input data. The X-ray spectrum at 90° is lower in energy than the spectrum at 0°. This lower energy radiation is described in terms of an incident electron energy that would, in effect, produce X-rays with similar transmission characteristics in the 0° direction. Transmission values or tenth-value layer values applicable to the lower energy selected may be used in the calculation of shielding thicknesses for the 90° beam.

From the problem information, an incident energy of 20 MeV corresponds to an effective energy of 10 MeV. In addition, the problem data for broad beam transmission of 10 MeV photons through concrete indicates that a transmission of 1.0×10^{-5} corresponds to a shielding thickness of 191 cm. Therefore, the minimum thickness required for the concrete wall is 191 cm.

Question 7.41

The required transmission factor (T) is 1.0×10^{-4} and the existing concrete wall is 76 cm thick. The additional lead thickness, required to complement the concrete wall, is determined from the information supplied in the problem information that indicates that the existing 76 cm concrete wall has a transmission of 0.01:

$$T_{\text{wall}} = 0.01$$

The required transmission factor is given by the relationship:

$$T = 1.0 \times 10^{-4} = T_{\text{wall}}\, T_{\text{Pb}}$$

where T_{Pb} is the additional transmission from added lead shielding:

$$T_{\text{Pb}} = \frac{T}{T_{\text{wall}}} = \frac{1.0 \times 10^{-4}}{1.0 \times 10^{-2}} = 0.01$$

The tenth-value layer for 20 MeV incident (10 MeV effective) electrons is provided in the problem data. Given that the TVL for an effective energy of 10 MeV is 5.6 cm of lead, the desired lead addition is determined from the relationship:

$$T_{\text{Pb}} = (1/10)^N = 0.01$$

where N is the number of tenth-value layers of lead. The value of N is determined by taking the log of both sides of the previous equation:

$$\log_{10}[(1/10)^N] = \log_{10}(0.01)$$
$$N \log_{10}[(1/10)] = -N = -2$$
$$N = 2$$

The require shielding thickness (t) is just N times the TVL:

$$t = N(\text{TVL}) = (2)\,(5.6\text{ cm}) = 11.2\text{ cm}$$

Question 7.42

Parameters of significant importance when estimating the emission of radiation from an accelerator include:

a. Target composition – determines the type of atoms activated or involved in the reaction. Typical electron accelerator bremsstrahlung yields increase as the target's atomic number increases. In proton and heavy-ion machines, radiation output significantly increases when the ion energy exceeds the ion-target's Coulomb energy.
b. Target mass – determines the number of atoms activated or involved in the reaction.
c. Composition of the beam dump – see items a and b.
d. Configuration of the beam dump – governs scattering of the incident beam
e. Beam energy – governs the type of reaction that occurs: elastic scattering, inelastic scattering, or particle production. As the beam energy increases, the radiation yield from the target normally increases and the number of

available reaction channels increases, permitting additional radiation types to be produced.
f. Type of beam – the accelerated particle determines the types of radiation that are produced. For example, low-energy electrons produce bremsstrahlung. As the electron beam energy exceeds about 6 MeV, neutrons are produced. At energies above 140 MeV, charged pion production is possible.
g. Accelerator type (electron, proton, or heavy ion) – determines the radiation produced and its energy.
h. Mode of accelerator operation (continuous or pulsed) – determines the particle fluence rate or flux.
i. Pulse repetition frequency – determines the beam current that determines the flux.
j. Beam current – determines particle flux. The radiation output varies linearly with the average beam current.
k. Accelerator condition – operating or shut down. The radiation output depends on the accelerator's operating status.
l. Reaction under investigation – determines the cross-section.
m. Materials in the target room – determines activation sources.
n. Time after beam shutdown – determines activity as a function of time.
o. Effectiveness of beam optics package – failures of either the electric or magnetic field generators alter the beam's trajectory and may cause the beam to enter unanticipated areas.
p. Secondary radiations produced – affects the radiation characteristics of the machine.
q. Beam exposure time – determines activity as a function of time.
r. Target thickness – radiation yields vary with the target thickness.

Question 7.43

Considerations for selecting shielding materials for an accelerator include:

a. Maximum accelerator terminal potential.
b. Range of accelerated ions.
c. Types of ion sources and their output.
d. Maximum beam energy.
e. Types of target materials.
f. Beam format – pulsed or continuous.
g. Location of personnel.
h. Layout of the accelerator complex.
i. Beam fluence rate.
j. Accelerator upgrades planned and their impact on key accelerator parameters.
k. Available space to install shielding.
l. Primary and secondary radiation types produced.
m. Energy of radiation types produced.

n. Accelerator type (electron, proton, or heavy ion).
o. Design accident conditions (e.g., failure of beam optics package, loss of beam control, over-current conditions in the ion source power supply, and impact of unauthorized experiments).
p. Effectiveness of beam optics package – failures of either the electric or magnetic field will alter the beam's trajectory and may cause the beam to enter unanticipated areas.
q. Cost.
r. Type of shielding materials available.

Question 7.44

For each of the following accelerator types, the principle radiations of concern for occupied areas include:

a.

Accelerator	Particle Accelerated	Beam Energy (MeV)	Principle Radiations
Potential Drop	Protons/Deuterons	1–10	1. scattered protons and deuterons 2. neutrons[a] 3. nuclear fragments 4. photons[a]

[a] Dominant radiation type.

The energy region below 6–8 MeV leads only to elastic scattering. The primary radiation types are the protons and deuterons in the beam and scattered protons and deuterons. In the 6–100 MeV region, in elastic scattering dominates. Neutrons and nuclear fragments are produced, and neutrons dominate the shielding considerations.

b.

Accelerator	Particle Accelerated	Beam Energy (MeV)	Principle Radiations
Electron Linear	Electrons	1–10	1. electrons 2. photons[a] 3. neutrons[a] 4. bremsstrahlung

[a] Dominant radiation type.

Below 6 MeV, the primary electron beam and the photons it produces are the dominant health physics concerns. Between 6 and 50 MeV electrons and photons produced by electrons excite the nucleus and produce neutrons. At lower energies, neutron production is small.

c.

Accelerator	Particle Accelerated	Beam Energy (MeV)	Principle Radiations
Electron Linear	Electrons	> 10	1. electrons 2. photons[a] 3. neutrons[a] 4. pions 5. muons[a] 6. bremsstrahlung

[a] Dominant radiation type.

Between 6 and 50 MeV electrons and photons produced by electrons excite the nucleus and produce neutrons. At lower energies, neutron production is small. Between 30 and 150 MeV neutrons and bremsstrahlung are the principle radiations produced. Beyond 150 MeV, pions, muons, and neutrons are produced.

d.

Accelerator	Particle Accelerated	Beam Energy (MeV)	Principle Radiations
Cyclotron	Protons/Deuterons	10–50	1. neutrons[a] 2. nuclear fragments 3. photons[a]

[a] Dominant radiation type.

The region between 6 and 100 MeV lies in the inelastic scattering reaction region. Neutrons dominate the shielding requirements and nuclear fragments are also produced. Photons result from de-excitation of nuclei created in nuclear reactions.

e.

Accelerator	Particle Accelerated	Beam Energy (MeV)	Principle Radiations
Betatron	Electrons	1–50	1. electrons 2. photons[a] 3. neutrons[a] 4. bremsstrahlung[a]

[a] Dominant radiation type.

Below 6 MeV, the primary electron beam and the photons it produces are the dominant health physics concerns. Between 6 and 50 MeV electrons and photons produced by electrons excite the nucleus and produce neutrons. At lower energies, neutron production is small.

Solutions for Chapter 8

Scenario 8.1

Question 8.1

Organs and tissues are composed of electrically polarized molecules and biological fluids that contain ionic species including dissolved electrolytes and macromolecules. When subjected to high-frequency electromagnetic (EM) radiation, both polar molecules and ionic species encounter electric (**E**) and magnetic (**H**) fields, which produce forces (**F**) whose strength depends on the net charge (q):

$$\mathbf{F}_E = q\,\mathbf{E}$$

$$\mathbf{F}_M = q\,\mathbf{v} \times \mathbf{B} = q\,\mu_0\,\mathbf{v} \times \mathbf{H} = q\,\mu_0\,|\mathbf{v}|\,|\mathbf{H}|\sin\theta$$

where μ_0 is the permeability of free space, v is the ion velocity, θ is the angle between **v** and **H**, and B is the magnetic induction ($B = \mu_0 H$).

Moving charges generate currents (I) and electric currents produce Joule heating. In a similar manner, molecular vibrations produce heating in tissue and fluids:

$$P = I\,V = I^2\,R$$

where P is the power (heat energy/time), V is the potential difference induced by the EM field, and R is the resistance of tissue to the flow of charge. Thermal effects are the major impact of microwave exposure on human tissue.

Factors affecting the extent of the injury from microwave heating of tissue include:

1. The frequency of the microwave radiation.
2. The intensity of the electric and magnetic fields of the microwave radiation.
3. Exposure time.
4. Density of the absorbing material (soft tissue, bone, etc.)
5. Specific heat of the absorbing material.

6. Ability of the irradiated tissue to liberate generated heat via conduction, convection, or evaporation cooling.
7. Resistance of the absorbing material.
8. Potential difference inducing the currents.
9. Orientation of the exposed individual with respect to the EM field.
10. Distance from the microwave radiation source.
11. Presence of any shielding material between the source and the irradiated individual.
12. Conductivity of the absorbing medium.
13. Permittivity of the absorbing medium.
14. Relative dielectric constant of the absorbing medium.
15. Permeability of the absorbing medium.
16. Absorption coefficient of the absorbing medium.

Question 8.2

Items of information you need to perform a hazard evaluation include:

1. Characteristics of the emitters and their operating parameters including:
   a. Operating frequency.
   b. Transmitter peak power.
   c. Pulse width.
   d. Pulse repetition frequency.
   e. Antenna gain or dB.
   f. Antenna dimensions.
   g. Beam width.
   h. Scan or rotation angle.
2. Purpose and use of the emitter.
3. Site configuration and terrain.
4. Procedures followed in all phases of operation including maintenance and testing.
5. Magnitude and power densities created by the emitter.
6. Extent of hazardous and potentially hazardous areas both within the facility and in areas adjacent to the facility.
7. Locations of hazardous materials (petroleum, gasoline, oil, lubricants, and explosives).
8. Location of ionizing radiation sources due to microwave generating equipment such as klystrons, wave guides, and free electron lasers.
9. Control techniques, which will effectively reduce potential hazards. ALARA principles should be applied.
10. Other hazard sources including noise, chemical, ventilation, heat, stored energy, steam, and high voltage.
11. Location of potentially exposed individuals.
12. Location of controlled and uncontrolled areas.

Question 8.3

The average power that the antenna radiates is defined by the relationship:

$$P_{avg} = D_f \; P_{peak}$$

where
P_{avg} = average power radiated by the antenna
D_f = duty factor for the antenna system = 0.001
P_{peak} = peak power = 1.0 MW

With this relationship and defining parameters, the average power is determined:

$$P_{avg} = (0.001)\ (1.0\ \text{MW})\ (1.0 \times 10^6\ \text{W/MW}) = 1000\ \text{W}$$

Question 8.4

The near- and far-field distances are defined in terms of the wavelength of the emitted electromagnetic radiation. This wavelength is obtained from the relationship:

$$c = \nu\,\lambda \rightarrow \lambda = \frac{c}{\nu}$$

where
λ = wavelength of the electromagnetic radiation
c = speed of light = 3.0×10^8 m/s
ν = frequency of the radiation = 3000 MHz

With these values, the wavelength of the emitted antenna radiation is:

$$\lambda = \frac{(3.0 \times 10^8\ \text{m/s})}{(3000\ \text{MHz})\ (1.0 \times 10^6\ \text{Hz/MHz})\ (1/\text{s-Hz})} = 0.1\ \text{m}$$

The near-field distance is defined in terms of the wavelength and the properties of the antenna:

$$r_{nf} = \frac{D^2}{4\,\lambda}$$

where
r_{nf} = near-field distance
D = diameter of the antenna = 10 m

With these values, the near-field distance is determined:

$$r_{nf} = \frac{(10.0\text{ m})^2}{4\,(0.1\text{ m})} = 250\text{ m}$$

In a similar fashion, the far-field distance is determined:

$$r_{ff} = \frac{0.6\,D^2}{\lambda} = \frac{(0.6)\,(10.0\text{ m})^2}{(0.1\text{ m})} = 600\text{ m}$$

Question 8.5

Near field
The power density is uniform in the near-field and does not vary with distance. In the near-field,

$$W_{nf} = \frac{4\,P_{avg}}{A} = 4\,W_0$$

where
A = antenna area
W_0 = average power density

Far-field
In the far-field, the average power density varies as the square of the distance:

$$W_{ff} = 2\,W_0\left(\frac{r_{ff}}{r}\right)^2$$

where
r = distance from the source in the far-field region.

Question 8.6

In the near-field, the electric and magnetic fields are not perpendicular to each other. Their exact orientation varies from point to point. Therefore, a simple relationship between power density and electric or magnetic field strength does not exist in the near-field. Moreover, true near-field conditions occur only near the radiator at distances considerably less than one wavelength. The wide variation in the orientation of the of the electric and magnetic fields and the short range of the near field complicate power density measurements.

The reader should note that the hazard assessment's near field extends well beyond the actual near field. This is a conservative approach utilized in hazard assessments.

In the far field, the electric and magnetic fields are orthogonal. This fixed orientation and wide spatial region in which it exists, permits meaningful power-density measurements.

Question 8.7

The maximum power density at the near field is:

$$W_{\text{nf}} = \frac{4\,P_{\text{avg}}}{A} = \frac{4\,(1000\ \text{W})}{\pi\,(10.0\ \text{m}/2)^2} = 51.0\ \text{W/m}^2$$

Question 8.8

From question 8.4, the far-field does not begin until 600 m while the near field ends at 250 m. The 300 m distance lies in the intermediate field (if). In the intermediate field, the power density decreases in a linear fashion with distance from the source:

$$W_{\text{if}} = 4\,W_0\left(\frac{r_{\text{nf}}}{r}\right) = \frac{4\,P_{\text{avg}}}{A}\frac{r_{\text{nf}}}{r}$$

$$= (51.0\ \text{W/m}^2)\left(\frac{250\ \text{m}}{300\ \text{m}}\right)\left(\frac{1\ \text{m}}{100\ \text{cm}}\right)^2\left(\frac{1000\ \text{mW}}{\text{W}}\right) = 4.25\ \text{mW/cm}^2$$

Therefore, the power density at 300 m does not exceed the 10 mW/cm^2 ANSI Standard for controlled areas.

Question 8.9

The distance of 250 meters lies in the near field. The near-field power density was determined previously in question 8.7 to be 51.0 W/m^2:

$$W_{\text{nf}} = (51.0\ \text{W/m}^2)\left(\frac{1\ \text{m}}{100\ \text{cm}}\right)^2\left(\frac{1000\ \text{mW}}{\text{W}}\right) = 5.1\ \text{mW/cm}^2$$

which exceeds the ANSI Standard of 2.0 mW/cm^2 for uncontrolled environments. Therefore, placement of the fence at 250 meters would lead to exceeding the ANSI Standard for continuous exposure.

Question 8.10

Steps you should take to investigate the over-exposure allegation include:

1. Interview the worker to determine the duration and location of his exposure. Video taping the recreation of the event will enable a detailed time and position analysis.
2. Compare the worker's positions and times with the operating history of the antenna. The 1992 ANSI Standard retained 6 minutes as the time over which exposures, values of the specific absorption rate, and input power are to be averaged.
3. Calculate near-field, intermediate-field, and far-field power densities and associated distances for the alleged locations.
4. Compare the calculated values with the 1992 ANSI Standard to determine if the standard was exceeded.
5. Have the results reviewed by a qualified health physicist.
6. If an overexposure occurred, consult with a physician to evaluate potential health effects.
7. Review the results with management including legal counsel.
8. Discuss the results of your evaluation with the worker. Having a physician present would ensure that the worker receives a complete description of any possible health effects and the physiological consequences of the exposure.

Scenario 8.2

Question 8.11

The radiometric quantities and their appropriate units are:

a. Radiant energy 3. J
Radiant energy is energy in the form of electromagnetic waves usually expressed in J (W-s).

b. Radiant power 6. W
Radiant power is the time rate of flow of radiant energy, which is expressed in watts.

c. Radiant intensity 2. W/sr
Radiant intensity is radiant power in a given direction or radiant flux emitted from the source per unit solid angle (steradian), in the direction of propagation which is usually expressed in watts/steradian.

d. Radiance 4. W/sr-cm^2
Radiance is radiant flux (radiant power) output per unit solid angle per unit area in units of W/sr-cm^2. Radiance is also known as radiometric brightness.

e. Radiant exposure 5. J/cm^2
Radiant exposure is the total energy per unit area incident upon a given surface in a given interval of time. This unit is used to express exposure to pulsed laser radiation and it is usually expressed in units of J/cm^2.

Question 8.12

The following definitions are consistent with ANSI Z136.1-2007.

a. Nominal Hazard Zone (NHZ) – describes the space within which the level of the direct, reflected, or scattered radiation during operation exceeds the applicable MPE. Exposure levels beyond the boundary of the NHZ are below the applicable MPE level.
The specification of the NHZ is more appropriate for indoor laser laboratories because direct, reflected, and scattered radiation occur. The unobstructed beam is not the only concern. Laboratory conditions require the consideration of direct as well as reflected and scattered radiation.

b. Nominal Ocular Hazard Distance (NOHD) – is the distance along the axis of the unobstructed beam from the laser to the human eye beyond which the irradiance or radiant exposure during operation is not expected to exceed the appropriate MPE. The NOHD is usually more appropriate for outdoor laser applications. Scattering and reflection are less likely outdoors than in a laboratory setting.

Question 8.13

The nominal hazard zone (NHZ) for this laser laboratory is obtained from the following relationship and by assuming that the intra-beam MPE applies:

$$\mathrm{NHZ} = \sqrt{\frac{\rho_\lambda \, \Phi \cos\theta_v}{\pi \, \mathrm{MPE}}}$$

where

ρ_λ = spectral reflectance or reflectivity factor at a specified wavelength
= 0.9
λ = laser wavelength = 540 nm = 0.540 μm
Φ = total power of the continuous wave laser (W) = 20 W
θ_v = viewing angle measured from the normal of a reflecting surface = 0°
MPE = maximum permissible exposure determined from the intra-beam viewing table

In selecting the MPE, it should be noted that the laser is in the visible region. Following ANSI Z136.1, the exposure duration (t) is taken to be the blink aversion response time of 0.25 s. Using these values, the radiant exposure MPE is:

$$\text{MPE}(H) = 1.8\, t^{3/4} \times 10^{-3}\ \text{J/cm}^2 = 1.8\,(0.25)^{3/4} \times 10^{-3}\ \text{J/cm}^2$$
$$= 6.36 \times 10^{-4}\ \text{J/cm}^2$$

The radiant exposure (H) is converted into the irradiance (E) by dividing by the exposure duration:

$$\text{MPE}(E) = \text{MPE}(H)/t = \frac{(6.36 \times 10^{-4}\ \text{J/cm}^2)\left(\frac{\text{W-s}}{\text{J}}\right)}{(0.25\ \text{s})}$$
$$= (2.54 \times 10^{-3}\ \text{W/cm}^2)\,(1000\ \text{mW/W})$$
$$= 2.54\ \text{mW/cm}^2$$

Using these values, the NHZ is determined:

$$\text{NHZ} = \sqrt{\frac{\rho_\lambda\ \Phi \cos\theta_v}{\pi\ \text{MPE}}} = \sqrt{\frac{(0.9)\,(20\ \text{W})\cos 0^o}{(3.14)\,(2.54 \times 10^{-3}\ \text{W/cm}^2)}}$$
$$= (47.5\ \text{cm})\left(\frac{1\ \text{m}}{100\ \text{cm}}\right) = 0.475\ \text{m}$$

The NHZ has a maximum range of 0.475 m, and has a profile as a function of θ_v specified by the relationship:

$$\text{NHZ}(\theta_v) = 0.475\ \text{m}\ \sqrt{\cos\theta_v}$$

Question 8.14

The diameter (d) of a laser beam is usually defined as the distance within which $1/e^2$ of the total power exists. This problem provides the $1/e$ beam diameter as 2 mm. The $1/e$ and $1/e^2$ beam diameters are related by the relationship:

$$d_{1/e^2} = \sqrt{2}\ d_{1/e} = \sqrt{2}\ (2\ \text{mm}) = 2.83\ \text{mm}$$

The diameter used in the OD calculation depends on the relationship of the $1/e^2$ diameter to the diameter of the limiting aperture. In this case, the beam diameter is less than the limiting aperture diameter (7 mm).

The optical density (OD) is defined by the relationship:

$$\text{OD} = \log_{10}\frac{E}{\text{MPE}(E)}$$

where

E = irradiance for the appropriate area.

Generally, any time a beam (0.4–1.4 μm) is smaller than the limiting aperture, the limiting aperture is used in OD calculations. This selection is based on the foundations of ANSI Z136.1. As a beam becomes smaller than the limiting aperture, aberrations in the focusing mechanisms of the eye cause the beam to broaden over a large area of the retina. For a visible laser, the limiting aperture corresponds to the fully dilated pupil or 0.7 cm. In addition, the MPE is defined with respect to the limiting aperture.

$$E = P/A$$

P = power output of the CW laser = 20 W
A = limiting aperture area = πr^2
r = limiting aperture radius = 0.7 cm/2 = 0.35 cm
A = (3.14) $(0.35\ \text{cm})^2 = 0.385\ \text{cm}^2$

With these values, the irradiance is determined:

$$E = \frac{P}{A} = \frac{(20\ \text{W})}{(0.385\ \text{cm}^2)} = (51.9\ \text{W/cm}^2)\,(1000\ \text{mW/W})$$
$$= 5.19 \times 10^4\ \text{mW/cm}^2$$

The other quantity needed to define the OD is the MPE:
MPE(E) = maximum permissible exposure obtained from the intrabeam viewing table with 0.540 μm and a 0.25 exposure time (t) determined in the previous question
= 2.54 mW/cm^2

With these values, the OD is determined:

$$\text{OD} = \log_{10}\frac{E}{\text{MPE}(E)} = \log_{10}\frac{(5.19 \times 10^4\ \text{mW/cm}^2)}{(2.54\ \text{mW/cm}^2)}$$
$$= \log_{10}(2.04 \times 10^4) = 4.31$$

Question 8.15

The minimum optical density is specified to be 5.0. From the problem statement, the available goggles at 540 nm are:

Specifications	Brand X	Brand Y	Brand Z
OD @ 540 nm	5	7	8
Luminous Transmission	35%	6%	< 1%

All brands meet the optical density requirement of 5. However, protective eyewear is only effective if the researcher uses it. If the luminous transmission is too low, the researcher will be unable to see and hence unable to perform the task safely. The researcher may occasionally lift the eyewear to view the experiment if the luminous transmission is too low. Therefore, use Brand X, which meets the OD requirement and has the largest luminous transmission of the candidate protective eyewear.

Question 8.16

The following FDA laser classes are best matched with the characteristic that describes the class:

1. Class I	b. is not an ocular hazard.
2. Class II	c. 0.25 second exposure threshold.
3. Class III	d. momentary intrabeam viewing is hazardous.
4. Class IV	a. can damage skin or eye from diffuse reflection.

Question 8.17

The three basic components of all lasers include:

1. An amplifying or active medium in which a population inversion occurs. Examples include a mixture of helium and neon gases or a ruby crystal.
2. A source of energy to drive or pump the lasing medium to produce a population inversion. Examples include passing an electrical current (D.C.) through a mixture of helium and neon gasses (HeNe laser). In a pulsed ruby laser, the pumping system is a flash lamp.
3. An optical cavity to support the amplification and propagation of the laser beam. This cavity contains the active medium, sets up standing waves, and contains reflective surfaces. Laser amplification and stimulated emission occur within the optical cavity.

Scenario 8.3

Question 8.18

If the energy difference between the normal (ground) and the first excited state of an atom is E, the atom is capable of absorbing a photon whose frequency ν is given by:

$$E = h\,\nu$$

where
h = Planck's constant

The absorption of a photon by a neutral atom leads to the atom residing in an excited state (A*). A short time later, spontaneous emission occurs and the excited atom decays by emitting a photon of frequency ν in a random direction. In 1917, Einstein when deriving the laws of blackbody radiation, postulated the existence of a process which he called stimulated emission.

Stimulated emission takes place when a photon of energy $h\nu$ encounters an excited atom whose excitation energy is also $h\nu$, and de-excites the atom. The net result is the emission of another photon of the same frequency, in the same direction, and in the same phase. The two photons exit the atom as coherent, monochromatic radiation emitted in a specific direction.

Question 8.19

The unique characteristics of laser radiation include:

1. Monochromatic wavelength (same frequency).
2. Spatial coherence (same phase).
3. Emission is in a specific direction.

Question 8.20

The ANSI Z136.1-2007 laser classes and their general descriptions include:

1. Class 1 lasers are incapable of producing damaging radiation levels during operation, and are exempt from any control measures or surveillance.
2. Class 1M lasers are incapable of producing hazardous exposure conditions during normal operation unless the beam is viewed with an optical instrument.
3. Class 2 lasers emit in the visible portion of the spectrum. Eye protection is normally afforded by the aversion response mechanism.
4. Class 2M lasers emit in the visible portion of the spectrum. Eye protection is normally afforded by an aversion response for unaided viewing. However, Class 2M is potentially hazardous if viewed with certain optical aids.
5. Class 3R lasers are hazardous under some direct and specular reflection viewing conditions if the eye is appropriately focused and stable. The probability of an actual injury is small, and these lasers will not pose either a fire hazard or diffuse reflection hazard.
6. Class 3B lasers may be hazardous under direct and specular reflection viewing conditions, but are normally not a diffuse reflection or fire hazard.
7. Class 4 lasers are a hazard to the eye and skin from the direct beam, and may pose a diffuse reflection or fire hazard. These lasers also produce laser-generated air contaminants and hazardous plasma radiation (i.e., UV and blue light).

Question 8.21

Biological damage from laser light includes both thermal effects caused by energy absorption and photochemical reactions. The damage depends on the wavelength of the laser radiation and the tissue being irradiated, and the most sensitive tissues are the skin and the eye.

Laser radiation incident on the eye traverses the cornea, lens, and finally impinges on the retina. The degree of damage depends on the absorption and transmission of the laser radiation by the various tissues. Measured absorption and transmission curves are the basis for answering this question. Table 8.2 provides a summary of these data. However, as noted in ANSI Z136.1-2007, there is a "certain spread among the data". Therefore, threshold effects for all wavelengths are not yet fully established.

The skin-damage mechanisms include thermal and photochemical effects. The threshold for skin injury is comparable to the corneal hazard level. Most visible and IR-A radiation is reflected by the skin. The epidermis is less sensitive to UV-A. However, UV-B, UV-C, IR-B, and IR-C radiation are highly absorbed by the epidermis. Given the wavelengths presented in this problem, the skin can be significantly affected by each of the considered lasers.

Based on these considerations, tissues significantly affected by the indicated laser types are summarized in the following table:

Laser Type	Wavelength (μm)	Tissues significantly affected[a,b]			
		Skin	Cornea	Lens	Retina
CO_2	10.6	Y	Y	N	N
Nd-YAG	1.06	Y[c]	N	Y	Y
Ar	0.325	Y[c]	N	Y	N
HeNe	0.633	Y[c]	N	N	Y

[a] Y = yes; N = no.
[b] Following the commentary in ANSI Z136.1-2007, some variation in assignments may be expected because of threshold effect uncertainties.
[c] Depends on the laser power and exposure duration.

Question 8.22

The Federal agency that promulgates performance standards for laser products is the Food and Drug Administration (FDA). The Occupational Safety and Health Administration (OSHA) regulates the use of lasers and specifies user qualifications, posting and labeling requirements, exposure levels, and safety equipment.

Question 8.23

The US Code of Federal Regulations (CFR) title containing the FDA's regulations is 21 CFR. OSHA regulations are promulgated in 29 CFR.

Question 8.24

Controls specified in the ANSI Z136.1-2007 standard for Class 4 Lasers include:

1. Use of a protective housing to limit accessible levels of laser radiation.
2. Use of interlocks on a removable protective housing.
3. Service access panel.
4. Key control.
5. Remote interlock connector.
6. Beam stop or attenuator to keep radiation levels below the maximum permissible exposure (MPE) when the laser is in a standby mode.
7. Activation warning system (e.g., an audible or visible alarm to warn that the laser is about to be activated).
8. Establishment of a laser control area to prevent exposures above the MPE.
9. Equipment labels.
10. Standard operating procedures.
11. Authorized personnel.
12. Education and training.
13. Alignment procedures.
14. Protective equipment and eyewear.

These controls (safety features) are similar to those of the Federal Government (protective housing, safety interlock, remote connector, key control, emission indicator, and a beam attenuator).

Other controls include:
1. Beam tubes to restrict laser radiation from entering occupied areas.
2. Beam shutters to exclude the beam from occupied areas if interlocks are activated.

Scenario 8.4

Question 8.25

1. c. C_A – Correction factor which increases the MPE in the near-infrared (IR-A) region (0.7–1.4 μm), based on reduced absorption properties of melanin pigment granules found in the skin and in the retinal pigment epithelium.
2. e. C_B – Correction factor that increases the MPE in the red end of the visible spectrum (0.45–0.60 μm), based upon greatly reduced photochemical hazards.

3. a. C_C – Correction factor that increases the MPE for ocular exposure, based on pre-retinal absorption of radiant energy in the spectral region 1.15–1.40 μm.
4. d. C_E – Correction factor used for calculating the extended source MPE for the eye from the point source MPE, when the source subtends a visual angle exceeding α_{min}.
5. b. C_P – Correction factor that reduces the MPE for repetitive-pulse exposure of the eye.

Question 8.26: c

The fundamental ($n = 1$) and first harmonic ($n = 2$) wavelengths for a Nd-YAG laser are 1.064 μm and 0.532 μm, respectively. For the nth harmonic ($n \geq 2$), the wavelength is:

$$\lambda_n = \frac{\lambda_1}{n}$$

Question 8.27

The required OD must reduce the radiant exposure (H) at 1 m to the maximum permissible exposure (MPE) value. In order to answer this question, the MPE value must be determined for the ruby laser. The MPE is determined from the requirements of ANSI Z136.1-2007. This Standard requires the computation of three MPE values and the selection of the smallest value from the calculated set. The three MPE values are determined from the rules:

Rule 1: Single Pulse MPE
MPE(1) uses the pulse width as the exposure time and the MPE is determined from the laser wavelength (694.3 nm) and the pulse width (10 μs) for the exposure time. Using the table provided in the problem statement yields MPE(1) = 5×10^{-7} J/cm^2.

Wavelength λ (μm)	Exposure Time t (s)	MPE
0.400 to 0.700	10^{-9} to 1.8×10^{-5}	5×10^{-7} J/cm^2

Rule 2: Average Power MPE for Thermal and Photochemical Hazards

$$\text{MPE(2)} = \frac{\text{continuous wave (CW) MPE for the same wavelength}}{\text{PRF}}$$

In order to determine the MPE(2) value, we must again refer to the ANSI Table and the given PRF value (2/min):

Wavelength λ (μm)	Exposure Time t (s)	MPE
0.400 to 0.700	10^4 to 4×10^4	C_B x 10^{-6} W/cm^2

$$\text{MPE}(2) = \frac{(C_B \times 10^{-6}\ \text{W/cm}^2)\,(60\ \text{s/min})\,(\text{J/W-s})}{2/\text{min}}$$

where

C_B = $10^{15(\lambda - 0.550)}$ for $\lambda = 0.550$ to 0.700 μm
= $10^{15(0.6943 - 0.550)} = 10^{15(0.1443)} = 146$

$$\text{MPE}(2) = \frac{(146 \times 10^{-6}\ \text{W/cm}^2)\,(60\ \text{s/min})\,(\text{J/W-s})}{2/\text{min}} = 4.38 \times 10^{-3}\ \text{J/cm}^2$$

Rule 3: Multiple Pulse MPE for Thermal Hazards

$$\text{MPE}(3) = \text{single pulse MPE} \times C_P$$

where

C_P = repetitive pulse reduction factor = $n^{-1/4} \leq 1$
n = number of pulses over the exposure duration
= PRF × exposure duration

Following ANSI Z136.1-2007, the exposure duration is taken to be 0.25 s for visible lasers unless purposeful staring into the beam is intended. Since purposeful staring is not intended, an exposure duration of 0.25 s is indicated. Therefore,

$$n = (2/\text{min})\,(1\ \text{min}/60\ \text{s})\,(0.25\ \text{s}) = 8.33 \times 10^{-3}$$

Given this value and the limitation that C_P is a reduction factor with a maximum value of 1, C_P =1. Since,

$$\text{MPE}(3) = C_P \times \text{MPE}(1)$$
$$= \text{MPE}(1)$$

Therefore, the limiting MPE value is 5×10^{-7} J/cm^2.

Given the MPE, the radiant exposure H at 1 meter is calculated:

$$H = e/A = e/\pi r^2$$

where

e = laser pulse energy = 20 J
A = area of the laser beam = πr^2

r = laser beam radius at 1 m which is determined using the beam-spreading relationship and methodology from ANSI Z136.1-2007
$r = (a^2 + d^2\ D^2/4)^{1/2}$
a = aperture radius = 1 mm = 0.1 cm
d = distance from the aperture = 1 m = 100 cm
D = divergence angle = 15 mrad = 0.015 rad

$$H = \frac{20\ \text{J}}{\pi\,[(0.1\ \text{cm})^2 + (100\ \text{cm})^2\ (0.015)^2/4]^{1/2}} = \frac{20\ \text{J}}{\pi\,[0.757\ \text{cm}^2]}$$
$$= 8.41\ \text{J/cm}^2$$

NOTE: The laser beam area at 100 cm is $\pi\,[0.757\ \text{cm}^2] = 2.38\ \text{cm}^2$ which corresponds to a radius of 0.870 cm or a diameter of 1.74 cm. Since the beam area is larger than that corresponding to the limiting aperture diameter (0.7 cm), the actual area is used in the radiant exposure calculation.

Specification of the MPE and the radiant exposure permits the optical density (OD) to be determined:

$$\text{OD} = \log_{10}[H/\text{MPE}] = \log_{10}[(8.41\ \text{J/cm}^2)/(5 \times 10^{-7}\ \text{J/cm}^2)] = 7.23$$

Question 8.28

The emergent irradiance (E) is the radiant flux density leaving the aperture's surface and is usually expressed in units of W/cm^2:

$$E = P/A$$

where
P = laser power = 50 mW
A = aperture area = $\pi\ r^2$ = (3.14) (0.15 cm)^2 = 0.0707 cm^2
r = aperture radius = 0.3 cm/2 = 0.15 cm

The emergent irradiance is now determined:

$$E = (50\ \text{mW} \times 1\ \text{W}/1000\ \text{mW})/0.0707\ \text{cm}^2 = 0.707\ \text{W/cm}^2$$

Question 8.29

The hazardous inter-beam viewing distance or the nominal ocular hazard distance (NOHD) is given by the ANSI Z136.1-2007 relationship:

$$\text{NOHD} = \frac{1}{\varphi}\sqrt{\frac{1.27\ \Phi}{\text{MPE}} - d_a^2}$$

where

φ = beam divergence = 0.3 mrad = 3×10^{-4} rad

Φ = beam power = 50 mW × 1 W/1000 mW = 0.05 W

d_a = aperture diameter = 3 mm = 0.3 cm

MPE = maximum permissible exposure for continuous viewing

= $C_B \times 10^{-6}$ W/cm^2

$C_B = 10^{15(\lambda - 0.550)} = 10^{15(0.633 - 0.550)} = 10^{1.245} = 17.6$

MPE = 17.6×10^{-6} W/cm^2 = 1.76×10^{-5} W/cm^2

These values determine the NOHD:

$$\text{NOHD} = \frac{1}{3 \times 10^{-4}} \sqrt{\frac{1.27\,(0.05\text{ W})}{1.76 \times 10^{-5}\text{ W/cm}^2} - (0.3\text{ cm})^2}$$

$$= (3333)\,(3608\text{ cm}^2 - 0.09\text{ cm}^2)^{1/2} = 2.00 \times 10^5\text{ cm} = 2\text{ km}$$

Scenario 8.5

Question 8.30

In order to determine the MPE, the exposure duration must be determined. The MPE is determined from the requirements of ANSI Z136.1-2007. This Standard requires the computation of three MPE values and the selection of the smallest value from the calculated set. The laser of interest is pulsed with a frequency of 0.5 Hz and it has a pulse width of 25 ns.

Ocular MPE:

Rule 1: Single Pulse MPE

MPE(1) uses the pulse width as the exposure time and the MPE is determined from the laser wavelength (1.06 μm = 1060 nm) and the exposure time(25 ns). Using the MPE Table provides an MPE(1) = 5 $C_C \times 10^{-6}$ J/cm^2 where $C_C = 1.0$. Therefore, MPE(1) = 5×10^{-6} J/cm^2.

Rule 2: Average Power MPE for Thermal and Photochemical Hazards

$$\text{MPE(2)} = \frac{\text{continuous wave (CW) MPE for the same wavelength (W/cm}^2\text{)}}{\text{PRF}}$$

For the 1060 nm wavelength and the expectation that purposeful staring into the beam is neither intended nor anticipated, 10 s may be taken as the limiting exposure duration. Accordingly, MPE(2) is:

$$\text{MPE(2)} = \text{MPE(10 s)/PRF}$$

where

MPE(CW) = MPE(10 s) = 9.0 $C_C\, t^{3/4} \times 10^{-3}$ J/cm^2
$C_C = 1.0$
t = continuous exposure time (s) = 10 s
MPE(10 s) = 9.0 (1.0) $(10)^{3/4} \times 10^{-3}$ J/cm^2 = 5.06×10^{-2} J/cm^2
PRF = pulse repetition frequency = 0.5/s

In order to calculate MPE(2) the MPE(10 s) is converted into the expected units for continuous exposure (W/cm^2) using the maximum exposure duration (10 s).

$$\text{MPE(10 s)} = (5.06 \times 10^{-2}\ \text{J/cm}^2)\,(\text{W-s/J})/10\ \text{s} = 5.06 \times 10^{-3}\ \text{W/cm}^2$$

Given the MPE(10 s) value, MPE(2) can be determined:

$$\text{MPE(2)} = \frac{(5.06 \times 10^{-3}\ \text{W/cm}^2)\,(\text{J/W-s})}{(0.5/\text{s})} = 1.01 \times 10^{-2}\ \text{J/cm}^2$$

Rule 3: Multiple Pulse MPE for Thermal Hazards

$$\text{MPE(3)} = \text{single pulse MPE} \times n^{-1/4} \text{ with } n^{-1/4} \le 1$$

where

n = number of pulses over the exposure duration
= PRF × exposure duration

Following ANSI Z136.1-2007, the exposure duration is taken to be 10 s for 1060 nm lasers unless purposeful staring into the beam is intended. Since purposeful staring is not indicated, an exposure duration of 10 s is used. Therefore,

$$n = (0.5\ \text{pulses/s})\,(10\ \text{s}) = 5\ \text{pulses}$$

$$\begin{aligned}\text{MPE(3)} &= \text{MPE(1)} \times n^{-1/4} \\ &= 5 \times 10^{-6}\ \text{J/cm}^2 \times (5)^{-1/4} = 5 \times 10^{-6}\ \text{J/cm}^2 \times (0.669) \\ &= 3.34 \times 10^{-6}\ \text{J/cm}^2\end{aligned}$$

The three required calculations are summarized:

$$\text{MPE(1)} = 5 \times 10^{-6}\ \text{J/cm}^2$$
$$\text{MPE(2)} = 1.01 \times 10^{-2}\ \text{J/cm}^2$$
$$\text{MPE(3)} = 3.34 \times 10^{-6}\ \text{J/cm}^2$$

Therefore, the ocular MPE is the smallest of the calculated values or 3.34×10^{-6} J/cm^2 by Rule 3.

Skin MPE:
The skin MPE calculation parallels the eye calculation noted above.

