80
8-10-65

MASTER

Argonne National Laboratory

RADIATION SAFETY TECHNICIAN TRAINING COURSE - PART I

by

H. J. Moe and S. R. Lasuk

APPROVED FOR PUBLIC RELEASE

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ANL-6991

Health and Safety
(TID-4500, 43rd Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60440

RADIATION SAFETY TECHNICIAN
TRAINING COURSE -- PART I

by

H. J. Moe and S. R. Lasuk

Industrial Hygiene and Safety Division

December 1964

Operated by The University of Chicago
under
Contract W-31-109-eng-38
with the
U. S. Atomic Energy Commission

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

TABLE OF CONTENTS

	<u>Page</u>
SECTION 1 - BASIC INFORMATION	11
A. Atomic Structure.	11
B. Ionization.	13
C. Isotopes.	13
D. Particle Dynamics.	14
E. Work and Energy.	14
F. Charged Particles	15
G. Relativistic Corrections	17
H. Charged Particle in a Magnetic Field	18
BIBLIOGRAPHY.	19
SECTION 2 - NATURAL RADIOACTIVITY	20
A. Early History.	20
B. Radioactive Decay	20
C. Half-life	23
D. Curie	24
E. Specific Activity	25
F. Decay Chains	25
G. Decay Curve of a Mixture	27
BIBLIOGRAPHY.	30
SECTION 3 - PROPERTIES OF ALPHA, BETA, GAMMA, X RAYS, AND NEUTRONS	31
A. Alpha Particles.	31
1. Specific Ionization.	32
2. Stopping Power.	33
3. Range	34
4. Relative Hazard	35
B. Beta Particles	36
1. Specific Ionization.	37
2. Stopping Power.	38

TABLE OF CONTENTS

	<u>Page</u>
3. Radiation Production - Bremsstrahlung	38
4. Range	39
5. Exponential β -particle Absorption	40
6. Relative Hazard	40
C. Wave Properties	40
D. Electromagnetic Waves	42
E. X and γ Radiation	44
1. Interactions with Matter	46
2. Photoelectric Effect	46
3. Compton Effect	48
4. Pair Production	49
5. Absorption of X and γ Rays	50
6. Half-value Layer	52
7. Mean Free Path	52
8. $\mu - \sigma_s$	52
9. Buildup Factor	53
10. Relative Hazard	54
F. Neutrons	54
1. Sources of Neutrons	55
2. Neutron Energy	56
3. Interactions with Matter	57
4. Cross Sections	59
5. Neutron Absorption	61
6. Removal Cross Section Σ_R	62
7. Neutron Activation	63
8. Relative Hazard	65
BIBLIOGRAPHY	66
SECTION 4 - CONCEPTS OF RADIATION UNITS AND DOSE DETERMINATIONS	69
A. Definition of Terms	69
B. Quality Factor	70
C. Exposure	70
D. Exposure Rate - Isotropic Point Source	71
E. Exposure Rate - Flux Density	73
F. Absorbed Dose	74

TABLE OF CONTENTS

	<u>Page</u>
G. Absorbed Dose Rate - Isotropic Point Source of Photons . . .	74
H. Absorbed Dose Rate - Flux Density of Photons	75
I. Absorbed Dose Rate - Isotropic Point Source - Alpha and Beta Radiation	75
J. Mass Stopping Power	75
K. Absorbed Dose Rate - Flux Density of Particles	77
L. Absorbed Dose for Neutrons	78
BIBLIOGRAPHY.	80
SECTION 5 - SHIELDING.	83
A. Factors Affecting Exposure in a Radiation Field	83
B. Alpha Radiation.	84
C. Beta Radiation	85
D. Gamma Radiation	85
1. Calculations of Shield Thickness	86
2. Buildup Factor	89
3. Effective Atomic Number	91
E. Neutrons	92
F. Shielding Materials	95
BIBLIOGRAPHY.	95
SECTION 6 - BIOLOGICAL EFFECTS OF RADIATION	97
A. The Cell - Basic Unit of Structure	97
B. Radiosensitivity	99
C. Radiation Damage	99
D. Somatic Effects.	99
1. Blood and Bone Marrow	101
2. Lymphatic System.	103
3. Digestive Tract	103
4. Reproductive Organs	103
5. Nervous System	104
6. Thyroid Gland	104

TABLE OF CONTENTS

	<u>Page</u>
7. Eyes	104
8. Lungs	105
9. Liver and Gall Bladder	105
10. Kidneys	105
11. Circulatory System	106
12. Skin	106
13. Hair	106
14. Bones	106
E. Late Effects	106
1. Neoplasms	106
2. Tissue Effects	107
3. Life-span	107
4. Growth and Development	107
F. Hereditary Effects	107
BIBLIOGRAPHY	108
SECTION 7 - BACKGROUND RADIATION	110
A. Cosmic Radiation	110
B. Radioactivity of the Earth	112
C. Radioactivity of Air	113
D. Radioactivity of Water	114
E. Radioactivity of the Human Body	114
F. Fallout	115
G. Effect of World-wide Fallout	117
BIBLIOGRAPHY	117
SECTION 8 - RADIATION PROTECTION STANDARDS	119
A. History of Protection Standards	119
1. Erythema Dose	119
2. Tolerance Dose	120
3. ICRU, ICRP, and NCRP	120
4. Permissible Dose	121
5. Total Accumulated Dose	122
6. Federal Radiation Council	123
7. Radiation Protection Guides	123
8. General Philosophy of the Federal Policy on Radiation Matters	124

TABLE OF CONTENTS

	<u>Page</u>
B. Regulating Agencies.	124
BIBLIOGRAPHY.	126
SECTION 9 - INTERNAL DOSE CALCULATIONS.	129
A. Main Factors Affecting Dose Calculations	129
B. Standard Man	132
C. Critical Organ	132
D. Effective Energy per Disintegration	133
1. The Energy Absorbed per Disintegration, E.	133
a. β^- Radiation	134
b. β^+ Radiation	134
c. γ Rays	134
d. Internal Conversion, e^-	134
e. α Radiation	135
f. Recoil Atoms.	135
g. K and L Capture X Rays.	135
2. Radioactive Chain Factor, F.	135
3. Relative Biological Effectiveness, RBE.	136
4. Relative Damage Factor, n.	137
E. Other Factors.	137
F. Maximum Permissible Body Burden.	138
G. Body Burden Based on Radium.	139
H. Body Burden Based on Dose Equivalent Rate	140
I. MPC in Air and Water - Single Radionuclide	140
J. MPC Formulae for a Radioactive Series	142
K. MPC of Noble Gases - Submersion Exposure.	143
L. MPC Values for a Mixture of Radionuclides	144
M. Dose Equivalent from Single Intake.	145
N. Sample Calculations of $(MPC)_a$ and $(MPC)_w$	146
BIBLIOGRAPHY.	152

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.1	The Stable Fluorine Atom (${}^9\text{F}^{19}$)	12
1.2	The Ionization of an Atom	13
1.3	The Electric Field E Is Perpendicular to the Plates.	16
1.4	A Charged Particle Traveling Perpendicular to a Magnetic Field Which Is Directed into the Plane of the Paper	18
2.1	Radiation Deflection in a Magnetic Field.	20
2.2	Radioactive Decay, Linear Plot.	21
2.3	Radioactive Decay, Semilog Plot.	21
2.4	Sample Decay Curve Using Example Data	28
3.1	Illustration of Alpha Tracks as Seen in a Cloud Chamber. . .	31
3.2	Electrostatic Interaction	32
3.3	Excitation.	32
3.4	The Ionization Increases as the Velocity of the Alpha Particle Decreases with Depth of Penetration.	33
3.5	Typical Beta Spectrum	36
3.6	The General Shape of the Specific Ionization vs. Energy Curve for β Particles	37
3.7	Bremsstrahlung.	39
3.8	Electromagnetic Wave.	42
3.9	Typical X-Ray Spectrum	44
3.10	The Photoelectric Effect	46
3.11	Process Whereby a Photon Ejects an Auger Electron	47
3.12	The Compton Effect	48
3.13	Pair Production and Annihilation.	49
3.14	Absorption of X and γ Rays	50
3.15	Wide Beam of γ Rays in an Absorber.	53
3.16	Elastic Collision. Energy and momentum are conserved. . .	57
3.17	Inelastic Collision	58
3.18	Impact Area	60
4.1	Isotropic Point Source.	71

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
4.2	Mass Stopping Power of Alpha Radiation as a Function of Energy	76
4.3	Mass Stopping Power of Beta Radiation as a Function of Energy	77
4.4	Absorbed Dose for Neutrons as a Function of Energy	78
5.1	Relationships for Calculating Shield Thicknesses	86

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
6.1	Summary of Clinical Effects of Acute Ionizing Radiation Doses	102
7.1	Body Tissue Dose Rates due to External and Internal Irradiation from Natural Sources of Radiation in "Normal Regions"	115

RADIATION SAFETY TECHNICIAN TRAINING COURSE - PART I

by

H. J. Moe and S. R. Lasuk

SECTION 1 - BASIC INFORMATION

A. Atomic Structure

An atom is the smallest unit of an element. A list of the different elements is given in the "Periodic Chart of the Atoms." There are 92 naturally occurring elements. In addition to these, modern science has increased the number to more than 100 by artificially producing elements.

The elements are usually designated by their chemical symbols, found in the periodic chart. Hydrogen, for example, has the symbol H; oxygen has the symbol O; etc.

An atom consists of a nucleus surrounded by a sufficient number of electrons to provide electrical neutrality. The nucleus is the massive, central, positively charged portion of the atom, composed of neutrons and protons. The neutron and the proton are particles of about the same mass, but the proton has a positive charge and the neutron is electrically neutral. A method of denoting atomic structure which is often used is ${}_Z^X A$, where X is the chemical symbol of the element. The quantity A is the mass number which gives the number of protons and neutrons in a nucleus, and Z is the atomic number of the element, which gives the number of protons (or positive charges) in the nucleus, and the number of surrounding electrons. The number of neutrons in the nucleus is then given by $A - Z$. Inasmuch as the proton and the neutron each have approximately unit mass on the scale of atomic weights, the mass number is also that integer closest to the atomic weight.

The radius of the nucleus is given approximately by

$$r \text{ (in cm)} = 1.5 \times 10^{-13} A^{1/3}.$$

The diameter of the nucleus is less than 1/10,000 that of the atom itself. Almost the entire mass of the atom is concentrated in the nucleus. The nuclear density is about 10^{14} gm/cc!

The atomic mass of an element may be expressed in atomic mass units (amu). The amu, on the physical scale, is 1/12 of the mass of a

carbon-12 atom. The mass of any atom, in grams, is equal to its atomic weight divided by Avogadro's number N_A ($= 6.025 \times 10^{23}$ atoms/gm-atom). The reciprocal of Avogadro's number ($1/N_A$) is equal to the mass, in grams, of 1 amu, i.e., $1 \text{ amu} = 1.66 \times 10^{-24} \text{ gm}$.

The number of atoms in a certain mass of an element is given by the expression

$$N = \frac{m}{A} N_A,$$

where m is the mass of the element in grams. For example, the number of atoms in one gram of U^{238} is calculated to be

$$N = \frac{m}{A} N_A = \frac{1}{238} (6.025 \times 10^{23}) = 2.53 \times 10^{21} \text{ atoms.}$$

The gram-atomic weight or gram-molecular weight is often used. These quantities simply mean a weight in grams of an element which is equal to its atomic or molecular weight. A gram-atomic weight of U^{238} is 238 gm of U^{238} ; a gram-molecular weight (often called a mole) of H_2O is 18 gm. A gram-atomic weight of an element contains 6.025×10^{23} atoms; a gram-molecular weight contains 6.025×10^{23} molecules.

The atom may be pictured as a planetary system of electrons in motion about the nucleus. An example, in this case of the fluorine atom, is shown in Fig. 1.1. The electrons rotate in elliptical and circular orbits about the nucleus. These are well-defined orbits (shells) which are called energy or quantum levels; the diameters of these orbits are large in comparison with the nuclear diameter. The shells are denoted by either a letter (K, L, M, N, O, P, Q) or quantum

Fig. 1.1

The Stable Fluorine Atom (${}^9F^{19}$).
The nucleus contains 9 protons
and 10 neutrons.

number (1, 2, 3, 4, 5, 6, 7). The energy state of each electron in a shell is completely described by four independent quantum numbers (n , l , m , and s). The Pauli Exclusion Principle states that no two electrons in any one atom can be identical, that is, have all four quantum numbers the same. This principle sets an upper limit to the number of possible electrons in each level. This limit is given by $N_e = 2n^2$, where n is the principal quantum number of the shell, which may take on integral values 1, 2, 3, 4, 5, 6, 7 for the corresponding shell of the atom.