Rule 1: Single-Pulse MPE
For a 25 ns pulse of 1.06 μm radiation, the skin MPE(1) is:

$$\text{MPE}(1) = 2\,C_A \times 10^{-2}\ \text{J/cm}^2$$

where
C_A = 5 for 1.06 μm

Using this value,

$$\text{MPE}(1) = 2\,(5) \times 10^{-2}\ \text{J/cm}^2 = 0.1\ \text{J/cm}^2$$

Rule 2: Average Power MPE for Thermal and Photochemical Hazards
For the 1.06 μm wavelength and the conditions of the problem, 10 s may be taken as the limiting exposure duration. Accordingly, MPE(2) is:

$$\text{MPE}(2) = \text{MPE}(10\ \text{s})/\text{PRF}$$

where
$\text{MPE}(2) = 1.1\,C_A\,t^{1/4}\ \text{J/cm}^2$
C_A = 5.0 for 1.06 μm
t = exposure time (s) = 10 s
$\text{MPE}(2) = 1.1\,(5.0)\,(10)^{1/4}\ \text{J/cm}^2 = 9.78\ \text{J/cm}^2$

In order to calculate MPE(2) the MPE(10 s) is converted into the expected units for continuous exposure (W/cm^2) using the maximum exposure duration (10 s).

$$\text{MPE}(10\ \text{s}) = (9.78\ \text{J/cm}^2)\,(\text{W-s/J})/10\ \text{s} = 0.978\ \text{W/cm}^2$$

Given the MPE(10 s) value, MPE(2) can be determined:

$$\text{MPE}(2) = \frac{(0.978\ \text{W/cm}^2)\,(\text{J/W-s})}{(0.5/\text{s})} = 1.96\ \text{J/cm}^2$$

Rule 3: Multiple-Pulse MPE for Thermal Hazards
Following ANSI Z136.1-2007 (Section 8.4.1), Rule 3 is not applicable to skin exposures.

The required skin MPE calculations are:

$$\text{MPE}(1) = 0.1\ \text{J/cm}^2$$
$$\text{MPE}(2) = 1.96\ \text{J/cm}^2$$
MPE(3) is not applicable.

The skin MPE is the smallest of the calculated values which is the MPE(1) value of 0.1 J/cm^2.

Comparing the ocular and skin MPE values demonstrates that the eye is limiting. Therefore, the ocular MPE by Rule 3 is the limiting value.

Question 8.31

The nominal ocular hazard distance (NOHD) is given by the relationship:

$$\text{NOHD} = \frac{1}{\varphi}\left(\frac{1.27\,\Phi}{\text{MPE}} - d_a^2\right)^{1/2}$$

where

φ = beam divergence = 0.001 rad
Φ = energy per pulse of the laser = 1.0 J

NOTE: The ANSI Standard defines Φ as the laser power. However, converting the laser energy to power would involve the same factor in the numerator and denominator of the NOHD equation. For simplicity, the extra time factors are not included in the NOHD equation [Energy (J) = Power (W) × Time (s)].

MPE = maximum permissible exposure = 7.0×10^{-6} J/cm^2
d_a = aperture exit diameter = 0.9 cm

Using the values noted above, the NOHD is determined:

$$\begin{aligned}\text{NOHD} &= \frac{1}{0.001}\left[\frac{(1.27)\,(1\text{ J})}{(7.0\times10^{-6}\text{ J/cm}^2)} - (0.9\text{ cm})^2\right]^{1/2}\\ &= (1000\text{ cm})\,(1.81\times10^5 - 0.81)^{1/2}\\ &= (4.26\times10^5\text{ cm})\left(\frac{1\text{ m}}{100\text{ cm}}\right)\left(\frac{1\text{ km}}{1000\text{ m}}\right) = 4.26\text{ km}\end{aligned}$$

In this problem, the following assumptions were made:

1. Air attenuation is neglected.
2. The beam spreads in a hyperbolic manner, which is the basis for the ANSI standard.
3. There is no atmospheric scattering of the beam.
4. Workers are wearing no device that provides eye protection. This would include personal protective equipment that would be common to an industrial environment, but not specifically designed for a laser environment (e.g., face shields, welding helmets, and safety glasses).

Question 8.32

In this question, you are to evaluate two CW lasers and determine the organs at risk for biological damage from unprotected exposure at 1) the exit port and 2) 2 kilometers (km). These are Ar [515 nm (CW) visible] and GaAs [905 nm (CW) IR-A] lasers. Both lasers are potentially eye (retinal) hazards since they are between 0.4 and 1.4 μm. At sufficient laser power, they may also present a skin hazard. Given the wavelengths, the cornea is not at risk. The risk of exposure will be based on the MPE values tabulated below.

ANSI Standard Z136.1-2007 supports 0.25 s (the aversion response time) for visible lasers and 10 s for Near-IR lasers (0.7–1.4 μm) for ocular exposure time durations for CW lasers if purposeful staring into the beam is not intended or anticipated. Using these exposure durations and their respective wavelengths, permit the determination of the appropriate MPE values.

Ar Laser (0.515 μm)

Ocular Exposure

$$\begin{aligned}\text{MPE (ocular)} &= 1.8\, t^{3/4} \times 1.0 \times 10^{-3}\ \text{J/cm}^2 \\ &= 1.8\,(0.25)^{3/4} \times 1.0 \times 10^{-3}\ \text{J/cm}^2 \\ &= 1.8 \times 0.354 \times 1.0 \times 10^{-3}\ \text{J/cm}^2 \\ &= 6.36 \times 10^{-4}\ \text{J/cm}^2\ \text{(radiant exposure)}\end{aligned}$$

The irradiance is obtained from the radiant exposure using the exposure time:

$$\text{MPE (ocular)} = \left(\frac{6.36 \times 10^{-4}\ \text{J/cm}^2}{0.25\ \text{s}}\right)\left(\frac{\text{W-s}}{\text{J}}\right) = 2.54 \times 10^{-3}\ \text{W/cm}^2$$

This MPE exceeds the tabulated irradiance at the aperture exit (2.0×10^{-3} W/cm^2) and at 2 km (2.9×10^{-4} W/cm^2). Therefore, the eye is not at risk from the Ar laser.

Skin Exposure

The skin MPE is calculated using the relationship

$$\text{MPE (skin)} = 1.1\, C_A \times t^{1/4}\ \text{J/cm}^2$$

for exposure times between 1×10^{-7} and 10 s. For exposure times between 10 and 3×10^4 s, the applicable relationship is:

$$\text{MPE (skin)} = 0.2\, C_A\ \text{W/cm}^2$$

Extended skin exposure times (compared to the 0.25 s aversion response) are more likely especially at the 2 km distance. At 515 nm, $C_A = 1.0$ and the following MPE values are determined:

$t = 0.25$ s

$$\text{MPE (skin)} = \frac{(1.1)\,(1.0)\,(0.25^{1/4}\ \text{J/cm}^2)}{0.25\ \text{s}}\left(\frac{\text{J}}{\text{W-s}}\right) = 3.11\ \text{W/cm}^2$$

$t = 10.0$ s

$$\text{MPE (skin)} = \frac{(1.1)\,(1.0)\,(10^{1/4}\ \text{J/cm}^2)}{10\ \text{s}}\left(\frac{\text{J}}{\text{W-s}}\right) = 0.196\ \text{W/cm}^2$$

$t = 10 - 3 \times 10^4$ s

$$\text{MPE (skin)} = (0.2)\,(1.0)\ \text{W/cm}^2 = 0.2\ \text{W/cm}^2$$

All of these MPE values exceed the tabulated irradiance at the aperture exit (2.0×10^{-3} W/cm^2) and at 2 km (2.9×10^{-4} W/cm^2). Therefore, the skin is not at risk from the Ar laser.

These results indicate that the MPE determination is insensitive to the skin exposure time (for exposure times of 10 s or greater). A conservative approach would suggest using a longer exposure time. The problem would need to provide additional data, including the nature of the task, the attire of personnel performing the task, and available monitoring equipment, to further justify the selection of the skin exposure time.

GaAs Laser (0.905 μm)

Ocular Exposure

$$\text{MPE (ocular)} = 1.8\,t^{3/4}\,C_\text{A} \times 1.0 \times 10^{-3}\ \text{J/cm}^2$$

where

C_A = $10^{2\,(\lambda - 0.7)} = 10^{2\,(0.905 - 0.7)} = 10^{0.41} = 2.57$

t = exposure duration = 10 s for the eye for the GaAs laser (905 nm)

MPE (ocular) = $1.8\,(10)^{3/4}\,(2.57)\,(1.0 \times 10^{-3})\ \text{J/cm}^2 = 2.60 \times 10^{-2}\ \text{J/cm}^2$

The irradiance is obtained from the radiant exposure using the 10 s exposure time:

$$\text{MPE (ocular)} = \left(\frac{2.60 \times 10^{-2}\ \text{J/cm}^2}{10\ \text{s}}\right)\left(\frac{\text{W-s}}{\text{J}}\right) = 2.60 \times 10^{-3}\ \text{W/cm}^2$$

This MPE is exceeded by the tabulated irradiance at the aperture exit (0.04 W/cm^2), but not at 2 km (6.3×10^{-7} W/cm^2). Therefore, the eye is at risk from the GaAs laser at the aperture but not at 2 km.

Skin Exposure

$$\text{MPE (skin)} = 1.1\, C_A \times t^{1/4}\ \text{J/cm}^2$$

where

$C_A = 10^{2(\lambda - 0.7)} = 10^{2(0.905 - 0.7)} = 10^{0.41} = 2.57$

$\text{MPE (skin)} = 1.1\,(2.57)\,(10.0)^{1/4}\ \text{J/cm}^2 = 5.03\ \text{J/cm}^2$

The irradiance is obtained from the radiant exposure using the 10 s exposure time:

$$\text{MPE (skin)} = \left(\frac{5.03\ \text{J/cm}^2}{10\ \text{s}}\right)\left(\frac{\text{W-s}}{\text{J}}\right) = 0.503\ \text{W/cm}^2$$

This MPE exceeds the tabulated irradiance at the aperture exit (0.04 W/cm^2) and at 2 km (6.3×10^{-7} W/cm^2). Therefore, the skin is not at risk from the GaAs laser.

Question 8.33

Only the GaAs laser presents an ocular hazard at the aperture but not at 2 km. To determine the extent of the GaAs hazard, the nominal ocular hazard distance (NOHD) is calculated:

$$\text{NOHD} = \frac{1}{\varphi}\left(\frac{1.27\,\Phi}{\text{MPE}} - d_a^2\right)^{1/2}$$

where

φ = beam divergence = 0.001 rad

Φ = power of the laser = 0.02 W

MPE = maximum permissible exposure (calculated in question 8.32) = 2.6×10^{-3} W/cm^2

d_a = aperture exit diameter = 0.8 cm

Using the values noted above, the NOHD is determined:

$$\text{NOHD} = \frac{1}{0.001}\left[\frac{(1.27)\,(0.02\ \text{W})}{(2.6 \times 10^{-3}\ \text{W/cm}^2)} - (0.8\ \text{cm})^2\right]^{1/2}$$

$$= (1000\ \text{cm})\,(9.77 - 0.64)^{1/2} = (3.02 \times 10^3\ \text{cm})\left(\frac{1\ \text{m}}{100\ \text{cm}}\right)$$

$$= 30.2\ \text{m}$$

The additional hazards associated with the lasers described in question 8.32 include:

1. The Ar laser is a flash-blinding hazard.
2. The use of optical devices (e.g., binoculars and periscopes) leads to an increase in the energy density to the eye.
3. Night vision or other infrared sensing devices could result in an increased eye detriment especially for the GaAs laser. The magnitude of the effect would depend on the characteristics of the IR device, particularly the transmission of IR radiation through the component's optical system.
4. Reflection, scattering, and refraction of laser beams from ground equipment and aircraft could divert the beam to the vicinity of ground or airborne personnel. These effects are wavelength and surface dependent phenomena.

Question 8.34

The question requests that the laser wavelength be matched with the tissue at risk. For specificity and for comparison with literature sources, the following classification scheme is noted:

Wavelength Range (μm)	Classification	Problem Wavelength	
		Number	Value
0.100–0.280	UV-C	3	0.28 μm
0.280–0.315	UV-B	3	0.28 μm
0.315–0.400	UV-A	5	0.36 μm
0.400–0.700	Visible	4	0.50 μm
0.700–1.400	IR-A	1	1.30 μm
1.400–3.000	IR-B	–	–
3.000–1.0×10^3	IR-C	2	5.00 μm

Wavelength (μm)	Tissues with Significant Energy Absorption
1. 1.3[a] (IR-A)	D. Skin, Retina, Cornea, and Lens
2. 5.0 (IR-C)	C. Skin and Cornea
3. 0.28 (UV-B/UV-C)	C. Skin and Cornea
4. 0.5 (visible)	B. Skin and Retina
5. 0.36 (UV-A)	E. Skin and Lens

[a] The 1.3 μm radiation is technically IR-A which impacts the lens and the retina. However, the delineation between IR-A and IR-B is near 1.3 μm so it is possible to find literature assigning this wavelength to IR-B, which impacts the cornea, retina, and skin. Given the data, answer D can be justified based on the proximity of the 1.3 μm radiation to the IR-A and IR-B boundary.

Scenario 8.6

Question 8.35

The absorption coefficient (b) is defined by the relationship:

$$b = \omega\left(\frac{\mu\,\varepsilon}{2}\right)^{1/2}\left\{\left[1+\left(\frac{\sigma}{\omega\,\varepsilon}\right)^{2}\right]^{1/2}-1\right\}^{1/2}$$

where

ν = operating frequency of the GP&L system = 2800 MHz

ω = angular frequency

= $2\,\pi\,\nu = (2)\,(3.14)\,(2800\times10^{6}/\text{s}) = 1.76\times10^{10}/\text{s}$

μ = permeability of body tissue

= $1.26\times10^{-6}\ \text{nt/A}^2 = 1.26\times10^{-6}\ (\text{kg-m/s}^2)/(\text{C/s}) = 1.26\times10^{-6}\ \text{kg-m/s-C}$

K = average relative dielectric constant of body tissue = 45

ε = dielectric constant

= $K\,\varepsilon_o = (45)\,(8.85\times10^{-12}\ \text{C}^2/\text{nt-m}^2) = 3.98\times10^{-10}\ \text{C}^2/\text{nt-m}^2$

= $3.98\times10^{-10}\ \text{C}^2/(\text{kg-m/s}^2)\text{-m}^2 = 3.98\times10^{-10}\ \text{C}^2\text{-s}^2/\text{kg-m}^3$

σ = conductivity of body tissue

= $2.2/\Omega\text{-m} = 2.2/(\text{kg-m}^2/\text{s-C}^2)\text{-m} = 2.2\ \text{s-C}^2/\text{kg-m}^3$

Inserting these values into the defining equation for the absorption coefficient yields:

$$b = (1.76\times10^{10}/\text{s})\left[\frac{(1.26\times10^{-6}\ \text{nt/A}^2)\left(\frac{\text{A}^2\text{-s}^2}{\text{C}^2}\right)(3.98\times10^{-10}\ \text{C}^2/\text{nt-m}^2)}{2}\right]^{1/2}$$

$$\times\left\{\left[1+\left(\frac{\left(\frac{2.2}{\Omega\text{-m}}\right)\left(\frac{\Omega\text{-s-C}^2}{\text{kg-m}^2}\right)}{(1.76\times10^{10}/\text{s})\,(3.98\times10^{-10}\ \text{C}^2\text{-s}^2/\text{kg-m}^3)}\right)^{2}\right]^{1/2}-1\right\}^{1/2}$$

$$= (1.76\times10^{10}/\text{s})\,(1.58\times10^{-8}\ \text{s/m})\,(1.048-1)^{1/2}$$

$$= (60.9/\text{m})\,(1\ \text{m}/100\ \text{cm}) = 0.61/\text{cm}$$

Question 8.36

The depth in tissue where the power density is reduced to 0.1 mW/cm^2 is determined from the relationship:

$$W(t) = W(0)\, e^{-2bt}$$

where

$W(t)$ = power density at a depth t into the tissue = 0.1 mW/cm^2
$W(0)$ = power density incident upon the absorber = 10 mW/cm^2
b = absorption coefficient = 0.61/cm
t = depth in absorbing medium

This equation is solved for the desired depth:

$$t = \left(-\frac{1}{2\,b}\right)\ln\left(\frac{W(t)}{W(0)}\right) = \left(-\frac{1}{(2)\,(0.61/\text{cm})}\right)\ln\left(\frac{0.1\ \text{mW/cm}^2}{10\ \text{mW/cm}^2}\right)$$
$$= (-0.82\ \text{cm})\,(-4.61) = 3.78\ \text{cm}$$

Question 8.37

The penetration depth (d_p) is defined as the reciprocal of the absorption coefficient:

$$d_p = \frac{1}{b} = \frac{1}{0.61/\text{cm}} = 1.64\ \text{cm}$$

Question 8.38

The maximum time an individual resides in the communications field, assuming that a limiting tissue temperature rise is 2 °C, is determined by calculating the energy absorbed by GP&L Reference Man. This energy becomes a heat input that causes the tissue's temperature to increase. In this problem, the GP&L Reference Man is exposed to the field according to the relationship:

$$P_{abs} = A\, W(t)$$

where

P_{abs} = heat energy absorbed by GP&L's Reference Man
A = effective area of absorber = 0.9 m^2
$W(t)$ = power density at the depth t in tissue
t = depth into tissue = 1.0 cm

The power density at a depth of 1 cm into tissue is obtained using the previously defined relationship:

$$W(1\ \text{cm}) = W(0)\, e^{-2bt} = (10\ \text{mW/cm}^2)\, e^{-2\,(0.61/\text{cm})\,(1\ \text{cm})} = 2.95\ \text{mW/cm}^2$$

With this value, the heat energy absorbed (P_{abs}) is determined by the power density at 1 cm depth and the area (A) of 0.9 m^2 of GP&L's Reference Man:

$$P_{abs} = A\,W(t) = (0.9\ m^2)\,(2.95\ mW/cm^2)\left(\frac{100\ cm}{m}\right)^2\left(\frac{1\ W}{1000\ mW}\right)$$
$$= 26.6\ W$$

This heat energy absorption and its associated temperature rise is given by the thermodynamic equality:

$$P_{abs}\,\tau = m\,c\,\Delta T$$

where
m = mass of GP&L reference man = $\rho\ V$
ρ = density of absorber = 1.0 g/cm^3
V = volume of absorber = $A\ x$
x = thickness of GP&L Reference Man = 2.0 cm
V = (0.9 m^2) (2.0 cm) (100 cm/m)^2 = 1.8×10^4 cm^3
m = $\rho\ V$ = (1 g/cm^3) (1.8×10^4 cm^3) (1 kg/1000 g) = 18 kg
c = specific heat of tissue = 4178 J/kg-°C
ΔT = temperature change (limit) caused by the absorption of the electromagnetic radiation = 2 °C
τ = time to reach the limiting temperature change

Using these values and the temperature change equation leads to the desired time:

$$\tau = \frac{m\,c\,\Delta T}{P_{abs}} = \frac{(18\ kg)\,(4178\ J/kg\text{-}°C)\,(2\ °C)}{(26.6\ W)}\left(\frac{W\text{-}s}{J}\right)$$
$$= (5.65 \times 10^3\ s)\,(1\ h/3600\ s) = 1.57\ h$$

Scenario 8.7

Question 8.39

The divergence angle (θ) of the laser beam is obtained from the diameter (D) and distance (d) measurements at 2 and 10 m:

$$\tan(\theta/2) = \frac{(1/2)\,[D(10\ m) - D(2\ m)]}{d(10\ m) - d(2\ m)}$$

An inspection of the diameter and distance values given in the problem statement indicates that the divergence is on the order of a milliradian. This permits the tangent to be expanded in a power series. With sufficient accuracy the tangent is approximated by the leading term of the expansion:

$$\tan x = x + \frac{x^3}{3} + \frac{2}{15}x^5 + \ldots \approx x$$

Using these two equations leads to the divergence angle:

$$\theta/2 = \frac{(1/2)\,[D(10\text{ m}) - D(2\text{ m})]}{d(10\text{ m}) - d(2\text{ m})}$$

$$\theta = \frac{[D(10\text{ m}) - D(2\text{ m})]}{d(10\text{ m}) - d(2\text{ m})} = \frac{(6\text{ mm} - 3\text{ mm})\,(1\text{ m}/1000\text{ mm})}{(10\text{ m} - 2\text{ m})}$$
$$= (3.75 \times 10^{-4}\text{ rad})\,(1000\text{ mrad/rad}) = 0.375\text{ mrad}$$

Question 8.40

The laser's duty cycle (D_C) is defined as the product of the pulse width (P_W) and the pulse repetition frequency (P_{RF}):

$$D_C = P_W\ P_{RF}$$

where
P_W = pulse width or pulse duration = 1 ms
P_{RF} = pulse repetition frequency = 100 pps

Using these values:

$$D_C = (0.001\text{ s})\,(100/\text{s}) = 0.1$$

Question 8.41

The average laser power is determined from the detector current and the detector calibration factor

$$\overline{P} = k\ I$$

where
$\overline{P}$ = average laser power
I = average current equivalent power measured with a photodiode detector = 40 μA
k = detector calibration factor = 2 μW/μA

With these values, the average laser power is determined:

$$\overline{P} = (2\ \mu\text{W}/\mu\text{A})\,(40\ \mu\text{A}) = 80\ \mu\text{W}$$

Question 8.42

The peak power (P_p) is determined from the average power ($\overline{P}$) and the duty cycle (D_C):

$$P_p = \frac{\overline{P}}{D_C} = \frac{80\ \mu W}{0.1} = 800\ \mu W$$

Question 8.43

The energy per pulse (E) is defined in terms of the peak power (P_p) and the pulse width (P_W):

$$E = P_p\ P_W$$

where
P_W = pulse width = 1 ms

With this relationship and the previously determined values, the energy per pulse is obtained:

$$E = P_p\ P_W = (800\ \mu W)\ (0.001\ s/pulse)\ (1.0 \times 10^{-6}\ W/\mu W)\ (J/W\text{-}s)$$
$$= 8.0 \times 10^{-7}\ J/pulse$$

Scenario 8.8

Question 8.44

In order to determine if the TLV is exceeded, assume the data is applicable to personnel performing the tasks associated with the light box and the BSC. Prior to considering the TLVs, note that the spectral effectiveness is reflected in these values. The spectral effectiveness values are defined relative to the 270 nm value. For example, TLV(270 nm)/TLV(i) = Spectral effectiveness for wavelength i.

Light box (300 nm):
The ratio of the radiant exposure (H) and TLV values is:

$$\frac{H(300\ nm)}{TLV(300\ nm)} = \frac{E(300\ nm)\ t}{TLV(300\ nm)}$$

where
E(300 nm) = irradiance at 300 nm = 1 $\mu W/cm^2$
t = exposure time = 20 min
TLV(300 nm) = 100 J/m^2

$$\frac{H(300\text{ nm})}{\text{TLV}(300\text{ nm})} = \frac{(1.0 \times 10^{-6}\text{ W/cm}^2)\,(20\text{ min})\,(60\text{ s/min})\,(100\text{ cm/m})^2\,(\text{J/W-s})}{(100\text{ J/m}^2)}$$

$$= \frac{12\text{ J/m}^2}{100\text{ J/m}^2} = 0.12$$

BSC (280 nm):
The ratio of the radiant exposure and TLV values is:

$$\frac{H(280\text{ nm})}{\text{TLV}(280\text{ nm})} = \frac{E(280\text{ nm})\,t}{\text{TLV}(280\text{ nm})}$$

where
$E(280\text{ nm})$ = irradiance at 280 nm = 0.6 μW/cm^2
t = exposure time = 30 min
$\text{TLV}(280\text{ nm})$ = 34 J/m^2

$$\frac{H(280\text{ nm})}{\text{TLV}(280\text{ nm})} = \frac{(0.6 \times 10^{-6}\text{ W/cm}^2)\,(30\text{ min})\,(60\text{ s/min})\,(100\text{ cm/m})^2\,(\text{J/W-s})}{(34\text{ J/m}^2)}$$

$$= \frac{10.8\text{ J/m}^2}{34\text{ J/m}^2} = 0.318$$

The sum of fractions (S) rule ensures exposures from multiple sources are within the limits:

$$S = \frac{H(280\text{ nm})}{\text{TLV}(280\text{ nm})} + \frac{H(300\text{ nm})}{\text{TLV}(300\text{ nm})} = 0.318 + 0.12 = 0.438 < 1$$

Since the sum of fractions is less than 1, the individual TLVs and total TLV were not exceeded.

Question 8.45

The assumption of irradiance at the peak wavelength is not an accurate approximation, because both sources emit a spectrum of wavelengths. A peak output approximation would be true if the devices were monochromatic sources such as lasers. The bulbs and boxes radiate over a spectrum of wavelengths. Therefore, all light output does not occur at the peak value.

Question 8.46

Methods to improve the assessment of the exposure to an individual working with the UV generating equipment include:

Method 1:
1. Measure the radiant exposure as a function of wavelength.
2. Using the values of the radiant exposure as a function of wavelength, calculate the sum of fractions for each source.

Method 2:
A more accurate method would be to measure the spectrum as noted in Method 1, and to account for the temporal and spatial variation of the worker. This method would be the application of the first method as a function of time and worker location.

Method 3:
The exposures could be determined using TLDs prepared and calibrated for use as UV detectors. The dosimeters could be placed on various body locations (e.g., head, hands, and trunk) to obtain reasonably accurate integrated exposure values.

Question 8.47

The skin and the eye are two biological tissues at risk from excessive UV light exposure. Excessive exposure would be manifest for each of these tissues in a variety of ways. In the eye, the cornea and lens are affected. For cornea surface UV absorption, conjunctivitis or inflammation of the corneal membrane occurs with recovery within 48 hours. For deeper corneal absorption, photochemical reactions (e.g., the cornea loses transparency and becomes milky in color) occur with an onset of 6–12 hours. Recovery requires a corneal graft. Exposure of the lens leads to a loss of elasticity and cataracts.

Exposure of the skin leads to erythema and photochemical reactions or tanning. Chronic exposures cause premature aging and increase the risk of cancer.

Question 8.48

Steps that could be implemented to lower the exposure to UV light in this lab include:

1. Use protective clothing (e.g., gloves and UV-absorbing safety glasses) to cover exposed skin and limit exposure to the eyes.
2. Adjust the light intensity to the minimum value required for process needs.
3. Replace manual operations with robotic techniques or processes that remove the worker from the light source.
4. Use light sources that only emit the process required wavelengths.
5. Use UV opaque shielding material whenever feasible.

Question 8.49

Precautions that could be followed to prevent harm to technicians working in the microwave laboratory include the following:

1. Limit access to the room when the microwave generator is powered.
2. Shield the generator and experimental area to limit stray fields.
3. Minimize X-ray output by shielding the high-voltage power supplies.
4. Construct access barriers to minimize worker access to the UV fields.
5. Interlock the beam output and the laboratory access control system to shutdown the UV sources if an individual inadvertently enters the controlled area.
6. Use microwave and UV absorbing materials on walls and surfaces to limit scattered radiation.

Question 8.50

Assuming far-field conditions, the maximum plane-wave free space power density at a distance (r) of 3.0 meters is obtained from the relationship:

$$W_{ff} = 2\, W_o \left(\frac{R_{ff}}{r}\right)^2$$

where

W_{ff} = far-field plane-wave free space power density (mW/cm^2)
W_0 = average power density = P_{avg}/A
P_{avg} = average power output of the antenna = $P_{peak}\, D_f$
P_{peak} = peak power output of the antenna = 2000 W
D_f = duty factor = $W_p\, P_{rf}$
W_p = pulse width = 0.001 s/pulse
P_{rf} = pulse repetition frequency = 200 pps
D_f = (0.001 s/pulse) (200 pulses/s) = 0.2
P_{avg} = $P_{peak}\, D_f$ = (2000 W) (0.2) = 400 W
A = area of microwave horn = $\pi\, (D/2)^2$
D = diameter or maximum horn dimension = 0.2 m
A = (3.14) (0.1 m)^2 = 0.0314 m^2
W_0 = P_{avg}/A = (400 W)/(0.0314 m^2) = 1.27×10^4 W/m^2
R_{ff} = far-field distance = $0.6\, D^2/\lambda$
λ = antenna output wavelength = c/ν
c = speed of light = 3.0×10^8 m/s
ν = frequency of the antenna radiation = 10 GHz = 1.0×10^{10}/s
λ = c/ν = (3.0×10^8 m/s)/(1.0×10^{10}/s) = 0.03 m
R_{ff} = (0.6) (0.2 m)^2/(0.03 m) = 0.8 m
r = distance of interest = 3.0 m

$$W_{ff} = 2\, W_o \left(\frac{R_{ff}}{r}\right)^2 = (2)\,(1.27 \times 10^4\ \text{W/m}^2)\left(\frac{0.8\ \text{m}}{3.0\ \text{m}}\right)^2$$

$$= (1.81 \times 10^3\ \text{W/m}^2)\left(\frac{1\ \text{m}}{100\ \text{cm}}\right)^2\left(\frac{1000\ \text{mW}}{\text{W}}\right) = 181\ \text{mW/cm}^2$$

Question 8.51

During this event, the lab worker was exposed to a free-space power density of 25 mW/cm^2 for two minutes. The ANSI Standard is 10 mW/cm^2 and the exposure standard allows averaging over a 6 minute time interval. Using the ANSI averaging approach, the worker's microwave exposure is:

$$E = (25\ \text{mW/cm}^2)\ (2\ \text{min}/6\ \text{min}) = 8.33\ \text{mW/cm}^2$$

Since this value is less than the recommended value, no exposure in excess of the microwave standard was received.

Question 8.52

Microwaves do not have sufficient energy (E) to produce ionization in tissue:

$$E = h\,\nu$$

where
h = Planck's constant = 6.626×10^{-34} J-s
ν = frequency of the radiation = 2.4×10^{9} Hz
E = $(6.626 \times 10^{-34}\ \text{J-s})\ (2.4 \times 10^{9}/\text{s})\ (1\ \text{MeV}/1.6 \times 10^{-13}\ \text{J}) = 9.94 \times 10^{-12}$ MeV

This energy is significantly less than the ionization potential for materials expected to be present in the facility. Therefore, microwaves will not damage DNA through ionization. However, the 2400 MHz microwave source may have a power supply and associated equipment that have high-voltage components. High-voltage equipment can produce X-rays, which do have the potential to cause ionization in tissue and the potential for subsequent DNA damage.

Question 8.53

The primary effect from the microwave device is the production of Joule heating within tissue. Tissue contains polar molecules and other ionic species that are subjected to electric and magnetic forces associated with the microwave electromagnetic field:

$$F_e = q\,E$$
$$F_m = \mu_o\,q\,v\,H\sin\theta$$

where
F_e = electric force
F_m = magnetic force
q = charge of the molecule or ion
E = magnitude of the microwave electric field

v = magnitude of the ion's velocity
H = magnitude of the microwave magnetic field
μ_0 = permeability of free space
θ = angle between the velocity vector and direction of the magnetic field

The electromagnetic force causes the charges, ions or polar molecules to move or vibrate. This motion produces a current, which leads to Joule heating within the medium according to the relationship:

$$P = I^2 R$$

where
P = power or heating produced in the medium
I = current caused by the moving charge or vibrating molecule
R = resistance of the medium

Question 8.54

The ALARA principle is applied to ionizing radiation to limit the risks of stochastic effects (e.g., cancer and hereditary disease), which are assumed to have no threshold. For nonionizing radiation, the deterministic effects of concern have a threshold power density or total exposure that must be exceeded for detrimental effects to be observed. Threshold Limit Values (TLVs) are used to prevent detrimental deterministic effects associated with nonionizing radiation sources.

The TLV value is the maximum concentration of a chemical or energy/power density of nonionizing radiation to which workers can be exposed for a fixed time (normally eight hours/day) without developing a physiological impairment. The TLV values are normally based on 95% confidence level values.

ALARA is based on maintaining exposures to values as low as reasonably achievable. Observable stochastic effects are limited by properly setting regulatory limits. The ALARA concept maintains exposures below these limits.

The ICRP currently adopts a linear nonthreshold approach for stochastic effects. If stochastic effects associated with ionizing radiation had a practical threshold dose and this threshold were recognized by international consensus bodies and regulatory agencies, then TLVs could be established for ionizing radiation. Once TLVs were accepted for ionizing radiation, the ALARA principle would no longer be an appropriate regulatory approach.

Scenario 8.9

Question 8.55

The anatomical structure(s) of the eye that are the most sensitive to damage for the selected lasers are:

a. Far infrared (IR-C), carbon dioxide laser (10.6 μm)
Cornea:
The absorption data suggests minimal 10.6 μm light is transmitted by the ocular media. Most of the CO_2 laser radiation is absorbed by the cornea because this wavelength is strongly absorbed by most organic molecules, including those in corneal tissue.

b. Visible, gold vapor laser (0.628 μm)
Retina:
The absorption data suggests that visible light is transmitted by the ocular media and is strongly absorbed by the retina and retinal pigment epithelium.

c. UV-A, nitrogen laser (0.337 μm)
Lens:
The transmission data suggests that transmission of light to the retina is minimal. At 337 nm, light is preferentially absorbed in the lens. This UV-A radiation is absorbed in the lens producing photochemical effects leading to possible cataracts and loss of elasticity.

d. UV-C, krypton fluoride excimer laser (0.248 μm)
Cornea:
The transmission data suggests that transmission of light to the retina is minimal. At 248 nm, the light is preferentially absorbed in the cornea. This UV-C radiation produces conjunctivitis for surface absorption and possibly corneal photochemical effects (e.g., loss of transparency) for deeper corneal absorption.

Question 8.56

Maximum permissible exposure limits vary considerably within a narrow range of the visible spectrum. For example, MPEs of 2.5×10^{-3} W/cm^2 and 1.7×10^{-5} W/cm^2 apply respectively to quarter-second exposures for wavelengths of 0.647 μm (red) and 0.530 μm (green). The primary source of these MPE differences is attributed to the greatly reduced photochemical hazard in the red portion of the visible spectrum. The reduced potential for photochemical reactions increases the MPE as the wavelength increases from the blue light region where photochemical effects occur. Since green light has a shorter wavelength than red, the MPE will be lower, i.e. it is closer to the blue wavelengths. ANSI Standard Z136.1 (applicable when this question appeared on the 2000 Examination) reflected this effect in the C_B correction factor:

$$\text{for} \quad 0.400\ \mu\text{m} \le \lambda \le 0.550\ \mu\text{m} \qquad C_B = 1.0$$

$$\text{for} \quad 0.550\ \mu\text{m} \le \lambda \le 0.700\ \mu\text{m} \qquad C_B = 10^{15\ (\lambda - 0.550)}$$

Question 8.57

The skin effect(s) associated with the following UV bands are:

1. UV-A (0.315 – 0.400 μm):
The main layers of the skin are the epidermis, dermis, and subcutaneous tissue. The skin reflects most visible and IR-A radiation. When considering the UV spectrum, the epidermis is less sensitive to UV-A. If the individual is photosensitive, abnormal skin reactions to UV-A can occur including phototoxicity and photoallergy. Skin tanning and erythema are potential minor effects.

2. UV-B (0.280 – 0.315 μm):
UV-B effects include photosensitivity (see the description for UV-A), dermal elastosis or aging, skin cancer (particularly melanoma), erythema (0.250–0.300 μm), and photochemical reactions (tanning). Chronic UV exposure causes premature aging of the skin and increases the risk of cancer.

3. UV-C (0.100 – 0.280 μm):
UV-C effects include erythema (0.250–0.300 μm), photochemical reactions (tanning), premature aging, and skin cancer.

Question 8.58

In this question, the beam divergence is defined to be negligible and the beam has a diameter of 3.5 mm. This beam diameter is smaller than the limiting aperture defined in ANSI Standard Z136.1. This standard defines the MPE value based on the limiting aperture. Since the beam area is smaller than the limiting aperture, the limiting aperture (0.7 cm) is used in subsequent calculations. The use of the actual beam area would provide a conservative result. For zero divergence, the area (A) of the laser beam having a diameter (d) of 7.0 mm is:

$$A = \pi\left(\frac{d}{2}\right)^2 = \pi\left(\frac{0.7\text{ cm}}{2}\right)^2 = 0.385\text{ cm}^2$$

The maximum allowable laser power (P) is written in terms of the MPE (0.01 J/cm^2) and beam area:

$$E = \frac{P}{A}$$

$$P = E\,A$$

where
E = irradiance (W/cm^2)

The MPE given in the problem is based on the radiant exposure (H) in J/cm^2 which can be written in terms of the irradiance and the maximum alignment or exposure time (t = 1.2 s):

$$E = \frac{H}{t} = \frac{(0.01\ \text{J/cm}^2)\,(1\ \text{W-s/J})}{(1.2\ \text{s})} = 8.33 \times 10^{-3}\ \text{W/cm}^2$$

With these values, the desired power output is determined:

$$\begin{aligned} P &= E\,A = (8.33 \times 10^{-3}\ \text{W/cm}^2)\,(0.385\ \text{cm}^2) \\ &= (3.21 \times 10^{-3}\ \text{W})\,(1000\ \text{mW/W}) = 3.21\ \text{mW} \end{aligned}$$

Question 8.59

You are to determine if the laser array is appropriate for use in a pre-game show. No optically aided viewing is to be assumed. The most direct solution is to calculate the nominal ocular hazard distance (NOHD) and compare this distance to the minimum distance from the laser to the audience (25 m). The NOHD is:

$$\text{NOHD} = \frac{1}{\phi}\sqrt{\frac{1.27\,\Phi}{\text{MPE}} - d_a^2}$$

where

ϕ = divergence angle = 0.2 mrad

Φ = average power = $P_w\,P_{rf}\,P_p$

- P_w = pulse width = 20 μs
- P_{rf} = pulse repetition frequency = 25 kHz
- P_p = peak power = 40 W

= $(20 \times 10^{-6}\ \text{s})\ (25 \times 10^3/\text{s})\ (40\ \text{W}) = 20\ \text{W}$

MPE = maximum permissible exposure = $1.8\,t^{0.75} \times 10^{-3}$ J/cm^2

- t = maximum exposure time in scanning the audience
- d_{pupil} = limiting aperture = fully dilated pupil diameter = 0.7 cm
- S = scanning rate = 20,000 cm/s

$$t = \frac{d_{pupil}}{S} = \frac{0.7\ \text{cm}}{2 \times 10^4\ \text{cm/s}} = 3.5 \times 10^{-5}\ \text{s}$$

d_a = beam aperture diameter = 0.2 cm

Using these values leads to the MPE:

$$\begin{aligned} \text{MPE} &= 1.8\,t^{0.75} \times 10^{-3}\ \text{J/cm}^2 = 1.8\,(3.5 \times 10^{-5})^{0.75} \times 10^{-3}\ \text{J/cm}^2 \\ &= 8.19 \times 10^{-7}\ \text{J/cm}^2 \end{aligned}$$

It is necessary to either convert the average power to average energy ($E = P\,t$) or convert the MPE to irradiance (W/cm^2). The inclusion of the time factor in the NOHD equation accomplishes this action. Given these values the NOHD is determined:

$$\begin{aligned}
\text{NOHD} &= \frac{1}{\phi}\sqrt{\frac{1.27\,\Phi}{\text{MPE}} - d_a^2} \\
&= \frac{1}{0.2 \times 10^{-3}}\sqrt{\frac{1.27(20\text{ W})\,(3.5 \times 10^{-5}\text{ s})\,(\text{J/W-s})}{(8.19 \times 10^{-7}\text{ J/cm}^2)} - (0.2\text{ cm})^2} \\
&= (1.65 \times 10^5\text{ cm})\,(1\text{ m}/100\text{ cm}) = 1650\text{ m}
\end{aligned}$$

The NOHD value exceeds the audience distance of 25 m from the laser. Therefore, this laser is not appropriate for the pre-game show.

Part IV
Appendices

Appendix I
Serial Decay Relationships

Many radioactive nuclides decay in a single transition to a stable nuclide. However, there are cases when the transition occurs to a system that is also unstable. These serial decays are encountered in neutron-induced fission products of U-235 and Pu-239 and in the natural decay series.

The decay relationships are usually derived on the assumption that at time $t = 0$ only parent atoms exist and all daughter activity is zero. The parent nuclide is labeled by the subscript a, and its daughters are labeled with b, c, d, and so on. N is the number of atoms at time t, and X is the decay constant.