B. Ionization

An electron can be removed fairly easily from an atom or molecule. One or more electrons may also be added to such entities. The resulting component has an electric charge (positive) because the number of protons exceed the number of electrons. The term ion is used to define atoms or groups of atoms carrying an electric charge. The charge each ion carries depends upon the number of electrons which have been added or removed. Ionization is a process whereby electrons may be removed from or added to a neutral atom or molecule. The removal of an electron from an atom or molecule (see Fig. 1.2 for the process of electron removal from an atom) results in an ion pair: the free electron and the positively charged residue.

Fig. 1.2
The Ionization of an Atom

C. Isotopes

It was mentioned that over 100 elements are known to man. Each element is characterized by its atomic number. However, an element may have atoms possessing different masses. The term isotope is used to denote elements with the same atomic number but different atomic masses. Since the atomic number is the same, isotopes have the same number of protons within the nucleus, but the number of neutrons vary. An element may have one or more isotopes.

Some isotopes are unstable, releasing energy in the form of particles and/or electromagnetic radiation by a process of decay or disintegration called radioactivity.

Within the nucleus, a repulsive force exists between the protons, the Coulomb force, which tends to pull the nucleus apart. Such forces are relatively long-range forces. In addition, there are very short-range attractive forces between the neutrons and protons which tend to hold the nucleus together. The net effect of these forces will determine whether a certain combination or ratio of neutrons and protons is stable. The lighter elements, in which the neutrons and protons are about equal, exhibit stability. Above $Z = 20$, an excess of neutrons is required for a nuclide to be stable.

D. Particle Dynamics

A particle has the property of inertia, that is, a resistance to change in its velocity. The mass of a particle is a measure of this inertia. The greater the resistance to a change in velocity, the greater will be the inertia, and thus the mass of the particle. The unit of mass in the cgs system is the gram. There are about 454 gm to a pound.

The velocity of a particle is a vector quantity in which both speed and direction must be specified. The magnitude of the velocity (that is, the speed) is given in cm/sec in the cgs system. The acceleration of a particle is defined as the time rate of change of velocity.

The force F which acts on a particle is defined in terms of the product of the acceleration a and the mass m of the object:

$$F = ma.$$

If the acceleration is constant,

$$F = \frac{m(v - v_0)}{t} = \frac{mv - mv_0}{t}.$$

This equation is known as the impulse equation. The quantity mv is called the momentum of the particle, so that one can measure the magnitude of a force from the time rate of change of the momentum. The unit of force in the cgs system is the dyne: a dyne is the force which will give to one gram an acceleration of 1 cm/sec^2 , that is, $1 \text{ dyne} = 1 \text{ gm-cm/sec}^2$.

E. Work and Energy

Work W is usually defined as the product of a force F and the distance d through which an object moves under the action of the force: $W = Fd$. The term energy is defined as the capacity for doing work. An object has energy by virtue of its position and by virtue of its motion. Potential energy is energy of position; e.g., a book resting on the floor has potential energy equal to zero if we consider the floor as a reference level.

In order to give the book potential energy we must do work on it to change its position. For example, we might pick it up and put it on a table. In raising the book to the table, the potential energy of the book will now be equal to the work we have done in raising it to that height. This work can be recovered if the book falls off the table and returns to its original position on the floor.

Kinetic energy is energy of motion. It is given by the equation

$$E_k = 1/2 mv^2.$$

The unit of work or energy in the cgs. system is the erg. The erg = dyne-cm = gm-cm²/sec². When we have the book on the floor or table, its kinetic energy will be zero. If the book falls, it will gain kinetic energy as it approaches the floor. During its flight, it will lose potential energy. However, at any point in its flight, the sum of the two energies will be constant and equal to the work done in raising the book to the table.

Two important concepts in physics are Conservation of Momentum and Conservation of Energy. When no external force acts on a system, the total momentum of the system remains constant. If no work is done on the system or by the system, the sum of the potential and kinetic energies of the system of objects in motion remains constant.

F. Charged Particles

The expression which gives the force of attraction or repulsion between two electric charges of magnitudes q_1 and q_2 separated by a distance r is called Coulomb's law and is expressed as

$$F = \frac{q_1 q_2}{kr^2} \text{ dynes,}$$

where k is the dielectric constant.

Charges which are alike, repel; charges which are unlike, attract. The constant k depends upon the medium between the charges but is taken as unity if the medium is air.

The unit of charge in the cgs. system is the electrostatic unit (esu) or statcoulomb, defined in terms of Coulomb's law: one esu is that charge which will repel an exactly equal charge with a force of one dyne when the two are one centimeter apart in a vacuum. The magnitude of the charge on the proton and electron is 4.8×10^{-10} esu. Although the statcoulomb is a small unit itself, it takes 2.08×10^9 electrons or protons to equal a charge of one esu.

When two parallel plates are connected to a battery (see Fig. 1.3), a negative charge appears on one plate and an equal positive charge on the other plate. In the region between the plates an electric field exists which will exert a force on a charge placed in the field. The field strength E is defined by the force exerted on a unit charge in the field:

$$E = F/q.$$

The term potential difference, symbolized by V , is defined as the work done in moving a unit charge from one plate to the other:

$$W = Fd = Eqd;$$

but

$$E = V/d$$

also, so that

$$W = \frac{V}{d} qd = Vq,$$

and

$$V = W/q.$$

If W is in ergs and q in esu, then V will be in statvolts. When V is expressed in Volts, the relationship

$$\text{Volts}/300 = \text{statvolts}$$

may be used.

Fig. 1.3

The Electric Field E Is Perpendicular to the Plates. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition (©) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

In the parallel-plate example, the electric field E is uniform, that is, the force on a charge in the field is constant, both in direction and magnitude. In the case of cylindrical electrodes, the field is not uniform, but varies with radius r . In this type of arrangement, one will

usually have a central wire of radius r_1 and an inner radius of the cylinder r_2 . The expression for the field is then

$$E = \frac{V}{r \ln(r_2/r_1)},$$

where V is the voltage across the electrodes. The field gets stronger as one approaches the central wire, so that the force on a charge increases as the charge nears the wire.

The usual unit of energy is the erg. In nuclear physics, however, the electron Volt (eV) is used as the energy unit. By definition, the electron Volt is the amount of work done in accelerating an electron by a potential difference of one Volt. It is related to the erg by: $1 \text{ eV} = 1.6 \times 10^{-12} \text{ erg}$.

The velocity of an electron is related to the potential difference V by the expression

$$Ve = \frac{1}{2} mv^2,$$

where Ve is expressed in ergs. This relationship is valid if the velocity of the electron is not comparable with that of light and the scattering of electrons is negligible (high vacuum). When the speed of the electron becomes greater than $1/10$ that of light, the mass of the electron will increase and the above relationship will not hold.

G. Relativistic Corrections

Among the new concepts which Einstein contributed to modern physics in his theory of general relativity are:

- 1) No measured velocity can exceed the speed of light c .
- 2) A body does not have a constant mass m . The mass of an object depends upon its speed v :

$$m = \frac{m_0}{\sqrt{1 - \beta^2}},$$

where $\beta = v/c$.

- 3) Mass and energy are interchangeable: $E = (\Delta m) c^2$, or, for our purposes, $E = mc^2$.

The relativity theory considers a rest mass m_0 and a mass m which an object has by virtue of its speed v . From statements 2) and 3), we get

$$E = mc^2 = m_0 c^2 / \sqrt{1 - \beta^2},$$

where E represents the total energy content of an object of rest mass m_0 moving with a speed v . The energy content of the rest mass is found by setting $v = 0$, namely, $E_0 = m_0 c^2$. The kinetic energy of the object is then

$$E_k = E - E_0 = m_0 c^2 \left[\frac{1}{\sqrt{1 - \beta^2}} - 1 \right].$$

H. Charged Particle in a Magnetic Field

A charged particle which moves through a magnetic field will experience a force acting upon it if the magnetic field has a component at right angles to the direction of motion of the particle. In particular, if the entire field is perpendicular to the direction of motion of the particle, the force acting on the particle will be:

$$F = Hev,$$

where H is the magnetic field strength, e is the charge on the particle, and v its speed. The force will be perpendicular to H and v and the particle will tend to move in a circle of radius r (see Fig. 1.4).

Fig. 1.4

A Charged Particle Traveling Perpendicular to a Magnetic Field Which is Directed Into the Plane of the Paper. (Richards, Sears, Wehr and Zemansky, Modern University Physics, 1960, Addison-Wesley, Reading, Mass.)

In considering circular motion the force is given by

$$F = mv^2/r.$$

Accordingly, for a magnetic field acting on a charged particle moving at right angles to the field,

$$Hev = mv^2/r.$$

BIBLIOGRAPHY

1. Glasstone, S., SOURCEBOOK ON ATOMIC ENERGY, D. Van Nostrand Co., Inc., Princeton, N. J. (1958).
2. Friedlander, G., and Kennedy, J. W., NUCLEAR AND RADIOCHEMISTRY, John Wiley and Sons, Inc., New York, N. Y. (1960).
3. Lapp, R. E., and Andrews, H. L., NUCLEAR RADIATION PHYSICS, Prentice-Hall, Inc., New York, N. Y. (1955).
4. Stranathan, J. D., THE "PARTICLES" OF MODERN PHYSICS, Blakiston, Philadelphia, Pa. (1942).
5. Lemon, H. B., and Ference, Jr., M., ANALYTICAL EXPERIMENTAL PHYSICS, The University of Chicago Press, Chicago, Ill. (1947).
6. Weber, R. L., White, M. W., and Manning, K. V., PHYSICS FOR SCIENCE AND ENGINEERING, McGraw-Hill Book Co., Inc., New York, N. Y. (1959).
7. Hoag, J. B., and Korff, S. A., ELECTRON AND NUCLEAR PHYSICS, D. Van Nostrand Co., Inc., New York, N. Y. (1948).

SECTION 2 - NATURAL RADIOACTIVITY

A. Early History

Henri Becquerel is credited with the discovery of natural radioactivity. While working with uranium salts and photographic plates in 1896, he found that the uranium emitted a penetrating radiation which seemed similar to that which Roentgen had produced a year earlier in experiments with a gas discharge tube. Further investigation showed that this property of emitting radiation is a specific property of the element and that certain quantitative relationships exist. In particular, the atoms producing these radiations are unstable and decay at characteristic rates to form new atoms.

By the use of magnetic fields, it was shown that there are three distinct types of radiation (see Fig. 2.1). These were given the designations: alpha (α), beta (β), and gamma (γ) radiation. The first two types can be deflected in magnetic fields, indicating that they are charged particles. Gamma radiation cannot be deflected in magnetic fields. It is not a particle but a form of electromagnetic radiation, similar to light.

Fig. 2.1

Radiation Deflection in a Magnetic Field. The field is perpendicular to and directed into the plane of the paper. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition (©) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

Since alpha particles were deflected in a different direction in the magnetic field than were beta particles, the two were oppositely charged. From the direction in which the alpha particle was deflected, it was shown to be positively charged, whereas the beta particle is negatively charged. Further investigation revealed that the alpha particle is actually a helium nucleus (${}^4_2\text{He}$ - two positive charges, mass number 4) and the beta particle is actually an electron (e^- - one negative charge, mass of 0.000549 amu).

B. Radioactive Decay

When an atom decays by the emission of an alpha particle, the atomic number Z decreases by 2 and the mass number A decreases by 4, so that

When the decay is by beta emission, the atomic number Z increases by 1, but the mass number A remains the same:

The emission of a gamma ray does not affect either the atomic number or the mass number.

When one deals with large numbers of atoms, it is found that all radioactive substances follow the same general decay pattern. For example, assume one has a radioactive source and some means of counting the number of atoms which disintegrate in a given time interval. If one plots the number of atoms present at time t versus the time on linear graph paper (see Fig. 2.2), the curve obtained will indicate an exponential or logarithmic relationship. If the number present versus the time is plotted on semilog paper (see Fig. 2.3), a straight line is obtained. This indicates that radioactive decay is an exponential (logarithmic) process, that is, there is a constant fractional decrease in the decay rate during equal units of time. Although the same number of atoms do not disintegrate during each unit of time, the same fraction of the atoms present decay during a unit of time. The disintegration rate is proportional to the number of atoms present:

$$\frac{\Delta N}{\Delta t} = \frac{N_1 - N_0}{t_1 - t_0} \propto N,$$

where N is the number of atoms present at any time t . By including a constant of proportionality, the expression becomes

$$\frac{\Delta N}{\Delta t} = -\lambda N,$$

Fig. 2.2. Radioactive Decay, Linear Plot

Fig. 2.3. Radioactive Decay, Semilog Plot

where λ , called the decay constant, is the constant of proportionality and the minus sign indicates a decrease in the decay rate as time increases. When the expression is integrated, we get the general exponential relationship for radioactive decay:

$$N = N_0 e^{-\lambda t},$$

where N is the number of atoms left at time t from a sample of N_0 radioactive atoms initially present.