The serial decay relationships can be derived for any number of daughters. We illustrate the decay relationships for the decay of the parent and first, second, and third daughters:

$$N_a = N_{a_0} \exp(-\lambda_a t) \tag{I.1}$$

$$N_b = N_{a_0} [a_1 \exp(-\lambda_a t) + a_2 \exp(-\lambda_b t)] \tag{I.2}$$

$$N_c = N_{a_0} [a_3 \exp(-\lambda_a t) + a_4 \exp(-\lambda_b t) + a_5 \exp(-\lambda_c t)] \tag{I.3}$$

$$N_d = N_{a_0} [a_6 \exp(-\lambda_a t) + a_7 \exp(-\lambda_b t) + a_8 \exp(-\lambda_c t) + a_9 \exp(\lambda_d t)] \tag{I.4}$$

where

N_{a_0} = number of parent atoms present at $t = 0$

$$a_1 = \lambda_a / (\lambda_b - \lambda_a) \tag{I.5}$$

$$a_2 = -\lambda_a / (\lambda_b - \lambda_a) \tag{I.6}$$

$$a_3 = \lambda_a \lambda_b / [(\lambda_b - \lambda_a)(\lambda_c - \lambda_d)] \tag{I.7}$$

$$a_4 = \lambda_a \lambda_b / [(\lambda_a - \lambda_b)(\lambda_c - \lambda_b)] \tag{I.8}$$

$$a_5 = \lambda_a \lambda_b / [(\lambda_a - \lambda_c)(\lambda_b - \lambda_c)] \tag{I.9}$$

$$a_6 = \lambda_a \lambda_b \lambda_c / [(\lambda_b - \lambda_a)(\lambda_c - \lambda_a)(\lambda_d - \lambda_a)] \tag{I.10}$$

$$a_7 = \lambda_a \lambda_b \lambda_c / [(\lambda_a - \lambda_b)(\lambda_c - \lambda_b)(\lambda_d - \lambda_b)] \tag{I.11}$$

$$a_8 = \lambda_a \lambda_b \lambda_c / [(\lambda_a - \lambda_c)(\lambda_b - \lambda_c)(\lambda_d - \lambda_c)] \tag{I.12}$$

$$a_9 = \lambda_a \lambda_b \lambda_c / [(\lambda_a - \lambda_d)(\lambda_b - \lambda_d)(\lambda_c - \lambda_d)] \tag{I.13}$$

The activity (A) is readily determined from these relationships because $A = \lambda N$.

In practical applications, the case involving a parent and two daughter nuclides frequently occurs. In general, the activity of the daughters is not always zero. The serial decay of parent nuclide a to a daughter b that subsequently decays to nuclide c is given by

$$a \xrightarrow{\lambda_a} b \xrightarrow{\lambda_b} c \tag{I.14}$$

$$N_b = \frac{\lambda_a N_{a_0}}{\lambda_b - \lambda_a} [\exp(-\lambda_a t) - \exp(-\lambda_b t)] + N_{b_0} \exp(-\lambda_b t) \tag{I.15}$$

where

N_b = number of b atoms at time t
N_{a_0} = number of a atoms present at time $t = 0$
N_{b_0} = number of b atoms present at time $t = 0$
t = time of interest

The activity follows from (I.15):

$$A_b = \frac{\lambda_b A_{a_0}}{\lambda_b - \lambda_a} [\exp(-\lambda_a t) - \exp(-\lambda_b t)] + A_{b_0} \exp(-\lambda_b t) \tag{I.16}$$

where A_{b_0} is usually assumed to be zero for simplicity.

For example, (I.16) is used in medical applications involving the milking or elution of a Mo-99 generator to obtain the desired quantities of Tc-99m. For cases such as a Mo-99 generator, both chemical and fractional yields must be considered and (I.16) must be modified to reflect the physical limitations of the generator and leaching technology

$$A_b = C Y \frac{\lambda_b A_{a_0}}{\lambda_b - \lambda_a} [\exp(-\lambda_a t) - \exp(-\lambda_b t)] + A_{b_0} \exp(-\lambda_b t) \tag{I.17}$$

where

C = chemical yield
Y = fractional yield

References

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Appendix II
Basic Source Geometries and Attenuation Relationships

The evaluation of radiation levels from a source is a fundamental problem in health physics. Common source configurations include the point, line, disk, and slab geometries. Although numerous computer codes are available for the assessment of these and more complicated geometries, it is often advantageous to have the capability to make rapid assessments using hand calculations before performing more detailed calculations. Knowledge of these basic geometries, as well as the capability to utilize them in dose estimates, will be valuable to the practicing health physicist.

Source Configurations – No Shielding

Point Source Geometry

The point source approximation is applicable whenever the dose is calculated at a distance that is at least three times the largest source dimension. The student should test this statement by performing sample calculations for various geometries. A comparison of point and disk geometries will be provided later in this appendix.

The exposure, absorbed dose, or dose rate $\dot{X}(r)$ at a location a distance r from a point source is given by

$$\dot{X}_0(r) = A\,G/r^2 \tag{II.1}$$

where

$\dot{X}_0(r)$ = unattenuated exposure rate (R/hr), absorbed dose rate (rad/hr), or dose rate (rem/hr) at point r. Although $\dot{X}(r)$ is normally expressed in terms of the Roentgen, the conversion to either rem or rad units is straightforward and these units are used rather loosely in external dose assessments.

A = source activity (Ci)

G = gamma constant (R-m^2/hr-Ci)

r = distance from the point source (m)

If the value of G is not provided, $\dot{X}_0$ can be estimated from the properties of the decaying nuclide using the approximate relationship

$$\dot{X}_0(r) = 0.5\, C\, E / r^2 \tag{II.2}$$

where

C = activity of the nuclide (Ci)
E = total weighted photon energy of the nuclide = $\sum_i E_i\, Y_i$
E_i = energy of the ith gamma ray from the nuclide
Y_i = yield of the ith gamma ray from the nuclide
G = 0.5 E

The generalization of this relationship to multiple source nuclides is achieved by summing over the various source nuclides.

The approximation of (II.2) is limited by how accurately the gamma constant is represented by the approximation $G = 0.5\, E$. In general, this approximation should be within 20% of the actual value of the gamma constant. For the case of Na-24, Co-60, and Cs-137, the approximation is within +12.0%, –5.3%, and –15.2%, respectively, of the actual gamma constant.

Equation (II.1) may be combined with the shielding relationships to account for attenuation and buildup through a shield. This will be explored in more detail later in this appendix.

For those situations in which the gamma constant is not available, or for which an accuracy greater than that given by the gamma constant approximation is desired, the total gamma dose equivalent at a distance r from the point source can be written in terms of the gamma energy of the source, source strength, mass energy absorption coefficient, and assumptions regarding the emission of the source. If an isotropic emission is assumed, the gamma dose rate for a source with a single gamma ray can be written as

$$\dot{H}_0 = (S / 4\,\pi\, r^2)\,(u_{\text{en}} / p)\, E_{\text{gamma}} \tag{II.3}$$

where

$\dot{H}_0$ = unattenuated gamma dose rate (rem/hr)
S = source strength (gammas/s)
r = distance from the source (cm)
(*Note:* This relationship can be generalized to line, disk, or slab sources by substituting the appropriate geometry factor.)
u_{en}/p = Mass energy absorption coefficient (cm^2/g)
p = density (g/cm^3)
E_{gamma} = weighted energy of the emitted gamma ray = $E\, Y$

For sources characterized by the emission of multiple gamma rays, the dose rate from a point source may be written in the form

$$\dot{H}_0 = \sum_i (S_i / 4\,\pi\, r^2) \sum_j (U^j_{\text{en}} / p)\, E_{ij}\, Y_{ij} \tag{II.4}$$

where

$\dot{H}_0$ = dose equivalent rate (rem/hr)
S_i = activity of isotope i (μCi or dis/s)
E_{ij} = energy of the jth gamma ray for the isotope i
Y_{ij} = yield of the jth gamma ray for the isotope i
u_{en}^{j} / p = mass energy absorption coefficient (cm^2/g) at energy E_{ij}

The total unattenuated neutron dose equivalent rate for a point source can be similarly defined:

$$\dot{H}_{\text{on}} = (S / 4\,\pi\, r^2)\, k \tag{II.5}$$

where

$\dot{H}_{\text{on}}$ = neutron dose rate (rem/hr)
S = source strength (neutrons/s)
r = distance from the source
(*Note:* This relationship can also be generalized to other source geometries.)
k = factor converting neutrons $\text{s}^{-1}\ \text{cm}^{-2}$ to rem/hr

Line Source Geometry

The unattenuated radiation field due to a line source of length L, illustrated in Figure II.1, is given by

$$\dot{X}_0(Q) = G\, C_{\text{L}}\, \theta / w \tag{II.6}$$

where

$\dot{X}_0(Q)$ = unattenuated exposure rate (R/hr) or dose rate (rem/hr) at point Q due to the line source $a\ b$.
w = perpendicular distance of the point Q to the line source
Q = point at which dose is to be determined
A = total source activity (Ci)
L = length of the line source
θ = angle subtended by the line source at the point of interest

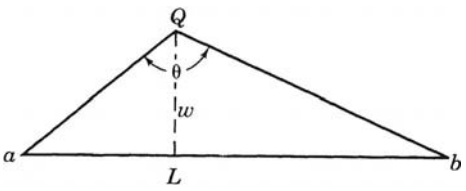


Figure II.1 Geometry for computing the gamma-ray dose field at point Q from a line source ($a\ b$).

C_L = source activity per unit length of the line source
= A/L (Ci/m, Ci/ft, Ci/cm, etc.)
G = gamma constant (R-m^2/hr-Ci)

This approximation is particularly useful for calculations involving piping, fluid lines, or resin columns containing radioactive material. For these calculations, the pipe wall thickness is often ignored without a significant loss in accuracy.

Disk Source Geometry

The unattenuated radiation field due to a thin disk source of radioactivity, illustrated in Figure II.2, is given by

$$\dot{X}_0(Q) = \pi\, G\, C_a \ln \frac{r^2 + h^2}{h^2} \tag{II.7}$$

where
$\dot{X}_0(Q)$ = unattenuated exposure rate (R/hr) or dose rate (rem/hr) at point Q lying an the axis of the disk
Q = lies a perpendicular distance h from the disk source
A = Total source activity (Ci)
a = area of the disk source
C_a = Source activity per unit area of the disk source
= A/a (Ci/m^2, Ci/ft^2, Ci/cm^2, etc.)
G = gamma constant (R-m^2/hr-Ci)
r = radius of the disk source

Disk relationships are useful to evaluate the radiation field associated with either (a) a spill of radioactive liquid, (b) the field above a resin bed, or (c) surface sources such as a radioactive pool.

A comparison of exposure rates from point and thin disk source geometries illustrates when a source may be reasonably approximated by a point source. This is illustrated by considering Scenario 60■, question 5.26, which evaluated the exposure rate a distance h from a 10-m-radius thin disk source containing 3.0 Ci of Co-60.

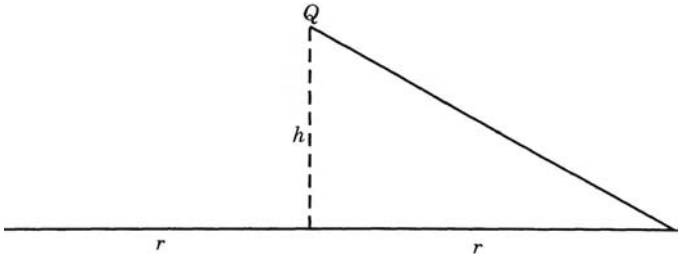


Figure II.2 Geometry for computing the gamma-ray dose field at point Q from a thin disk source of radius r.

Table II.1 summarizes the exposure rates from 3.0-Ci point and disk sources at a point h meters from the source. This table suggests that point and disk dose rates are within 1% of each other when the distance from the source is at least three times the largest source dimension. For the disk source, this occurs at a distance three times the disk diameter – that is, $h/d = 3$.

Table II.1 Comparison of Disk and Point Source Approximations for a 10-Meter-Radius Thin Disk Source.

h (m)	Exposure Rate (mR/hr)		
	Disk	Point	h/d
2	127	975	0.1
10	27	39	0.5
20	8.70	9.75	1.0
30	4.11	4.33	1.5
40	2.36	2.44	2.0
50	1.53	1.56	2.5
60	1.07	1.08	3.0
70	0.788	0.796	3.5
80	0.605	0.609	4.0
90	0.479	0.481	4.5
100	0.388	0.390	5.0

Stab Source Geometry

Consider a slab of uniformly distributed radioactive material characterized by a linear absorption coefficient u. The slab source is illustrated in Figure II.3. If the slab is defined to be of cylindrical shape with a radius R and thickness t, the dose rate at point Z a distance h above the centerline of the slab is

$$\dot{X}_0(h) = \pi G (C_v / u)(1 - e^{-u t}) \ln \frac{R^2 + h^2}{h^2} \tag{II.8}$$

where

$\dot{X}_0(h)$ = exposure rate (R/hr) or dose rate (rem/hr) at point Z an the axis of the slab at a distance h from the surface
G = gamma constant (R-m^2/hr-Ci)
C_v = activity concentration (Ci/cm^3) = A/v
A = total activity of the slab
v = volume of the source (cm^3) = $\pi R^2 t$

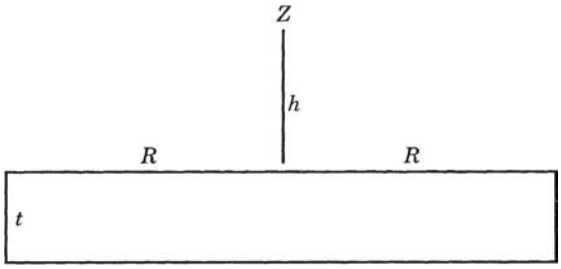


Figure II.3 Geometry for computing the gamma-ray dose field at point *Z* from a slab source of radius *R*.

This relationship is applicable to assessments of dose rates from distributed bodies such as spent-fuel pools, contaminated slabs, or tanks of contaminated fluids which contain uniformly distributed activity. The choice of using a disk or slab geometry is governed by a number of factors including the accuracy desired, knowledge of the radionuclide concentration, distribution, content, and distance of the receptor from the source.

Attenuation by a Shield Without Buildup

The attenuation of gamma rays by a shielding medium can be approximated by considering shield attenuation. For thin shields with inherently small buildup factors of nearly unity, the attenuation may be represented by

$$I(x) = I_0 \exp(-u\,x) \tag{II.9}$$

where

$I(x)$ = attenuated quantity (dose rate, fluence rate, flux, or activity) due to a thickness x of shielding material
I_0 = unattenuated initial value of the quantity
u = attenuation coefficient (cm^{-1}) = 1/relaxation length
x = shield thickness (cm)

Attenuation by a Shield With Buildup

For thicker shields, where the buildup factor is greater than unity, a better approximation to the attenuation of gamma rays by a shielding medium is given by

$$I(x) = I_0\, B\,(E, z, u\,x)\, e^{-u\,x} \tag{II.10}$$

where the buildup factor (B) is a function of the gamma-ray energy (E), the atomic number (Z) of the shield material, the attenuation coefficient (u), and the thickness

(x) of the shield material. This relationship also applies to the various geometries considered in this chapter.

The ratio $I(x)/I_0$ defines the transmission factor (T):

$$T = I(x)/I_0 = B \exp(-u\ x) \tag{II.11}$$

The reader should note that u may be expressed in a variety of ways such as the mass attenuation coefficient (cm^2/g) times the material density (g/cm^3) or as an attenuation coefficient with units of reciprocal length.

A common problem involves the determination of the shield thickness required to reduce the radiation level to a desired value. The desired thickness may be obtained by solving (II.10) for the required thickness:

$$x = (-1/u) \ln[I(x)/I_0\ B] \tag{II.12}$$

If the source emits multiple gamma rays during its decay, the photon fluence reduction due to a shield $I(x)$ is given by

$$I(x) = I_0 \sum_i B_i\ f_i \exp(-u_i\ x) \tag{II.13}$$

where

B_i = buildup factor for energy E_i
u_i = attenuation coefficient of the shield for energy E_i
x = thickness of the shield
E_i = energy of the i th gamma ray
f_i = fraction of I_0 contributed by E_i or the partial yield for the i th gamma ray
= $Y_i / \sum_i Y_i$
Y_i = yield of the i th gamma ray

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Appendix III
Neutron-Induced Gamma Radiation Sources

Neutron sources or beams, encountered in accelerator or power reactor environments, generate additional radiation sources as they penetrate matter. The neutron interactions produce activated material that remains a hazard long after the neutron source is terminated. The neutron-induced interactions include activation, capture, and fission reactions. Each of these mechanisms present a significant radiation hazard.

Activation Sources

The activation of a material as a function of time depends upon the material being activated and the beam activating the material. The activity buildup consists of both production and decay terms and may be written as

$$A = N\,\sigma\,\phi\,[1 - \exp(-\lambda\, t_{\text{irrad}})]\exp(-\lambda\, t_{\text{decay}}) \qquad \text{(III.1)}$$

where

A = activity of the sample as a function of time
N = number of atoms in the sample that are activated $N = m\,\overline{A}/\text{GAW}$
m = mass of the sample
(If multiple isotopes are activated, the mass of each constituent must be considered.)
$\overline{A}$ = 6.023×10^{23} atoms/GAW (Avogadro's number)
GAW = gram atomic weight or mass of a mole of material (g)
σ = cross-section for the reaction induced by the flux ϕ (barns/atom)
ϕ = fluence rate or flux (neutrons/cm^2-s)
λ = decay constant of the activated material
t_{irrad} = time the sample was irradiated or exposed to the flux
t_{decay} = decay time or time the sample was removed from the flux

For material that is activated for a long time relative to its half-life, the activity reaches a constant value or saturates. Saturation occurs as the irradiation time becomes much larger than the decay half-life, and the decay time is short relative

to the decay half-life. Under these circumstances, the activity approaches A_{sat}, which is the saturation activity

$$A_{sat} = N \sigma \phi \tag{III.2}$$

Determining the saturation activity is an important exercise because it represents a bounding case for dose-rate assessments. Design work often utilizes the saturation activity to ensure that the design will bound any operating condition.

Many common activation gamma sources involve the absorption of a neutron with the emission of a gamma ray. Usually this process involves thermal neutrons. Other reactions involve high-energy or fast neutrons and produce high-LET protons via (n, p) reactions. Examples of activation sources are contained in Table III.1. These sources are frequently produced in reactor or accelerator environments.

During activation, the absorbed neutron produces a radioactive nuclide that decays with a characteristic half-life. This may be contrasted with capture reactions which involve nearly an instantaneous gamma-ray emission following the capture.

Table III.1 Activation Gamma Sources.

Reaction	Activation Cross-Section (barns)	Half-life	Energy (MeV)	Yield Gammas/ Decay
$^{23}Na(n, \gamma)^{24}Na$	0.534	14.96 hr	1.369 2.754	1.00 1.00
$^{54}Fe(n, p)^{54}Mn$	1.0	314 days	0.835	1.00
$^{55}Mn(n, \gamma)^{56}Mn$	13.3	2.576 hr	0.847 1.811 2.11	0.99 0.29 0.15
$^{59}Co(n, \gamma)^{60}Co$	37.2	5.263 years	1.173 1.332	1.00 1.00
$^{58}Fe(n, \gamma)^{59}Fe$	1.2	45.6 days	1.095 1.292	0.56 0.44
$^{58}Ni(n, p)^{58}Co$	1.0	71.3 days	0.51 0.81 0.865 1.67	0.30 0.99 0.014 0.006
$^{94}Zr(n, \gamma)^{95}Zr$	0.075	65.5 days	0.724 0.756 0.765	0.49 0.49 1.00

Source: O. J. Wallace, WAPD-TM-1453.

Capture Gamma Sources

Capture gamma sources involve the absorption of a neutron with the nearly instantaneous emission of a gamma ray. Capture reactions are often defined in terms of bulk materials and play an important role in design calculations. Calculations of nuclear radiation heat generation and primary or biological shield design must properly account for capture sources. Examples of capture gamma sources are provided in Table III.2.

Table III.2 Capture Gamma Sources.

Reaction	Absorption Cross Section (barns)	Density (g/cm^3)	Energy (MeV)	Yield Gammas/ Capture
Hydrogen capture	0.33	8.988×10^{-4}	2.2	1.0
Iron capture	2.53	7.874	1.0 7.0	1.85 0.86
Zirconium capture	0.180	6.53	1.0 5.0	0.7 1.42
Uranium capture	7.68	18.95	0.7 2.5	3.63 1.2
Water capture	0.66	0.998	2.2	1.0
Hafnium capture	105.0	13.29	1.0 3.0	0.75 2.366
Stainless steel capture	0.0332	7.9	1.0 7.0	1.69 0.93
Inconel capture	0.0462	8.51	0.5 5.5	2.0 1.51

Source: O. J. Wallace, WAPD-TM-1453.

Fission Gamma Sources

Fission gamma sources are produced by the neutron spectrum of a reactor. Table III.3 summarizes the characteristics of two common fission gamma sources. The N-16 source is derived from the O-16 (n, p) reaction with the subsequent decay of the short-fived N-16 system. The prompt and delayed fission gamma source represents a collective summary of the uranium fission gamma spectrum. These sources tend to have more varied energies and yields when compared with either the activation or capture sources for specific isotopes.

Table III.3 Fission Gamma Sources.

Source	Energy (MeV)	Yield Gammas/Fission
^{16}N gammas	2.74	0.01
	6.13	0.69
	7.12	0.05
Prompt and delayed fission gammas	0.80	7.22
	2.0	2.26
	4.0	1.02

Source: O. J. Wallace, WAPD-TM-1453.

References

Bevelacqua, J. J., *Basic Health Physics, Problems and Solutions*, John Wiley & Sons, Inc., New York (1999).

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Appendix IV
Selected Topics in Internal Dosimetry

As a health physicist, you may be required to evaluate the dose to internal organs and to the whole body as a result of the intake of radioactive material. This intake may occur either intentionally or as the result of an unplanned activity, in a variety of ways, including inhalation, injection and ingestion. Internal dosimetry involves the use of numerous calculational models to evaluate the consequences of the intake. This appendix will present a variety of approaches that introduce the essential elements of internal dosimetry and its radiological basis.

The calculation of internal dose by any model has a number of common elements. The dose depends upon the activity deposited in the organ (q), the energy of the emitted radiation (E) absorbed by the organ, and the mass of the organ (m) at risk. Because dose (D) is defined in terms of energy deposition per unit mass, internal dose rate $(\dot{D})$ formulas contain factors of the form

$$\dot{D} = \frac{k\, q\, E}{m} \tag{IV.1}$$

where k is a conversion factor which expresses the dose rate in the desired units.

Internal dose is dominated by alpha and beta radiation because these radiations deposit their energy in distances shorter than typical organ sizes. Gamma rays contribute to the internal dose, but to a lesser extent than either beta or alpha radiation.

Internal Dose Assessment Models

An anatomical model, used to approximate the human body, must be formulated in order to calculate internal doses. The model will consist of mathematical equations that define the sizes and shapes of organs and tissue that are assembled together to form a representation of the human body. The degree of sophistication of the internal dosimetry models varies from simplistic singlecompartment models to the more complex lung model currently under development. The current lung model has a strong resemblance to the actual body tissue and is considerably more complex than the simplistic geometric forms used in the initial infernal dosimetry models.

Internal Dosimetry Definitions

Before considering specific models, several terms will be defined to ensure that the subsequent discussion is clear. These terms include intake, uptake, deposition, and content.

Intake is the quantity of a radionuclide taken into the body by inhalation, ingestion, or injection. An uptake is that quantity of a radionuclide that is absorbed into systemic circulation by injection directly into the blood or by absorption via the respiratory or gastrointestinal (GI) tract. Deposition is that quantity of a radionuclide that is deposited into an organ. Finally, content refers to the quantity of a radionuclide that resides in the compartment of interest.

In the subsequent discussion, the focus will be upon the more common internal dosimetry models. These models include the ICRP-2/10, MIRD, ICRP-26/30, ICRP-60/66, and ICRP-103/66/100 internal dosimetry formulations. As a matter of notation, ICRP-103/66/100 refers to the baseline document (ICRP-103) and major supporting documents (ICRP-66 for the human respiratory tract model and ICRP-100 for the human alimentary tract model). ICRP-26 is supported by ICRP-30 and ICRP-60 is supported by ICRP-66. ICRP-30 also supports ICRP-60. Each of the aforementioned internal dosimetry models is addressed in subsequent discussion.

Regulatory Adoption of Internal Dosimetry Models

The ICRP internal dosimetry models were adopted by most nations, after their publication. The transition from ICRP-60 to ICRP-103 is the most recent example.

ICRP-26/30 is currently the basis for U.S. Nuclear Regulatory Commission (10CFR20) and U.S. Department of Energy (10CFR835) regulations. The Department of Energy is in the process of transitioning from ICRP-26/30 to ICRP-60/66, and that transition will occur during the 2007–2010 timeframe.

ICRP-2/10 Methodology

Internal Dose Assessment – Single-Compartment Model

An early internal dosimetry model was described in the 1959 ICRP Committee 2 Report, which utilized the concept of a body and organs characterized by effective radii. This approximation made it possible to calculate the dose to an organ from the activity residing within that organ. The concept of a single-compartment model was developed to address the absorption and elimination of radioactive material from that organ.

Although simplistic by today's standards, the single-compartment model has considerable utility and will be addressed in terms of its key properties and

dosimetry relationships. Moreover, it will serve as a basis for formulating more complex models describing both chronic and acute internal exposures.

The following methodology is based upon the single-compartment model in which the radionuclide is uniformly distributed in the organ of interest. Both chronic and acute exposures will be considered, and dose rate and integrated dose relationships will be provided.

Single Uptake – Single-Compartment Model

An initial uptake of activity (q_{i_0}) of radionuclide i results in a deposition $(q_{i_0}\ f_2)$ in the organ of interest. The activity $q_i(t)\, f_2$ post uptake is given by

$$q_i(t)\ f_2 = q_{i_0}\ f_2 \exp(-\lambda_e\ t) \tag{IV.2}$$

where
λ_e = effective decay constant
f_2 = fraction of the material entering the organ of interest

The initial dose equivalent rate $(\dot{D}_0)$ to the organ of interest due to an initial deposition of $q_0 f_2$ is

$$\dot{D}_0 = q_0 f_2\ E/m \tag{IV.3}$$

where
E = effective energy deposition
m = mass of the organ

The effective energy, for alpha and beta emitters, may be written as

$$\begin{aligned} E &= E_{\text{alpha}} \qquad \text{(alpha emitters)} \\ E &= E_{\max}\ f/3 \qquad \text{(beta emitters)} \end{aligned} \tag{IV.4}$$

where
E_{alpha} = energy of the emitted alpha particle
$E_{\max}$ = maximum beta energy
f = probability that a beta is emitted

If q_0 is expressed in μCi, E in MeV, and m in grams, (IV.3) takes the form for the initial dose rate in rad/hr:

$$\begin{aligned} \dot{D}_0 = {} & [q_0\ (\mu\text{Ci})\ f_2\ E\ (\text{MeV})/m\ (\text{g})]\ (3.7 \times 10^4\ \text{dis}/\mu\text{Ci-s}) \\ & \times (1.6 \times 10^{-6}\ \text{erg/MeV})\ (1\ \text{rad}/100\ \text{erg/g})\ (3600\ \text{s/hr}) \end{aligned} \tag{IV.5}$$

$$\dot{D}_0 = 2.13\ q_0\ f_2\ E/m \tag{IV.6}$$

The dose equivalent rate $\dot{D}(t)$ at a time t post uptake to the organ of interest containing a uniform deposition of $q_0\, f_2$ is

$$\dot{D}(t) = \dot{D}_0 \exp(-\lambda_e\, t) \tag{IV.7}$$

The total dose equivalent accumulated in the organ of interest from the single intake at $t = 0$ to any time t post uptake is obtained by integrating (IV.7) from $t = 0$ to time t:

$$D = (\dot{D}_0 / \lambda_e)\,[1 - \exp(-\lambda_e\, t)] \tag{IV.8}$$

or

$$D = 2.13\,[A_0\, E / m\, \lambda_e]\,[1 - \exp(-\lambda_e\, t)] \tag{IV.9}$$

where $A_0 = f_2\, q_0$.

For the case of long times after deposition, $t \to \infty$, the total dose due to the intake is

$$D = (\dot{D}_0 / \lambda_e) \tag{IV.10}$$

or

$$D = 2.13\, A_0\, E / m\, \lambda_e \tag{IV.11}$$

The form of this equation assumes the deposition of a single radionuclide. The deposition of multiple radionuclides may be assessed by performing a calculation for each isotope.

The calculation of the activity of a nuclide distributed throughout the body or organ can be estimated from the definition of activity:

$$A = \lambda\, N = (0.693 / T_{1/2})\, N \tag{IV.12}$$

where

A = activity in the organ
λ = disintegration constant
$T_{1/2}$ = half-life
N = number of atoms of the isotope = $m\, G/M$
m = mass of the isotope in the organ
M = gram atomic weight (GAW) of the radionuclide in the organ
G = Avogadro's number = 6.02×10^{23} atoms/GAW

Constant Rate of Uptake – Single-Compartment Model

A constant rate of uptake (P) of a particular radionuclide by the organ of interest (μCi/hr) results in a burden of $q(t)$ which increases with time t:

$$q(t)\, f_2 = (P / \lambda_e)\,[1 - \exp(-\lambda_e\, t)] \tag{IV.13}$$

where P has the units of activity per unit time such as μCi/day. An approach similar to that for a single uptake leads to the following relationship for the dose rate delivered to the organ of interest from a continuous uptake P:

$$\dot{D}(t) = [(P/\lambda_e)\, E/m]\,[1 - \exp(-\lambda_e\, t)] \tag{IV.14}$$

It also leads to the relationship for the dose equivalent delivered to the organ of interest during the time interval from $t = 0$ to the time t:

$$D = \frac{(P/\lambda_e)\, E\, t}{m}\left[1 - \frac{1 - \exp(-\lambda_e\, t)}{\lambda_e\, t}\right] \tag{IV.15}$$

Variation of $q\,f_2$ After the Cessation of the Uptake

The deposition ($q\,f_2$) in an organ of interest at the end of a continuous uptake interval may be considered an initial burden at that instant. This activity burden will subsequently decrease in the same manner as the initial deposition from a single uptake. If t_1 defines the interval of continuous uptake and t_2 defines the time interval post uptake, then the deposition's variation is given by

$$q(t_2)\, f_2 = (P/\lambda_e)\,[1 - \exp(-\lambda_e\, t_1)]\exp(-\lambda_e\, t_2) \tag{IV.16}$$

The dose equivalent rate at the end of the uptake interval is

$$\dot{D}(t_1) = (P/\lambda_e)\,(E/m)\,[1 - \exp(-\lambda_e\, t_1)] \tag{IV.17}$$

and the dose equivalent rate during the decay period post intake is

$$\dot{D}(t_2) = \dot{D}(t_1)\exp(-\lambda_e\, t_2) \tag{IV.18}$$

The dose equivalent accumulated by an organ of interest during the uptake interval from 0 to t_1 is obtained by integrating (IV.14) from $t = 0$ to t_1:

$$D(0 \text{ to } t_1) = \frac{(P/\lambda_e)\, E\, t_1}{m}\left[1 - \frac{1 - \exp(-\lambda_e\, t_1)}{\lambda_e\, t_1}\right] \tag{IV.19}$$

Similarly, the dose equivalent accumulated by an organ of interest during the time post uptake from $t_2 = t_1$ to infinity is derived from (IV.18):

$$D(t_1 \text{ to } \infty) = \frac{(P/\lambda_e)\, E\, t_1}{m}\left[\frac{1 - \exp(-\lambda_e\, t_1)}{\lambda_e\, t_1}\right] \tag{IV.20}$$

Adding Eqs. (IV.19) and (IV.20) leads to the total dose equivalent accumulated in the organ from intake to complete decay:

$$D(0 \text{ to } \infty) = \frac{(P/\lambda_e)\, E\, t_1}{m} \tag{IV.21}$$

Mird Theory

The Committee an Medical Internal Radiation Dose (MIRD) of the Society of Nuclear Medicine developed a methodology to perform radiation absorbed dose calculations. These calculations are performed to assess the risks of the application of radiopharmaceuticals to medical studies involved in imaging, therapy, or metabolic applications.

The MIRD technique (schema) is a computational framework that facilitates these absorbed dose calculations to specified target organs from radioactive decays that occur within source organs. The source organs contain the radioactive isotope, and the target is the organ in which the dose is computed. The target and source organ may be the same tissue.

Simplified MIRD Equation

To introduce the methodology, a simplified MIRD dose equation will be derived. Initially, a single radiopharmaceutical, single radiation type, and single source organ will be considered.

To define the MIRD schema, it is necessary to define several terms. The mean energy emitted per transition (Δ), in Gy-kg/Bq-s or rad-g/μCi-hr, is given by the product of the mean particle energy (E) (in MeV or joules) and the number of particles emitted per nuclear transition (n):

$$\Delta = K\, E\, n \tag{IV.22}$$

where K is a conversion factor.

Particles are defined to be either photons, beta particles, or positrons within the simplified MIRD model.

The cumulated activity or the total number of nuclear transitions occurring within the source organ $(\tilde{A})$ from $t = 0$ to some time t is given by

$$\tilde{A} = \int_0^t A(t)\, \mathrm{d}t \tag{IV.23}$$

The activity as a function of time is

$$A(t) = A(0) \exp(-\lambda_{\text{eff}}\, t) \tag{IV.24}$$

Using (IV.24), the expression for $\tilde{A}$ takes a closed form if the integration interval is taken from $t = 0$ to infinity. For this case, $\tilde{A}$ takes the form

$$\tilde{A} = A(0)\, T_{\text{eff}} / \ln(2) = 1.44\, T_{\text{eff}}\, A(0) \tag{IV.25}$$

where

$$T_{\text{eff}} = (T_{\text{physical}} \times T_{\text{biological}})/(T_{\text{physical}} + T_{\text{biological}}) \tag{IV.26}$$

The reader should note that the initial activity in the organ [$A(0)$] is not the intake activity but is only a fraction (f_2) of the total activity of radionuclide i in the body (q):

$$A(0) = q\, f_2 \tag{IV.27}$$

where f_2 is the fraction of the intake activity reaching the organ of interest.

The total energy emitted by the source organ is the product of Δ and the cumulated activity. However, only a fraction (f) of this energy will be deposited in the target organ, which is the location of interest in the dose assessment.

With these quantities and a knowledge of the mass of the target organ (m_T), the mean absorbed dose ($\bar{D}$) can be defined:

$$\bar{D} = \tilde{A}\, \Delta f / m_T \tag{IV.28}$$

The MIRD methodology also defines the specific absorbed fraction (F):

$$F = f / m_T \tag{IV.29}$$

where

$$f = \frac{\text{energy absorbed by the target}}{\text{energy emitted by the source}} \tag{IV.30}$$

and m_T is the target mass. The specific absorbed fraction represents the mean target dose per unit energy emitted by the source. Therefore, the mean absorbed dose can be written as

$$\bar{D} = \tilde{A}\, \Delta F \tag{IV.31}$$

The MIRD schema defines the mean dose to the target per unit cumulated activity (S):

$$S(T \leftarrow S) = \Delta f / m_T = \Delta F \tag{IV.32}$$

which permits another mean absorbed dose relationship to be written:

$$\bar{D} = \tilde{A}\, S(T \leftarrow S) \tag{IV.33}$$

where

S = mean absorbed dose per unit cumulated activity in mGy/MBq-s or rad/μCi-hr

In (IV.33), most of the metabolic factors are contained in the $\tilde{A}$ term, which depends on uptake by the source organ and biological elimination of the radio-

pharmaceutical by the source organ. The S factor represents much of the physical decay process including the decay characteristics of the radionuclide, the range of the emitted radiations, and the organ size and configuration. If a standard anatomy is utilized, S can be calculated and tabulated for a variety of radionuclides and source-target combinations. MIRD Publication 11 provides a tabulation of these S values.

Alternate MIRD Equation

The basic MIRD equation can be rewritten in terms of the activity administered to the patient (A_0):

$$\bar{D}/A_0 = (\tilde{A}/A_0)\,S \tag{IV.34}$$

where both sides of (IV.33) have been divided by A_0. The residence time (T) of a radiopharmaceutical in the source organ is defined as

$$T = \tilde{A}/A_0 \tag{IV.35}$$

The mean dose per unit administered activity $(\bar{D}/A_0)$ is expressed as

$$\bar{D}/A_0 = T\,S \tag{IV.36}$$

MIRD Equation

The MIRD schema can be expanded to include a single radiopharmaceutical which leads to multiple source organs and multiple radiation types. In order to define the more general MIRD equation, a number of terms need to be defined.

For radiation type i, the mean energy emitted per nuclear transition Δ_i is defined as

$$\Delta_i = K\,E_i\,n_i \tag{IV.37}$$

where E and n were defined previously. The total mean energy emitted per nuclear transition from all radiation types is expressed as

$$\Delta = \sum_i \Delta_i \tag{IV.38}$$

The absorbed fraction f_i is defined as

$$f_i(r_T \leftarrow r_S) = b/B \tag{IV.39}$$

where b is the i th-type radiation energy emitted in the source r_S and absorbed in target r_T. The label r defines an anatomical region, with r_S defining the source organ and r_T defining the target organ. B is the i th-type radiation emitted in the source. MIRD Pamphlet No. 3 defines the f_i values as summarized in Table IV.1.

In a similar fashion, the specific absorbed fractions F_i and the mean dose per unit cumulated activity $S(r_T \leftarrow r_S)$ are defined as

$$F_i(r_T \leftarrow r_S) = \frac{f_i(r_T \leftarrow r_S)}{m_T} \tag{IV.40}$$

$$S(r_T \leftarrow r_S) = \sum_i \Delta_i \, F_i(r_T \leftarrow r_S) \tag{IV.41}$$

Equations (IV.40) and (IV.41) are defined analogously to the single target equations.

These definitions permit the general MIRD equation describing the mean absorbed dose to the target organ $\bar{D}_T$ to be defined as a sum of dose contributions from all source organs (S):

$$\bar{D}_T = \sum_S \tilde{A}_S \sum_i K \, E_i \, n_i \, f_i(r_T \leftarrow r_S)/m_T \tag{IV.42}$$

Equation (IV.42) may be rewritten in a variety of ways to suit the particular dose calculation:

$$\bar{D}_T = \sum_S \tilde{A}_S \sum_i \Delta_i \, F_i(r_T \leftarrow r_S) \tag{IV.43}$$

$$\bar{D}_T = \sum_S \tilde{A}_S \, S(r_T \leftarrow r_S) \tag{IV.44}$$

and

$$\frac{\bar{D}_T}{A_0} = \sum_S T_S \, S(r_T \leftarrow r_S) \tag{IV.45}$$

Table IV.1 f_i Values.

Radiation Type	$f_i(r_T \leftarrow r_S)$
Penetrating radiations (photons with energy > 10–20 keV)	0 to 1
Nonpenetrating radiations (beta particles, positrons, alpha particles, and photons with energies < 10–20 keV):	
When $S = T$	1
When $S \neq T$	0

Stochastic and Nonstochastic Effects

Stochastic effects are those in which the probability of the effect increases with increasing dose without threshold. That is, any dose has a probability of causing the effect. Stochastic effects may result from injury to a small number of cells.

Cancer and hereditary effects are examples of stochastic effects. Once these types of effects are induced, their severity is already determined. The severity of cancer or hereditary effects does not increase with dose. Only the probability of their incidence increases with dose.

Nonstochastic effects are those in which the severity of the effect varies with the dose. That is, if the dose is kept below a given threshold, no effect will be observed. Nonstochastic effects result from the collective damage to a large number of cells in an organ or tissue. Examples of nonstochastic effects include cataracts, impairment of fertility, and depletion of blood-forming cells in bone marrow.

ICRP-26/30 Dose Limits

In the late 1970s, ICRP-26/30 established a dose-limitation system designed to prevent nonstochastic effects and to limit the stochastic effects of radiation. The system of dose control and limitation recommended in 1977 was intended to be applied to controllable activities encountered during normal facility operations.

The ICRP dose-limit recommendations for occupational exposure are formulated to prevent stochastic and nonstochastic effects. To prevent nonstochastic effects, the ICRP recommends that exposures be limited to 50 rem/year (0.5 Sv/year) to all tissues except the lens of the eye and that the lens of the eye be limited to 15 rem/year (0.15 Sv/year). To limit stochastic effects, 5 rem (50 mSv) in 1 year for uniform whole-body irradiation is recommended.

The nonstochastic limits are based on the 50-year organ doses ($H_{50,T}$), and the stochastic limits depend upon the effects of the organ doses upon the whole body:

$$H_E = \sum_T w_T \, H_{50,T} \tag{IV.46}$$

where

H_E = effective dose equivalent (rem)
T = tissue
w_T = weighting factor for tissue (T)

The tissue weighting factors, derived on the basis of stochastic risk, are defined as follows:

Tissue	w_T	Tissue	w_T
Gonads	0.25	Thyroid	0.03
Breast	0.15	Bone surfaces	0.03
Red bone marrow	0.12	Remainder (five highest other organs)	0.30
Lung	0.12		

As in ICRP-2, the period of integration of the dose equivalent is the entire working lifetime, which is selected to be 50 years. The committed dose equivalent (CDE) to an organ ($H_{50,T}$) is defined as the total dose equivalent in an organ after the intake of a radionuclide into the body.

In order not to exceed either the stochastic or nonstochastic limits for internal exposure, the concepts of the annual limit an intake (ALI) and derived air concentration are introduced. Following ICRP-30, the ALI is defined to be the largest value of I which satisfies both of the following inequalities:

$$I \sum_{T} w_T \, H_{50,T} \leq 0.05 \text{ Sv} \qquad \text{for stochastic effects} \tag{IV.47}$$

$$I \, H_{50,T} \leq 0.5 \text{ Sv} \qquad \text{for nonstochastic effects} \tag{IV.48}$$

It should be noted that $H_{50,T}$ is specified as the dose per unit intake (Sv/Bq) which yields the correct units for I.