If we take the logarithm of each side of this equation, we get

$$\ln N = \ln N_0 - \lambda t.$$

This is similar to the slope-intercept form of a straight line, i.e., $y = a + bx$. Thus, if one uses semilog paper and plots the values of N on the logarithmic scale versus the time on the linear scale, the slope of the resulting straight line will be $-\lambda$ and the intercept (the value of N at $t = 0$) will be N_0 .

As an example, consider the data:

<u>Number of Atoms</u>	<u>Time (min)</u>
80,000	0
29,432	1
10,824	2
3,984	3
1,464	4
536	5
200	6

Plot a curve on semilog and on linear paper, and determine the decay constant λ for the radioactive element.

To plot on semilog paper: Take three-cycle semilog paper (this will cover the ranges 100-1000, 1000-10,000, and 10,000-100,000), and plot the number of atoms on the logarithmic scale and the corresponding value of the time on the linear scale. The plot will be a straight line.

To find the slope of the line ($-\lambda$): Take the ratio of the number of atoms for two times; find the natural log of this ratio, and divide this log by the difference in time for the values chosen. For example, choose the readings for $t = 0$ and $t = 2$ min. Take the ratio of the number of atoms at the later time to the number of atoms at the earlier time. This ratio is 0.1353. The natural log of this ratio is $\ln 0.1353 = \ln (1.353 \times 10^{-1}) = \ln 1.353 + \ln 10^{-1} = \ln 1.353 - \ln 10 = 0.3026 - 2.3026 = -2$. The time difference for the two values is 2 min. We now obtain:

$$\ln N = \ln N_0 - \lambda t;$$

$$\ln N - \ln N_0 = -\lambda t;$$

$$\ln N/N_0 = -\lambda t;$$

$$-2 = -\lambda(2 \text{ min});$$

$$1 \text{ min}^{-1} = \lambda.$$

To plot on linear paper: Plot number of atoms as ordinate (y axis) and the time as abscissa (x axis).

The decay constant λ expresses the probability that a single atom will decay in a unit of time. The larger the value of λ , the more rapidly a radioelement decays.

One is usually more interested in the decay rate of a given sample rather than the number of atoms present. The activity of a sample is expressed as

$$\frac{dN}{dt} = A = \lambda N.$$

In other words, the activity A of the sample can be found by multiplying N , the number of atoms present at any time, by the decay constant λ of the element. Accordingly,

$$N = N_0 e^{-\lambda t};$$

$$\lambda N = \lambda N_0 e^{-\lambda t};$$

$$A = A_0 e^{-\lambda t},$$

where A_0 is the activity (decay rate) of the sample at $t = 0$.

C. Half-life

The decay constant λ is closely related to the concept of the half-life of a radioisotope. The half-life $T_{1/2}$ is defined as the time required for a given isotope to decay to one-half of its initial value. Consider an initial activity A_0 . At some time $t = T_{1/2}$, the activity will be equal to $A_0/2$. From our general relationship, then,

$$A = A_0 e^{-\lambda t},$$

$$\frac{1}{2} A_0 = A_0 e^{-\lambda T_{1/2}},$$

$$\frac{1}{2} = e^{-\lambda T_{1/2}};$$

$$\ln 1 - \ln 2 = -\lambda T_{1/2};$$

$$\lambda T_{1/2} = \ln 2;$$

$$T_{1/2} = \frac{\ln 2}{\lambda} = \frac{0.693}{\lambda}.$$

Each radioactive atom has its own unique pattern of decay. Two aspects which characterize the radioactive decay of a given element are:

- 1) half-life of emission,
- 2) energy of the emission.

The identification of a particular radioisotope will normally depend upon how well one can determine both of these factors. Many radioisotopes have half-lives which are nearly the same, but the energy of their emissions differs greatly. On the other hand, many radioisotopes have similar energy of emission, but their half-lives differ greatly.

D. Curie

The activity of a radioactive substance is often designated by the Curie (abbreviated Ci). The Curie is defined as the activity of a radioactive nuclide disintegrating at the rate of 3.7×10^{10} dis/sec. The Curie is not a measure of dose; it merely states the amount of a particular radioisotope disintegrating per unit time.

The definition of the Curie refers to disintegrations per unit time. Oftentimes, the number of disintegrations will differ greatly from the number of radiations emitted, as for example, in the case of Co^{60} . For each Co^{60} atom which disintegrates, a beta particle and two gamma rays are emitted. In this case, then, a Curie of Co^{60} would emit 3.7×10^{10} β /sec and 7.4×10^{10} γ /sec. If the activity of a sample is to be accurately calculated from experimental data, the decay scheme of the radioisotope must be taken into account.

In counting a β emitter, some of the γ rays will also be counted. If one merely takes the counting rate and converts this to a disintegration rate, without considering the decay scheme, one will usually overestimate the activity of the source. For example, consider a counter which counts all the betas and 2% of the gammas. In counting a $1\text{-}\mu\text{Ci}$ Co^{60} source, the observed counting rate would be 3.7×10^4 β counts/sec + $0.02 (7.4 \times 10^4)$ γ counts/sec = 3.8×10^4 counts/sec. If one converts this activity to Curies without regard to the decay scheme, the activity would come out as $1.04 \mu\text{Ci}$.

In counting an α emitter, the situation is usually different. Since a proportional counter is normally used, the γ rays associated with an α emitter would not be counted, and thus the activity can be determined from the counting rate.

For purposes of internal dose calculations, the National Committee on Radiation Protection defines a Curie of natural uranium and natural thorium in a different manner in NBS Handbook #69. A Curie of recently extracted uranium is equal to the sum of 3.7×10^{10} dis/sec from U^{238} , 3.7×10^{10} dis/sec from U^{234} , and 1.7×10^9 dis/sec of U^{235} . By this definition, then, a Curie of natural uranium is equal to 7.57×10^{10} dis/sec. A Curie of recently extracted thorium is equal to the sum of 3.7×10^{10} dis/sec from Th^{232} and 3.7×10^{10} dis/sec from Th^{228} . Thus, a Curie of natural thorium is equal to 7.4×10^{10} dis/sec.