The derived air concentration (DAC) in Bq/m^3 for any radionuclide is that concentration in air which, if breathed by reference man for a working year under conditions of light activity, would result in the uptake of one ALI by Inhalation:

$$\text{DAC} = \text{ALI}/200 \text{ hr/year} \times 1.2 \text{ m}^3/\text{hr} = \text{ALI}/2400 \text{ m}^3 \tag{IV.49}$$

The factor of 2400 m^3 is obtained from the assumption of breathing 1.2 m^3 per hour for 2000 working hours per year.

ICRP-30 Methodology

Metabolic Models

The metabolic models summarized within the ICRP-30 framework have a number of common characteristics. These characteristics include the following:

1. Simple, linear first-order equations adequately describe the translocation of material.
2. Contents mix in an instantaneous, uniform manner.
3. Organ deposition from the transfer compartments occurs rapidly. Removal is governed by specific retention times.
4. Direct excretion occurs from the organs and tissues after deposition.
5. Metabolic properties of a nuclide are described by a set of rate constants that are fixed for the 50-year deposition period.
6. Daughter atoms born in the body following deposition of the parent are usually metabolized with the properties of the parent.

These properties form the basis of the ICRP internal dosimetry models.

ICRP-30 Lung Model

The mathematical model used to describe clearance from the respiratory tract consists of a nasopharyngeal (NP) region, a tracheobronchial (TB) region, a pulmonary region (P), and the lymph (L) nodes. Deposition is governed by either the activity median aerodynamic diameter (AMAD) or the mass median aerodynamic diameter (MMAD) of the inhaled aerosol.

The model describing the deposition of particles into the various respiratory tract regions is summarized in ICRP-30. The percentage of activity or mass of an aerosol which is deposited in the NP, TB, and P regions is given as a function of the AMAD of the aerosol distribution.

The deposition model is applicable to aerosols having an AMAD between 0.2 and 10 μm and with geometric standard deviations of 4.5 or less. The model does not apply to aerosol distributions with an AMAD of 0.1 μm or less. For distributions having an AMAD greater than 20 μm, complete NP deposition can be assumed.

The retention model describes the clearance from the respiratory region. Table IV.2 summarizes the key parameters of this model. The values for the removal half-times (T) and compartmental fractions (F) are tabulated for each of the three classes of retained material.

The respiratory tract clearance model assigns particles to one of three classes: D, W, and Y. The chemical composition of the aerosol will govern the class assignment which is similar to the ICRP-2 assignment of material in terms of its solubility characteristics. The ICRP-30 classes describe clearance half-times for material deposited in the lung which are on the order of days (D), weeks (W), and years (Y). Class D, W, and Y aerosols have clearance half-times less than 10 days, between 10 and 100 days, and greater than 100 days, respectively.

The values given for D_{NP}, D_{TB}, and D_P (left column), given in Table IV.2, are the regional depositions for an aerosol with an AMAD of 1 μm. ICRP-30 presents a method for correcting the dose calculations when the AMAD is not 1 μm.

Values of $H_{50,T}$ per unit intake, ALI, and DAC values given in ICRP-30 for individual radionuclides are given for an AMAD of 1 μm. ICRP-30 also provides the fractions (f_{NP}, f_{TB}, and f_P) of the committed dose equivalents in the reference tissue resulting from depositions in the NP, TB, and P regions, respectively. The $H_{50,T}$ value for aerosols other than 1 μm are determined from

$$H_{50,T}(x) = H_{50,T}(1\ \mu\text{m}) \sum_i f_i \frac{D_i(x)}{D_i(1\ \mu\text{m})} \tag{IV.50}$$

where

i = 1, 2, and 3 refer to the NP, TB, and P lung regions, respectively
D_i = deposition probabilities in the respiratory region for a given AMAD
x = aerosol size for which the dose is required

ICRP-30 recommends that if the AMAD is unknown, then data for 1 μm should be used.

The schematic drawing (Figure IV.1) and Table IV.2 identify the various clearance pathways from compartments a–j in the four respiratory regions (NP, TB, P, and L).

Table IV.2 ICRP-30 Respiratory Model's Compartment Fractions and Clearance Half-Times.

Region	Compartment	Class					
		D		W		Y	
		T (day)	*F*	*T* (day)	*F*	*T* (day)	*F*
NP ($D_{NP} = 0.30$)	a	0.01	0.5	0.01	0.1	0.01	0.01
	b	0.01	0.5	0.40	0.9	0.40	0.99
TB ($D_{TB} = 0.08$)	c	0.01	0.95	0.01	0.5	0.01	0.1
	d	0.2	0.05	0.2	0.5	0.2	0.99
P ($D_P = 0.25$)	e	0.5	0.8	50	0.15	500	0.05
	f	NA[a]	NA	1.0	0.4	1.0	0.4
	g	NA	NA	50	0.4	500	0.4
	h	0.5	0.2	50	0.05	500	0.15
L	i	0.5	1.0	50	1.0	1000	0.9
	j	NA	NA	NA	NA	∞	0.1

Source: ICRP-30, Part 1.
[a] NA, not applicable.

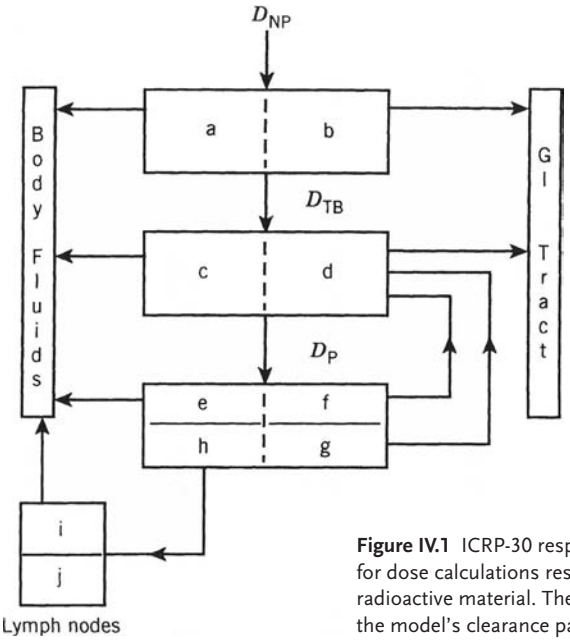


Figure IV.1 ICRP-30 respiratory tract clearance model for dose calculations resulting from an intake of radioactive material. The schematic drawing identifies the model's clearance pathways. (From ICRP-30, Part I.)

Table IV.3 ICRP-30 Absorption and Translocation Processes.

Pathway	Initial Location	Final Destination	Description
a	NP	Systemic blood	Rapid uptake from NP directly to the systemic blood.
b	NP	GI tract	Rapid clearance by ciliary-mucous transport.
c	TB	Systemic blood	Rapid absorption from TB directly to the systemic blood.
d	TB	GI tract	Analogous to pathway b.
e	P	Systemic blood	Direct translocation from P to the systemic blood.
f	P	TB and then GI tract via pathway d	Relatively rapid clearance of pulmonary region to TB followed by a ciliary-mucous transport to the GI tract.
g	P	TB and then GI tract via pathway d	Slower but analogous to pathway f. The P translocation is governed by the material characteristics.
h	P	Lymph nodes	Slow removal from P via lymph nodes.
i	Lymph	Systemic blood	Partial or complete particle dissolution impacts the transport to the systemic blood.
j	Lymph	Lymph	Insoluble material is retained indefinitely.

The various absorption and translocation processes associated with the compartments are designated by the labels a through j. These processes describe the movement of material from the major respiratory tract regions to the gastrointestinal tract, systemic blood, and the lymph nodes. A brief summary of the pathways is provided in Table IV.3.

ICRP-30 Ingestion Model

The ingestion or inhalation of radioactive material leads to the transport of this material into the GI tract. The GI tract is divided into the four regions illustrated in Figure IV.2 and characterized in Table IV.4. The tract is divided into the stomach (S), small intestine (SI), the upper large intestine (ULI), and the lower large intestine (LLI). Absorbed doses are modeled as averages over the particular tract section.

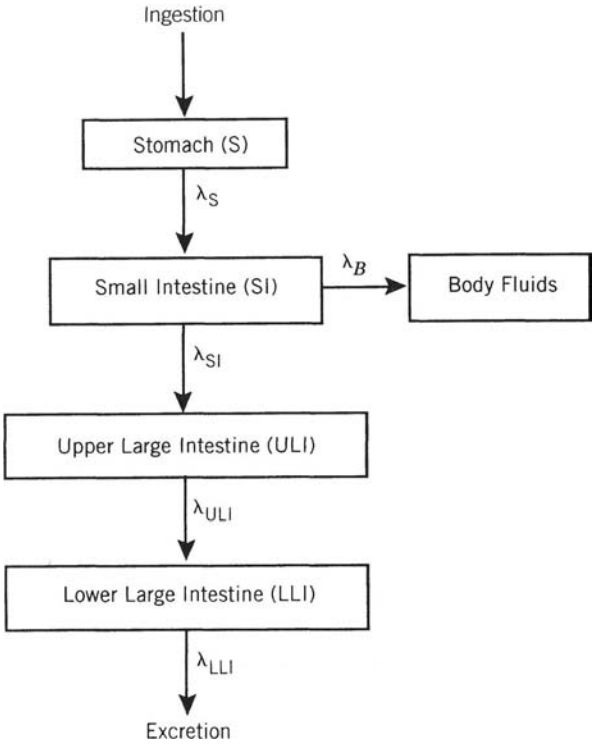


Figure IV.2 ICRP-30 gastrointestinal tract model for dose calculations resulting from an intake of radioactive material. The schematic drawing identifies the model's clearance pathways. (From ICRP-30, Part 1.)

Table IV.4 Gastrointestinal Tract Model for Reference Man.

Section of GI Tract	Mass of Walls (g)	Mass of Contents (g)	Mean Residence Time (day)	λ (1/day)
Stomach (S)	150	250	1/24	24
Small intestine (SI)	640	400	4/24	6
Upper large intestine (ULI)	210	220	13/24	1.8
Lower large intestine (LLI)	160	135	24/24	1

Source: ICRP-30, Part 1.

Table IV.4 specifies the rate constant for each of the sections of the GI tract.

The rate constant for removal of material from the small intestine into the body fluids (λ_B) is defined in ICRP-30 by

$$\lambda_B = \frac{f_1 \lambda_{SI}}{1 - f_1} \tag{IV.51}$$

where f_1 is the fraction of the stable element reaching the body fluids after ingestion.

Table IV.4 provides the model parameters for each section of the GI tract. These parameters include the mass of the tissue walls, the mass of the contents (food) residing in the tissue, and the mean residence time of the food in each tract section.

Each of the sections in the GI tract is a single compartment, and the movement of material from one compartment to the next is represented by first-order kinetics and their associated linear differential equations. These equations are provided in ICRP-30.

The ICRP model for dosimetry in the GI tract assumes that material entering each segment of the tract instantaneously mixes uniformly with the contents and is removed from the segment at a constant instantaneous fractional rate. An insoluble radionuclide's total removal rate constant for a particular segment is given simply by the sum of the radionuclide's decay constant and the transfer rate constant for the segment determined from the mean residence time for the segment:

$$k^{(i)} = \lambda + \frac{1}{T^{(i)}} \tag{IV.52}$$

where

λ = decay constant of the radionuclide
$T^{(i)}$ = mean residence time in segment i

For specificity $i = 1, 2, 3$, and 4, refer to the stomach, small intestine, upper large intestine, and lower large intestine, respectively. The dose equivalent delivered to the walls of a segment of the GI tract is calculated from one-half the energy spatial equilibrium dose delivered to the contents by particulate radiation emitted by the radionuclide in the segment.

Calculation of Doses

Following a methodology similar to that used in the development of the MIRD dose equation, the 50-year committed dose equivalent to a target organ (T) is calculated. Estimates are made of committed dose equivalents in a number of target organs from the activity in a given source organ. For each type of radiation i, $H_{50,T}$ in target organ T resulting from radionuclide j in source organ S is the product of two factors. The first factor is the total number of transformations of radionuclide j in S over a period of 50 years after intake. This factor is analogous to the MIRD $\tilde{A}$ factor. The second factor is the energy absorbed per gram in T, suitably modified with a quality factor, from radiation of type i per transformation of radionuclide j in S. This factor is similar to the MIRD S-factor. For each radiation type i from radionuclide j, the committed dose equivalent is

$$H_{50,T}(T \leftarrow S)_i = (1.6 \times 10^{-10})\, U_S\, \mathrm{SEE}(T \leftarrow S)_i \tag{IV.53}$$

where

$H_{50,T}(T \leftarrow S)_i$ = 50-year committed dose equivalent for radiation type i of radionuclide j (Sv)

U_S = the number of transformations of radionuclide j in S over the 50 years following intake of the radionuclide

$\mathrm{SEE}(T \leftarrow S)_i$ = the specific effective energy (MeV/g-transformation) for radiation type i, suitably modified by a quality factor, absorbed in T from each transformation in S

Equation (IV.53) may be generalized to include all types of radiation (i) emitted by radionuclide j :

$$H_{50,T}(T \leftarrow S)_j = \left[\sum_i H_{50,T}(T \leftarrow S)_i\right]_j \tag{IV.54}$$

$$H_{50,T}(T \leftarrow S)_j = (1.6 \times 10^{-10})\left[U_S \sum_i \mathrm{SEE}(T \leftarrow S)_i\right]_j \tag{IV.55}$$

If radionuclide j has a radioactive daughter j, then we obtain

$$H_{50,T}(T \leftarrow S)_{j+j'} = 1.6 \cdot 10^{-10}\left\{\left[U_S \sum_i \mathrm{SEE}(T \leftarrow S)_i\right]_{j'}\right\} \tag{IV.56}$$

For the intake of any mixture of radionuclides, the dose $H_{50,T}$ in the target from activity in the source S is given by

$$H_{50,T} = \sum_j H_{50,T}(T \leftarrow S)_j \tag{IV.57}$$

$$= 1.6 \times 10^{-10} \sum_j \left[U_S \sum_j \mathrm{SEE}(T \leftarrow S)_i\right]_j \tag{IV.58}$$

where the sum over j is over all the radionuclides involved.

For the general case in which target T is irradiated by multiple radiation sources, (IV.58) becomes

$$H_{50,T} = 1.6 \times 10^{-10} \sum_S \sum_j \left[U_S \sum_i \mathrm{SEE}(T \leftarrow S)_i\right]_j \tag{IV.59}$$

where i represents the sum over radiation type, j labels the radionuclide, and S labels the source organ.

ICRP-60/66 Methodology

In 1991, the ICRP-26 recommendations regarding the biological effects of ionizing radiation were revised by ICRP-60. The ICRP-60 methodology incorporated additional epidemiological data including BEIR V and additional dose assessments including the 1986 Dosimetry System (see Appendix V). In 1994, ICRP-66 provided a revised lung model. However, ICRP-60 retained the ICRP-30 ingestion model.

ICRP-60 also addressed the methodology used to define the risk of ionizing radiation as a result of periodic reviews of basic and derived radiological data. Based upon these series of reviews, the ICRP concluded that the risks of radiation-induced cancers are a factor of 3–4 greater than those published in 1977.

The ICRP-60 recommendations include situations where there is a probability of exposure, including accident situations and the disposal of solid radioactive waste. The recommendations and ongoing Task Group efforts also include those situations in which the source is not under control, including radon intrusion into homes. These recommendations and Task Group efforts are applicable to nearly all situations encountered by the health physics professional.

Detriment

There is considerable uncertainty regarding the best manner to characterize the consequences of the effects of exposure to low levels of ionizing radiation. These effects include the various forms of cancer and hereditary or stochastic effects resulting from exposures of a few tens of mSv/year (few rem/year). The detrimental effects include severe hereditary diseases or stochastic effects and nonstochastic cancers.

The ICRP considers health detriment effects resulting from low exposures to ionizing radiation to the organs and tissues of the body. The first consideration is the probability of a radiation-induced cancer death. The weighted probability of a nonfatal radiation-induced cancer or of a severe hereditary disease occurring in the children of irradiated individuals was also considered by the ICRP. Finally, the ICRP considered the time of the appearance of these detrimental effects following

Table IV.5 ICRP-60 Risk Factors from Exposure to Low Doses of Ionizing Radiation.

Detrimental Effect	Risk Coefficient (Detriment) (1.0×10^{-4} effects/rem)	
	Public	Radiation Workers
Fatal cancers	5	4
Nonfatal cancers (weighted)	1.0	0.8
Hereditary effects	1.3	0.8
Total detriment	7.3	5.6

Source: ICRP-60.

the exposure to ionizing radiation. The contributions of these factors to the risk is summarized in Table IV.5.

The total health detriment following exposure to low doses of radiation is 7.3×10^{-4}/rem for a nominal population. The total detriment is slightly less for a working-age population of radiation workers whose ages range from 18 to 64 years. The differences between the public and the radiation worker risk coefficients are influenced by the omission of younger, more radiosensitive persons and the shorter period of potential reproduction during occupational exposure for persons commencing work at age 18.

Terminology

Although the exact nature of radiation and tissue interactions is not fully understood, ICRP-60 focuses on doses averaged over tissue rather than the dose at a point. The average absorbed dose in the tissue or organ (T) of interest, due to all radiations (R), weighted by a radiation weighting factor w_R is defined to be the equivalent dose (H_T):

$$H_T = \sum_R w_R \, D_{T,R} \tag{IV.60}$$

where $D_{T,R}$ is the average absorbed dose in tissue T due to radiation of type R. The radiation weighting factor replaces the quality factor which is related to the physical effects of the radiation at a point. Values of the radiation weighting factor as a function of the radiation type and energy are defined in Table IV.6. The equivalent

Table IV.6 Radiation Weighting Factors.[a]

Type and Energy Range[b]	Radiation Weighting Factor
Photons (all energies)	1
Electrons and muons (all energies)[c]	1
Neutrons	
< 10 keV	5
10 keV to 100 keV	10
> 100 keV to 2 MeV	20
> 2 MeV to 20 MeV	10
> 20 MeV	5
Protons, other than recoil protons (> 2 MeV)	5
Alpha particles, fission fragments, and heavy nuclei	20

[a] All values relate to the radiation incident an the body or, for internal sources, emitted from the source.
[b] The choice of values for other radiations is discussed in Annex A, ICRP-60.
[c] Excluding Auger electrons emitted from nuclei bound to DNA.

dose is significantly different than the ICRP-26/30 dose equivalent, which is a point quantity. The effective dose (E), analogous to the effective dose equivalent, is also defined in ICRP-60. The effective dose expresses the relative detriment associated with each irradiated tissue or organ, and its response is expressed as if the whole body were irradiated. The tissue weighting factor (w_T) is obtained by expressing the detriment of each tissue-specific cancer or hereditary disease relative to the total aggregated detriment. With these definitions, the effective dose is defined as

$$E = \sum_T w_T \, H_T \tag{IV.61}$$

which can be written in terms of the double sum of the average absorbed dose in tissue T due to the radiation R:

$$E = \sum_R w_R \sum_T w_T \, D_{T,R}$$

or

$$E = \sum_T w_T \sum_R w_R \, D_{T,R} \tag{IV.62}$$

A comparison of the tissue weighting factors for ICRP-26, UNSCEAR-1988, ICRP-60, and ICRP-103 is provided in Table IV.7. The reader should note the wide variation in the selection of both organs and values of the weighting factors. Clearly, the selection of organs and weighting factors will continue to evolve as better data become available. The 10 remainder tissues for ICRP-60 include the following organs: adrenals, brain, small intestine, spleen, kidneys, muscle, pancreas, upper large intestine, thymus, and uterus. In the exceptional case in which one of the remainder tissues or organs receives an equivalent dose in excess of the highest dose in any of the 12 organs for which a weighting factor is assigned, a weighting factor of 0.025 should be applied to that remainder organ or tissue. A weighting factor of 0.025 should also be assigned to the average dose in the rest of the remainder.

ICRP-60 defines subsidiary dosimetric quantities including committed effective dose $E(t)$ and the committed equivalent dose $H(t)$. For intakes of radionuclides, the ICRP defines the committed equivalent dose as the time integral of the equivalent dose rate:

$$H_T(d) = \int_{t_0}^{t_0+d} \dot{H}(t) \, \mathrm{d}t \tag{IV.63}$$

where

t_0 = time that the intake occurs
d = integration time following the intake

Table IV.7 Tissue Weighting Factors for Various Models.

Tissue	ICRP-26 (1977)	UNSCEAR (1988)	ICRP-60 (1991)	ICRP-103 (2007)
Gonads	0.25	–	0.20	0.08
Breast	0.15	0.05	0.05	0.12
Bone marrow (red)	0.12	0.17	0.12	0.12
Lung	0.12	0.17	0.12	0.12
Thyroid	0.03	–	0.05	0.04
Bone surfaces	0.03	–	0.01	0.01
Stomach	–	0.18	0.12	0.12
Colon	–	0.09	0.12	0.12
Esophagus	–	0.04	0.05	0.04
Bladder	–	0.05	0.05	0.04
Ovary	–	0.03	–	–
Skin	–	—	0.01	0.01
Liver	–	–	0.05	0.04
Multiple myeloma	–	0.03	–	–
Brain	–	–	–	0.01
Salivary glands	–	–	–	0.01
Remainder	0.30	0.19	0.05[a]	0.12[b]

[a] The ICRP-60 remainder tissues are: adrenals, brain, small intestine, spleen, kidneys, muscle, pancreas, upper large intestine, thymus, and uterus.
[b] The ICRP-103 remainder tissues are: adrenals, extrathoracic (ET) region, gall bladder, heart, kidneys, lymphatic nodes, muscle, oral mucosa, pancreas, prostate (♂), small intestine, spleen, thymus, and uterus/cervix (♀).
Source: ICRP-26 (1977), ICRP-60 (1991), UNSCEAR (1988), and ICRP-103 (2007).

For occupational dose, d is taken to be 50 years; for nonoccupational exposures, 70 years is selected.

For intakes of radionuclides, the ICRP defines the committed effective dose $E(t)$ as the time integral of the effective dose rate. The integration times are the same as defined for the committed equivalent dose.

The committed effective dose $E(50)$ for the worker is explicitly written as

$$E(50) = \sum_{T=1}^{12} w_T \, H_T(50) + w_{\text{remainder}} \frac{\sum_{t=13}^{22} m_T \, H_T(50)}{\sum_{T=13}^{22} m_T} \tag{IV.64}$$

where

$H_T(50)$ = committed equivalent dose
m_T = mass of the remainder tissues
$W_{\text{remainder}}$ = 0.05

In (IV.64), the first sum is over the 12 organs/tissues with assigned weighting factors and the second sum is over the 10 remainder organs/tissues.

For the exceptional case, noted above for the remainder organs/tissues, the committed effective dose $E(50)$ for the worker is computed as

$$E(50) = \sum_{T=1}^{12} w_T \, H_T(50) + 0.025 \, H_{T'}(50) + 0.025 \frac{\sum_{T=13}^{22} m_T \, H_T(50) - m_{T'} \, H_{T'}(50)}{\sum_{T=13}^{22} m_T - m_{T'}} \tag{IV.65}$$

where $m_{T'}$ is the mass of the remainder tissue or organ in which the committed equivalent dose is calculated to be higher than that in any of the 12 specified tissues/organs with assigned weighting factors, and $H_{T'}(50)$ is the committed equivalent dose in that remainder tissue/organ.

Framework

The recommendations of the ICRP are based upon the prevention of deterministic effects by keeping equivalent or effective dose limits below the relevant threshold for these effects. Implicit is the demand that all reasonable actions are taken to reduce the incidence of stochastic effects to acceptable levels.

Exposure to a radiation source involves the transit of radiation from within the source, through a series of media (air, shielding, worker clothing, etc.) and into the tissues of the irradiated person. The radiation source may originate in a variety of locations including the environment, industrial facility, medical establishment, public building, or individual residence. Although radiation may be controlled at any of these locations, the most effective way of controlling radiation exposure is at the source.

The ICRP defines activities related to radiation exposure and control. Activities that increase radiation exposures or risks are defined as “practices”, and those activities that reduce exposure are “interventions”. For both practices and interventions, some radiation exposure will occur. The magnitude of the exposure is predictable with a reasonably defined error. The possibility also exists that, although there is the potential for an exposure, the exposure may not occur. The ICRP calls such exposures “potential exposures”. Within the framework of these new terms, the ICRP formulates a system of radiation protection whose requirements are as follows:

1. Any practice should do more good than harm, and the practice should produce a net benefit to the exposed individual or to society.
2. Doses or the likelihood of being exposed should be as low as reasonably achievable. Exposures and risks should be optimized.
3. Individuals should be subject to dose limits or to the control of risk in the case of potential exposures.

Any radiation protection program should be judged in terms of its effectiveness. Assessments should be based upon the distribution of measured doses in the exposed population and on the effectiveness of actions taken to limit the probability of potential exposures.

Practices

For practices encountered during a facility's normal operation, the ICRP radiation protection system includes the requirement to optimize the radiation protection program elements to maximize the benefit of exposure to ionizing radiation. This requirement involves professional judgment, supported by quantitative procedures such as a cost benefit analysis, to determine the point at which any further improvements can be obtained only through the use of excessive resources. Optimization is a source-related process, and all sources of radiation are to be considered in applying the ICRP's limits.

Constraint

The ICRP-60 recommendations also include the concept of a "constraint" which indicates a restriction to be applied during optimization of individual doses resulting from a single source of exposure. The constraint represents an upper bound that must not be exceeded. Any option causing the constraint to be exceeded would be excluded from further consideration.

A constraint is not a dose limit which relates to the total dose received from all sources. However, the ICRP views the constraint as a regulatory requirement and not a design, target, or investigation level. Accordingly, it should be established by the regulatory agency based on industry judgment or experience at a well-managed operation performing the task under review.

The concept of a constraint differs from the ICRP's 1977 recommendations which used the dose limit to restrict the completion of a practice. The use of a constraint provides more control for those practices where a dose limit would be too lenient and therefore could lead to more worker exposure than warranted.

Intervention

An intervention is intended to improve the existing radiological conditions and, therefore, lower the collective exposures. Each intervention should be justified to ensure the proper balance of risk and benefit.

Two major examples of intervention are the actions taken following the discovery of elevated levels of radon in homes and following a potential or actual accidental release of radioactive material to the environment. In neither of these cases does the ICRP provide a recommendation about numerical values. For both subjects, task groups have been established. Until these task groups complete their work, ICRP Publications 39 and 40 remain applicable.

Dose Limits

It is unlikely that the dose limit will be the limiting factor during the conduct of operational activities. However, these limits are needed to ensure control of the total exposure from facility practices, to restrict the choice of dose constraints, and to protect against judgment errors during optimization.

For occupational exposures, an effective does limit of 20 mSv/year (2 rem/year) to be averaged over a period of 5 years is established. No more than 50 mSv (5 rem) should be received in a single year. Public effective dose limits should be limited to 1 mSv/year (100 mrem/year) averaged over 5 years. The ICRP does permit a higher public limit in a single year, provided that the average over 5 years does not exceed 1 mSv/year.

For the skin, the recommended annual occupational organ equivalent limit is 500 mSv (50 rem) averaged over 1 cm^2, regardless of the area exposed. The nominal depth for the dose evaluation is 7 mg/cm^2. The annual occupational dose limit for the hands and feet is also 500 mSv. The public dose limit for the skin, hands, and feet is 50 mSv. The annual limit for the lens of the eye is 150 mSv and 15 mSv for occupational and public exposure limits, respectively.

The basis for control of occupational exposure of women who are not pregnant is the same as that for men. However, if a woman is or may be pregnant, additional controls are required to protect the unborn child. Once pregnancy has been declared, the embryo/fetus should be protected by applying a supplementary equivalent dose limit of 2 mSv to the surface of the woman's abdomen for the remainder of the pregnancy, and internal exposure is restricted to 1/20 of an ALI.

ICRP-60 also recommends criteria for exemptions from regulatory control. Exemptions are recommended if individual doses are unlikely to exceed 10 μSv/year (1 mrem/yr), and the collective dose is not more than 1 person-Sv per year of practice.

ICRP-103/66/100 Methodology

ICRP-103 documents the 2007 recommendations of the International Commission on Radiological Protection, and it replaces the ICRP-60 recommendations. The ICRP-103 recommendations incorporate the results of the BEIR VII report and supporting Dosimetry System 2002 (DS02). The linear-no-threshold assumption is integral to ICRP-103. The 2007 recommendations utilize the ICRP-66 lung model and the ICRP-100 ingestion model.

ICRP-100 provides a human alimentary tract model (HATM) to complement the human respiratory tract model (HRTM) presented in ICRP-66. The HRTM and HATM are replacement models for the predecessor models of ICRP-30. The ICRP-30 models were developed for the calculation of occupational doses. An improvement of both the HATM and the HRTM is their capability to calculate doses from intakes by children as well as male and female adults. Therefore, the new models are applicable to environmental as well as occupational exposures. Another important development in the HATM and HRTM is the calculation of doses to target regions containing cells considered susceptible to cancer induction.

Radiation Effects, Tissue Weighting Factors, and Radiation Weighting Factors

The linear-no-threshold (LNT) model is a fundamental assumption used in the formulation of ICRP-103. It is the basis for the averaging and summing of doses, the effective-dose concept, the collective-dose concept, individual dosimetry, and keeping dose records.

Table IV.8 summarizes the revised ICRP-103 risk coefficients and compares them to the ICRP-60 values. The most significant change from ICRP-60 is the 6–8 fold decrease in the risk coefficient for hereditary effects. This change arises because ICRP-103 chose to express these risks up to the second generation rather than a theoretical equilibrium value because the ICRP-60 methodology is no longer sustainable. There are no substantial differences between genetic risks expressed at 2 and 10 generations.

The use of the DS02 dosimetry system in BEIR VII further reduces the risk coefficients by about 10% in comparison to the ICRP-60 values that were based on BEIR V and the DS86 dosimetry system.

The nominal radiation risk estimates are slightly smaller than the ICRP-60 values. The overall risk of 0.05/Sv continues to be appropriate for radiological protection purposes. This risk value is also consistent with BEIR VII and ICRP-60.

The ICRP revised the tissue weighting factors and significant revisions occurred for the breast, gonads, and treatment of remainder tissues (Table IV.7). For the purposes of radiological protection, the tissue weighting factors are assumed valid for both sexes and modeled age groups.

Table IV.8 ICRP-103 Detriment Adjusted Nominal Risk Coefficients for Cancer and Hereditary Effects (10^{-2}/Sv).

Exposed Population	Cancer		Hereditary Effects		Total	
	ICRP-103	ICRP-60	ICRP-103	ICRP-60	ICRP-103	ICRP-60
Whole	5.5	6.0	0.2	1.3	5.7	7.3
Adult	4.1	4.8	0.1	0.8	4.2	5.6

Source: ICRP-103.

Table IV.9 ICRP-103 Recommended Radiation Weighting Factors.[a]

Radiation Type	Radiation Weighting Factor (w_R)
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, and heavy ions	20
Neutrons	[b]

[a] All values relate to the radiation incident on the body or, for internal sources, emitted from the incorporated radionuclide(s).
[b] A continuous function of the neutron energy (see (IV.66))
Source: ICRP-103.

Biological and dosimetric considerations were evaluated in the reappraisal of the ICRP-60 radiation weighting factors. Most relative biological effectiveness values were derived from high doses and were extrapolated to the low-dose regime appropriate for radiological protection purposes. The w_R values for photons, beta particles, and alpha particles remain the same as the ICRP-60 values. The proton radiation weighting factor for energies greater than 2 MeV was revised because the ICRP-60 value was judged to be a significant overestimate of the biological effectiveness. For neutrons, the w_R values are energy dependent and based on the continuous function:

$$w_R = \begin{cases} 2.5 + 18.2\, e^{-[\ln(E_n)]^2/6} & E_n < 1 \text{ MeV} \\ 5.0 + 17.0\, e^{-[\ln(2E_n)]^2/6} & 1 \le E_n \le 50 \text{ MeV} \\ 2.5 + 3.25\, e^{-[\ln(0.04E_n)]^2/6} & E_n > 50 \text{ MeV} \end{cases} \quad \text{(IV.66)}$$

where
E_n = neutron kinetic energy

The ICRP-103 radiation weighting factors are summarized in Table IV.9.

Sex-Averaging

The effective dose is calculated from the equivalent dose or from coefficients of the effective dose or equivalent dose. These coefficients are represented by convention as e and h for the effective dose and equivalent dose, respectively.

Given the uncertainties in the dose assessment, the ICRP calculated a single value for effective dose for both sexes. Therefore, the tissue weighting factors of Table IV.7 are sex-averaged values for all tissues and organs including the male and female breast, testis, and ovary. The effective dose (E) is calculated from the

equivalent dose (H) for tissue T of the Reference Male (H_T^M) and Reference Female (H_T^F), including the remainder tissues using the equation:

$$E = \sum_{T=1}^{27} w_T \left[\frac{H_T^M + H_T^F}{2} \right] \tag{IV.67}$$

where the sum includes organs with assigned weighting factors ($T = 1, 14$) and remainder tissues ($T = 15 - 27$).

The equivalent dose to the tissues of the remainder of the Reference Male and Reference Female is computed as the arithmetic mean:

$$H_{\text{remainder}}^M = \frac{1}{13} \sum_{T=15}^{27} H_T^M \quad \text{and} \quad H_{\text{remainder}}^F = \frac{1}{13} \sum_{T=15}^{27} H_T^F \tag{IV.68}$$

where the sum is over the 13 tissues of the remainder listed in footnote b of Table IV.7. For radiation protection purposes, the effective dose is based on the mean dose to the organs/tissues in the ICRP-89 Reference Male and Reference Female. Following ICRP-103, the weighting factors are mean values representing an average over many individuals of both sexes. For males, remainder tissues are: adrenals, extrathoracic region, gall bladder, heart, kidneys, lymphatic nodes, muscle, oral mucosa, pancreas, prostate, small intestine, spleen, and thymus. For females, the sum excludes the prostate and includes the uterus/cervix.

Assessment of Occupational Dose

The total effective dose for demonstrating compliance with dose limits and constraints is defined in terms of the committed effective dose $E(50)$ from internal exposure and the personal dose equivalent from external exposure $H_p(10)$:

$$E \cong H_p(10) + E(50) \tag{IV.69}$$

and

$$E(50) = \sum_j e_{j,\text{inh}}(50) \cdot I_{j,\text{inh}} + \sum_j e_{j,\text{ing}}(50) \cdot I_{j,\text{ing}} \tag{IV.70}$$

where the personal dose equivalent is evaluated at a depth of 10 mm, $e_{j,\text{inh}}(50)$ is the committed effective dose coefficient for inhalation intakes of radionuclide j, $I_{j,\text{inh}}$ is the inhalation activity intake of radionuclide j, $e_{j,\text{ing}}(50)$ is the committed effective dose coefficient for ingestion intakes of radionuclide j, and $I_{j,\text{ing}}$ is the ingestion activity intake of radionuclide j. The $e(50)$ values are sex-averaged effective dose coefficients for the intake of specified radionuclides and are calculated using the relationship:

$$e(50) = \sum_{T=1}^{27} w_T \left[\frac{h_T^M(50) + h_T^F(50)}{2} \right] \tag{IV.71}$$

where w_T is the tissue weighting factor for tissue T, and h_T^M and h_T^F are the committed equivalent dose coefficients for tissue T of the male or female, respectively. The sum of (IV.71) also includes the remainder tissues in both the male and female.

Use of Effective Dose and Collective Dose

Effective Dose is used mainly for planning in prospective situations and for compliance with dose restrictions to control stochastic effects. It is not intended for detailed retrospective dose and risk assessments after exposure of individuals or epidemiological studies.

Collective Dose is used for optimization and for comparing radiological technologies and protection options. It is not a tool for epidemiological risk assessments or for predicting the number of cancer deaths due to trivial exposures to large populations.

Structure of the ICRP-103 System of Protection

Radiological protection deals with deterministic and stochastic effects. High doses cause deterministic effects or tissue reactions that occur above a threshold value. Both high and low doses produce stochastic effects that may be observed as a statistically significant increase in the incidence of these effects occurring long after exposure. The probabilistic nature of stochastic effects obscures a clear distinction between "safe" and "dangerous", and this introduces uncertainty in defining the control of radiation risks. To avoid this uncertainty, the ICRP introduces assumptions to create a specific model. For example, the LNT hypothesis combined with a dose and dose-rate effectiveness factor are used by the ICRP as the model for radiological protection at low doses and low dose rates. The validity of the model ultimately depends on the viability of its assumptions and formulation.

The ICRP adopts Principles of Protection that include both source-related and individual approaches. An individual may be exposed to several sources of radiation and an assessment of the total exposure is required. This assessment is called individual related. It is also necessary to consider the exposure of all individuals by a source or collection of sources. This procedure is called source related. Source-related principles include the concepts of justification and optimization and apply in all situations. The principle of justification requires that any decision that alters the radiation exposure situation should do more good than harm. Optimization of protection means that the number of exposed individuals and their dose should be kept as low as is reasonably achievable considering economic and societal factors.

Individual related principles utilize the application of dose limits and apply in planned situations. The total dose to an individual from planned exposure situations (excluding medical exposure of patients) should not exceed the dose-limit recommendations.

Exposure Situations

ICRP-103 recognizes three types of exposure situations (i.e., planned, emergency and existing). Planned exposures are situations involving the scheduled introduction and operation of sources including decommissioning and waste-disposal activities. Practices in operational activities are planned exposure situations.

Emergency exposures are unexpected situations that occur during the operation of a planned situation or from a malicious act. An emergency exposure requires urgent action.

Existing exposures are situations that already exist when a decision on control selection has to be taken. These exposure situations include natural background radiation and residual exposure from past practices.

Existing exposure situations can involve large distributions of individual exposures. These situations often affect the public and are usually controllable through pathway analysis. Time is a key parameter, and the level of exposure is driven by individual behavior.

Dose Constraints and Reference Levels

Dose constraints and reference levels are values that should not be exceeded. A dose constraint is a prospective and source-related restriction in planned exposure situations that serves as an upper bound on the dose in the optimization of protection for a source. Medical exposure of patients is not included in formulation of a constraint.

Reference levels are restrictions on dose or risk applied during optimization in emergency or existing (excluding medical exposure of patients) exposure situations. Planned exposures should not exceed the dose constraint or the diagnostic reference level. Emergency and existing exposures should be kept below the reference level.

The selected value of a constraint or reference level depends on the specific exposure circumstances. In addition, the value of the constraint or reference level does not represent a demarcation between a safe or unsafe exposure or reflect a step change in the associated health risk. Table IV.10 illustrates the types of dose restrictions (limits, constraints, and reference levels) used in the ICRP-103 system of protection in terms of the type of exposure situations.

The use of the optimization principle, source-related constraints, and individual-related dose limits has resulted in a reduction in worker and public doses. Constraints and reference levels apply to all situations, and the specific value depends on the circumstances associated with the exposure. Constraints and reference levels should optimize protection at the source. If a relevant constraint or reference level is not met, further protection options are considered.

For planned situations, dose constraints provide a basic level of protection. The constraints are set below the dose limits of each source to ensure that the limit is not exceeded. For occupational exposure, the dose constraint can be set by the facility operator or by regulatory authorities. Dose constraints are usually set by

Table IV.10 Dose Constraints and Reference Levels Used in the ICRP-103 System of Protection.

Exposure Situation	Category of Exposure		
	Occupational Exposure	**Public Exposure**	**Medical Exposure**
Planned Exposure	Dose limit Dose constraint	Dose limit Dose constraint	Diagnostic Reference Level[c] (Dose Constraint)[d]
Emergency Exposure	Reference level[a]	Reference level	Not applicable
Existing Exposure	Not applicable[b]	Reference level	Not applicable

[a] Long-term recovery operations should be treated as part of planned occupational exposure.
[b] Exposures resulting from long-term remediation or from protracted employment in affected areas should be treated as part of planned occupational exposure, even though the source of radiation is 'existing'.
[c] Patients.
[d] Comforters, care givers, and volunteers in research only.
Source: ICRP-103.

regulatory authorities for public exposures. For patients, comforters, and caregivers, dose constraints are usually set by medical professionals.

Reference levels are used for existing and emergency exposure situations. Prospectively, reference levels are used as a level of ambition. Retrospectively, they are used for assessing the effectiveness of protection, but not a mandatory level that must be achieved.

For emergency exposure situations, ICRP-60 and ICRP-63 recommend no response below the action levels. Recommended action-level values for the averted dose are appropriate for protective actions where intervention is usually justified. These protective actions include sheltering, administration of stable iodine, evacuation, and relocation. ICRP-103 recommends optimization below the reference levels. The ICRP recommends an upper value of the projected dose or reference level received from all pathways below which optimization is applied.