E. Specific Activity

The specific activity is defined as the activity of one gram of a substance. It is usually expressed as Ci/gm of the radioelement. The shorter the half-life of the emitter, the greater is found to be its specific activity. The specific activity can be calculated from the equation

$$SP. A. = \lambda N,$$

where λ is the decay constant and N is now the number of atoms in one gram of the radioelement. Co^{60} has a half-life of 5.24 yr. Calculate the specific activity:

$$\begin{aligned} SP. A. &= \lambda N = \frac{0.693}{T_{1/2}} \frac{1 \text{ gm}}{A} N_A \\ &= \frac{0.693}{5.24 \text{ yr}} \frac{(1) 6.025 \times 10^{23}}{\left(3.1 \times 10^7 \frac{\text{sec}}{\text{yr}}\right) 60} = 4.22 \times 10^{13} \frac{\text{dis}}{\text{sec-gm}} \\ &= \frac{4.22 \times 10^{13}}{3.7 \times 10^{10}} = 1.14 \times 10^3 \frac{\text{Ci}}{\text{gm}} \end{aligned}$$

F. Decay Chains

In general, most radioactive substances do not decay to form a stable isotope, that is to say, the daughter nucleus is also radioactive and decays with its own characteristic half-life. The problem of determining the amount of the daughter present at any time depends therefore upon both half-lives. The daughter will be produced at a certain rate from the parent but will decay with its own rate.

Suppose that at a given time there are N_1^0 parent atoms, with decay constant λ_1 , and no daughter atoms. After a certain interval of time, Δt , the increase in the number of daughter atoms, ΔN_2 , is given by

$$\Delta N_2 = (\text{decay rate of parent} - \text{decay rate of daughter}) \Delta t.$$

The decay rate of the parent is actually the formation rate of the daughter since whenever a parent atom decays it becomes a daughter atom. The decay rate of the parent is $\lambda_1 N_1$, where N_1 is the number of parent atoms present at any time. Similarly, the decay rate of the daughter is $\lambda_2 N_2$. The expression for the rate of change of daughter atoms per unit time becomes

$$\frac{\Delta N_2}{\Delta t} = \lambda_1 N_1 - \lambda_2 N_2.$$

From the general relationship, the number of parent atoms at any time is

$$N_1 = N_1^0 e^{-\lambda_1 t}.$$

Substituting this into the above expression and integrating the equation, one arrives at

$$N_2 = \frac{N_1^0 \lambda_1}{\lambda_2 - \lambda_1} (e^{-\lambda_1 t} - e^{-\lambda_2 t}),$$

where N_2 is the number of daughter atoms present at any time.

When the parent half-life is long compared with that of the daughter, then $\lambda_1 < \lambda_2$, and the term $e^{-\lambda_2 t}$ becomes negligible compared with $e^{-\lambda_1 t}$ after a sufficiently long time. Then the last equation reduces to

$$N_2 = \frac{N_1^0 \lambda_1}{\lambda_2 - \lambda_1} e^{-\lambda_1 t}.$$

However, since

$$N_1 = N_1^0 e^{-\lambda_1 t},$$

we find

$$N_2 = \frac{N_1 \lambda_1}{\lambda_2 - \lambda_1}.$$

A condition is thus reached in which the ratio N_2/N_1 remains constant. This state is called transient equilibrium. In this case, the daughter activity decays at the same rate as the parent activity.

When the parent activity is extremely long-lived, as is the case with U^{238} ($T_{1/2} = 4.5 \times 10^9$ yr), then $\lambda_1 \ll \lambda_2$, $e^{-\lambda_1 t} \approx 1$, and $\lambda_2 - \lambda_1 \approx \lambda_2$, so that the original expression for N_2 reduces to

$$N_2 = \frac{N_1^0 \lambda_1}{\lambda_2} (1 - e^{-\lambda_2 t}).$$

After a sufficient time, $e^{-\lambda_2 t}$ approaches zero, and the amount of the daughter product is then given by

$$N_2 = N_1^0 (\lambda_1 / \lambda_2).$$

This state is known as secular equilibrium.

If the parent has a shorter half-life than the daughter, $\lambda_1 > \lambda_2$, and the daughter rises to a maximum and decays with its own characteristic half-life. In this case, no equilibrium is reached.

G. Decay Curve of a Mixture

The problem of identifying the half-life of a radioisotope from a decay curve can be complicated by the fact that the experimental curve may be a composite of the contributions from a mixture of radioisotopes. When the activity values are plotted on semilog paper, the curve obtained from such a mixture will contain: 1) an initial straight portion, followed by 2) a curved portion, which is then followed by 3) a final straight portion. This type of decay curve is explained by the presence of at least two radioactive products in the sample being counted.

The initial straight portion of the curve represents the sum of the activities of all the components, whereas the final straight portion is due only to the activity of the longer-lived component of the mixture. The shorter-lived products contribute greatly to the decay rate initially; but as these decay out, the total activity will not decrease as rapidly as in the beginning. This accounts for the curvature in the plot.

Since the final straight portion represents the activity of the long-lived component, this line can be extrapolated back to zero time. The initial activity of the long-lived component is the intercept value at $t = 0$ of the extrapolated line. In addition, this line gives the decay curve of the long-lived component. When a series of values along this curve are subtracted from corresponding values on the original curve, the remainders will represent the activity of the short-lived components at the corresponding times.

If the mixture contains only two components, then a plot of the remainder values will give a straight line. As before, the activity value at $t = 0$ for this new straight line will represent the initial activity of the short-lived component.

The half-life for each component can then be determined graphically from each of the straight lines.

If three or more components are present, this method can be extended provided the original data are sufficiently accurate.

Example: Using the data in the table, plot a curve on semilog paper (see Fig. 2.4) and determine the half-lives for all the activities in the sample.