The application of reference levels is essentially the same approach as for constraints in planned situations. This approach involves (1) characterizing the exposure situation, (2) setting a reference level, and (3) optimizing protection accounting for the specific circumstances. This is an iterative process for implementing optimization, and yields an improvement in the level of protection for existing and emergency situations.

Factors Influencing the Choice of Source-Related Dose Constraints and Reference Levels

The process of selecting source-related dose constraints and reference levels involves the optimization process. Table IV.11 provides a framework for this

Table IV.11 Framework for Source-Related Dose Constraints and Reference Levels.

Band of Constraints and Reference Levels[a] (mSv)	Characteristics of the Exposure Situation	Radiological Protection Requirements	Examples
> 20 – 100[b,c]	Individuals are exposed to sources that are not controllable or where actions to reduce doses would be disproportionately disruptive. Exposures are usually controlled by action on the exposure pathways.	Consideration should be given to reducing doses. Increasing efforts should be made to reduce doses as they approach 100 mSv. Individuals should receive information on radiation risk and on the actions to reduce doses. Assessment of individual doses should be undertaken.	The reference level is set for the highest planned residual dose from a radiological emergency.
> 1–20	Individuals will usually receive direct benefit from the exposure situation but not necessarily from the exposure itself. Exposures may be controlled at the source or by action in the exposure pathways.	General information should be provided to enable individuals to reduce their doses. For planned situations, individual assessment of exposure and training should occur.	Constraints are set for occupational exposure in planned situations. Constraints are set for comforters and caregivers of patients treated with radiopharmaceuticals. A reference level is set for the highest planned residual dose from radon in dwellings.
1 or less	Individuals are exposed to a source that gives them little or no direct benefit, but benefits society. Exposures are usually controlled by action taken on the source for which radiological protection requirements can be planned.	General information on the level of exposure should be available. Periodic checks should be made on the exposure pathways to determine the level of exposure.	Constraints are set for public exposure in planned situations.

[a] Acute or annual dose.

[b] In exceptional situations, informed volunteer workers may receive doses above this band to save lives, prevent severe radiation-induced health effects, or prevent the development of catastrophic conditions.

[c] Actions are required for situations in which the dose threshold for deterministic effects in relevant organs or tissues could be exceeded.

Source: ICRP-103.

process. The specific value selected for the dose constraint or reference level is established by an optimization process that considers national or regional considerations and accepted good practices.

The banding of constraints and reference levels summarized in Table IV.11 applies to the three exposure situations of Table IV.10, and refers to the projected dose over an appropriate time. Constraints for planned exposures and reference levels in existing exposure situations are expressed as an annual effective dose. In an emergency exposure situation, the reference level is the total residual dose an individual would plan not to exceed during a single acute exposure or on an annual basis for a protracted exposure.

The 1 mSv or less band applies to exposure situations (usually planned) that may be of no direct benefit to the individual, but the exposure may benefit society. These doses represent a marginal increase above natural background levels.

The greater than 1 mSv but not more than 20 mSv band applies to circumstances where the individual receives a direct benefit from the exposure situation. Constraints and reference levels exist where there is individual dose monitoring/assessment. Exposure situations in this band can involve high levels of natural background radiation or stages in post-accident recovery.

The > 20 mSv – 100 mSv band applies to unusual or extreme exposure situations where actions to reduce exposure would be disruptive. Action to reduce exposures during a radiological emergency is an example of an event in this category. ICRP-103 suggests that doses approaching 100 mSv will usually justify protective action.

Dose Limits

Dose limits apply only in planned exposure situations. These limits are not applicable to medical exposure of patients. The ICRP concludes that the level of protection provided by the ICRP-60 dose limits continue to provide an appropriate level of protection. ICRP-103 dose limits in planned exposure situations are summarized in Table IV-12.

The planned dose limit for pregnant women for the remainder of the pregnancy was changed from the ICRP-60 approach. ICRP-60 limited the effective dose to 2 mSv to the surface of the abdomen or 1 mSv from the intake of radionuclides. ICRP-103's pregnancy limit is based on limiting the embryo/fetal dose to 1 mSv.

Planned dose limits do not apply in situations where an informed, exposed person is involved in volunteer life-saving actions or in the prevention of a catastrophic event. Emergency exposure situations are summarized in the next section.

Table IV.12 ICRP-103 Recommended Dose Limits in Planned Exposure Situations.[a]

Type of Limit	Occupational	Public
Effective Dose	20 mSv/y, averaged over defined periods of 5 years[e]	1 mSv in a year[f]
Annual Equivalent Dose:		
Lens of the eye[b]	150 mSv	15 mSv
Skin[c,d]	500 mSv	50 mSv
Hands and feet	500 mSv	–

[a] Limits on effective dose are for the sum of the relevant effective doses from external exposure in the specified time period, and the committed effective dose from intakes of radionuclides in the same period. For adults, the committed effective dose is computed for a 50-year period after intake. For children, it is calculated for the period up to age 70.
[b] This limit is currently under review by an ICRP Task Group.
[c] The limitation on effective dose provides sufficient protection for the skin against stochastic effects.
[d] Averaged over 1 cm^2 of skin regardless of the area exposed.
[e] With the provision that the effective dose does not exceed 50 mSv in any year. Additional restrictions apply to the occupational exposure of pregnant women.
[f] In special circumstances, a higher value of effective dose could be allowed in a single year. However, the average over 5 years is limited to 1 mSv/y.

Source: ICRP-103.

Comparison of Radiological Protection Criteria

The ICRP-103 recommended values for protection criteria are compared in Tables IV.13–IV.15 with those provided in ICRP-60 and its derivative publications. In general, the current recommendations encompass the previous values but are broader in scope. The reader should note that the new recommendations may cite values in terms of different quantities. For example, in emergency exposure situations the criteria in ICRP-60 are specified in terms of averted dose (intervention levels), but the ICRP-103 recommendations are defined in terms of incremental dose (reference levels).

The ICRP emphasizes the need for optimization and justification of protection strategies for application during an emergency. Optimization is influenced by the reference levels. Protective actions and dose evaluations are part of the optimization process.

The dose that is expected to occur from the emergency event, should no protective actions be utilized, is called the projected dose. Residual dose is the dose that results following the implementation of a protection strategy. Each protective measure eliminates or avoids a certain dose, which is called the averted dose. As noted in Table IV.13, optimization of protective measures that comprise the overall protection strategy is a dynamic process that includes a number of considerations.

Table IV.13 Emergency Exposure Situations.

Categories of Exposure (ICRP Publication No.)	ICRP-60 and Subsequent Publications	ICRP-103
	Intervention Levels[a,b,c]	Reference Levels[a,c]
Occupational exposure (60, 96)		
Life-saving (informed volunteers)	No dose restrictions[d]	No dose restrictions if benefit to others outweighs the rescuer's risk[e]
Other urgent rescue operations	~500 mSv; ~5 Sv (skin)[d]	1000 or 500 mSv[e]
Other rescue operations	Not provided	≤ 100 mSv[e]
Public exposure (63, 96)		
Foodstuffs	10 mSv/y[f]	–
Distribution of stable iodine	50–500 mSv (thyroid)[f,g]	–
Sheltering	5–50 mSv in 2 days[f]	–
Temporary evacuation	50–500 mSv in 1 week[f]	–
Permanent relocation	100 mSv first year or 1000 mSv[f]	–
All countermeasures combined in an overall protection strategy	–	In planning, typically 20–100 mSv/y according to the situation.

[a] Effective dose unless otherwise specified.
[b] Averted dose.
[c] Intervention Levels refer to averted dose for specific countermeasures, and remain valuable for optimization of individual countermeasures when planning a protection strategy. As a supplement to Reference Levels for evaluation of protection strategies, these levels refer to residual dose.
[d] ICRP-60.
[e] ICRP-96. Effective doses below 1000 mSv should avoid serious deterministic effects, and effective doses below 500 mSv should avoid other deterministic effects.
[f] ICRP-63.
[g] Equivalent dose.
Source: ICRP-103.

Existing exposure situations include ambient radon and environmental source terms. Table IV.14 provides action levels and reference levels for radon. Naturally Occurring Radioactive Material (NORM) reference levels are provided in Table IV.15.

Table IV.14 Existing Exposure Situations – Radon.

Categories of Exposure (ICRP Publication No.)	ICRP-60 and Subsequent Publications	ICRP-103
	Action Levels[a]	Reference Levels[a,b]
Radon (65)		
Home	3–10 mSv/y (200 – 600 Bq/m^3)	< 10 mSv/y (< 600 Bq/m^3)
Work	3–10 mSv/y (500 – 1500 Bq/m^3)	< 10 mSv/y (< 1500 Bq/m^3)

[a] Effective dose unless otherwise specified.
[b] Reference Levels refer to residual dose and are used to evaluate protection strategies. The previously recommended Intervention Levels referred to averted doses from individual protective actions.
Source: ICRP-103.

Table IV.15 Existing Exposure Situations – Naturally Occurring Radioactive Material.

Categories of Exposure (ICRP Publication No.)	ICRP-60 and Subsequent Publications	ICRP-103
	Generic Reference Levels	Reference Levels[a,b]
NORM, natural background radiation and radioactive residues in human habitat (82)		
Interventions:		
Unlikely to be justifiable	< ~ 10 mSv/y	
May be justifiable	> ~ 10 mSv/y	Between 1 and 20 mSv/y according to the situation
Almost always justifiable	towards 100 mSv/y	

[a] Additional restrictions apply to the occupational exposure of pregnant women. For the intake of radionuclides, the committed effective dose is the relevant dosimetric quantity.
[b] Reference Levels refer to residual dose and are used to evaluate protection strategies. The previously recommended Intervention Levels referred to averted doses from individual protective actions.
Source: ICRP-103.

Protection of the Environment

Previous ICRP environmental policy was based on the premise that protection of the general public would ensure that other species were not placed at risk, and the Commission continues to believe this is valid. However, the ICRP now believes that it is necessary to consider a wider range of environmental situations irrespective of any human connection.

Human radiological protection has been facilitated by the creation of anatomical and physiological reference models. The ICRP concludes that a similar approach would be useful in developing recommendations for the protection of other species. In this spirit, the ICRP is developing a small set of reference animals and plants and their relevant databases for species that are typical of the major environments. This development is not yet complete, and was initially addressed in ICRP-91. However, the ICRP does not propose to recommend dose limits for environmental protection. The intention is to compare environmental radiation doses to doses having known or expected biological effects.

Human Respiratory Tract Model (HRTM)

In 1994, the ICRP recommended an improved model for aerosol deposition in the respiratory tract, particle clearance behavior, dose calculations, and the absorption of materials from different regions of the respiratory tract into the blood. ICRP-66 extended the ICRP-30 dosimetric model to include both workers and members of the public. Characteristic breathing rates for various groups of people (males, females, children, and adults) and ages were also defined in ICRP-66. ICRP-30 was based on the ICRP-23 Reference Man Philosophy.

Table IV.16 Respiratory Tract Region Comparison.

Formulation		Anatomy Modeled
ICRP-30	**ICRP-66**	
Nasopharyngeal Region	Extrathoracic Region (ET)	
	ET_1	Anterior nose
	ET_2	Posterior nasal passages Larynx, Pharynx, Mouth
Tracheobronchial Region	Bronchial Region (BB)	Trachea Bronchi
	Bronchiolar Region (bb)	Bronchioles Terminal bronchioles
Pulmonary Region	Alveolar-Interstitial Region (AI)	Respiratory bronchioles Alveolar ducts and sacs Alveoli Interstitial connective tissue
Lymph	[a]	

[a] All ICRP-66 regions contain lymphatic tissue or components of it.
Fluid collected in the interstitial connective tissue is collected in lymph capillaries, from which it flows and drains via lymph nodes (LN):
LN_{ET} drains the extrathoracic region; and
LN_{TH} is located in the BB region and drains the BB, bb, and AI regions.

Source: ICRP-30 and ICRP-66.

ICRP-66 further subdivides the ICRP-30 regions and models the lymph nodes as connections to each region. A summary of the modeled regions is provided in Table IV.16.

Absorption

The ICRP-66 respiratory tract model includes absorption rates that vary with time following deposition in the lung. If experimental data are available, specific absorption behavior is used to obtain the most realistic assessment of systemic uptake.

For the cases in which only a generic knowledge of the material is available, the ICRP adopts a methodology that is similar to the Class D, W, and Y assumptions used in ICRP-30. ICRP-66 adopts the use of three types of absorption (Types F, M, and S).

For highly soluble material or material that is fast-absorbed (Type F), it is assumed that the absorption from each region of the respiratory tract occurs at the constant rate of 100/d, except for materials deposited in the nostrils, which are lined by impervious skin. Type F absorption behavior corresponds to ICRP-30's Class D.

For materials that are only moderately soluble (Type M), ICRP-66 recommends a more complex absorption characterization which is equivalent to approximately 10% of the deposited material absorbed at a rate of 100/d and the remaining 90% absorbed at a rate of 0.005/d. This solubility lies roughly between the ICRP-30 Class W and Class Y absorption.

The most slowly absorbed materials are classified as Type S. Type S materials are relative insoluble in the respiratory tract (slow solubilization). For insoluble materials such as actinide oxides, the absorption rate is typically much slower. Fuel cycle materials, such as UO_2 or PuO_2, are represented with an assumed 0.1% absorbed at a rate of 100/d and 99.9% absorbed at 0.0001/d.

Particle Sizes

ICRP-66 extended the ICRP-30 range of particle size (0.2 μm to 10 μm) to particles of atomic dimensions (0.6 nm diameter) through aerosols of course particles (100 μm AMAD). The default occupational particle size in ICRP-66 is 5 μm as compared with the 1 μm size in ICRP-30. ICRP-66 more realistically treats the effects of particle density, particle shape, and their absorption from the different regions of the respiratory tract.

Dose Assessments

Internal dosimetry using the ICRP-30 methodology is performed via hand-calculations and a variety of computer codes. The methodology is suited to hand calculations once the reference parameters and quantities are provided. The number of parameters required to perform a calculation is relatively small.

In ICRP-66, the deposition of an aerosol in the regions of the respiratory tract is modeled by treating the various lung regions as a series of filters. Each region has a deposition efficiency, which depends on both the aerodynamic and thermodynamic parameters of the particle. Algebraic expressions for deposition efficiency are provided in ICRP-66.

An ICRP-66 HRTM calculation requires specification of the intake and particle properties. The calculation inputs include (1) the fractional deposition in each region of the respiratory tract (specified directly or calculated from specific particle size and physiological parameters), (2) the transport of deposited particles from one region of the respiratory tract to another, (3) the rates at which the particles are cleared from the respiratory tract, (4) the rates of particle dissolution and subsequent uptake to the blood, (5) radionuclide data derived from the ICRP-38 nuclear decay properties, and (6) biokinetic information specified directly or from the models defined in the ICRP-30 metabolic descriptions. Although hand calculations can be performed using ALI and DAC values, more complex ICRP-66 HRTM calculations (e.g., particle size and shape sensitivity studies) are most easily implemented through codes such as the Lung Dose Evaluation Program (LUDEP) published by the National Radiation Protection Board (U.K.). Additional comparisons between ICRP-30 and ICRP-66 are provided in Table IV.17.

Table IV.17 Comparison of the ICRP-30 and ICRP-66 Lung Models.

Model Parameter/Quantity	ICRP-30	ICRP-66
Biokinetic model basis	Reference Man (ICRP-23)	Age-specific models (ICRP-89)
Dose coefficients availability	Reference Man only	Age-specific dose coefficients
Dosimetry capability	Occupational only	Occupational and environmental
Clearance	Pulmonary clearance (D, W, and Y)	Blood absorption (Type F, M, and S) Translocation from initial to transformed state (Type F, M, and S)
Physiology basis	Reference Man (ICRP-23)	Updated physiology/ transport models (ICRP-89)
Dose recommendation basis	ICRP-26	ICRP-103
Default occupational particle size	1 μm	5 μm
Particle size range	0.2–10 μm particles	0.6 nm – 100 μm particles
Particle shapes allowed	Spherical only	User-defined particle shapes and sizes

Additional Model Details

The ICRP-66 HRTM model contains considerable flexibility in terms of the character and shape of the inhaled aerosol, transformation of aerosol characteristics once it is deposited in the lung, and in the translocation of material within the lung. In the ICRP-66 model, a radioactive aerosol is deposited in the respiratory tract into the Particles in Initial State (PIS) subsystem and into the ET_1 region. The ICRP-66 model assumes that each compartment in the PIS subsystem has a mirror compartment in the Particles in Transformed State (PTS) subsystem. The PTS subsystem permits a particle entering the lung with a given type classification (e.g., Type M) to be altered by interaction with the lung environment and be transformed into another type (e.g., Type F or S). The transformed aerosol would behave differently in the PTS subsystem than how it did in the PIS subsystem.

There are 13 specific compartments in the PIS and PTS subsystems. These compartments are labeled as: 1 (AI_1), 2 (AI_2), 3 (AI_3), 4 (bb_1), 5 (bb_2), 6 (bb_{seq}; seq = sequestered), 7(BB_1), 8 (BB_2), 9 (BB_{seq}), 10 (LN_{TH}), 11 (ET_2), 12 (ET_{seq}), and 13 (LN_{ET}).

The PIS subsystem also includes the ET_1 compartment that connects to the environment. Both PIS and PTS Compartment 11 (ET_2) connect to the GI Tract. The model allows the following transport pathways:

$AI_1 \rightarrow bb_1$
$AI_2 \rightarrow bb_1$
$AI_3 \rightarrow bb_1$
$AI_3 \rightarrow LN_{TH}$
$bb_1 \rightarrow BB_1$
$BB_1 \rightarrow ET_2$
$ET_2 \rightarrow$ GI tract
$ET_1 \rightarrow$ Environment (PIS only)
$bb_2 \rightarrow BB_1$
$BB_2 \rightarrow ET_2$
$BB_{seq} \rightarrow LN_{TH}$
$bb_{seq} \rightarrow LN_{TH}$
$ET_{seq} \rightarrow LN_{ET}$
PIS $\rightarrow$ Body fluids
PTS $\rightarrow$ Body fluids
PIS $\rightarrow$ PTS

From each PIS compartment the material is transferred into the body fluids at an absorption rate s_p. As warranted, it is also simultaneously transferred from the PIS block at a rate s_{pt} to the corresponding PTS compartment (i.e., the material is transferred from PIS compartment i to PTS compartment i where $i = 1$ to 13). Each compartment in PIS has a counterpart in PTS. In each PTS compartment, the isotope is transferred at a constant rate s_t into the body fluids.

As an example, the total transfer rate (K) for Compartment 2 (AI_2) in PIS is:

$$K(AI_2 \text{ in PIS}) = k_{PiS}(2,4) + s_{pt} + s_p \quad \text{(IV.72)}$$

where

$k_{PIS}(2,4)$ = transfer rate from PIS Compartment 2 (AI_2) to PIS Compartment 4 (bb_1)
s_p = transfer rate from PIS to the body fluids
s_{pt} = transfer rate from PIS to PTS

The total transfer rate for Compartment 2 (AI_2) in PTS is:

$$K(AI_2 \text{ in PTS}) = k_{PTS}(2,4) + s_t \quad \text{(IV.73)}$$

where

$k_{PTS}(2,4)$ = transfer rate from PTS Compartment 2 to PTS Compartment 4
s_t = transfer rate from PTS to the body fluids

The absorption rates (s_{pt}, s_p, and s_t) are determined by the chemical form of the inhaled radioactive isotope.

HRTM Revisions

Revisions to the HRTM are under consideration. These possible revisions include simplification of the treatment of vapors; changes to the treatment of particle transport in the nasal passages and pharynx, the bronchial airways, and the alveolar region of the lungs; and revision of the default assumptions for absorption to blood for Types F, M, and S materials. Additional revisions of the HRTM are likely as the model is refined and compared with additional data.

Human Alimentary Tract Model (HATM)

ICRP-100 presented a revised dosimetric model for the human alimentary tract to replace the ICRP-30 Ingestion Model. The model is fully consistent with the anatomical and physiological data given in ICRP-89. ICRP-100 provides examples of radionuclide behavior and doses to alimentary tract regions. Comprehensive calculations of ingestion-dose coefficients using the HATM model will be given in subsequent ICRP publications.

The revised HATM model consists of the following regions: mouth, esophagus, stomach, small intestine, right colon, left colon, and rectosigmoid. These sections are modeled as a hollow tube, extending from the pharynx to the anus, and are comprised of four concentric layers. From the lumen outward, these layers are the mucosa, submucosa, muscularis, and adventitia or serosa.

The HATM includes the following processes:

1. Entry of a radionuclide into the oral cavity by ingestion or into the esophagus following mechanical clearance from the respiratory tract. After entering the oral cavity sequential transfer occurs through the esophagus, stomach, small intestine, and segments of the colon, followed by elimination as feces.
2. Radionuclide deposition and retention on or between the teeth and return to the oral cavity.
3. Deposition and retention in the oral mucosa or walls of the stomach and intestines.
4. Transfer from the oral mucosa or walls of the stomach and intestines back into the lumenal contents or into blood (absorption).
5. Transfer from various secretory organs or blood into the contents of certain segments of the alimentary tract (secretion).

First-order kinetic processes are assumed as the basis for the HATM. Although this is a considerable simplification of the complex processes involved in the transfer of material through the lumen of the alimentary tract, it is judged by the ICRP to provide a reasonably accurate representation of the mean residence time of a radionuclide in each segment of the tract.

For computational purposes, each parameter value of the model is represented by a transfer coefficient that describes the rate of outflow of a substance from a compartment. The transfer coefficient or rate constant is defined as the instantaneous fraction of the contained substance leaving the compartment per unit time.

A comparison of the ICRP-30 and ICRP-100 ingestion models is provided in Table IV.18. Transfer coefficients for the movement of HATM contents between regions are summarized in Table IV.19. The degree of precision of the values given in Table IV.19 is for computational purposes, and does reflect the accuracy of the value.

Absorption to Blood

Retention and absorption to blood is permitted in all regions except the esophagus, but the possibility of lymphatic transport is excluded. For most radionuclides, absorption occurs in the small intestine, and limited or no information is available on retention in alimentary tract tissues or absorption into the blood from other HATM regions.

While absorption occurs predominantly in the small intestine, the HATM includes absorption in the oral cavity, stomach, or the three segments of the colon. Absorption from the oral cavity is modeled as the transfer from the oral cavity contents to the oral mucosa and then direct transfer from the oral mucosa to blood. Absorption from other segments of the HATM is modeled as the transfer from the contents to the wall of that segment, followed by transfer to blood in

Table IV.18 Comparison of the ICRP-30 and ICRP-100 Ingestion Models.

Model Parameter/Quantity	ICRP-30	ICRP-100
Absorption of radionuclides into blood	Occurs only in the small intestine	Absorption occurs in all compartments of the HATM with the exception of the esophagus, but the possibility of lymphatic transport is excluded.
Applicability	Occupational doses	Occupational and environmental doses
Biokinetic model basis	Reference Man (ICRP-23) only	Provides age-dependent parameter values based on ICRP-89 for the alimentary tract regions and associated transit times for the movement of materials through these regions. For adults, gender-dependent parameter values are given for dimensions and transit times.
Dose coefficient availability	Reference Man only	Age and gender-specific dose coefficients
Dose recommendation basis	ICRP-26	ICRP-103
Dosimetry calculations	Doses are calculated to the walls and contents of the modeled tissues	Doses are calculated to targets for cancer induction (stem cells) in the oral cavity, esophagus, stomach, small intestine, right colon, left colon, and rectosigmoid. ICRP-103 tissue weighting factors are included for the stomach, colon, and esophagus. The small intestine and oral mucosa are included in the remainder.
Point of entry into alimentary tract	Stomach	Oral cavity, esophagus, and stomach
Transit times	Independent of age, gender, and the type of material ingested	Provides age- and gender-specific transit times for all segments of the HATM. For the upper segments of the HATM (oral cavity, esophagus, and stomach) provides material-specific transit times.

Source: ICRP-100.

the portal vein with the potential for direct uptake by the liver prior to entry into the general circulation. The model does not include the slow transfer from the alimentary tract to the blood via the lymphatic system, which is not expected to significantly contribute to the total absorption.

Table IV.19 Transfer Coefficients (per day) for the Movement of Alimentary Tract Contents between Regions.

Region/Component	Age 3 months	Age 1 year	Age 5–15 years	Adult male	Adult female
Mouth					
Solids	–	5760	5760	5760	5760
Liquids	43 200	43 200	43 200	43 200	43 200
Total Diet	43 200	7200	7200	7200	7200
Esophagus (fast)					
Solids	–	10 800	10 800	10 800	10 800
Liquids	21 600	17 280	17 280	17 280	17 280
Total Diet	21 600	12 343	12 343	12 343	12 343
Esophagus (slow)					
Solids	–	1920	1920	1920	1920
Liquids	2880	2880	2880	2880	2880
Total Diet	2880	2160	2160	2160	2160
Stomach					
Solids	–	19.2	19.2	19.2	13.71
Caloric Liquids	19.2	32	32	32	24
Non-caloric Liquids	144	48	48	48	48
Total Diet	19.2	20.57	20.57	20.57	15.16
Small Intestine					
Small Intestine	6	6	6	6	6
Right Colon					
Right Colon	3	2.4	2.182	2	1.5
Left Colon					
Left Colon	3	2.4	2.182	2	1.5
Rectosigmoid					
Rectosigmoid	2	2	2	2	1.5

Source: ICRP-100.

In forthcoming ICRP reports supporting the use of the HATM, information for each element will be given in terms of fractional absorption (f_A). These values replace the f_1 values of ICRP-30. The f_A value defines the total absorption to blood in the HATM and represents the fraction of the material entering the alimentary tract that is absorbed in the absence of radioactive decay or endogenous input to the tract. It is defined by the sum of the fractions of the material entering the alimentary tract (f_i) absorbed in all of the regions (i) of the alimentary tract:

$$f_A = \sum_i f_i \tag{IV.74}$$

In most situations, data is only available on the total absorption of an element to the blood with no information on regional absorption, and the ICRP-100 model assumes that this absorption takes place entirely from the small intestine (SI). In this case, the fractional transfer from the small intestine to blood, f_{SI}, is equal to f_A. If an element is absorbed from the stomach (ST) as well as from the small intestine (SI), f_A is:

$$f_A = f_{ST} + f_{SI} \tag{IV.75}$$

In order to perform calculations, transfer coefficients describing the uptake from the HATM to the blood are required. In the absence of retention in the walls, teeth and oral mucosa, the transfer coefficient ($\lambda_{i,B}$), describing uptake to the blood (B) from compartment i of the HATM is:

$$\lambda_{i,B} = \frac{f_i \, \lambda_{i,i+1}}{1 - \sum_i f_i} \tag{IV.76}$$

where

f_i = fraction of the intake assumed to be absorbed from compartment i

$\lambda_{i,\,i+1}$ = HATM transfer coefficient for movement of material from compartment i to compartment $i + 1$ of the alimentary tract

For the most common situation involving uptake only from the small intestine, the transfer coefficient for uptake from the small intestine to blood ($\lambda_{SI,B}$) is:

$$\lambda_{SI,B} = \frac{f_{SI} \, \lambda_{SI,RC}}{1 - f_{SI}} \tag{IV.77}$$

where

$\lambda_{SI,RC}$ = coefficient for transfer from the small intestine to the right colon

A second example is more complex with blood uptake occurring from both the small intestine and stomach (S). In this case, the transfer coefficients to blood are:

$$\lambda_{ST,B} = \frac{f_{ST}\,\lambda_{ST,SI}}{1 - f_{ST}} \tag{IV.78}$$

$$\lambda_{SI,B} = \frac{f_{SI}\,\lambda_{SI,RC}}{1 - f_{SI} - f_{ST}} \tag{IV.79}$$

Similar expressions, based on (IV.76), can be written for uptake from other compartments of the HATM.

Dose Calculations

The HATM dosimetry approach is to evaluate the dose to target regions within the mucosal lining of the alimentary tract. Epithelial stem cells are usually the targets for cancer induction. Applications of the HATM will be published in a subsequent ICRP report on dose assessment and monitoring for occupational intakes of radionuclides. It is the intent of the ICRP to replace ICRP-30, -54, -68, and -78 by a series of publications that provide dose coefficients for radionuclide inhalation and ingestion. Additional specificity of the ICRP-103/66/100 approach to dose calculations must await forthcoming publications.

Calculation of Organ Doses from Intakes of Radioactive Material

A common practical problem in health physics is the estimate of a worker's intake. This is often required for conditions in which an unanticipated intake occurred. The following relationships are not tied to specific ICRP recommendations, but are useful methods to assess intakes resulting from abnormal situations.

Normally, organ doses are assessed using bioassay or whole-body counting. However, there are times when a quick assessment is required and only air sample data are available. When bioassay or whole-body counting cannot be performed in a timely manner, an estimate of the intake can be obtained by knowing the average air concentration to which a worker was exposed. Knowledge of the intake is needed for the assessment of organ dose. The intake can be determined once the exposure time, air concentration, and worker breathing rate are determined:

$$I = k\,t(\mathrm{BR})\,C \tag{IV.80}$$

where

I = intake (μCi)
t = exposure time (s)
BR = breathing rate of individual exposed (liters/min)
C = average air concentration (μCi/ml)
k = (1 min/60 s) (1000 ml/liter)

An alternate method of calculating the intake is based upon the use of lapel air samplers to determine the air concentration in the breathing tone of the worker. The activity on the air sampler's filter paper (A) may be obtained from the filter paper count rate (CR) and counter efficiency (e):

$$A = (\mathrm{CR})/e\,k \tag{IV.81}$$

where
A = activity on the filter (μCi)
CR = count rate from the lapel air sampler filter (cpm)
e = counting efficiency (cpm/dpm)
k = conversion factor (2.22×10^6 dpm/μCi)

The average concentration inhaled by the worker $(\overline{C})$ is determined from the breathing Zone filter's activity:

$$\overline{C} = A/f\,t \tag{IV.82}$$

where
$\overline{C}$ = average concentration inhaled (μCi/liter)
f = lapel air sampler flow rate (liters/min)
t = exposure duration (min)

The intake may be defined in terms of these quantities

$$I = (\mathrm{CR})\,(\mathrm{BR})/e\,k\,f\,(\mathrm{DF}) \tag{IV.83}$$

$$= \overline{C}\,t(\mathrm{BR})/\mathrm{DF} \tag{IV.84}$$

where
I = intake (μCi)
BR = worker's breathing rate (liters/min)
DF = protection factor of the worker's respirator

The calculation of organ doses follows from the intake:

$$D = I \sum_i f_i\,(\mathrm{DCF}_i) \tag{IV.85}$$

where
D = organ dose equivalent (rem)
i = number of isotopes in the air mixture
I = intake (μCi)
f_i = activity fraction of nuclide i in the inhaled air which can contain multiple radionuclides
DCF_i = dose conversion factor (rem/μCi intake)

Specific examples of the use of these relationships are presented in Chapter 4.

Releases

Unanticipated uptakes also occur when radioactive material is released into an area which may be followed by a release to the environment. Releases to a room may occur in both ventilated and unventilated areas. Releases of this type include rupture of a glove box with the subsequent release of its contents to a laboratory or work area, ruptures of process lines or vessels, or failures of glove box or process ventilation systems. These types of events can occur in any area that contains radioactive material.

Release of Activity into a Room without Ventilation

Assuming an instantaneous, uniform distribution of the activity into the room's volume and no dilution by the ventilation system, the dose to an individual in the room is expressed as

$$D = \overline{C}(\mathrm{BR})\, t(\mathrm{DCF}) \tag{IV.86}$$

where

D = dose to the individual in the room (rem)
$\overline{C}$ = average concentration in the room ($\mathrm{Ci/m^3}$) = fA/V
V = room volume ($\mathrm{m^3}$)
A = total activity that could be released (Ci)
(It is assumed that the half-life of the radionuclide is much longer than the individual's residence time in the room.)
f = fraction of activity that is released and can be inhaled by an individual in the room
BR = breathing rate of the worker ($\mathrm{m^3/s}$)
t = exposure time
DCF = dose conversion factor (rem/Ci inhaled)

The maximum dose that could be delivered to a person outside the room can be calculated by assuming that all activity is released to the environment in a short time and that the person stays at a given location for the entire release. The location is assumed to be an the plume centerline. With these conditions, the dose delivered to the individual is expressed as

$$D = f\, A(X/Q)\,(\mathrm{BR})\,(\mathrm{DCF}) \tag{IV.87}$$

where

D = dose to the individual outside the room (rem)
X/Q = atmospheric diffusion factor at the receptor ($\mathrm{s/m^3}$)
A = total activity that could be released (Ci)
f = fraction of activity that is released to the environment
BR = breathing rate of the worker ($\mathrm{m^3/s}$)
DCF = dose conversion factor (rem/Ci inhaled)

Release of Activity into a Room with Ventilation

A ventilation system's design criteria specifies the number of complete air changes per hour in the room. If the ventilation system works as designed, the activity in the room will be reduced as a result of the air changes.

This situation is similar to the previous example. The major difference is that the ventilation system reduces the concentration of the room air. Following the release, the worker's dose is still given by (IV.86).

Assuming a uniform distribution of the radionuclide and exponential removal by the ventilation system, the concentration of activity in the room's air $C(t)$ as a function of time is

$$C(t) = C_0 \exp(-r\,t) \tag{IV.88}$$

where

C_0 = airborne concentration assuming uniform mixing with the room air
r = ventilation system removal rate (hr^{-1})

Again, the assumption is made that the half-life is much longer than the duration of the accident. If this were not the case, an effective half-life (air removal + physical decay) would be employed in the calculations. Chapter 7 illustrates the use of effective half-lives and clearance of radionuclides by room air changes.

The average concentration $(\overline{C})$ in the room during the time of exposure is obtained by integrating $C(t)$ from time = 0 to t and by dividing by the associated time interval:

$$\overline{C} = C_0\,[1 - \exp(-r\,t)]/r\,t \tag{IV.89}$$

The dose is readily obtained from (IV.86) once the average air concentration is known.

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Appendix V
Radiation Risk and Risk Models

Previous material focused upon the various approaches used to calculate dose. Although dose is an important quantity, it is becoming more important to relate the long-term effects of receiving a radiation exposure to its impact upon the body. These impacts usually include the probability of the radiation exposure inducing cancer or death. The concept of risk via risk coefficients has been introduced to assess these effects and to compare the risk of radiation exposure to risks in other commercial and industrial activities. This appendix will outline the concept of risk and the various risk models.

Risk

Radiation is one of the most thoroughly studied agents associated with biological impacts. These impacts are quantified in terms of stochastic and nonstochastic effects and their associated health risks. The risk is often quantified in terms of a risk coefficient expressing excess-radiation-induced risk per unit radiation dose. Accordingly, the risk of the radiation exposure is often determined from the relationship

$$\text{Risk} = (\text{Risk Coefficient})\ (\text{Dose}) \tag{V.I}$$

where the dose is the radiation exposure received and under evaluation. The risk coefficient varies depending upon the data under evaluation and the underlying modeling assumptions. Risk estimates are also influenced by the radiation characteristics (dose, dose rate, fractionalization, and radiation quality), biological characteristics (age, sex, genetic background, and nature of the tissue or organ), and the approach to the analysis (dose-response model, projection model, and risk model).

In view of these factors, it is not surprising that there is considerable variance in risk estimates. For example, the ICRP-26 risk coefficient is 2×10^{-4} excess cancer deaths/rem, while BEIR V with its 8×10^{-4} coefficient yields a larger characterization of the risk. A summary of risk coefficients derived from major studies is provided in Table V.1.

Equation (V.I) is often applied carelessly. This equation is only valid for a large ensemble of subjects (10 000–100 000) who have each received at least 10 rem of acute radiation exposure.

The total risk coefficient (r) is the sum of the risk coefficients for the organs or tissues (T) composing the modeled human body:

$$r = \sum_{T} r_T \tag{V.2}$$

Table IV.7 summarizes the various organs that are assumed in the ICRP-26, UNSCEAR 88, ICRP-60, and ICRP-103 formulations. The formulations do not contain the same organs or level of organ risk. This table also provides the values of the organ weighting factors (w_T) for these models:

$$w_T = r_T / r \tag{V.3}$$

where the weighting factor is a dimensionless number with a value between zero and unity.

The value of Tables V.1 and IV.7 illustrate the wide modeling variations encountered in the risk estimates. The ICRP-60 and -103 models include more organs with specified organ weighting factors than the ICRP-26 formulation and also include a set of specified organs to be included in the remainder. An examination of Table IV.7 illustrates the model dependence (number of organs and assigned weighting factors) of the various ICRP internal dosimetry formulations. For example, the tissue weighting factor for the gonads changed significantly in the ICRP-26 (0.25), ICRP-60 (0.20), and ICRP-103 (0.08) formulations.

Table V.1 Ionizing Radiation Risk Coefficient Summary.

Year	Report	Risk Coefficient ($\times 1.0 \times 10^{-4}$) (Radiation Induced Effects/rem)
1972	BEIR I	1
1976	ICRP-26	2
1980	BEIR III	2
1985	EPA NESHAP	4
1988	NRC BRC Policy	5
1990	BEIR V	8
1991	ICRP-60	7

Basic Epidemiology

Studies of radiation risk utilize epidemiological input which requires a sample size dependent on the magnitude of the radiation exposure. The sample size, required for statistically meaningful results, is 5×10^4, 5×10^8, and 5×10^{12} for acute exposure of 10 rad, 100 mrad, and 1 mrad, respectively. These values illustrate how the size of the required exposed group varies with the absorbed whole-body dose. Meaningful results are possible for larger exposures, but the population size required for typical occupational exposures is prohibitive.

The BEIR III Committee did not know whether dose rates of gamma or X-rays of about 100 mrad/year are detrimental to humans. Somatic effects at these doses would be masked by environmental or other factors that produce the same types of health effects as does the ionizing radiation. Clearly, assessments of the impact of doses on the order of magnitude of 100 mrad or less are not physically possible.

Epidemiological studies must also consider a number of factors including sex, age, time since exposure, and the age at exposure. They are also of long duration because of the time required to follow the exposed population and control group.

The number of cancers expected in a cohort (E) is given by the sum

$$E = \sum_x c(x)\, r(x) \tag{V.4}$$

where

$r(x)$ = annual incidence (morbidity) per person at age x per year
$c(x)$ = sum of all years spent by cohort members at age x

Once the expected incidence is determined, the number of excess cancers (EC) is readily obtained:

$$\mathrm{EC} = O - E \tag{V.5}$$

where

O = observed cancer incidence in the risk population

The excess cancers per population year per incident exposure (Z) is given by

$$Z = (O - E)/N \tag{V.6}$$

The quantity N has the units of population year-Gy:

$$N = d_i\, \gamma_i \tag{V.7}$$

where

d_i = dose to the i th group
γ_i = number of years the i th group is observed

Therefore, Z is expressed in excess cancers per population year-Gy.

With these definitions, the following terms commonly utilized in epidemiology can be defined: the relative risk (RR), standard mortality ratio (SMR), and excess relative risk (ERR):

$$\mathrm{RR} = O/E \tag{V.8}$$

$$\mathrm{SMR} = 100\,\mathrm{RR} \tag{V.9}$$

and

$$\mathrm{ERR} = \mathrm{RR} - 1 \tag{V.10}$$

Dose Response Relationships

Dose response relationships describe how an effect varies with dose. Currently, the two most popular dose response relationships are the linear and linear quadratic models. In the linear mode!, the effect increases linearly with dose and there is no threshold below which effects do not occur:

$$f(d) = a_1\, d \tag{V.11}$$

where
d = dose
f = effect (such as excess cancers) being observed
a_1 = constant

The BEIR V Report suggests that a linear relationship is applicable for all cancers except for leukemia.

BEIR V suggests that a linear quadratic model is most representative of the leukemia data:

$$f(d) = a_2\, d + a_3\, d^2 \tag{V.12}$$

Risk Models

There are two general types of risk models: the absolute risk model and the relative risk model. The absolute risk model assumes that the cancer risk from radiation exposure adds an increment to the natural incidence. In relative risk models the cancer risk from radiation exposure increases in an amount proportional to the natural incidence.

Age-specific models for cancer risk can be explicitly written as

$$r(d) = r_0\,[1 + f(d)\,g(B)] \qquad \text{(for relative risk)} \qquad \text{(V.13)}$$

or

$$r(d) = r_0 + f(d)\,g(B) \qquad \text{(for absolute risk)} \qquad \text{(V.14)}$$

where

$$F(d) = a_1\,d \qquad \text{(linear)} \qquad \text{(V.15)}$$

or

$$f(d) = a_2\,d + a_3\,d^2 \qquad \text{(linear-quadratic)} \qquad \text{(V.16)}$$

and $g(B)$ is the excess risk function. Table V.2 summarizes the BEIR V models used for a variety of cancer types.