<u>cpm</u>	<u>t (hr)</u>	<u>cpm</u>	<u>t (hr)</u>
10,000	0	370	14
5,778	1	258	16
4,078	2	181	18
3,148	3	127	20
2,516	4	89	22
1,666	6	62	24
1,125	8	44	26
770	10	31	28
531	12

Fig. 2.4
Decay Curve Using
Example Data above

Plot the data on 3-cycle semilog paper. From the shape of the curve one would surmise that there are at least two components. Extrapolate the final straight portion of the curve back to zero time. This line intersects the $t = 0$ axis at 4000 cpm. The half-life is determined directly from the graph: find the time at which the activity has dropped to 2000 cpm. This occurs at 4 hr. The half-life can be determined mathematically by choosing two points on the curve, finding the ratio of these two points, and dividing the natural log of this ratio by the difference in time between these two points. This gives the value of λ . Take the points: $t = 0$, $\text{cpm} = 4000$; $t = 12$ hr, $\text{cpm} = 500$.

$$\ln \frac{N}{N_0} = \ln \frac{1}{8} = -\ln 8 = -12\lambda;$$

$$\lambda = \frac{\ln 8}{12} = \frac{3 \ln 2}{12} = \frac{\ln 2}{4} \text{ hr}^{-1}.$$

But

$$T_{1/2} = \frac{\ln 2}{\lambda} = \frac{\ln 2}{\ln 2} (4) \text{ hr} = 4 \text{ hr}.$$

Take points on the extrapolated curve and subtract their values from the corresponding points on the experimental curve. Plot these differences versus the corresponding times. Consider the points for $t = 0$: the value on the experimental curve is 10,000 cpm, the value on the extrapolated curve is 4000 cpm. The difference, 6000 cpm, is plotted for $t = 0$. Similarly, for the point $t = 1$ hr, the value on the experimental curve = 5778 cpm, and the value on the extrapolated curve = 3400 cpm. The difference, 2378 cpm, is plotted for $t = 1$ hr. Proceeding in this way, one can determine enough points to draw the curve.

In this case the curve is not a straight line, it is a curved line. Once again extrapolating the final straight portion back to $t = 0$ gives an initial activity of 2000 cpm. The half-life of this activity as determined from the graph is 2 hr.

When points on this extrapolated curve are subtracted from the corresponding points on the second curved line and the differences are plotted, a straight line is obtained. The initial activity of this component is 4000 cpm, and its half-life as determined from the graph is 30 min.

BIBLIOGRAPHY

1. Glasstone, S., SOURCEBOOK ON ATOMIC ENERGY, D. Van Nostrand Co., Inc., Princeton, N. J. (1958).
2. Stranathan, J. D., THE "PARTICLES" OF MODERN PHYSICS, Blakiston, Philadelphia, Pa. (1942).
3. Halliday, D., INTRODUCTORY NUCLEAR PHYSICS, John Wiley and Sons, Inc., New York, N. Y. (1955).
4. Lapp, R. E., and Andrews, H. L., NUCLEAR RADIATION PHYSICS, Prentice-Hall, Inc., New York, N. Y. (1955).
5. Friedlander, G., and Kennedy, J. W., NUCLEAR AND RADIO-CHEMISTRY, John Wiley and Sons, Inc., New York, N. Y. (1960).
6. Richards, J. A., Sears, F. W., Wehr, M. R., and Zemansky, M. W., MODERN UNIVERSITY PHYSICS, Addison-Wesley Publishing Company, Inc., Reading, Mass. (1960).
7. Bleuler, E., and Goldsmith, G. J., EXPERIMENTAL NUCLEONICS, Rinehart and Company, Inc., New York, N. Y. (1952).
8. MAXIMUM PERMISSIBLE BODY BURDENS AND MAXIMUM PERMISSIBLE CONCENTRATIONS OF RADIONUCLIDES IN AIR AND IN WATER FOR OCCUPATIONAL EXPOSURE, HANDBOOK 69, published by National Bureau of Standards (June 5, 1959).

SECTION 3 - PROPERTIES OF ALPHA, BETA, GAMMA, X RAYS, AND NEUTRONS

There are four main types of radiation (we include γ and X rays as one type) which must be dealt with in health physics. The properties of these radiations are important in the determination of the relative hazard which they present. These properties will be treated in the following discussions.

A. Alpha Particles

The alpha particle is a helium nucleus (${}_2\text{He}^4$) emitted with a discrete energy and a characteristic half-life from each alpha emitter. Rutherford and Royds showed that the alpha particle is a helium nucleus by collecting α particles from radon in a glass tube with electrodes. After allowing time for build-up, a source of high voltage was connected to the electrodes. When a discharge in the tube occurred, the spectrum obtained showed characteristic helium lines.

Examination of cloud-chamber photographs (see Fig. 3.1) have revealed these properties of α particles:

- 1) Most alphas have the same range in a given gas.
- 2) The alphas travel along straight tracks.
- 3) Some are scattered near the end of their path.
- 4) A few are scattered at distances closer to the source.

The first property mentioned above shows that almost all the α particles emitted by a given radioisotope have about the same discrete energy. The other properties imply that scattering of the alphas occurs infrequently, most often near the end of the path of the particle.

Fig. 3.1

Illustration of Alpha Tracks as Seen in a Cloud Chamber.
(Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation
Physics, second edition. (C) 1954, by permission of
Prentice-Hall, Inc., Englewood Cliffs, N. J.)

Alpha particles are ejected from naturally occurring radioactive atoms with speeds of the order of one-twentieth that of the speed of light. Because of their large mass and relatively high speed, they have large kinetic energies. Alpha particles from naturally radioactive nuclei usually have energies in the range from 4 to 9 MeV.

The energy of an alpha particle is lost by ionization and excitation of the atoms of the traversed substance. Ionization occurs by two processes: (1) direct collision with an orbital electron, and (2) interaction of the electrostatic fields of the α particle and the orbital electrons of the absorbing medium (see Fig. 3.2). Excitation occurs when the energy transferred to the electron is not enough to remove it from the atom (see Fig. 3.3). In this case an ion pair does not result from the loss of energy by the alpha. Consequently, the average energy expended by an alpha particle in creating an ion pair in a given substance is usually greater than the ionization potential of the medium.

Fig. 3.2. Electrostatic Interaction

Fig. 3.3. Excitation

1. Specific Ionization

The W value is defined as the average energy needed to create an ion pair in a given substance. This value is different for alpha particles in various gases, ranging from 22 in Xenon to 46 eV/ion pair (ip) in pure helium, but is relatively constant in a given gas for different α energy. The W value for α particles in air is 35 eV/ip.

Because of the double positive charge and its large mass, an α particle forms a rather high number of ion pairs per unit path length. The specific ionization is defined as the number of ion pairs formed per cm of path. The actual value of the specific ionization for α particles is not a constant. The specific ionization depends upon the velocity of the alpha particle, as well as on the atomic number and mean ionization potential of the absorbing substance. The range of specific ionization in air extends from about 10,000 to 70,000 ip/cm.