As an example, consider the BEIR V Committee's preferred risk model for leukemia. BEIR V supports a relative risk model that is dependent upon the following factors: dose, dose squared, age at exposure, time after exposure, and interaction effects. There is an observed two-year latency period. According to BEIR V, the preferred leukemia model is

Table V.2 BEIR V Preferred Relative Risk Model.[a]

Cancer Type	Dose Response Model	Comments
Leukemia	Linear + quadratic	Minimum latency of 2 years.
Breast	Linear	Highest risk in women under age 15 at the time of exposure. Risk is low for women if exposed after age 40.
Respiratory	Linear	Minimum latency of 10 years. Risk decreases with time after exposure. Relative risk for females is twice that for males.
Digestive	Linear	About seven times the risk if exposure occurs at age 30 or less. Risk does not change with time post exposure.
Other	Linear	Contributes significantly to total risk. No age or sex effects have been noted. Insufficient data to permit detailed modeling.

[a] BEIR V does not support the absolute risk model.

$$f(d) = a_2\, d + a_3\, d^2 \tag{V.17}$$

$$g(B) = \exp[B_1\, I(T \le 15) + B_2\, I(15 < T \le 25); \quad E \le 20 \tag{V.18}$$

$$g(B) = \exp[B_3\, I(T \le 25) + B_4\, I(25 < T \le 30)]; \quad E > 20 \tag{V.19}$$

where
T = number of years after the exposure
E = age at the time of the exposure

The I factors have the following values:

$$I(T \le x) = \begin{cases} 0 & \text{for } T > x \\ 1 & \text{for } T \le x \end{cases} \tag{V.20}$$

For leukemia, $x = 15$ years.
The coefficients specifying the BEIR V leukemia model are

$$\begin{aligned} a_2 &= 0.243 \quad a_3 = 0.271 \\ B_1 &= 4.885 \quad B_2 = 2.380 \\ B_3 &= 2.367 \quad B_4 = 1.638 \end{aligned} \tag{V.21}$$

As noted in Table V.2, the BEIR V models have been applied to a variety of cancer types. An example of their application to leukemia and nonleukemia cancers will further illustrate the difficulties encountered by the BEIR III and BEIR V Committees in attempting to assess radiation risk.

BEIR III and BEIR V Comparisons

Leukemia is one of the more likely forms of radiation-induced cancer. Japanese atomic bomb survivors showed an increase in leukemia incidence about 2–3 years following exposure. The incidence increased until about 7 years post exposure, and then it decreased until the leukemia rate reached normal levels in surviving populations during the early 1970s.

While the age at exposure has an important impact on risk, there was no evidence to support earlier beliefs that those exposed before the age of 10 had a greater risk than those exposed between the ages of 10 and 20.

The variation in radiation-induced cancer risks is illustrated by a comparison of the BEIR III and BEIR V estimates. Table V.3 illustrates the variation for both leukemia and nonleukemia cancers. The nonleukemia cancers include respiratory, digestive, breast, and other cancer types. For leukemia, BEIR V leads to a factor of 4–5 greater risk. A similar increase of about 3–5 occurs for nonleukemia cancers if relative risk models are considered.

Table V.3 Lifetime Cancer Risk Estimates (Deaths/100 000 Persons).

Cancer Type	Continuous Lifetime Exposure 1 mGy/year (100 mrad/year)		Instantaneous Exposure 0.1 Gy (10 rad)	
	Male	Female	Male	Female
Leukemia				
BEIR III	15.9	12.1	27.4	18.6
BEIR V	70	60	110	80
BEIR V/BEIR III	4.4	5.0	4.0	4.3
Nonleukemia				
BEIR III (absolute)	24.6	42.4	42.1	66.5
BEIR III (relative)	92.9	118.5	192	213
BEIR V (relative)	450	540	660	730
BEIR V/BEIR III (relative)	4.8	4.6	3.4	3.4
BEIR V/BEIR III (absolute)	18.3	12.7	15.7	11.2

These factors are considerably larger (11–19) if the BEIR III absolute risk model is compared to BEIR V's relative risk model. Because recent data support the relative risk model, these larger factors do not appear to be reasonable.

BEIR VII

The BEIR VII Report is consistent with BEIR V. The key elements of BEIR VII and their comparison with BEIR III and BEIR V are summarized in Table V.4.

The BEIR VII total cancer mortality and leukemia risk estimates from radiation exposure have not changed significantly from BEIR V. BEIR VII's risk estimates are based on expanded epidemiological data including cancer incidence data and 15 years of additional mortality follow-up for atomic bomb survivors. Studies involving occupational and environmental exposure were also evaluated in the BEIR VII Report.

In formulating its risk models, the BEIR VII Report used the revised Dosimetry System 2002 (DS02) for atomic bomb survivors as a portion of the basis for evaluation of the dependence of risk on dose. The risk models were developed from atomic bomb survivors and persons exposed for medical reasons.

BEIR VII also reviewed the dose response model and its functional dependence, the emergence of hormesis as a positive consequence of the radiation dose, and the existence of a threshold for radiation-induced effects. According to BEIR VII, the updated molecular and cellular data from studies of radiation exposure do not support the postulate that low doses of low-LET radiation are more harmful than predicted by the LNT model. That is, the contention that the dose response curve

Table V.4 BEIR III, V, and VII Comparison.

Parameter/Quantity	BEIR III (1980)	BEIR V (1990)	BEIR VII (2006)
Dose Response Model – solid tumors	LQ[a]	L[a]	L
Dose Response Model – leukemia	LQ	LQ	LQ
Preferred Risk Model	Absolute	Relative	Various[c,d]
Dosimetry system[b]	T65D	DS86	DS02
DDREF[e] (range)	–	2–10	1.1–2.3
DDREF (adopted)	–	–	1.5 for Linear Models

[a] L = linear LQ = linear-quadratic.
[b] T65D = Tentative 1965 Dosimetry.
DS86 = Dosimetry System 1986.
DS02 = Dosimetry System 2002.
[c] For solid cancers other than lung, breast, and thyroid, the preferred risk model is a weighted average (on a logarithmic scale) of relative and absolute risk models with relative risk given a weight of 0.7 and absolute risk a weight of 0.3.
These weights are reversed for lung cancer.
The preferred breast cancer model is based on the absolute risk model.
The preferred thyroid cancer model is based on the relative risk model.
[d] For leukemia the preferred risk model is a weighted average (on a logarithmic scale) of relative and absolute risk models with relative risk given a weight of 0.7 and absolute risk a weight of 0.3.
[e] Dose and dose rate effectiveness factor.

exhibits supra-linearity is not supported. In addition, the updated molecular and cellular data from studies of radiation exposure do not support hormesis. BEIR VII re-affirms the LNT hypothesis and concludes that there is cellular level evidence for the LNT. Thresholds were considered, but not endorsed as representing the best scientific view of low-dose risk. NCRP-136 also supported the LNT approach.

BEIR VII also noted that a number of effects were observed to exist, but were small. In particular, BEIR VII concluded that the genetic risks of low-dose, low-LET radiation are very small compared to the baseline frequencies of genetic disease. In addition, a dose response for non-cancer mortality in atomic bomb survivors has been demonstrated, but data are not sufficient to determine if this effect exists at low doses and dose rates. BEIR VII does not provide risk estimates for non-cancer mortality.

Although BEIR VII does not provide an excess cancer risk coefficient, a public risk coefficient for all cancers can be developed from the report's data. The number of excess cancer deaths (ecd) from exposure to 0.1 Gy to males is 410 (200, 830) ecd in an exposed population of 100 000. The values in brackets are the 95% confidence intervals. For females, the number of excess deaths from exposure to 0.1 Gy is 610 (300, 1200) ecd in an exposed population of 100 000. Averaging over age and sex (410 ecd + 610 ecd)/2 produces a value of 510 ecd. This data can be used to obtain a corresponding risk coefficient:

$$r = \frac{\left(\frac{410\ \text{ecd} + 610\ \text{ecd}}{2}\right)}{(100\,000\ \text{persons})\,(0.1\ \text{Gy/person})} \frac{1\ \text{Gy}}{100\ \text{rad}} \frac{1\ \text{rad}}{1\ \text{rem}} \qquad \text{(V.22)}$$
$$= 5 \times 10^{-4}\ \frac{\text{ecd}}{\text{rem}}$$

As a comparison, BEIR V derived a value of 695 ecd/100 000 persons exposed to 0.1 Gy (no DDREF utilized). This is again averaged over males and females [(660 ecd + 730 ecd)/2 = 695 ecd]. If the BEIR VII DDREF is applied to the BEIR V data, 695 ecd/1.5 provides a value of 463 ecd. Using the methodology of (V.22) and keeping one significant figure, leads to a public, excess cancer death risk coefficient for BEIR V of 5×10^{-4} ecd/rem. Therefore, BEIR V, BEIR VII, and ICRP-60 have the same excess cancer risk coefficient of 5×10^{-4} ecd/rem. This calculation illustrates the consistency of these reports.

Reports such as BEIR VII are important because they refine the internal dosimetry models and impact the risk estimates. Consequently, conclusions of BEIR VII carry significant weight and ideally are clear, unambiguous, and widely accepted.

The BEIR VII Report's conclusion regarding the LNT hypothesis has been challenged by a number of professional organizations including two French Academies. The author views the LNT hypothesis as an expedient regulatory model, but feels that scientific evidence has not yet fully resolved this issue.

Doubling Dose

The qualitative relationship between radiation dose and the probability of a mutation is often described in terms of the doubling dose. The doubling dose is the radiation dose that would lead to a doubling of the mutation rate. Table V.5 summarizes the doubling dose from BEAR and from BEIR I, III, V, and VII.

Probability of Causation

Public Law 97-414, the Orphan Drug Act of 1984, directed the Secretary of Health and Human Services to construct radioepidemiological tables providing the probability that certain cancers could result from prior exposure to radiation. The probability of causation (PC) is defined as a number that represents the probability that a given cancer, in a specific tissue, has been caused by a previous exposure or series of exposures to a carcinogenic agent such as ionizing radiation.

The PC tables are based upon the BEIR III (1980) report and are influenced by NCRP-71, which established the groundwork for the PC concept for radiogenic tumors. The original PC tables are somewhat outdated because BEIR III was based on an absolute-risk model which has been superseded by BEIR V.

Table V.5 Doubling Dose.

Formulation	Doubling Dose (rem)
BEAR (1956)	5–100
BEIR I (1972)	20–100
BEIR III (1980)	50–250
BEIR V (1990)	< 100
BEIR VII (2006)	100

The PC has the form

$$\mathrm{PC} = R/(1 + R) \tag{V.23}$$

where
R = relative excess

In the case of a single exposure of short duration to an individual representative of the U.S. population, the relative excess is given by

$$R = F\,T\,K \tag{V.24}$$

In this equation, F is the exposure factor which characterizes the dependence of R on the radiation dose to the risk organ. The use of whole-body exposure from film or TLD packages is not appropriate because absorbed tissue dose in units of rad is the desired quantity. The appropriate value of F is defined as a function of the absorbed tissue dose (D), measured in rads. The factors T and K are defined in the subsequent discussion.

The specific functional form for F depends on the radiation quality and cancer site. For example, consideration of Ra-224 irradiating the bone and leading to bone cancer results in the simple relationship

$$F_{\mathrm{Bone}} = D \tag{V.25}$$

for high-LET alpha radiation. For low-LET radiation, the values of F for thyroid, breast, and other cancers are

$$F_{\mathrm{thyroid}} = D \tag{V.26}$$

$$F_{\mathrm{Breat}} = D \tag{V.27}$$

and

$$F_{\mathrm{other}} = D + (1/116)\,D^2 \tag{V.28}$$

The second factor (T) in the definition of relative excess represents the relative likelihood that a cancer induced at age A_1 will be diagnosed after Y years. For diagnosis times between Y and $Y + 1$ years, Y is utilized in the computation. Under the constant relative risk model, which is used for cancers other than leukemia and bone cancer, T depends only on Y and has a value that increases with Y. For $Y = 0$–4 years, $T = 0$ and it rises to a value of unity for $Y \geq 10$ years. T values of 0.25, 0.5, and 0.75 occur at about $Y = 6$, 7, and 8 years, respectively.

The constant relative risk model has not been assumed to hold for bone cancer and leukemia. For these two cancer types, T is a conditional probability which assumes that the cancer has been caused by an exposure at age A_1 and will be diagnosed Y years later. For these cases, T is calculated as the lognormal probability that a cancer is detected between years Y and $Y + 1$ after exposure at age A_1. The PC tables compile T for the various forms of cancer.

The final factor defining the relative excess is K, and it provides the dependence of R on age and baseline cancer incidence for persons of age A_2 and sex (S) for exposure at age A_1:

$$K = K(A_1, A_2, \text{and } S) \tag{V.29}$$

The reader has by now drawn the conclusion that the PC concept is not precise. A qualitative estimate of the uncertainties of the PC concept may be illustrated by a few examples. If the PC is calculated to be 2% or less, the true PC could be as large as 7% even if we have an accurate knowledge of all the input parameters. If the PC is within the 5–10% range, the true PC could lie within the 1–30% range. Finally, if the PC is calculated to be a least 20%, the true PC could be in the 5–40% range.

A final complication of the PC concept lies in its ties to the BEIR III methodology. The differences between BEIR III and BEIR V suggest that a review of the current PC approach and its underlying assumptions is in order.

Energy Employees Occupational Illness Compensation Program Act

The probability of causation concept has been revised, and it is now used as the basis for determining the legal standard for resolving radiation related claims associated with the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA) as outlined in 42CFR81, 42CFR82, and 42CFR83. EEOICPA models incorporate relevant epidemiology, BEIR reports, and ICRP reports available at the time of its enactment.

The methods for calculating internal dose from the intake of radioactive material use the ICRP-66 respiratory tract model and the ICRP-30 ingestion model. In addition, supporting radionuclide data (e.g., ICRP-56, -67, and -68) are utilized in the internal dose assessment. The EEOICPA permits calculation methods to be updated to reflect new reports and science as they become available.

This Act established that a lump-sum payment and medical benefits can be awarded as compensation to covered employees suffering from designated

illnesses (e.g., cancer resulting from radiation exposure) incurred as a result of their performance of duty for the Department of Energy (DOE) and designated contractors. Under EEOICPA, an employee seeking compensation for cancer is eligible if the cancer has a 50% or greater probability of being caused by radiation doses incurred in the performance of duty or the employee is included in a specified cohort.

The risk models address a number of cancer types and most types of radiation exposure are relevant to employees covered by the EEOICPA. These models include the employee's cancer type, year of birth, year of cancer diagnosis, exposure information, and the dose received from gamma radiation, X-rays, alpha radiation, beta radiation, and neutrons. In addition, the risk model for lung cancer includes the worker's smoking history, and the risk model for skin cancer incorporates race and ethnicity. None of the risk models explicitly includes exposure to other occupational, environmental, or dietary carcinogens. Models incorporating chemical agents have not yet been developed.

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Appendix VI
Key Health Physics Relationships

This appendix provides a summary of important ionizing radiation, and non-ionizing radiation relationships. These equations represent a set of key health physics relationships that are utilized throughout the text. Applications of these equations are provided in the Chapter 1–8 problems.

Within this appendix, the following notation is used:

A	=	Source activity
		Laser beam area
		Hot-particle activity
ALI	=	Annual limit on intake
ALI_{NS}	=	Non-stochastic annual limit on intake
ALI_S	=	Stochastic annual limit on intake
A_V	=	Activity per unit volume
$A(0)$	=	Initial activity
$\tilde{A}$	=	Cumulated activity
B	=	Magnetic Induction
		Buildup factor
C	=	Capacitance
C_a	=	Activity per unit area
C_v	=	Activity per unit volume
CDE	=	Committed dose equivalent
CEDE	=	Committed effective dose equivalent
D	=	Absorbed dose
		Displacement current
		Divergence angle
$\dot{D}_o$	=	Initial absorbed dose rate
$\overline{D}$	=	Mean organ dose
DAC	=	Derived air concentration
DF	=	Duty factor
		Hot-particle dose factor
DRCF	=	Dose rate conversion factor
$D_{T,R}$	=	Average absorbed dose in tissue T due to radiation of type R

E	=	Electric field
		Electric field strength
		Effective dose
		Energy
		Energy/disintegration deposited in an organ/tissue
		Irradiance
$\dot{E}$	=	Effective dose rate
E_o	=	Rest energy
$E(50)$	=	50 year committed effective dose
F	=	Force
GB	=	Gaussian beam
H	=	Effective dose (used to avoid confusion when the energy appears in an equation defining the effective dose)
		Magnetic field
		Magnetic field strength
		Radiant exposure
H_E	=	Effective dose equivalent
H_T	=	Equivalent dose
$H_{50,T}$	=	50 year committed dose equivalent
$H_T(50)$	=	50 year committed equivalent dose
$H_{T'}(50)$	=	50 year committed equivalent dose in the remainder tissue/organ (Exceptional Case)
$\dot{H}$	=	Effective dose rate
I	=	Current
		Measured radiation quantity (e.g., absorbed dose, flux, exposure, effective dose, and dose)
		Intake
J	=	Current Density
L	=	Inductance
M	=	Magnetization
MPE	=	Maximum permissible exposure
N	=	Number of atoms
NHZ	=	Nominal hazard zone
NOHD	=	Nominal ocular hazard distance
OD	=	Optical Density
P	=	Polarization
		Power
		Pressure
PRF	=	Pulse repetition frequency
PW	=	Pulse width
Q	=	Heat
R	=	Release Rate
		Idea gas constant
		Radiation type
		Resistance
		Radius of disk source

S	=	Source organ Source strength Poynting Vector
S	=	Mean dose per unit cumulated activity
SEE	=	Specific effective energy
STP	=	Standard temperature and pressure
T	=	Exposure time Half-life Kinetic energy Target organ Temperature Tissue/organ
T'	=	Tissue or organ in which the committed equivalent dose is calculated to be higher than in any of the 12 specified organs (ICRP-60) with an assigned weighting factor
T_b	=	Biological half-life
T_{eff}	=	Effective half-life
T_p	=	Physical half-life
V	=	Voltage Volume Potential energy function
U_S	=	Mean number of transformations in the source organ over 50 years
W	=	Work
Y	=	Yield
Z	=	Impedance
a	=	Acceleration Aperture radius Area
c	=	Speed of light Heat capacity
d	=	Daughter Distance from laser aperture
e	=	Energy stored in an electric field per unit volume Energy (used to avoid confusion when the energy appears in an equation defining a related quantity such as the radiant exposure and irradiance)
h	=	Release height Distance from disk source (on-axis) Energy stored in a magnetic field per unit volume
i	=	summation index
k	=	Conversion factor (value depends on the units selected and the particular relationship)
l	=	Angular momentum

m = Mass
= Organ mass
m_o = Rest mass
m_T = Mass of the ICRP-60 remainder tissue
$m_{T'}$ = Mass of the remainder tissue or organ in which the committed equivalent dose is calculated to be higher than in any of the 12 specified tissues/organs with assigned weighting factors (ICRP-60)
n = Number of moles
p = Momentum
Parent
q = Charge
$q(0)$ = Initial activity in the organ
r = Radius of circular orbit
Radius of laser beam
Distance from radiation source
s = Distance
t = Shield thickness
Time
Thickness of disk source
t_d = Decay time
t_{irr} = Irradiation time
u = Mean wind speed
v = Velocity
w = Perpendicular distance from line source
$w_{remainder}$ = Remainder weighting factor
w_R = Radiation weighting factor
w_T = Tissue weighting factor
x = Downwind distance
Shield thickness
Vector cross product [$\mathbf{A} \times \mathbf{B} = |\mathbf{A}||\mathbf{B}| \sin\theta$]
y = Cross-wind distance
z = Elevation

∇ = Gradient operator
Γ = Dose Factor or gamma constant (Sv-m^2/MBq-hr or R-m^2/Ci-hr)
ΔE = Uncertainty in energy or the width of an energy level
Δp = Uncertainty in momentum
Δt = Uncertainty in time or the lifetime of an energy level
ΔT = Change in temperature
Δx = Uncertainty in position
β = velocity relative to the speed of light
γ = Lorentz factor
ε = permittivity

θ = Angle between the two vectors involved in the cross-product
Included angle which the point of interest makes with the ends of a line source

θ_v = viewing angle from the normal to a reflecting surface

μ = Attenuation coefficient
Permeability of a medium

μ_{en} = Energy absorption coefficient

ν = Frequency

λ = Disintegration constant
Wavelength

λ_b = Biological removal rate

λ_{eff} = Effective removal rate

λ_p = Physical removal rate
= Disintegration constant for the parent radionuclide

ρ = Density
Charge density
Gas density in an ionization chamber
Physical density

ρ_λ = spectral reflectance of a diffuse or specular object at wavelength λ

σ = Microscopic cross-section

σ_y = Horizontal standard deviation

σ_z = Vertical standard deviation

ϕ = Activating flux or fluence rate

χ = Concentration of radioactive material in a plume

χ/Q = Dispersion factor (s/m^3)

$\chi u/Q$ = Dispersion factor ($1/m^2$)

ψ = Wavefunction

ω = Angular frequency

$\vec{a}$ = An arrow over a variable indicates it is a vector quantity

Key Relationships

Activation

$$A = N\,\sigma\,\phi\,(1 - e^{-\lambda\,t_{\mathrm{irr}}})\,e^{-\lambda\,t_{\mathrm{d}}}$$

$$A_{\mathrm{V}} = \rho\left(\frac{\mu}{\rho}\right)\phi\,(1 - e^{-\lambda\,t_{\mathrm{irr}}})\,e^{-\lambda\,t_{\mathrm{d}}}$$

Activity

$$A = \lambda\,N$$

$$A(t) = A(0)\,e^{-\lambda\,t}$$

$$A_{\mathrm{d}}(t) = \frac{\lambda_{\mathrm{d}}\;A_{\mathrm{p}}(0)}{\lambda_{\mathrm{d}} - \lambda_{\mathrm{p}}}\,(e^{-\lambda_{\mathrm{p}}\,t} - e^{-\lambda_{\mathrm{d}}\,t})$$

Attenuation

$$I(x) = I(0)\,B\,e^{-\mu\,x}$$

$$B(\mathrm{Fe}) = 1 + \mu\,x \quad \text{for small } \mu x$$

$$B(\mathrm{Pb}) = 1 + \mu\,x/3 \quad \text{for small } \mu x$$

Duty Factor

$$\mathrm{DF} = \frac{I_{\mathrm{average}}}{I_{\mathrm{peak}}} = \frac{P_{\mathrm{average}}}{P_{\mathrm{peak}}} = \mathrm{PW} \times \mathrm{PRF}$$

External Dosimetry

Dose – Point Source

$$\dot{E} = \frac{A\,\Gamma}{r^2}$$

$$\dot{H} = \frac{S}{4\,\pi\,r^2}\left(\frac{\mu_{\mathrm{en}}}{\rho}\right)\sum_i E_i\;Y_i$$

Dose – Line Source

$$\dot{E} = \frac{A\,\Gamma\,\theta}{w}$$

Dose – Thin Disk Source

$$\dot{E} = \pi\, C_a\, \Gamma \ln\left(\frac{R^2 + h^2}{h^2}\right)$$

Dose – Thick Disk Source

$$\dot{E} = \frac{\pi\, C_v\, \Gamma\, [1 - \exp(-\mu\, t)]}{\mu} \ln\left(\frac{R^2 + h^2}{h^2}\right)$$

Gamma Constant or Dose Factor

$$\Gamma = k \sum_i E_i\, Y_i = \frac{1}{2} \sum_i E_i\, Y_i \quad \left(\text{Conventional Units: } \frac{\text{R m}^2}{\text{Ci-hr}}\right)$$

Hot-Particle Absorbed Dose

$$D = \frac{A(\text{DF})\, t}{a}$$

Ionization Chamber Dose-Current Relationship

$$I = \rho\, V\, \frac{T_{STP}}{T}\, \frac{P}{P_{STP}}\, \dot{H}$$

Internal Dosimetry

Absorbed Dose

$$\begin{aligned} D &= \int_0^T \dot{D}(t)\,dt = \int_0^T \dot{D}_o\, \exp(-\lambda_{eff}\, t)\, dt \\ &= \int_0^T k\, \frac{q(0)\, E}{m} \exp(-\lambda_{eff}\, t)\, dt \\ &= k\, \frac{q(0)\, E}{m} \left(\frac{1 - \exp(-\lambda_{eff}\, T)}{\lambda_{eff}}\right) \end{aligned}$$

Absorbed Dose Rate

$$\dot{D}(t) = \dot{D}_o\, \exp(-\lambda_{eff}\, t)$$

Committed Dose Equivalent (50 y)

$$H_{50,T} = 1.6 \times 10^{-10} \frac{\text{Sv-g}}{\text{MeV}} U_S \text{ SEE}(T \leftarrow S)$$

$$\text{CDE} = H_{50,T} = \frac{I}{\text{ALI}_{\text{NS}}} 0.5 \text{ Sv} = 1.6 \times 10^{-10} \frac{\text{Sv-g}}{\text{MeV}} U_S \text{ SEE}(T \leftarrow S)$$

Committed Effective Dose

$$E(50) = \frac{I}{\text{ALI}} 0.02 \text{ Sv} = \sum_{T=1}^{12} w_T \, H_T(50) + w_{\text{remainder}} \frac{\sum_{T=13}^{22} m_T \, H_T(50)}{\sum_{T=13}^{22} m_T}$$

Committed Effective Dose (Exceptional Case)

$$E(50) = \sum_{T=1}^{12} w_T \, H_T(50) + 0.025 \, H_{T'}(50) + 0.025 \frac{\sum_{T=13}^{22} m_T \, H_T(50) - m_{T'} \, H_{T'}(50)}{\sum_{T=13}^{22} m_T - m_{T'}}$$

Committed Effective Dose Equivalent

$$\text{CEDE} = H_E = \sum_T w_T \, H_{50,T} = \frac{I}{\text{ALI}_S} 0.05 \text{ Sv}$$

Cumulated Activity

$$\tilde{A} = \int_0^T A(T) \, dt = \frac{A(0)}{\lambda_{\text{eff}}} (1 - e^{-\lambda_{\text{eff}} T})$$

Derived Air Concentration

$$\text{DAC} = \text{ALI}/2400 \text{ m}^3$$

Effective Dose

$$E = \sum_T w_T \, H_T = \sum_R w_R \sum_T w_T \, D_{T,R} = \sum_T w_T \sum_R w_R \, D_{T,R}$$
$$= \frac{I}{\text{ALI}} 20 \text{ mSv}$$

Effective Half-Life

$$T_{\text{eff}} = \frac{T_{\text{p}}\, T_{\text{b}}}{T_{\text{p}} + T_{\text{b}}}$$

Effective Removal Rate

$$\lambda_{\text{eff}} = \lambda_{\text{p}} + \lambda_{\text{b}}$$

Equivalent dose

$$H_T = \sum_R w_R\, D_{T,R}$$

Initial Absorbed Dose Rate

$$\dot{D}_{\text{o}} = k\, \frac{q(0)\, E}{m}$$

Mean Absorbed Dose (MIRD Methodology)

$$\bar{D} = \tilde{A}\, S$$

Total Cumulated Activity

$$\tilde{A} = \int_0^\infty A(T)\, \mathrm{d}t = \int_0^\infty A(0) \exp(-\lambda_{\text{eff}}\, t)\, \mathrm{d}t = \frac{A(0)}{\lambda_{\text{eff}}} = \frac{A(0)\, T_{\text{eff}}}{\ln(2)} = 1.44\, T_{\text{eff}}\, A(0)$$

Dispersion Relationships

Dispersion Theory: Pasquill–Gifford Equation

$$\chi = \frac{Q}{\pi\, \sigma_y\, \sigma_z\, u} \exp\left[-\frac{1}{2}\left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_z^2}\right)\right]$$

$$= \frac{Q}{2\, \pi\, \sigma_y\, \sigma_z\, u} \exp\left[-\left(\frac{y^2}{2\, \sigma_y^2}\right)\right]\left[\exp\left(-\frac{(z-h)^2}{2\, \sigma_z^2}\right) + \exp\left(-\frac{(z+h)^2}{2\, \sigma_z^2}\right)\right]$$

Dispersion Dose Rate

$$\dot{E} = Q\left(\frac{\chi\, u}{Q}\right)\frac{1}{u}(\text{DRCF}) = Q\left(\frac{\chi}{Q}\right)(\text{DRCF})$$

Electromagnetic Relationships

Constants/Units

Permittivity of Free Space:	$\varepsilon_o = 8.854 \times 10^{-12}$ F/m
Permeability of Free Space:	$\mu_o = 4\ \pi \times 10^{-7}$ N/A^2
Speed of light:	$c = 3.0 \times 10^8$ m/s $= (\varepsilon_0\ \mu_0)^{-1/2}$
Impedance of Free Space:	$Z_0 = (\mu_0/\varepsilon_0)^{1/2} = 376.7$ ohms
Charge:	1 C = 1 A/s
Potential:	1 V = 1 J/C
Magnetic Field:	1 T = 1 N/A-m $= 1.0 \times 10^4$ gauss

Capacitance

$$C = q/V$$

Constitutive Equations

$$\mathbf{D} = \varepsilon_0\ \mathbf{E} + \mathbf{P}$$

$$\mathbf{H} = \vec{B}/\mu_0 - \mathbf{M}$$

Constitutive Equations in a Linear Medium

$$\mathbf{D} = \varepsilon\ \mathbf{E}$$

$$\mathbf{H} = \mathbf{B}/\mu$$

Current

$$I = q/t$$

Electric Field Strength

$$E = F/q$$

Energy

$$E = q\ V$$

Energy Stored in an Electromagnetic Field per Unit Volume

$$E = \frac{1}{2}\,\varepsilon_0\,E^2$$

$$h = \frac{1}{2}\,\mu_0\,H^2$$

Forces

Electric Force

$$\mathbf{F} = q\,\mathbf{E}$$

$$F = \left(\frac{1}{4\,\pi\,\varepsilon_0}\right)\frac{q_1\,q_2}{s^2}$$

Magnetic Force

$$\mathbf{F} = q\,\mathbf{v} \times \mathbf{B} = q\,|\overline{v}|\,|\mathbf{B}|\sin\theta$$

$$\mathbf{B} = \mu\,\mathbf{H}$$

Lorentz Force

$$\mathbf{F} = q\,(\mathbf{E} + \mathbf{v} \times \mathbf{B})$$

Impedance (Alternating Current)

$$V = Z\,I$$
$$= V_o\,\sin\omega\,t$$

$$Z = \left[R^2 + \left(\omega\,L - \frac{1}{\omega\,C}\right)^2\right]^{1/2}$$

Ohm's Law (Direct Current)

$$V = I\,R$$

Power

$$P = I\,V = I^2\,R$$

Poynting Vector

$$\mathbf{S} = \mathbf{E} \times \mathbf{H} = |\mathbf{E}|\,|\mathbf{H}|\sin\theta$$

Maxwell Equations

$$\nabla \cdot \mathbf{D} = \rho$$

$$\nabla \times \mathbf{H} - \frac{\partial \mathbf{D}}{\partial t} = \mathbf{J}$$

$$\nabla \cdot \mathbf{B} = 0$$

$$\nabla \times \mathbf{E} + \frac{\partial \mathbf{B}}{\partial t} = 0$$

Mechanics Relationships

Angular Momentum

$$l = m\, v\, r$$

Centrifugal Force

$$F = m\, v^2 / r$$

Force

$$F = m\, a$$

Heat

$$Q = m\, c\, \Delta T$$

Ideal Gas

$$\mathrm{PV} = n\, \mathrm{RT}$$

Kinetic Energy

$$T = \frac{1}{2} m\, v^2 = p^2 / 2\, m$$

Momentum

$$\mathbf{p} = m\, \mathbf{v} = \gamma\, m_\mathrm{o}\, \mathbf{v}$$

Total Energy

$$E = m\, c^2 = \gamma\, m_\mathrm{o}\, c^2$$

$$E^2 = p^2\, c^2 + m_\mathrm{o}^2\, c^4 = (m_\mathrm{o}\, c^2 + T)^2$$

Relativistic Mass

$$m = m_o \gamma$$

Relativistic Notation

$$\beta = v/c$$

$$\gamma = \frac{1}{\sqrt{1-\beta^2}}$$

Rest Energy

$$E_o = m_o c^2$$

Wavelength

$$c = \nu \lambda$$

Work

$$W = F s$$

Non-Ionizing Radiation Relationships

Gaussian Beam Radius

$$r_{GB} = (a^2 + d^2 D^2/4)^{1/2}$$

Irradiance

$$E = \frac{P}{A}$$

Nominal Hazard Zone (NHZ)

$$\text{NHZ} = \sqrt{\frac{\rho_\lambda P \cos\theta_v}{\pi\,(\text{MPE})}}$$

Nominal Ocular Hazard Distance (NOHD)

$$\text{NOHD} = \frac{2}{D}\sqrt{\frac{P}{\pi\,(\text{MPE})} - a^2}$$

Optical Density

$$\mathrm{OD} = \log_{10}\left[\frac{H}{\mathrm{MPE}}\right] = \log_{10}\left[\frac{E}{\mathrm{MPE}}\right]$$

Radiant Exposure

$$H = \frac{e}{A}$$

Production Equations

See Appendix VII.

Quantum Mechanics

Schröedinger Equation

$$\left(-\frac{\hbar^2}{2\,m}\nabla^2 + V\right)\psi = E\,\psi$$

Uncertainty Relationships

$$\Delta E\,\Delta t \geq \hbar$$

$$\Delta p\,\Delta x \geq \hbar$$

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Appendix VII
Production Equations in Health Physics

Introduction

The mathematical framework for the accumulation or production of radioactive material is governed by linear differential equations having exponential solutions. In many health physics applications, the solutions of these equations have a similar mathematical framework that leads to a common structure in the resultant relationships. This structure leads to common solution types even if additional constraints are imposed.

For example, the assumption that radioactive material enters a system at a constant rate leads to a set of production equations that describe a broad class of phenomena encountered by health physicists. Equations governing activation, the buildup of radioactive material on a filter or demineralizer, deposition of material on a surface from a radioactive plume, and the release of material into a room, are examples of phenomena described in a consistent manner by production equations. This appendix describes production equations and their applications in a variety of health physics areas.

Theory

In health physics applications, the rate of change of radioactive material in a system is described by first-order linear differential equations that have exponential solutions. Since exponential forms appear throughout the field, it is expected that phenomena describing the accumulation of radioactive material have a similar mathematical structure. This text refers to these structures as production equations.

In order to formulate a general form of production relationship, consider the time rate of change of activity $\dot{A}$ associated with the continuous introduction of a radionuclide into a system or structure. For a given radionuclide:

$$\dot{A} = P\, e^{-Kt} \tag{VII.1}$$

where

P = production term or the rate at which activity is added to the system (e.g., a room, accelerator target, or filter paper)
K = total removal rate of the radionuclide from the system
t = time from the start of production

In order to simplify the equation resulting from the integration of (VII.1), it is assumed that P is constant. The production term has units of activity per unit time (Bq/s). Examples of the production term for a variety of physical phenomena are provided in Table VII.1.

When using (VII.1), it is important that the production equation be applied separately for each radionuclide of interest. The quantities P and K depend on the radionuclide half-life as well as its physical and chemical properties.

The total removal rate (K) has numerous components. The most common components are derived from radioactive decay (λ), biological decay (λ_b), or ventilation (λ_v). Explicit forms for these removal rates are

$$\lambda = \ln(2)/T_{1/2} \tag{VII.2}$$

$$\lambda_b = \ln(2)/T^b_{1/2} \tag{VII.3}$$

$$\lambda_v = F/V \tag{VII.4}$$

where

$T_{1/2}$ = physical half-life
$T^b_{1/2}$ = biological half-life
F = ventilation flow-rate of the system
V = free air volume of the system

The total removal rate

$$K = \lambda + \lambda_b + \lambda_v + \ldots \tag{VII.5}$$

is the sum of the individual removal rates as they apply to the problem of interest. Not all terms in (VII.5) appear in each application. The specific application of removal rates is addressed in subsequent discussion.

Equation (VII.1) can be integrated with respect to time from $t = 0$ to $t = T$ where the time T is the end of the production interval:

$$\int_0^T \dot{A}\, dt = A(T) = \int_0^T P\, e^{-Kt}\, dt = P \int_0^T e^{-Kt}\, dt \tag{VII.6}$$

In (VII.6), we assume that no activity is initially present in the system ($A(0) = 0$). Using this condition leads to the result

$$A(T) = \frac{P}{K}(1 - e^{-KT}) \tag{VII.7}$$

Table VII.1 Examples of Production Terms in Health Physics Applications.

Physical Phenomena	*P* (Bq/s)	Definition of Terms (units)
Activation of material in an accelerator	$N \sigma \varphi \lambda$	N = number of target atoms of the nuclide being activated (atoms) σ = activation cross-section for the specific activation reaction (b/atom or cm^2/atom) φ = activating flux of a beam of particles (particles/cm^2-s) λ = radioactive disintegration constant (1/s)
Activation of material in a reactor	$N \sigma \varphi \lambda$	N = number of target atoms of the nuclide being activated (atoms) σ = activation cross-section for the specific activation reaction (b/atom or cm^2/atom) φ = activating flux of neutrons (neutrons/cm^2-s) λ = radioactive disintegration constant (1/s)
Deposition of radioactive material in a demineralizer bed	$C F e$	C = influent activity concentration of an isotope entering the demineralizer (Bq/m^3) F = flow rate of fluid through the demineralizer (m^3/s) e = isotope-specific removal efficiency of the demineralizer bed
Deposition of radioactive material in a filter	$C F e$	C = influent activity concentration of an isotope entering the filter (Bq/m^3) F = flow rate of fluid through the filter (m^3/s) e = isotope specific removal efficiency of the filter
Surface deposition from a radioactive plume	$w S$	w = ground deposition rate (Bq/m^2-s) S = surface area of the deposition (m^2)
Inhalation of radioactive material	$C r$	C = air concentration of radioactive material (Bq/m^3) r = breathing rate (m^3/s)
Surface deposition from a leaking radioactive fluid	$C F$	C = activity concentration of the isotope in the fluid leaking onto the surface (Bq/m^3) F = leak rate of the fluid onto the surface (m^3/s)
Airborne entry of ^{222}Rn into a home	$C F$	C = air concentration of ^{222}Rn entering the home (Bq/m^3) F = air infiltration rate entering the home (m^3/s)
Release of radioactive material from a stack	$C F$	C = air concentration of radioactive material being released (Bq/m^3) from a stack F = stack flow rate (m^3/s)
Release of radioactive material into a room	Q	Q = release rate of airborne radioactive material into the room (Bq/s)

Equation (VII.7) provides a relationship describing the buildup of activity during the time that the production term is active. For $K\,T \gg 1$, the system activity reaches its maximum value. Accordingly, (VII.7) is written as:

$$A(\infty) = A_{\text{eq}} = \frac{P}{K} \tag{VII.8}$$

The saturation or equilibrium activity is the maximum activity that can be achieved in the system.

If T is defined as the time the production term is active and t describes the time after the production ceases, (VII.7) is rewritten to describe the activity variation following the production interval and during the subsequent decay period:

$$A(t) = \frac{P}{K}\left(1 - e^{-K\,T}\right) e^{-k\,t} \tag{VII.9}$$

where

k = total removal rate post production, i.e., during the decay time t

As a matter of specificity, $t = 0$ corresponds to the time that production ceases.

Examples of Production Equations

A number of examples are provided to illustrate the utility of the general production equation. These examples include: (1) the activation of a target by an accelerator beam or reactor neutron source, (2) buildup of activity on a filter or demineralizer, (3) buildup of activity in a pond, and (4) release of activity into a room.

Activation

Activation is a process described by the reaction $C(c, d)\ D$ during which radiation of type c strikes a target nucleus C and produces nucleus D and radiation of type d. Examples of activation reactions include: $^{59}\text{Co}(n, \gamma)^{60}\text{Co}$, $^{16}\text{O}(n, p)^{16}\text{N}$, $^{27}\text{Al}(n, \alpha)^{24}\text{Na}$, and $^{3}\text{H}(p, n)^{3}\text{He}$.

Using the generalized production equation (VII.9) and the production term from Table VII.1, leads to a relationship that describes the activity in the target as a function of time:

$$A = N\,\sigma\,\varphi\left(1 - e^{-\lambda\,T}\right) e^{-\lambda\,t} \tag{VII.10}$$

where N, σ, and φ are defined in Table VII.1. For non-gaseous products, the removal rates (K and k) are equal to the physical decay constant (λ). T is the irradiation time; the time the target is irradiated by the accelerator's beam or the time the material to be activated is exposed to the reactor's neutron fluence rate (flux).

The time after the reactor is shut down or the accelerator beam is terminated is t. The steady state (saturation) or equilibrium activity is $N \sigma \varphi$.

The application of (VII.10) is further illustrated by considering the activation of ^{59}Co by thermal neutrons. In this example, N is the number of ^{59}Co atoms in the target, σ is the microscopic cross-section for the $^{59}\text{Co}(n_{\text{thermal}}, \gamma)^{60}\text{Co}$ reaction, φ is the number of thermal neutrons per cm^2-s, and λ is the ^{60}Co decay constant. Equation (VII.10) is applied separately for each activated species.

Demineralizer Activity

Ion exchange is a process used in a variety of nuclear facilities to reduce the radioactive material concentration in water by removing radioactive ions and replacing them with nonradioactive ions. The device in which the ion exchange occurs is commonly called a demineralizer.

The activity that accumulates within a demineralizer bed is also obtained from (VII.9) and Table VII.1:

$$A = \frac{C\,F\,e}{\lambda}\left(1 - e^{-\lambda T}\right) e^{-\lambda t} \tag{VII.11}$$

Equation (VII.11) is also to be applied individually for each isotope trapped in the demineralizer bed. In (VII.11), C, F, and e are defined in Table VII.1, λ is the physical decay constant of the trapped material, T is the time the demineralizer is on-line (valved-in) and removing radioactivity from the influent stream, and t is the time after the demineralizer is no longer in-service (valved-out). For the demineralizer application, the total removal rate is just the physical decay constant.

Equation (VII.11) also applies to filters. The saturation activity for both filters and demineralizers is $C\,F\,e/\lambda$.

Surface Deposition

The deposition of radioactive material onto a surface from an airborne plume is also described by a production equation. Again, using Table VII.1 and (VII.9), the activity deposited onto a surface is:

$$A = \frac{w\,S}{K}\left(1 - e^{-K T}\right) e^{-k t} \tag{VII.12}$$

and $w\,S$ is defined in Table VII.1. The removal rates k and K are discussed below. Equation (VII.12) is used to illustrate the versatility of the production equation.

Assuming a continuous release of radioactive material from a plume, an expression for the equilibrium activity that is removed from the plume and deposited on a surface of area S is written as:

$$A_{\text{eq}} = \frac{w\,S}{K} \tag{VII.13}$$

If it is also assumed that the material deposits on the surface of a stationary body of water, such as a pond, then (VII.13) still applies and

$$K = \lambda + \lambda_b \tag{VII.14}$$

where
λ_b = biological removal rate from the pond

If the radionuclide deposited onto the surface of the pond is also soluble in the pond water and instantaneous mixing of the radionuclide within the pond occurs, then the equilibrium concentration of the radionuclide in the pond water (C_{eq}) is determined from the relationship

$$C_{eq} = \frac{A_{eq}}{V} \tag{VII.15}$$

where
V = volume of water in the pond

Using (VII.13) and (VII.15), the equilibrium concentration of a radionuclide in a pond is determined:

$$C_{eq} = \frac{w\,S}{K\,V} \tag{VII.16}$$

The production concept can also be extended to calculate the equilibrium concentration in an organism, such as a fish, living in the pond. The equilibrium activity concentration per unit mass (Bq/kg) in the fish ($C_{eq\text{-}fish}$) is written as

$$C_{eq\text{-}fish} = \frac{I\,C_{eq}}{K'} \tag{VII.17}$$

where
I = intake of pond water by the fish (m^3/kg(fish)-s)
K' = total removal rate of the isotope from the fish

$$K' = \lambda + \lambda_b' \tag{VII.18}$$

where
λ_b' = biological removal rate from the fish

A careful examination of (VII.18) indicates that the term $C_{eq}\,I$ is just P per unit mass of the fish. Equation (VII.18) is another application of the production equation (VII.9).

Release of Radioactive Material into a Room

The release of airborne radioactive material into a room is obtained from (VII.9) and Table VII.1:

$$A = \frac{Q}{K}\left(1 - e^{-KT}\right) e^{-kt} \tag{VII.19}$$

where removal of radioactive material includes both physical decay and ventilation terms

$$K = k = \lambda + \frac{F}{V} \tag{VII.20}$$

In (VII.20), the ventilation rate is assumed constant during the production and post-production periods.

Conclusions

The use of production equations provides a unified explanation for a variety of phenomena encountered in health physics. The specific application determines the *P*, *K*, and *k* values, but the form of the equation remains the same. The use of production equations greatly simplifies the understanding of a variety of health physics concepts that appear to involve dissimilar phenomena.

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Appendix VIII
Mathematical Review

Overview

A sound understanding of basic mathematical relationships is an essential element supporting health physics theory. In this appendix, the basic mathematical fundamentals in the areas of algebra, geometry, trigonometry, calculus, and special functions are summarized. Health physics applications of the mathematical concepts are provided.

Some students may find this material somewhat rudimentary. However, there are health physicists that have difficulty with mathematical concepts. This attachment is specifically designed for them.

Algebra

Numerous health physics applications, including the calculation of dose, shielding thickness, and decay time require the ability to manipulate algebraic equations. These equations may involve polynomials, exponentials, logarithms, and trigonometric functions. One of the most fundamental equations is the linear equation, which forms the basis for the linear-non-threshold dose-response model.

Linear Equations

Linear equations are frequently encountered in health physics applications. They are used to describe dose response and risk models, are useful in the interpolation of quantities such as buildup factors, and can be used to describe the expansion of a laser beam's diameter as it exits an aperture. In view of their use in a range of health physics applications, it is important to understand the characteristics of linear equations.

The linear equation involves an independent variable x and a dependent variable $y(x)$ which are related by the relationship:

$$y = m\,x + b \tag{VIII.1}$$

where
b = y-intercept
m = slope or derivative of y with respect to x

$$m = \frac{dy}{dx} \tag{VIII.2}$$

If the coordinates (x, y) at two points are known, the equation of the straight line connecting these points is determined from (VIII.1). Knowledge of the resultant linear equation permits the determination of the value of other points on the line. If the two points are defined by the coordinates (x_1, y_1) and (x_2, y_2), the linear equation passing through these points is derived from (VIII.1).

Using (VIII.1), the following two equations are defined at the points of interest:

$$y_1 = m\,x_1 + b \tag{VIII.3}$$

$$y_2 = m\,x_2 + b \tag{VIII.4}$$

By subtracting (VIII.3) from (VIII.4), the slope is determined:

$$m = \frac{y_2 - y_1}{x_2 - x_1} \tag{VIII.5}$$

In a similar fashion, the y-intercept (b) is obtained from (VIII.3):

$$b = y_1 - m\,x_1 \tag{VIII.6}$$

Equations (VIII.5) and (VIII.6) may be combined:

$$b = y_1 - \frac{y_2 - y_1}{x_2 - x_1}\,x_1 \tag{VIII.7}$$

The desired result is obtained by using the values of m from (VIII.5) and b from (VIII.7) in (VIII.1):

$$y = y_1 + \frac{x - x_1}{x_2 - x_1}\,(y_2 - y_1) \tag{VIII.8}$$

Equation (VIII.8) is also useful in interpolation calculations involving buildup factors and attenuation coefficients.

Example 1
The BEIR VII report characterizes solid tumor incidence $[f(d)]$ in terms of a linear nonthreshold dose response model. If a threshold is included, this model becomes:

$$f(d) = r\,d + b$$

where
r = risk coefficient (solid tumors/Sv)
d = effective dose (Sv)
b = incidence at zero dose

What is the threshold effective dose (d_{th}) for solid tumors?

Solution
At threshold, no solid tumors are observed or $f(d) = 0$. With this value, the dose response model becomes:

$$0 = r\,d_{th} + b$$

$$d_{th} = -\frac{b}{r}$$

Example 2
The buildup factor (B) is a function of the optical thickness (μx) where μ is the linear attenuation coefficient and x is the shield thickness. What is the value of the buildup factor at $\mu x = 14.0$ given the following data? Use linear interpolation in your solution.

Data

Point	B	μx
1	17.5	10.0
2	30.6	15.0

Solution

$$B(\mu x) = B_1 + \frac{u\,x - (\mu x)_1}{(\mu x)_2 - (\mu x)_1}(B_2 - B_1)$$

$$B(14) = 17.5 + \frac{14.0 - 10.0}{15.0 - 10.0}(30.6 - 17.5) = 28.0$$

Solution of Equations

Decay and attenuation equations are often encountered in practical health physics applications. These relationships must be solved for the various terms as governed by the particular application. The basic attenuation equation defines the change

in a variable I (dose, dose rate, fluence rate, etc.) as it is attenuated by material of thickness x. For photons, this relationship is:

$$I(x) = I(0)\, B\, e^{-\mu x} \tag{VIII.9}$$

where

$I(0)$ = unattenuated quantity at a specific location
$I(x)$ = attenuated quantity at that location
B = build-up factor
μ = attenuation coefficient
x = thickness of attenuating material

Determining the required shielding thickness to reduce a quantity such as the dose rate to a pre-established value is a frequent application of (VIII.9). This requires that (VIII.9) be solved for x. In order to obtain x, first isolate the exponential term by dividing both sides of (VIII.9) by $I(0)\ B$:

$$\frac{I(x)}{B\, I(0)} = e^{-\mu x} \tag{VIII.10}$$

Next, take the natural logarithm of each side of (VIII.10):

$$\ln\left[\frac{I(x)}{B\, I(0)}\right] = \ln\left[e^{-\mu x}\right] = -\mu\, x \tag{VIII.11}$$

Finally, divide both sides of the equation by $-\mu$:

$$x = -\frac{1}{\mu} \ln\left[\frac{I(x)}{B\, I(0)}\right] \tag{VIII.12}$$

Example 3
The radioactive decay relationship $A(t) = A(0)\ e^{-\lambda t}$, describing a radioactive parent decaying to a daughter, applies to the decay of nuclides such as H-3, C-14, and Co-60. Solve this equation for (1) the initial activity $A(0)$ and (2) the time (t) it takes for an initial activity to decay to activity $A(t)$.

Solution
Initial Activity $A(0)$:

$$A(t) = A(0)\, e^{-\lambda t}$$

$$A(t)\, e^{+\lambda t} = A(0)\, e^{-\lambda t}\, e^{+\lambda t} = A(0)$$

$$A(0) = A(t)\, e^{+\lambda t} = \frac{A(t)}{e^{-\lambda t}}$$

Time (t) to Decay to Activity $A(t)$:

$$A(t) = A(0)\, e^{-\lambda t}$$

$$e^{-\lambda t} = \frac{A(t)}{A(0)}$$

$$\ln[e^{-\lambda t}] = -\lambda\, t = \ln\left[\frac{A(t)}{A(0)}\right]$$

$$t = -\frac{1}{\lambda} \ln\left[\frac{A(t)}{A(0)}\right]$$

Geometry

Basic geometric relationships are required to perform numerous health physics calculations. For example, the calculation of the dose rate from a circular spill of a radioactive liquid requires the determination of the spill area. Airborne concentration calculations require the calculation of room or tank volumes. A number of important geometric relationships and common applications are summarized in Table VIII.1.

Trigonometry

Basic trigonometric relationships are frequently used in health physics applications to determine distances in dose calculations, perform set point calculations, determine shielding thicknesses, and to determine the diameter of a laser beam after it exits the aperture. The three basic trigonometric functions (sine, cosine, and tangent) are defined in terms of a right triangular. If the sides of the right triangle are defined to be a, b, and c (where c is the hypotenuse), and angle A is opposite side a and angle B is opposite side b, then

$$\sin B = \frac{\text{opposite}}{\text{hypotenuse}} = \frac{b}{c} \tag{VIII.13}$$

$$\cos B = \frac{\text{adjacent}}{\text{hypotenuse}} = \frac{a}{c} \tag{VIII.14}$$

$$\tan B = \frac{\text{opposite}}{\text{adjacent}} = \frac{b}{a} \tag{VIII.15}$$

where

$$a^2 + b^2 = c^2 \quad \text{(Pythagorean theorem)} \tag{VIII.16}$$

Table VIII.1 Geometric Relationships and Their Health Physics Applications.

Geometric Configuration	Relationship	Application/Approximation Example
Circumference (C) of a circle or radius r	$C = 2\,\pi\,r$	Definition of a radian. Calculation of dose rate from a pipe run of circular shape.
Area (A) of a circle	$A = \pi\,r^2$	Approximates laser beam cross-sectional area. Approximates parabolic antennae area. Approximates radioactive liquid spill. Approximates accumulation of radioactive material on a surface (e.g., surface of a demineralizer bed). Determination of the average concentration per unit area for the calculation of dose from a spill, surface, or thin disk source. Calculation of surface contamination for a plane surface (e.g., floor and wall). Calculation of dose rate from a thin disk (spill source). Calculation of power density for antennas. Calculation of power density for lasers.
Area of sphere of radius r	$A = 4\,\pi\,r^2$	Definition of steradian. Calculation of surface contamination area for a spherical surface (e.g., tank and room).
Area of a cylinder of radius r and height h	$A = 2\,\pi r^2 + 2\,\pi\,r\,h$	Calculation of surface contamination area for a cylindrical surface (e.g., tank and power reactor containment building).
Volume (V) of a cylinder of radius r and height h	$V = A\,h = \pi r^2\,h$	Approximation of tanks. Approximation of resin columns. Approximation of activated rod. Approximation of fuel pin. Approximation of ion chamber volume. Volume determination for containments, tanks, and ion chambers. Calculation of activity concentration in a cylindrical structure (e.g., slab, pool, or demineralizer bed). Calculation of air or liquid concentrations of radioactive materials.
Volume of sphere of radius r	$A = \frac{4}{3}\,\pi\,r^3$	Volume determination for spherical tanks and rooms. Calculation of the concentration of a fluid in a spherical structure. Semi-infinite cloud model.

These relationships can be remembered using the mnemonic SOHCAHTOA where S, C, and T refer to the sin, cos, and tan functions; and O, A, and H refer to opposite, adjacent, and hypotenuse.

In evaluating trigonometric functions, it is important to select the proper set of units. Two commonly used angular units are degrees and radians. A radian is defined to be the angle subtending a circular arc that is equal to the radius of the circle. There are 2 π radians in a circle or π radians in 180°.

Angles are encountered in equations such as the line source dose rate equation:

$$\dot{D} = \frac{A_L \, G \, \theta}{w} \tag{VIII.17}$$

Example 4 illustrates the calculation of the included angle (θ) which appears in (VIII.17).

When an angle appears as a factor in most equations, it must be expressed in radian units. Angles must also be expressed in radians whenever a series is used to approximate a trigonometric function.

Series expansions are useful whenever equations involving trigonometric functions are encountered and the function's argument is small. For example, calculations of a laser beam's diameter and laser compliance with protection standards utilize trigonometric functions involving the divergence angle of the laser. Since divergence angles are small and on the order of a milliradian, these trigonometric functions may be approximated by the first term of their series expansions.

The series expansions for $\sin(x)$, $\cos(x)$, and $\tan(x)$ are:

$$\sin(x) = x - \frac{x^3}{3!} + \frac{x^5}{5!} - \frac{x^7}{7!} + \ldots \qquad x^2 < \infty \tag{VIII.18}$$

$$\cos(x) = 1 - \frac{x^2}{2!} + \frac{x^4}{4!} - \frac{x^6}{6!} + \ldots \qquad x^2 < \infty \tag{VIII.19}$$

$$\tan(x) = x + \frac{x^3}{3} + \frac{2}{15} x^5 + \frac{17}{315} x^7 + \ldots \qquad x^2 < \frac{\pi^2}{4} \tag{VIII.20}$$

where

x = angle in radians

For small angles, the sin, cos, and tan relationships reduce to:

$$\sin(x) \approx x \tag{VIII.21}$$

$$\cos(x) \approx 1 \tag{VIII.22}$$

$$\tan(x) \approx x \tag{VIII.23}$$

Example 4

A line source is 7 meters long. What is the angle subtended at a point 1.0 meter from the end of the line source at a perpendicular distance of 3.0 meters?

Solution

The line source included angle θ is obtained from two triangles which have a base/height/included angle of 1.0 m/3.0 m/A and 6.0 m/3.0 m/B, respectively. The total angle subtended by the line source θ is the sum of the angles A and B. Each angle is determined from its respective tangent function:

$$\tan A = \frac{1\ m}{3\ m} = \frac{1}{3} \qquad \tan B = \frac{6\ m}{3\ m} = 2$$

$$\tan^{-1}(\tan A) = A = \tan^{-1}\left(\frac{1}{3}\right) \qquad \tan^{-1}(\tan B) = B = \tan^{-1} 2$$

$$A = 18.4^\circ \qquad B = 63.4^\circ$$

Therefore,

$$\theta = A + B = (18.4^\circ + 63.4^\circ)\left(\frac{\pi \text{ radians}}{180^\circ}\right) = 1.43 \text{ radians}$$

Functions

Exponential and logarithm functions are frequently encountered in health physics equations. Basic health physics relationships involve linear differential equations whose solutions involve exponential functions. Exponential functions have relatively simple properties and understanding these properties is essential to properly evaluating health physics phenomena. Logarithms are encountered in a variety of applications including basic shielding calculations. Basic rules that facilitate using these functions in health physics calculations are summarized below.

Exponentials

Exponential functions are encountered in decay and attenuation problems. A few rules simplify the solution of these problems:

$$e^a\ e^b = e^{a+b} \tag{VIII.24}$$

$$\frac{e^a}{e^b} = e^{a-b} \tag{VIII.25}$$

Exponential functions can also be expressed as a series expansion:

$$e^{-\lambda t} = 1 - \lambda t + \frac{(\lambda t)^2}{2!} - \frac{(\lambda t)^3}{3!} + \frac{(\lambda t)^4}{4!} + \ldots \quad (\lambda t)^2 < \infty \tag{VIII.26}$$

For small values of the argument, the function is approximated to sufficient accuracy by the leading terms:

$$e^{-\lambda t} \approx 1 - \lambda\, t \tag{VIII.27}$$

Logarithms

The solution of shielding equations can result in an equation involving logarithms. Several rules simplify the solution of these equations.

The following formulae provide useful logarithmic relationships:

$$\log(a\, b) = \log(a) + \log(b) \tag{VIII.28}$$

$$\log\left(\frac{a}{b}\right) = \log(a) - \log(b) \tag{VIII.29}$$

$$\log(a^n) = n \log(a) \tag{VIII.30}$$

$$\log(a^{1/n}) = \frac{\log(a)}{n} \tag{VIII.31}$$

These relationships apply to both base 10 and natural logarithms.

Example 5
The production equation for activation has the form:

$$A(t) = A_{\text{sat}}\, (1 - e^{-\lambda\, t_{\text{irr}}})\, e^{-\lambda\, t_{\text{d}}}$$

Simplify this relationship for the case when $\lambda\, t_{\text{irr}} \ll 1$ and when $\lambda\, t_{\text{irr}} \ll 1$ and $\lambda\, t_{\text{d}} \ll 1$.

Solution

$$\lambda\, t_{\text{irr}} << 1$$

$$e^{\lambda\, t_{\text{irr}}} \approx 1 - \lambda\, t_{\text{irr}}$$

$$A(t) = A_{\text{sat}}\, (1 - e^{-\lambda\, t_{\text{irr}}})\, e^{-\lambda\, t_{\text{d}}} \approx A_{\text{sat}}\, [1 - (1 - \lambda\, t_{\text{irr}})\,] e^{-\lambda\, t_{\text{d}}}$$
$$= A_{\text{sat}}\, \lambda\, t_{\text{irr}}\, e^{-\lambda\, t_{\text{d}}}$$

$$\lambda\, t_{\text{irr}} << 1 \quad \text{and} \quad \lambda\, t_{\text{d}} << 1$$

$$e^{\lambda\, t_{\text{irr}}} \approx 1 - \lambda\, t_{\text{irr}}$$

$$e^{-\lambda\, t_{\text{d}}} \approx 1 - \lambda\, t_{\text{d}}$$

$$A(t) = A_{sat}\,(1 - e^{-\lambda\, t_{irr}})\, e^{-\lambda\, t_d} \approx A_{sat}\,[1 - (1 - \lambda\, t_{irr})]\,(1 - \lambda\, t_d)$$
$$= A_{sat}\,\lambda\, t_{irr}\,(1 - \lambda\, t_d)$$

Calculus

Appendix VII summarized the use of production equations that required the integration of a linear differential equation to determine the activity of a system as a function of time. Calculus is also utilized in internal dosimetry to obtain key input quantities such as the cumulated activity and mean number of transformations in a source organ over a specified time.

Health physics applications of calculus include both the integration and differentiation operations. Each of the applications is examined in terms of its health physics use.

Integrals

An integral represents the area under a curve or the volume under a surface. The integral has a numerical value as well as units. In health physics applications, the integral's magnitude and units are both important.

Many expressions frequently encountered in health physics applications, involve the exponential function. Decay relationships, attenuation equations, and clearance functions are typically written in terms of exponential functions. The integral of an exponential function is also a useful quantity. As an example, calculation of cumulated activity involves the integral of an exponential.

The general expression for the integral of an exponential, between the limits a and b is:

$$\int_a^b e^{-q\,x}\,dx = \left.\frac{e^{-q\,x}}{-q}\right|_a^b = \frac{e^{-q\,b} - e^{-q\,a}}{-q} = \frac{e^{-q\,a} - e^{-q\,b}}{q} \qquad \text{(VIII.32)}$$

An integral of more significant health physics importance is the integral of activity with respect to time or the cumulated activity $(\tilde{A}(T))$:

$$\tilde{A}(T) = \int_0^T A(t)\,dt = \int_0^T A(0)\,e^{-\lambda\, t}\,dt$$
$$= A(0)\int_0^T e^{-\lambda\, t}\,dt = A(0)\left.\frac{e^{-\lambda\, t}}{-\lambda}\right|_0^T = \left[\frac{A(0)}{\lambda}\right]\left[1 - e^{-\lambda\, T}\right] \qquad \text{(VIII.33)}$$

This integral forms the basis for the MIRD and ICRP internal dosimetry methodologies. When $T = \infty$, the integral represents MIRD's total cumulated activity. When $T = 50$ y, $\tilde{A}(T)$ represents the ICRP's mean number of transformations in the source organ over 50 y (U_S).

Derivatives

In addition to the integral, familiarity with derivatives of exponential and other functions is important. Derivatives of exponentials and functions of health physics interest include:

$$\frac{d}{dx} e^{-\lambda t} = -\lambda\, e^{-\lambda t} \tag{VIII.34}$$

$$\frac{d}{dx} x^{n} = n\, x^{n-1} \tag{VIII.35}$$

$$\frac{d}{dx} x^{-n} = -n\, x^{-(n+1)} \tag{VIII.36}$$

Differentiation applications are less frequently encountered than integration. However, differentiation permits the determination of the rates of various decay functions.

Example 6
An activity of 2 MBq enters an organ and remains there until it completely decays. If this activity is due to an isotope with an effective disintegration constant of 0.2/hr, what is the total cumulated activity?

Solution

$$\tilde{A}(\infty) = \frac{A(0)}{\lambda} = \frac{2\ \text{MBq}}{0.2/\text{h}} = 10\ \text{MBq-h}$$

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Appendix IX
Selected Data on Radionuclides of Health Physics Interest

Although there are over 1000 known radionuclides, it is important to become familiar with the fundamental characteristics of those nuclides commonly encountered in the radiation protection field. These characteristics include the decay mode, type of radiation emitted, energy of the emitted radiation, half-life, and production mode.

Table IX.1 outlines the fundamental characteristics of selected radionuclides. These radionuclides include those emphasized in the American Board of Health Physics Examination Preparation Guide.

Table IX.1 Fundamental Characteristics of Commonly Encountered Radionuclides.

Nuclide	Major Radiation Emitted		Half-life	Production Modes
	Type	Energy (MeV)		
H-3	β^-	0.018591 (max)	12.32 y	$^{2}H(n, \gamma)^{3}H$ $^{3}He(n, p)\ ^{3}H$ $^{6}Li(n, \alpha)\ ^{3}H$ $^{10}B(n, 2\alpha)\ ^{3}H$ Atmospheric spallation from cosmic rays
Be-7	γ	0.4776	53.3 d	$^{10}B(p, \alpha)^{7}Be$ Atmospheric spallation from cosmic rays
C-11	β^+	0.960 (max)	20.36 m	$^{12}C(\gamma, n)^{11}C$ $^{12}C(n, 2n)^{11}C$
	γ	0.511		$^{14}N(p, \alpha)^{11}C$
N-13	β^+	1.190 (max)	9.97 m	$^{12}C(p, \gamma)\ ^{13}N$ $^{13}C(p, n)^{13}N$
	γ	0.511		$^{14}N(\gamma, n)^{13}N$ $^{14}N(n, 2n)^{13}N$ $^{16}O(p, \alpha)^{13}N$

Nuclide	Major Radiation Emitted		Half-life	Production Modes
	Type	Energy (MeV)		
C-14	β^-	0.157 (max)	5715 y	$^{13}C(n, \gamma)^{14}C$ $^{14}N(n, p)^{14}C$ $^{17}O(n, \alpha)^{14}C$
O-15	β^+	1.72 (max)	2.037 min	$^{12}C(\alpha, n)^{15}O$ $^{14}N(p, \gamma)^{15}O$
	γ	0.511		$^{16}O(\gamma, n)^{15}O$ $^{16}O(n, 2n)^{15}O$
N-16	β^-	4.27 (max) 10.44 (max)	7.13 s	$^{15}N(n, \gamma)^{16}N$ $^{16}O(n, p)^{16}N$ $^{19}F(n, \alpha)^{16}N$
	γ	6.129 7.115		
	α	1.85		
F-18	β^+	0.635 (max)	1.8295 h	$^{18}O(p, n)^{18}F$ $^{19}F(n, 2n)^{18}F$
	γ	0.511		$^{16}O(^3He, p)^{18}F$
Na-22	β^+	0.546 (max)	2.604 y	$^{19}F(\alpha, n)^{22}Na$ $^{23}Na(n, 2n)^{22}Na$
	γ	0.511 1.2745		
Na-24	β^-	1.391 (max)	14.95 h	$^{23}Na(n, \gamma)^{24}Na$ $^{24}Mg(n, p)^{24}Na$
	γ	1.3686 2.7540		$^{27}Al(n, \alpha)^{24}Na$
P-32	β^-	1.709 (max)	14.28 d	$^{31}P(n, \gamma)^{32}P$ $^{32}S(n, p)^{32}P$ $^{35}Cl(n, \alpha)^{32}P$
S-35	β^-	0.1674 (max)	87.2 d	$^{34}S(n, \gamma)^{35}S$ $^{35}Cl(n, p)^{35}S$
K-40	β^-	1.33	1.27×10^9 y	Naturally occurring
	γ	1.4608		
Ar-41	β^-	1.198 (max)	1.83 h	$^{40}Ar(n, \gamma)^{41}Ar$ $^{41}K(n, p)^{41}Ar$
	γ	1.2936		$^{44}Ca(n, \alpha)^{41}Ar$

Nuclide	Major Radiation Emitted		Half-life	Production Modes
	Type	Energy (MeV)		
Fe-55	e^-	0.000610 0.005190	2.73 y	$^{54}Fe(n, \gamma)^{55}Fe$ $^{58}Ni(n, \alpha)^{55}Fe$
Co-58	β^+	0.474 (max)	70.88 d	$^{57}Co(n, \gamma)^{58}Co$ $^{59}Co(n, 2n)^{58}Co$
	γ	0.8108		$^{58}Ni(n, p)^{58}Co$
Co-60	β^-	0.318 (max)	5.271 y	$^{59}Co(n, \gamma)^{60}Co$ $^{60}Ni(n, p)^{60}Co$
	γ	1.1732 1.3325		$^{63}Cu(n, \alpha)^{60}Co$
Zn-65	β^+	0.325 (max)	243.8 d	$^{64}Zn(n, \gamma)^{65}Zn$ $^{66}Zn(n, 2n)^{65}Zn$
	γ	1.1155		
Kr-85	β^-	0.687 (max)	10.76 y	Fission product $^{84}Kr(n, \gamma)^{85}Kr$
	γ	0.514		
Sr-90	β^-	0.546 (max)	28.78 y	Fission product $^{88}Sr + n \rightarrow {}^{89}Sr + n \rightarrow {}^{90}Sr$
Y-90	β^-	2.281 (max)	2.67 d	Fission product Sr-90 Daughter $^{89}Y(n, \gamma)^{90}Y$ $^{90}Zr(n, p)^{90}Y$ $^{93}Nb(n, \alpha)^{90}Y$
Tc-99m	e^-	0.0022 0.1427	6.01 h	$^{98}Tc(n, \gamma)^{99m}Tc$ $^{98}Mo(n, \gamma)^{99}Mo \xrightarrow{\beta^-} {}^{99m}Tc$
	γ	0.1405		
I-125	γ	0.03549	59.4 d	Fission product $^{124}Xe + n \rightarrow {}^{125}Xe \xrightarrow{\beta^+} {}^{125}I$
I-129	β^-	0.15 (max)	1.57×10^7 y	Fission product $^{128}Te + n \rightarrow {}^{129}Te \xrightarrow{\beta^-} {}^{129}I$
	γ	0.0396		
I-131	β^-	0.606 (max)	8.02 d	Fission product $^{130}Te + n \rightarrow {}^{131}Te \xrightarrow{\beta^-} {}^{131}I$
	γ	0.3645		

Nuclide	Major Radiation Emitted		Half-life	Production Modes
	Type	Energy (MeV)		
Xe-133	β^-	0.346 (max)	5.243 d	Fission product
	γ	0.08099		$^{132}Xe(n, \gamma)^{133}Xe$
Cs-137	β^-	0.514 (max)	30.07 y	Fission pproduct $^{136}Xe + n \rightarrow {}^{137}Xe \xrightarrow{\beta^-} {}^{137}Cs$
	γ	0.6617		
Tl-201	γ	0.1353 0.1674	3.039 d	$^{203}Tl(p, 3n)^{201}Pb \xrightarrow{\beta^+} {}^{201}Tl$
Pb-214	β^-	0.67 (max) 0.73 (max)	27 min	U-238 Decay series
	γ	0.242 0.2952 0.3519		
Bi-214	β^-	1.51 (max) 1.54 (max) 3.27 (max)	19.9 min	U-238 Decay series
	γ	0.6093 1.1203 1.7645		
	α	5.450 5.513		
Po-214	γ	0.799	163.7 μs	U-238 Decay series
	α	7.6869		
Po-218	γ	0.510	3.10 min	U-238 Decay series
	α	6.0024		
Rn-220	γ	0.5497	55.6 s	Th-232 Decay series
	α	6.2882		
Rn-222	γ	0.510	3.8235 d	U-238 Decay series
	α	5.4895		

Nuclide	Major Radiation Emitted		Half-life	Production Modes
	Type	Energy (MeV)		
Ra-226	γ	0.1862	1599 y	U-238 Decay sSeries
	α	4.602 4.7844		
Th-232	γ	0.06381 0.14088	1.4×10^{10} y	Naturally occurring
	α	3.947 4.012		
	SF[a]			
U-238	γ	0.0496	4.47×10^{9} y	Naturally occurring
	α	4.147 4.197		
	SF[a]			
Pu-239	γ	0.0516	2.41×10^{4} y	$^{238}\mathrm{U} + n \rightarrow {}^{239}\mathrm{U} \xrightarrow{\beta^-} {}^{239}\mathrm{Np} \xrightarrow{\beta^-} {}^{239}\mathrm{Pu}$
	α	5.105 5.144 5.156		
	SF[a]			
Am-241	γ	0.0595 0.0263–0.955	432.7 y	$^{239}\mathrm{Pu} + n \rightarrow {}^{240}\mathrm{Pu} + n \rightarrow {}^{241}\mathrm{Pu} \xrightarrow{\beta^-} {}^{241}\mathrm{Am}$
	α	5.4430 5.4857		
	SF[a]			
Cf-252	γ	0.0434 0.1002	2.646 y	Multiple neutron capture from a variety of nuclides (e.g., U-238, Pu-239, and Cm-244)
	α	6.0756 6.1181		
	SF[a]			

[a] Spontaneous Fission (SF) produces a variety of radiation types including fission fragments, neutrons, gamma rays, beta particles, positrons, neutrinos, and daughter radiations.

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Subject Index*

a
abdomen dose 109, 388
absolute risk 630 ff.
absorbed dose 78, 645
– neutrons 371
– point source approximation 565
absorbed fraction 110, 412, 583
absorption
– biological systems 278, 575
– HATM (blood) 617 f.
– ICRP 590, 613 ff.
– laser radiation 303
– tissues 545
– tritium 88
accelerators 237–272, 510
– activation 655
– beam containment 257
– physics 48
– radiation 516 ff.
– universities 97
accessible emission limit (AEL) 289
accuracy of instrument response 325
acoustic waves 283
actinides 123, 211
actinon 202
activated charcoal 475
activation 644
– accelerator/reactors 655
– air 252
– components 162
– hadronic cascade 247
– production equations 656
– soil 252
– sources 158, 573
– water 251
activation equation 162
activation products
– ALARA associated with fusion confinement methods 102
– effluents 171
– fusion energy research 99
– ITER 101
– power reactors 161, 165
active safety systems 158
activity 562, 644
– constant intake 478
– counting samples 92
– fuel cycle 142
– Kr-88 194, 470
– plutonium 144
– ponds/surfaces accumulation 208
– Rb-88 194, 468
– single-compartment model 579
activity balance relationship 164
activity median aerodynamic diameter (AMAD) 588
acute exposures 33, 108, 141, 358 f., 579, 607 f., 628
acute radiation syndrome 134
additive risk model 360
Adler's relationship 244
aerosol inhalation 588
age determination, C-14 method 311
age-specific cancer risk 631
agricultural/environmental research 93, 214
air
– gamma doses 211
– kerma 64, 80
– monitoring 143
– residual radioactivity 251
– samples 142
– samplers 10

* This index was developed by Wiley-VCH with limited author input.

– supply systems 90, 100
see also activation, derived air concentrations
air ionization chamber 12
airborne hazards 87 ff.
airborne radioactivity 390
Al-28 decay 500
ALARA 170
– associated with fusion confinement methods 102
– collective doses 185
– muon colliders 250
– plasma-confinement 102
– reactor accidents 180
– spent-fuel pool 190
alarm setpoint 190
algebraic equations 661
alignment error 257
alimentary tract 618
alpha activity 321
alpha contamination 106, 143
alpha decay, U/Th series 121, 202 f.
alpha emitters 53 f., 579
alpha energy 11, 120
alpha particles 595
– hadronic cascade 247
– radiation weighting factors 602
alpha track detectors 475
alternate MIRD equation 584
aluminum 20
alveolar–interstitial region 612
Am-241 data 106, 410, 677
American Conference of Government Industrial Hygienists (ACGIH) 284
American National Standard for Dosimetry (ANSI) 13
– antennas 525
– microwave radiation 552
– Z136.1-2007 284–302, 527–542
anatomical model 577
angiography 75
annual doses 217 f., 358
annual effective dose equivalent 201
annual equivalent dose 609
annual exposure 459
annual gamma air dose 217
annual limit on intake (ALI) 7, 444, 473, 639
– Cs-137/Ba-137m 351
– uranium cycle 127
annual meteorological conditions 207
annual neutrino effective dose 248
annual occupational organ equivalent limit 600
annual organ dose
– ground surface 220
– ingestion 221
– inhalation 221
annual public dose 96
annual skin dose 219 f.
annual total body dose equivalent 218 f.
antennas 275, 523 f.
anticipated fusion facility structural materials 99
antihadron probes 56
antineutrino exposures 315
antiparticles 238, 245
aperture 276, 286, 556
aquatic foods 213
Ar-41 data 96, 674
area radiation monitor (ARM) 190
argon laser 282, 532, 541 ff.
atmospheric dispersion 138 ff., 179, 193, 667
atmospheric spallation 673
atmospheric stability 138
atmospheric transport 477
atomic bomb fallout 485
atomic vapor laser isotope separation (AVLIS) 130, 147
attenuation 341, 505, 644
– biological systems 278
– brachytherapy 57
– equations 565–572, 663
– lead 181
– leakage 63
– medical accelerators 48
– neutrons 440
– shielding 255, 570
– Tc-99m gammas 75
– X-ray beams 51
Au-197 114, 423
average absorbed dose 596
average airborne concentration 472
average number of working level months (AWLM) 205
average power density 276
average power MPE 534

b

Ba-137m data 27, 348
background counting 318
background radiation 273, 485
barium contrast materials 51
barriers 60 ff.
basal cell layer 359
basic solutions 89
basic source geometries 565–572

basics 639–652
Be-7 data 251, 673
beam attenuation 293
see also attenuation
beam current 258, 268
beam dump configuration 510
beam leakage 97
beam restriction device 58
beam therapy 47
beam tubes 95
beam-path control 245
BEIR risk models 34 f., 207, 627–638
bending magnets 243
Bergonie–Tribondeau law 30
beryllium reflectors 96
beta doses 325
– effluents 211
– noble gas releases 218
– uranium 121
beta radiation 53 f., 511
– AVLIS 131
– fuel cycle 119
– fusion energy research 99
– hadronic cascade 247
– MIRD schema 582
– shielding 422
– single-compartment model 579
– U/Th series 202 f.
– uranium 121
– waste 133
beta–gamma survey instrument 12
betatrons 519
BF_3 proportional counter 505
Bi-214 11, 324, 676
binary proton reactions 240
bioaccumulation factor 498
bioassay counting 110
biokinetic model 618
biological damage
– antennas 279
– cells 586
– eyes 169, 284, 297
– forebrain 110
– ionizing radiation 29, 33
– laser radiation 282 f., 301 f., 532
– medical accelerators 48
– neutrons 370
– radium 47
– risk models 627 ff.
– skin 284, 297
– triated water 29
biological half-life 491
biological waste treatment 175
blackbody radiation 273, 531
blanket shield 100
blasting 174
blistering 471
blowdown monitors 177, 192
boiling-water reactors (BWRs) 158, 163, 231
bone marrow 48
borated concrete 101
boron 137
boron neutron capture therapy (BNCT) 50
bounding neutrino effective dose 248
brachytherapy 47, 53, 57, 71
brain tumors 50
braking radiation 240
breath analysis 88
breathing rate 25
– fuel cycle 151
– radon 476
– reference man model 369
bremsstrahlung 386, 502, 511
– accelerators 237–244, 519
– gaseous effluent 216
bronchial region 612
buildup factors 341, 570
burial, radioactive waste 173

c

C-11 data 251, 503 ff., 673
C-14 674
– age determination 311
– biomedical tracers 88
– decay modes 20
– dose factors 330
– effluents 171
– glucose compound labeling 72
– uptake 88
cadmium 137
calculus, internal dosimetry 670
cancer 28
– BEIR VII report 34
– ICRP 594, 601
– internal dosimetry 585
– radiotherapy 48
– radon 204, 209
– risk models 355, 627–638
– UV light exposure 551
capture gamma sources 575
capture therapy 50
carbon dating 4
carbon dioxide
– blasting 174
– lasers 555
– total committed effective dose equivalent 379

catalytic extraction 175
cataract formation 30
cell survival curves 31
cellulose filters 320
centrifuge 125, 127 ff.
cephalocaudal orientation 73
CERN shielding 256
Cf-252 data 88 f., 677
chain reactions 133
characteristics
– electromagnetic waves 274
– laser radiation 282
– radionuclides 673 ff.
charcoal filters 103, 324
chemical compounds 53, 127
chemical fuel cycle form 141
chemical processing
– radioactive waste 132
– uranium ore 124 f.
chemical toxicity 124, 141
chest image room shielding 79
chromosome aberrations 38, 370
chronic exposure
– fuel cycle 141
– laser radiation 283
– university health physics 107
– uranium cycle 127
circular muon collider 248
cladding failures 95 f., 163
class W compounds 127
cleanup 391
clearance functions 670
clearance half-times 589
closed fuel cycles 211
closed window beta/gamma measurements 325
cloud model, semi-infinite 178, 216
CO_2 laser 532
Co-56 256
Co-58 675
– accelerator shielding 256
– DACs 196
– outages 176
Co-60 675
– accelerator shielding 256
– activity 372, 413
– DACs 196
– decay modes 20
– demineralizer 455
– dose factor 183, 330, 369
– exposure 89
– half-life 39
– intake 473
– irradiation sterilizers 21
– neutron activation 459
– outages 176
– primary coolant 181 ff.
– radiation damage 88
– teletherapy 53
Code of Federal Regulations (CFR) 289, 533
– 10 CFR 20 120, 578
– 10 CFR 50 211
– 10 CFR 835 120, 578
– 42 CFR 81 637
– 42 CFR 82 637
– 42 CFR 83 637
collective doses 159, 604
colon transfer coefficients 619
committed dose equivalent (CDE) 26, 639
– 50 years 646
– I-125 427 f.
– ICRP 587
– radon 205
committed effective dose 367
committed effective dose equivalent (CEDE) 26, 344, 348, 367, 391
– brachytherapy 58
– Co-60 inhalation 372
– ICRP 597
– radon 208
– risk models 639, 645
– Ru-106 490
compaction 173
compartment fractions 589
components replacement 99
Compton scattering 314, 369, 501
computed tomography 52
concrete
– accelerators 268 f.
– attenuation 71, 376
– chest image room 80
– fuel cycle 137
– gamma rays 340
– initial fusion power facility 101
– shielding 19 ff., 255, 503
– X-rays 61
condenser air ejector 177
conductivity 278
confinement 100 ff.
constitutive equations 648
constraints, ICRP 599
contact dose rates 95
containment building 100, 161, 466
contamination
– control 171, 390
– environmental 224, 482
– fuel cycle 138
– hospitals 399

– power reactors 161, 164
– skin 458
– hot particle 183
continuous air monitor (CAM) 143
continuous samples 475
continuous wave emitters 274
continuous wave lasers 291 ff., 541
contrast diagnostic materials 51
conversion factor
– counting samples 92
– internal dosimetry 577
– muon colliders 249
– radon 205
coolant systems 96 ff., 157, 161, 176
cornea 283, 532, 555
correction factors 299, 533
corrosion products 176 f.
cosmic radiation 230, 485, 673
Coulomb barrier 240, 244
count rate 622
counting efficiency 91
counting room 9
cow's milk pathway 488
critical energy 243
critical mass 135
criticality
– fuel cycle 133, 138
– point source 441
cross-wind distance 138
Cs-137 348, 676
– brachytherapy 377
– closed fuel cycles 211
– decay scheme 26 f.
– dose factors 330
– gamma rays 408
– gynecological implants 71
– intake 448
– release 497
Cu-64 256
cumulated activity 582
cycle time 259
cyclotrons 53, 237, 519

d

D_2O systems 96
damage
– eyes 169
– food irradiation 89
– forebrain 110
– fuel 177
– laser radiation 282 f., 301 f.
– materials 94, 101 f., 178
– triated water 29
see also biological damage, material damage
daughter radioisotopes 5
DCF 436
D–D fusion process 102
death
– BEIR V 34
– collective dose 604
– IRCP 594
– radon 207 f., 229
decay equations 663
decay modes 20, 573
– active 352
– Cs-137 27
– low-energy elementary particles 238
– Mo-99 4
– U/Th series 202 f.
decay products 119, 142
decontamination 172
decontamination agents 472
decontamination blasting 174
decontamination technologies 173
decontamination/decommissioning (D&D) 336
deep dose equivalent (DDE) 344, 447
demineralizer 161, 655 f.
– Co-60 455
– power reactors 161
– pressurized water reactor 181
– production equations 657
– radioactive waste 173
Department of Energy (DOE) 120, 448
Department of Health and Human Services (HHS) 28
deposition pathways 488, 578 f.
depth dose profiles 49, 546
derivatives 671
derived air concentration (DAC) 639, 646
– Co intake 474
– Cs-137/Ba-137m 351
– exposure time 430
– IRCP-26 587
– power reactors 196
desquamation 358, 471
detectors 103, 331 ff.
– accelerators 259
– fuel cycle 142
– research reactors 103
deterministic effects 604
detriment, ICRP 594
deuterium 99 f.
deuteron acceleration 518
dewatering 175
diabetics liver metabolism 72
diagnostic exposure 344
diagnostic nuclear medicine 51

diagnostic protocols 79
diffuse reflections 286 f., 290
diffusion cascade 125
dioctylphthalate 91
diodes 280 f.
dipoles 280 f.
direct doses 400
direct exposure 161, 296 ff.
discharge parameters 494
disk source approximation 462, 568
– doses 645
– power reactors 157, 163
– primary system decontamination 189
dispersion 647
– gas 138 ff.
– particulates 138 ff.
– radon 208
disposal methods 228, 443
dissociation, UF_6 130
divergence 48, 286
door interlocks 293
dose calculations 592, 618, 621
dose constraint 605 ff.
dose conversion factors 345
– I-125 115
– neutrons 505
dose equivalent 339, 377
– accelerators 244, 258
– aquatic foods 213 ff.
– potable water 212
dose factors 330, 645
– point source 182
see also gamma constant
dose fractionation sequences 359
dose limits 31, 607
– effluent pathways 211
– ICRP 600
dose rate
– equations 179, 398
– point source approximation 565
– power reactors 163
dose rate conversion factor (DRCF) 179
dose rate effectiveness factor 604, 634
dose response relationships 630 ff.
doses 644
– atmospherically released radionuclides 220
– constraints 605 ff.
– deposition 6, 55
– effluents 212, 216
– environmentals 201
– risk models 630
– therapeutic 53
dosimeter configurations 365
dosimetry 645 ff.
– gastrointestinal tract model 591
– ICRP 613, 618
– X-ray diffraction 97
doubling dose 635
drilling and spalling decontamination technique 175
dry application ice 175
dry desquamation 471
D–T fusion process 102
duty cycles 548
duty factor 644
– accelerators 259
– nonionizing radiation 274

e

ear's hot particles dose 169
effective dose 367, 604, 646
– ICRP 596, 609
– muon colliders 248
– therapy patients 69
– university health physics 110
effective half-life 647
effective release 451, 492
effective removal rate 647
efficiency calculations 403
effluents 171, 488
– isotopes 132
– pathways 211
– releases 201 f., 208
elastic scattering
– proton accelerators 239
– shielding 255
– tissue dose 369
electret (E-perms) 475
electric fields 288
electromagnetic antennas 274
electromagnetic cascade 240 ff.
electromagnetic energy 288
electromagnetic fields measurement 280
electromagnetic force 552
electromagnetic cascade 247
electromagnetic radiation 521
electromagnetic relationships 648
electromagnetic waves 275
electron accelerators 240, 503, 519
electron backscatter 49
electron capture 6, 99
electron linear accelerators (linacs) 48, 237, 261
electrons
– density 48
– energy range 595
– low-energy elementary particles 238

– radiation weighting factors 602
– rest mass 7
embryo 108
see also pregnancy, fetal exposure
emergency actions, environmental 224
emergency core cooling systems (ECCS) 180
emergency exposures 605, 610 ff.
emergency operations facility (EOF) 192
emitter characteristics 522
energy absorption 544
energy deposition 6
energy employees occupational illness compensation program act (EEOICPA) 637
energy loss 7, 278
engineering controls 87, 90, 293
enrichment
– fissile isotopes 136
– fuel cycle 119, 141
– laser radiation 302
– uranium cycle 125 f.
environmental health physics 201–236
environmental monitoring programs 207
environmental protection 611 ff., 616
Environmental Protection Agency (EPA) 207
epidemiology 629
epidermis 532
epilation 471
epithelium 283
epithermal neutrons 50
equilibrium 5, 476
equivalent dose 366, 595
see also annual equivalent dose
errors 12
– accelerator beam containment 257
– memory effect 325
erythema
– acute radiation exposure 358
– skin irradiation 471
– UV light exposure 551
esophagus transfer coefficients 619
evacuation
– environmental health physics 224
– ICRP 610
excess relative risk (ERR) 630
excimer lasers 555
excitation energy 55
exhaust systems 90
exponential functions 668
exposure 170, 488
– Co-60 373
– environmental health physics 201, 224
– fuel cycle 141 ff.
– ICRP 594
– laser radiation 282, 287, 292 f., 301
– microwave heating 521
– nonionizing radiation 279
– photon beamline 386
– planned 604
– power reactors 160
– pregnancy 368
– pulsed laser 527
– risk models 627–638
– university health physics 107
– uranium cycle 127
– UV generating equipment 550
exposure rate
– angiography 75
– brachytherapy 57
– disk source approximation 568
– point source approximation 565
– power reactors 166 ff.
– primary system decontamination 188
– shielding 60
– X-rays 61, 97
external beam therapy 55
external dosimetry 644
extranuclear cascade 247
eyes
– annual dose limit 358, 609
– doses 328, 609
– hot particles 169
– laser radiation 283, 294, 532
– P-32 protection 88
– UV light exposure 551

f

F-18 data 674
facility design 58
facility workload 59 f.
family members 69, 390
far infrared laser radiation 287, 555
far-field distance 523 f.
far-field power densities 277
fast neutron therapy (FNT) 50
fast neutron tissue doses 313, 369
FDA laser classes 530
Fe-55 data 675
Fe-59 256
federal regulations 289
feedwater 158
feet, annual equivalent dose 609
fertile material 157
fetal exposure 389, 410
fifty year deposition period 587, 592
film contrast 51
filtered viewing devices 293

filters
– activity 161, 194
– collection efficiency 10
– degradation 465
– reactors 103, 180
– university health physics 91
fire hazard 289
fish 481
fissile materials 135, 157
fission gamma sources 575
fission neutrons 369
fission products 675
– barriers 466
– energy range 595
– power reactors 161, 165
– radiation weighting factors 602
– radioactive waste 132
– radionuclides effluents 172
– research reactors 96
– uranium fuel cycle 121
fluence buildup factors 21
fluence per unit dose equivalent 371
fluorine compounds, uranium 141
fluoroscopy 51
flux-to-dose conversion factor 19
food
– environmental monitoring 481
– irradiation experiments 89
– radionuclide ingestion 221
Food and Drug Administration (FDA) 289, 532
foodstuffs 610
fractional absorption 620
free electron lasers (FEL) 288, 243
free-standing stacks ≥80 meters 218
fuel
– power reactors 160
– radioactive waste 132
– repositories 123
– research reactor 96
fuel cycle 121
fuel cycle facilities 140
fuel cycle health physics 119–156
fuel element cladding 161 ff., 466
fuel failures 159 ff.
fuel-handling accidents (FHAs) 178 f.
fuel reprocessing 123, 131 ff.
fusion energy research 98
fusion process types 102

g
GaAs laser 282, 542
gamma activation sources 574
gamma activity 403
– diagnostic radionuclides 52
– radioactive waste 133
gamma constant
– Co-60 188
see also dose factor 645
gamma discrimination 505
gamma dose 439
– demineralizer 455
– effluent pathways 211
– noble gas releases 217
gamma dose equivalent rate 423
gamma isotopes 89
gamma radiation
– biological effectiveness 370
– Cs-137 314
– delayed 511
– emission rate 6
– fuel cycle health physics 119
– fusion energy research 99
– radium 71
– shielding 335 f.
– sources 573 ff.
gas centrifuges 125 ff.
gas channel monitors 486
gas-cooled fast reactors (GFRs) 123, 159
gas-filled detectors 17
gas generation
– radiolytic 415
– toxic 253
gas ionization chambers 331
gaseous diffusion 125
gaseous effluents 96, 216, 488
gaseous products 503
gaseous radioactive waste 132
gaseous releases 94
gastrointestinal (GI) tract 352
– hot particles 169
– ICRP 590 ff.
– internal dosimetry 578
Gaussian beam 286, 651
Gaussian dispersion model 233
Gaussian plume model 138, 450 f., 491
Ge(Li) detector 103, 143
Geiger-Mueller thin-window detector 142
generation I–IV reactors 123, 158 ff.
geologic repository 210
geometric relationships 136, 665
geometry factor 325
germanium detector 409
giant resonance 241
glass fiber filters 320
glioblastoma 50
glucose 378
gold foil activation analysis 114

gold vapor laser 555
grab samples 475
graphite 137
graphite moderated thermal reactors (VHTRs) 123
graphite reflectors 96
gravitational settling 140
greases, activation 251
green salt 124
grit blasting 174
ground temperature 493
ground-level concentration 138
groundwater 480
gypsum transmission curve 382
gypsum wallboard 80

h

H_2 laser 282
H-3 673
– biomedical tracer studies 88
– decay modes 20
– dose factors 330
– effluents 171
– uptake 88
– water/soil activation 251
hadron collider 245 ff.
hadronic cascade 247
hafnium 137, 575
half-life 8, 11, 24, 573
– air activation products 252
– biological 491
– Co-60 188
– diagnostic radionuclides 52 ff.
– internal dosimetry 580
– physical 491
– radionuclides 673 ff.
– research reactors 104
– Th series 202 f.
– thyroid dose 345
– U series 202 f.
half-value layer (HVL)
– fetal exposure 410
– leakage radiation 66
– NCRP-155 376
– radionuclide therapy 69
hands, annual equivalent dose 609
health physics 3–86
– accelerators 237–272
– basic concepts 639–652
– environmental 201–236
– fuel cycles 119–156
– medical 47–86
– nonionizing radiation 273–308
– power reactors 157–200
– program elements 170
– university/research 87–118
healthy tissue irradiation 54
heat energy absorption 547
heat exchange systems 99, 158
heating mechanisms 279
heavy ion accelerators 244
heavy ion therapy 51, 56
heavy ions
– human risk 359
– radiation weighting factors 602
– water 56
heavy water systems 96
HeCd laser 282
HeNe laser 298, 532
HF laser 282
Hg-203 activity 405
high-efficiency particulate air (HEPA) filter 154
high-energy interactions 237
high-energy elementary particles 359
high-level waste 133, 210 f.
high-pressure washing 174
high-temperature helium-cooled graphite moderated thermal reactors 159
high-voltage power supplies 253
high-volume air sampler 10
hospital staff, NCRP-155 dose recommendations 70, 401 f.
hot particles 457, 465
– absorbed dose 645
– activated corrosion product 165
– personnel contamination 183
– skin dose 161, 165
household water supply, radon 206
HTO data 406 f., 421
see also tritium oxide
human alimentary tract model (HATM) 601, 616, 620
human respiratory tract model (HRTM) 601, 612
hydraulic cutter 190
hydroblasting 174
hydrogen
– capture sources 575
– fusion energy research 99
– radiolytic decomposition of water 414
– university health physics 111
– uranium fuel cycle 124

i

I-125 89, 675
– brachytherapy 377
– dose conversion factors 115

– dosimetry 421
– intake relationships 406
– medical research/therapeutic treatments 88
– thyroid 407, 413
– uptake 110
I-129 data 675
I-131 89, 675
– committed dose equivalent 444
– gamma constant 389
– medical research/therapeutic treatments 88
– optimum bioassay approach 360
– oral therapeutic dose 77
– reference man model 369
– thyroid dose 82, 467
ICRP methodology 391, 584–620, 670
– counting samples 92
– fuel cycles 140
– organ doses 413
– radon 205
ICRP-2/10, internal dosimetry 578
ICRP-23 Reference Man 369, 546, 591, 603
ICRP-26 23 f.
– brachytherapy 58
– dose limits 586
– risk coefficient 355, 627 ff.
– weighting factors 39
ICRP-30
– dose limits 586
– gastrointestial tract parameters 28
– ingestion models 617
– methodology 587 ff.
ICRP-49, pregnancy 412
ICRP-60 36, 460
– doses 367
– fuel cycle 120
– risk models 628
ICRP-66 596–614
ICRP-100
– human alimentary tract 613
– ingestion models 617
ICRP-89 603, 614
ICRP-103
– environmental health physics 225
– fuel cycle 120
– risk models 628
– system of protection 604
imaging equipment 59
immersion exposure 488
immersion model 216
in vivo counting 485
incident electron beam 240
incident report 151, 420
Inconel capture sources 575
indoor radon measurements 226
induction accelerators 288
inelastic scattering 239, 314
inertial confinement 102
ingestion
– aquatic foods 213
– ICRP-30 590, 617
– ICRP-100 600, 617
– radionuclides in food 221
inhalation 488
– diagnostic medicine 51
– fuel cycle 141
– ICRP 588 ff.
– plutonium 144
– power reactors 196
– radioactive material 655
– radon 204
– tritium 88
– uranium 146
initial dose equivalent rate 92, 579
initial fusion power facility 100
injections 392, 578
inleakage 206
insect sterilization 93
instantaneous grab samples 475
intake 92, 578, 593
– activity 435
– C-14 88
– Co-60 372, 473
– fractions 39
– ICRP 392
– ICRP-30/100 621
– H-3 103
– MIRD 392
integrals 670
integrated samples 475
intentional dispersal of radioactive materials (IND) 222
interaction properties 273
internal dosimetry 577–626, 645 ff.
– hospitals 400
– ICRP 391
– MIRD 391
internal intake *see* intake
International Commission on Radiological Protection (ICRP) 23
internal dose assessment models 577
International Thermonuclear Experimental Reactor (ITER) 100
intervention 599
intestine 352 f., 590, 619
intrabeam exposures 285
intranuclear cascade 247

iodine
– contrast materials 51
– ICRP 610
– partitioning factor 193
– reactor accidents 180
– thyroid 89, 115
see also I-125, I-129, I-131
ion exchange demineralization 161, 173
ion exchange production equations 657
ion pairs 6
ionization chamber 9, 387, 645
ionization detector 142
ionizing radiation 92, 438 f.
– ALARA principle 554
– electron accelerators 241
– ICRP-60/66 594
– risk coefficients 628
iron capture sources 575
iron components 251, 255
irradiance 294, 651
irrigated foods 214 f.
isotopes
– activation products 99
– air concentrations 446
– production 94
– research reactors 103
– uranium fuel cycle 125, 128
ITER general radiological characteristics 101

j

Joule heating 521
– biological systems 278
– microwave radiation 553 f.

k

K-40 data 230, 343, 485, 674
kaons 245
kerma 64, 80
key health physics relationships 639–652
key switch 293
klystrons 253, 274
Kr-85 487, 675
– effluents 171
– environmental release 163, 230
Kr-88 data 194, 470
krypton laser 282, 555

l

Large Hadron Collider (LHC) 140, 245 ff.
laser basics 281–289, 304, 530 ff.
laser isotope separation 128
lead
– attenuation coefficients 181
– chest image room 80
– point source dose buildup factors 19
– shielding 61, 396
lead-bismuth cooled fast reactors (LFRs) 123, 159
leakage 63, 68
– chest image room 80
– facility design 58 ff.
– primary coolant system 164
– radioactive fluids 655
lens (eye) 283, 532
letdown system 176, 192
lethal range doses 237
leukemia 631
Li-6-based dosimeters 327
LiF thermoluminescent dosimeter 36
life-saving, ICRP 610
lifetime 8
– low-energy elementary particles 238
lifetime cancer risk 633
light amplification 281
light sources 243
light water reactors 96, 111
limiting aperture 292
line source approximation 461, 567, 668
– Co-60 414
– doses 644
– power reactors 157, 163
– primary systems 189
linear accelerator (LINAC) 53, 265, 288, 502
linear attenuation coefficient 21, 341, 376
linear differential equations 653
linear energy transfer (LET) 237, 369
linear equations 661
linear muon collider 248
linear no threshold (LNT) model 601
liquid effluents 97, 488
liquid metal coolants 99
liquid scintillation counter 422
logarithm functions 668
Lorentz factor 242, 288
loss of coolant accidents (LOCAs) 178 f.
low-energy elementary particles 238
lower large intestine (LLI) 590
lower limit of detectability (LLD) 416
low-pressure liquid chromatography (LPLC) 115
low-pressure wet treatment 174
Lucite 421
luminous transmission 529
lung model 588, 614 f.
– ICRP-30 590, 617
– ICRP-66 596–614

lymph nodes 588
lymphocytes 38

m
macroscopic reaction cross-sections 55
magnet failure 257
magnetic confinement system 100 f.
magnetic fields 280, 521, 554
maintenance outage 196
mammography 51, 73
mass, low-energy elementary particles 238
mass attenuation coefficients 19, 71, 343
mass defects 314
mass energy absorption coefficient 19
mass median aerodynamic diameter (MMAD) 588
material characteristics, uranium fuel cycle 141
material damage
– Co-60 activity 415
– research reactors 94
see also damage
material removal 591
material research 97 f.
material translocation 587
mathematical review 661–672
maximum allowed occupational concentration limit (MAOC) 355
maximum dose equivalent 25
maximum permissible exposure (MPE) 529, 534, 555 f., 640
– laser radiation 284–299
– nonionizing radiation 279
– X-ray shielding 61
maximum survivable radiotherapy exposures 48
Maxwell equations 650
mean absorbed dose (MIRD) 647
mean quality factors, neutrons 371
mechanical shock wave 283
mechanics relationships 650
medical accelerator physics 48
medical exposure 605
medical health physics 47–86
medical isotope generators 112
mediolateral orientation 73
melanin pigment granules 533
members of the public 69
membrane filters 320
metabolic models 587
microtrons 288
microwave radiation 273, 521
milling 132, 175
minimum critical parameters 136
minimum detectable amount (MDA) 233
mining 124, 132
MIRD
– internal dosimetry 578, 582, 592
– integrals 670
– methodologies 391
mitigation
– environmental release 201, 224, 484
– radon 206
– reactor accidents 180
– human risk 359
mixed reflection 286
MLIS laser isotope separation 130
Mn-54 data 20, 256
Mo-99 384 f.
– activity 312
– generator 4, 75, 112
– package 418
model parameters, environmental 222
moderator materials 96, 136
moist desquamation 471
molten salt epithermal reactors (MSRs) 123, 159
momentum conservation 245
monochromatic emissive power 273
mouth transfer coefficients 619
multiple exit channel particles 240
multiple gamma rays 566, 571
multiple organ failure 134
multiple pulse MPE 535
multiplicative risk model 360
muon colliders 237, 248 f.
muons 246 f.
– electron acceleration 241, 519
– energy range 595
– hadronic cascade 247
– large hadron collider 245
– low-energy elementary particles 238
– particle accelerators 97
– penetrating radiations 499
– radiation weighting factors 602
mutations 359

n
N-13 data 251, 503, 673
N-16 data 103, 488, 575, 674
Na-22 data 251, 256, 674
Na-24 data 20, 96, 674
NaI detector 408, 507
NaI-125 labeling 113
nasopharyngeal region 588, 612
natural radiation 230, 273
naturally occurring radioactive material (NORM) 201, 610

Nb-95 dose factor 183
NCRP 375, 393
– NCRP-49 61, 68
– NCRP-84 31
– NCRP-91 31 f., 412
– NCRP-93 201
– NCRP-116 32, 36
– NCRP-130 169
– NCRP-147 64
– NCRP-151 66
– NCRP-155 69, 376
Nd-YAG laser 532 f.
near infrared laser radiation 287
near-field maximum power density 276
necrosis 471
needle scaling 174
negative temperature gradient 493
neptunium 127
net radiation impact, environmental 227
neutrinos 237 f., 248 f.
neutron absorbers 137
neutron activation 94, 190, 500
neutron bubble detectors 329
neutron capture cross-sections 137
neutron dose equivalent rate 258, 567
neutron doses 439
neutron emission strength 18
neutron energy loss 255
neutron fluence 162, 165
neutron induced gamma radiation 573 ff.
neutron therapies 50
neutron track 15
neutrons 5, 246
– biological effectiveness 370
– electron accelerators 240 f.
– energy range 595
– fuel cycle 119, 136
– heavy-ion accelerators 244
– human risk 359
– ITER 101
– penetrating radiations 499
– proton accelerators 239
– quality factor 329
– radiation weighting factors 602
nitrogen 266
nitrogen laser 555
noble gases 96, 180, 211, 216
nominal hazard zone (NHZ) 287, 527, 651
nominal ocular hazard distance (NOHD) 286, 527–557, 640, 651
nonfatal cancer 594
nonionizing radiation (NIR) 273–308, 651
nonstochastic effects 585
nuclear fuel cycle 119 ff.
nuclear medicine 17, 51 ff.
nuclear power plants 18, 459
Nuclear Regulatory Commission (NRC)
– effluent pathways 211
– fuel cycle health physics 120
nuclear track detectors 475
nucleon evaporation 247
nucleon–antiproton interactions 245
number of nuclear transformations 27
number of working level months (NWLM) 204

o

O-15 data 251, 503, 674
occupancy factor
– mammographic suite 74
– radon 204
– shielding 59 f.
occupational annual dose equivalents 32
occupational cumulative effective dose 32
occupational dose 603
occupational exposures 120, 597–610
Occupational Safety and Health Administration (OSHA) 289, 532
ocular exposure (MPE) 284, 537, 541
see also eyes
ocular transmission 305
Ohm's law 649
oil's residual radioactivity at accelerators 251
open and closed fuel cycles 123, 210
open window measurements 325
operating temperature 160
operational characteristics 94
optical cavity 282
optical density 294 ff., 528, 652
oral ingestion 51
ore mining 119
ore processing 119
organ burden 58
organ damage 521
organ doses 52, 433, 587, 621
outages 176 f.
ovaries dose 109
oxygen nucleus–pion reaction 260
oxygen, radiolytic decomposition of water 414
ozone 266 f., 512

p

P-32 406, 421, 674
– beta shielding 422
– biochemical labeling 88
– university health physics 105

– uptake 88
Pa-234m data 119
pair annihilation 53
parabolic antenna 276
particle accelerators 97
see also accelerators
particles
– ICRP 613
– in initial/transformed state (PIS/PTS) 615
– MIRD schema 582
– production 237 f., 258
particles fluence 55
particulates
– continuous source 140
– environmental monitoring 480
– ITER 101
– reactors 96, 165, 180
Pasquill–Gifford equation 138, 647
Pasquill stability class 193
passive environmental radon monitors (PERMs) 324
pathway selection, environmental 207, 222
patient doses 69, 400
Pb-214 11, 324, 676
penetrating radiation 55, 499, 546
permanent relocation 610
permeability 278
permittivity 278
personal dose equivalent 603
personnel-access structures 510
personnel doses 14, 256
personnel errors 257
personnel monitoring 17, 170
personnel protective equipment 294
photochemical hazards 534
photochemical laser radiation 282, 287
photoelectric effect 51
photoionization 129
photometric terms/units 281
photons
– accelerators 48, 251
– attenuation equations 663
– beam scattering 22
– bremsstrahlung 240, 515
– energy range 595
– exposure rate 386
– gaseous effluent 216
– light sources 243
– MIRD schema 582
– penetrating radiations 499
– radiation weighting factors 602
– synchrotron radiation 242
physical form, fuel cycle 141
physical sampling 464
pigments 283
pions
– accelerators 97, 241, 259, 519
– capture 499
– large hadron collider 245
– low-energy elementary particles 238
– radiation weighting factors 602
planned special exposures 120
plant management, ALARA support 171
plasma parameters 100 ff.
plexiglass 421
plutonium 121, 127, 143, 154
Po-210 113, 419
Po-214 11, 324, 676
Po-218 11, 324, 676
point source approximation 339, 440 f., 460, 565
– attenuation 505
– doses 19, 182, 644
– power reactors 157, 163, 189
– transmission 514
poison material 137
polyethylene 137, 149
pond water concentration 232
positron emission tomography (PET) 52, 56, 237
positron emitters 507
positron–electron pair annihilation 53
positrons 99
– MIRD schema 582
potable water, effluent pathways 212
potential fusion reactions 98
power densities 276
power intensities 242
power reactor containment 329
power reactors 157–200
Poynting vector 274, 649
practices, ICRP 599
pregnancy 368, 388, 410 f.
– I-125 428
– university health physics 108
– X-ray beams 76
premature aging 551
pressurized-water reactors (PWRs) 158, 165
primary air kerma 395
primary barriers 61 ff., 67, 396
primary beams 237
primary component maintenance 161
primary coolants 163, 181
primary radiation sources 58
primary systems 176–188
probability of causation (PC) 28, 635

process line rupture 151
process requirements 134
processing radioactive waste 173
production equations 209, 653–660
production scale facilities 100
production terms 573
propagation of RF and laser radiation 274, 282
Protective Action Recommendations (PAR) 466
protection standards 607 ff.
– electromagnetic fields 280
– environmental 224 ff.
– nonionizing radiation 279
protective agents 359
protective barriers 60
protective clothing
– hands/face 470
– hospitals 400
– laser radiation 294
– power reactors 167 f., 195
– UV light exposure 551
protective housing 293
proton acceleration 51, 239, 507, 518
proton reactions 245
proton recoil badge 327
protons
– energy range 595
– large hadron collider 245
– radiation weighting factors 602
Pu-238 intake 448
Pu-239 677
– activity 321, 452 f.
– release 10
– series 210
Pu-241 beta decay 410
public effective dose 600–610
pulmonary lung 436
pulmonary region 588, 612
pulse repetition frequency 292, 548
pulsed emitters 274
pulsed laser output 291
pulsed radiation fields 259
pulsing in research reactors 95

q

qualified expert certifications 398
quality factors 369 ff.
quantum mechanics 652
Q-value 7, 99

r

Ra-226 data 508, 677
rabbit transfer system 94, 114
radar systems 294
radiant exposure 285, 652
radiant parameters 526
radiation 170
– brachytherapy 57
– Co-60 181
– damage studies 89
– fuel cycle 119, 142
– ICRP 601
– medical use 47 ff.
– neutrinos 250
– oncology 48
– protection 3 ff., 399 ff.
– sources 253
– types 239
– weighting factors 601
radiation fields
– accelerators 250 f.
– disk source geometry 568
– line source geometry 567
radiation instrumentation systems 176
radiation monitoring 176
radiation risk 627–638
radiation work permit (RWP) 144, 195
radiation-induced mutations 359
radioactive dust 101
radioactive gases
– accelerators 263
– continuous source 138
– irradiation cell 253
radioactive implants 48
radioactive particulates 140
radioactive plume 655
radioactive release 87
radioactive waste 132, 151, 172
– disposal 120, 228
– processing 161 f.
radiobiological factors 359
radiofrequency accelerators 288
radiofrequency radiation sources 253, 273
radioiodine 96
see also I-125, I-129, I-131
radioisotopes 5 ff.
– accelerators 237
– administration 53
– I-126 7
radiological characteristics 101, 115
radiological dispersal device (RDD) 222
radiological hazard 141, 178
radiolytic decomposition of water 414
radiolytic gas generation 415
radiolytic hydrogen 111
radiometric terms/units 281
radionuclides 324, 673–678

– accelerators 237, 251 ff.
– administration 53
– environmental release 210
– high-level waste 152
– hot particles 166
– nuclear medicine 52
– potable water 212
– research 87, 96
– shoreline deposits 214
– sources 58
– station personnel 183
– therapy 47
radiopharmaceuticals 51, 75, 582
radioresilient tissues 33
radiotherapy 47
radium 71
radon
– decay 321, 475 f.
– environmental health physics 201 ff., 226
– fuel cycle 119
– ICRP 611
– risk assessments 207
– series 142, 476 f.
Rb-88 data 194, 468
reaction geometry, ALARA 102
reactors
– accidents 178 ff.
– activation 162, 655
– coolant system 111, 162 ff., 466
– effluents 96
– fusion energy 99
– material enrichment 119
– power 160
– radioactive waste 132
– radionuclide production 96
– research 94 ff.
– transients 177
real-time imaging 51
real-time radiation data 176
recoil track-etch 15
recommended exposure limit 279
rectosigmoid transfer coefficients 619
recycling of spent fuel 123, 173
reduction factor 277
reference levels 605 ff.
Reference Man, ICRP-23 369, 546, 591, 603
refining, fuel cycle 132
reflecting materials 136
reflectors 96
refueling 190, 196, 463
regulatory adoption of internal dosimetry models 578
Regulatory Guide 1.109 211, 222
relative biological effectiveness (RBE) 369
relative risk 630
relativistic beams 288
relativistic electrons 242
release
– fuel cycle 138, 153
– internal dosimetry 623 ff.
– reactors 94, 180
– stack 655
relocation 224
rem-ball 15
remote firing/monitoring 293
removal rate 336, 592, 647, 654
repetition rate 259
reprocessing facilities 123
research 17
– radionuclides 87
– reactors 93 ff.
see also university
residence time 582
residual radioactivity 251
respiratory tract 25
– hot particles 169 f.
– ICRP-103 588, 612
retina 283, 532
risk coefficients 33, 355
– BEIR I 627 f.
– BEIR III 627 f.
– BEIR V 627 f., 635
– BEIR VII 634 f.
– EPA NESHAP 628
– ICRP-60 584
– ICRP-103 601
– NRC BRC 628
– radon 209
risk models 627–638
Rn-220 676, 475
Rn-222 11, 676, 475
rotating antennas 277
routine swipes 142
Ru-106 data 232, 489
ruby laser 298

S

S-35, data 674
safety standards 245, 284, 294
see also standards, protective standards
sample counting 91, 103, 318, 485
sampling 464
saturation activity 573
scabblers 175
scarification 175
scattering 5, 68
– accelerators 518
– chest image room 80

– facility design 58 ff.
– medical accelerators 49
– transmission factor 62
scenarios
– accelerator health physics 259–271
– basic concepts 4–39, 71–83
– environmental health physics 222
– fuel cycle health physics 143–154
– medical health physics 47–86
– nonionizing radiation health physics 295–305
– power reactor health physics 181–196
– solutions 309–559
– university health physics 103–115
Schrödinger equation 652
scintigraphy 51
scintillation/photomultiplier detectors 259
scintillators 505
Se-75 activity 404
secondary air kerma 397
secondary barriers 62, 68
secondary coolant contamination 164
secondary radiations 237
secondary system monitors 177
sediments 214, 481, 495
selective photoexcitation 147
self-sustaining chain reaction 133
semi-infinite cloud model 178, 216
serial decay relationships 316, 561 ff.
sex averaged effective dose 602
shallow dose equivalent fraction 195
sheltering 224, 610
shielded dose rate 455
shielding 59, 255, 335 ff., 664
– accelerators 239 ff., 268 f.
– attenuation 570
– calculations 65
– initial fusion power facility 100
– materials 80, 517 f., 551
– NCRP 375, 395
– penetrating radiations 499 f.
– research reactors 95
– source configurations 565
– transmission 511 ff.
– X-ray diffraction 97
shoreline deposits 214, 488
side scatter 80
single photon emission computed tomography (SPECT) 52
single-compartment model 578 f.
single-pass recycling 123
single-pulse MPE 534, 537
skin
– acute radiation exposure 358
– annual occupational organ equivalent limit 600, 609
– doses 328, 421
– effluent pathways 211
– hot particles 166, 457
– laser radiation 284 ff., 532, 539 f., 301
– noble gas releases 219
– radioresilience 33
– tritium absorption 88
– UV light exposure 551
slab source approximation 157, 163, 569
sodium hydroxide decontamination 174
sodium-cooled fast reactors (SFRs) 123, 159
soil 251, 481
solid radioactive waste 132, 173
solid tumors 634, 662 f.
solutions, scenarios 309–558
solvents 135
source image distance (SID) 79
source strengths 206
source-related dose constraints 606
spallation 175, 252
specialized areas, theory/problems 45–308
specific absorbed fraction 583
specific activity, C-14 4, 311
specific effective energy (SEE) 26, 392
spectral effectiveness 294
spectral weighting functions 303
specular reflections 286
spent-fuel pool 131, 463
sponge blasting 174
spontaneous emission 531
Sr-90 data 8, 211, 675
stability classes 139
stack flow patterns 320
stack height 451
stack release 486, 655
stainless steel 91, 137, 575
standard mortality ratio (SMR) 630
standards, laser radiation 284
stationary antennas 275
steady-state concentration 209
steam generators 161, 178 f., 192
stimulated emission 531
stochastic effects 32, 585 f., 604
risk models 627
stomach 590, 619
stopping power 55
storage rings 243, 288
structural materials 99
submersion exposure 488
superconducting accelerators 288
supercritical water-cooled reactors (SWCRs) 123, 159

surface contamination calculation 164
surface deposition 164
– radioactive plume 655 ff.
surface dose rates 121
surface topology 286
surface water 480
synchrotron radiation 240 ff.
synergistic agents 359
system filters 91

t
tau particles 238
Tc-99m 384 f., 675
– activity 312
– diagnostic radionuclide 53
– dose factors 330
– generator 75, 112
– package 418
– ring badges 330
– specific effective energy 393
– uptake 28
temperature effects 282
tenth-value layer (TVL) 66
terminology, ICRP 595
tertiary production 240
Th-232 series 202, 677
therapeutic radionuclides 53 f.
thermal hazards 279 ff., 534, 537
thermal neutrons 5
– accelerators 50, 251
– Au-197 424
– tissue dose 313, 369
thermocouples 280
thermoluminescence dosimeter (TLD) 15, 89, 327 ff., 418
thorium fuel cycle 131
thorium series 230, 485
thoron 142, 202
threshold limit values (TLV) 303, 547, 554
thymidine, tritiated 29
thyroid
– ALI 314
– committed dose equivalent 444
– counting 35
– doses 345, 467
– function test 24
– I-125 407, 428
– I-131 therapy 82, 360
– initial activity 344
– metastasis 77
– reference man model 369
– university health physics 110
– weighting factor 377
$TiCl_4$ smoke sticks 91
tissues
– absorbency 282, 544
– depth 15, 48
– doses 5, 313
– weighting factors 586, 597 f., 601
Tl-201 data 78, 393, 676
Tokaimura criticality 134
tomography 52
total body dose 211, 219, 487
total committed effective dose equivalent 58, 379
total dose equivalent 580
total effective dose equivalent (TEDE) 344, 373, 445
total effective removal rate constant 392
total instantaneous power 241
total removal rate 654
total risk coefficient 628
total transfer rate constant 352
toxic gases 253, 263, 504
toxic materials 101, 121 ff.
tracer isotopes 53, 94
tracheobronchial region
– ICRP-30 588, 612
– radon 205, 476
track edge detectors 324
trajectory change 240
transfer coefficients 619
transit times 618
translocation processes 590
transmission 335, 511 f., 571
– accelerators 268 f.
– concrete 376
– luminous 529
– X-ray 61 ff., 80
transport index 417
transuranic (TRU) wastes 133
Tribondeau, Law of Bergonie and 30
trigonometric relationships 665
tritiated water 113
tritium
– accelerators 265
– breeding 100
– effluents 171
– fusion energy research 99 f.
– handling 25
– incorporation 29
– ITER 101
– production 96 ff.
– release 96 ff.
– university health physics 88, 105
– uptakes 88
– ventilation system 346
tritium oxide (HTO) intake 406 f.

tumor irradiation 499 ff.
two-source rule 66

u
U-233 131
U-234 121, 436
U-235 series 436
– DCF 436
– environmental health physics 203
– fuel cycles 121, 157
U-238 series 676 f.
– DCF 436
– environmental health physics 202
– fuel cycle 121
UF_6 dissociation 130
UF_6 enrichment facility feed material 128
ulceration 471
ultrasonic cleaning 174
ultraviolet radiation exposure 303
unattached fraction 476
uncertainty relationships 652
underwater surveys 190, 464
undulators 243
university health physics 87–118
unshielded air kerma 64, 395 ff.
upper large intestine 590
uptake
– brachytherapy 58
– internal dosimetry 578 ff.
see also C-14 uptake, H-3 uptake, I-125 uptake, P-32 uptake, Tc-99m uptake, Uranium uptake
uranium 485
– capture sources 575
– chemical compounds 141
– chemical processing 124 f.
– environmental health physics 210, 230
– fuel cycle 119 ff.
– radiation levels 23
– uptake 146
uranium hexafluoride 124
uranium tetrafluoride 124
urinanalysis 88
use factor 59 f., 74
UV lasers 555

v
vacuum vessel 100 ff.
van de Graaff accelerators 288
ventilation 10, 25, 70
– fuel cycle 153
– hospitals 400
– internal dosimetry 623
– irradiation cell 253
– P-32 88
– production equations 659
– radon 204
– tritium 346
– university health physics 89 ff.
visible gold vapor laser 555
visible laser radiation 287
volume parameters, critical concentration 135

w
warning devices 293
waste, radioactive 99, 172
waste disposal 391
– fuel cycle 119
– university/research 111
– uranium cycle 123
waste gas decay tank ruptures 178 f.
waste oil 150, 442
water
– activation 251
– capture sources 575
– contamination 442
– cooling 157
– radiolytic decomposition 414
– environmental monitoring 480
– heavy ion ranges 56
– ingestion 488
– radionuclide migration 210
– residual radioactivity 251
– shielding 94
– tritiated 29
– uranium fuel cycle 137
wavelengths 281, 286, 544
weapons fabrication 119
wear particulate material 165
whole-body counting 409
– I-131 intake 360
– Ra-226 508
whole-body dose 328
whole-body dose equivalent 15, 48
– fuel cycles 119
– research 89, 95, 107 ff.
– glucose 379
wigglers 243
wildlife parameters 232
wind speed 493
work area survey 195
UF_6 enrichment facility feed material 128
working level radon concentration 204
working level exposure unit 11
workload calculation 73, 80

x
Xe-133, data 676
Xe_2, laser 282
xenon release 163, 171
X-ray diffraction techniques 97 f.
X-rays 17, 51
– accelerators 254, 263
– electron accelerators 241
– gaseous effluent 216
– medical accelerators 48
– shielding 59 f.

y
Y-90, data 675
yellowcake 124
yield
– neutrons 365
– gamma 325, 340 f.

z
zirconium capture sources 575
zirconium/niobium particles 165
Zn-65 dose factor 183, 675
Zr-95 dose factor 183