If the ionization produced by an alpha particle is plotted against the distance of penetration in a substance, the specific ionization gradually increases as the particle loses energy (see Fig. 3.4). Finally, the ionization reaches a peak value and drops to zero as all the alpha energy is lost. An alpha whose energy is entirely lost in a substance will pick up two electrons and become a neutral helium atom.

Fig. 3.4

The Ionization Increases as the Velocity of the Alpha Particle Decreases with Depth of Penetration. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The specific ionization (SP. I.) well within the range of the alpha particle is given by

$$\text{SP.I.} = \frac{dE/dx}{W},$$

where dE/dx is the linear stopping power of the medium (eV/cm). The linear stopping power gives the energy transferred as ionization and excitation along the path of the charged particle. The mathematical expression for the linear stopping power fails for alpha particles of very low energy. Therefore, one cannot express the specific ionization near the end of the path of the alpha particle.

The average specific ionization can be found from the expression:

$$\overline{\text{SP.I.}} = \frac{\text{No. of ip formed}}{\text{cm of path}} = \frac{E_{\alpha}}{WR} \frac{\text{ip}}{\text{cm}}$$

Here, E_{α} is the energy of the α in eV, and R is the range of the α in cm of the substance.

2. Stopping Power

The linear stopping power for heavy particles of charge ze is given by the expression

$$\frac{dE}{dx} = \frac{4\pi e^4 z^2}{m_0 v^2} NB,$$

where N is the atoms/cc of the medium, B the atomic stopping number (which is a function of the speed v of the particle, the atomic number, and the mean ionization potential of the medium). The linear stopping power is

often expressed in units of MeV/cm. The mass stopping power, S/ρ , is the linear stopping power divided by the density ρ of the substance:

$$\frac{S}{\rho} = \frac{1}{\rho} \frac{dE}{dx} \frac{\text{MeV}}{\text{gm/cm}^2}$$

In health physics work, a quantity of importance is the ratio of two stopping powers or the relative stopping power. For alpha particles, air is often taken as the reference substance, and other media are compared to air. The relative mass stopping power $(S/\rho)_m$ of a medium m with respect to air is then

$$\left(\frac{S}{\rho}\right)_m = \frac{S/\rho \text{ of the medium}}{S/\rho \text{ of air}} = \frac{(B/A)_m}{(B/A)_{\text{air}}}$$

where A is the effective atomic weight.

3. Range

Because of large specific ionization of α particles, they use up their energy in short distances. For instance, the range in air will be only a few centimeters. Most of the alphas from a given source lose their energy in about the same distance. The range R of alpha particles in air at 15°C and 76 cm Hg can be found with the aid of the relationships

$$\begin{aligned} R_a (\text{in cm}) &= 0.56 E \text{ (for } E < 4 \text{ MeV)} \\ &= 0.318 E^{3/2} \text{ (for } 4 < E < 7 \text{ MeV).} \end{aligned}$$

The range of alpha in media other than air can be found approximately by the Bragg-Kleeman relationship:

$$R_m = \frac{\rho_a R_a}{\rho_m (S/\rho)_m}$$

where $(S/\rho)_m$ is the relative mass stopping power of the medium with respect to air. Bragg showed that for a number of substances

$$B_m/B_{\text{air}} \approx \frac{\sqrt{A_m}}{\sqrt{A_a}}$$

From this relationship, then, since

$$\left(\frac{S}{\rho}\right)_m = \frac{B_m A_a}{B_a A_m} \approx \frac{\sqrt{A_m}}{\sqrt{A_a}} \frac{A_a}{A_m} = \frac{\sqrt{A_a}}{\sqrt{A_m}},$$

we find for the range in any substance

$$R_m(\text{in cm}) \approx \frac{3.2 \times 10^{-4} \sqrt{A_m}}{\rho_m} R_a,$$

where $\sqrt{A_a} = 3.82$ and $\rho_a = 1.226 \times 10^{-3} \text{ gm/cc}$ at 15°C and 76 cm of Hg . This relationship is usually good within $\pm 15\%$. The quantity $\sqrt{A_m}$ for a compound or a mixture can be found from the expression

$$\sqrt{A_m} = \frac{n_1 A_1 + n_2 A_2 + n_3 A_3 + \dots}{n_1 \sqrt{A_1} + n_2 \sqrt{A_2} + n_3 \sqrt{A_3} + \dots},$$

where n_i is the atomic fraction of the element i of atomic weight A_i .

Since the mass stopping power in tissue, $(S/\rho)_t$, for alphas is almost equal to the mass stopping power in air, $(S/\rho)_t \approx 1$, and

$$R_{\text{air}} \rho_{\text{air}} \approx R_{\text{tissue}} \rho_{\text{tissue}}.$$

Because there is such a great difference in density between a solid and a gas, the range of alphas in solids will be very small. Find the range of U^{238} α particles ($E = 4.195 \text{ MeV}$) in UO_2 ($\rho = 10.9 \text{ gm/cc}$).

$$\sqrt{A_m} = \frac{\frac{1}{3}(238) + \frac{2}{3}(16)}{\frac{1}{3}\sqrt{238} + \frac{2}{3}\sqrt{16}} = \frac{90}{7.81} = 11.52.$$

$$R_a = 0.318 E^{3/2} = 0.318 (4.195)^{3/2} = 2.73 \text{ cm in air}.$$

$$R_{\text{UO}_2} = \frac{3.2 \times 10^{-4} (11.52) 2.73}{10.9} = 9.25 \times 10^{-4} \text{ cm} = 0.009 \text{ mm}.$$

4. Relative Hazard

The fact that alpha particles have short ranges in dense substances makes the alpha particle much less of a hazard to humans than other external radiations. The alpha particle of highest energy emitted by natural radioactive substances will just penetrate the dead layer of skin on the human body. Consequently, for most α emitters, no living tissue is damaged and no external hazard will exist.

On the other hand, once inside the body, the short range of the α particle becomes very important. In this case, the alpha source will be surrounded by living tissue. This means that the damage will be highly localized near the point of origin of the alphas. Thus, great damage can be done to small essential organs of the body if an α source is lodged in them, since all the α energy will be absorbed in that particular organ and not spread out over a larger volume of tissue. For this reason, alpha particles are a very real concern as an internal hazard.

B. Beta Particles

Beta particles were found to be high-speed electrons emitted from the nucleus of the unstable atom. Further investigation has shown that

either positive electrons (positrons) or negative electrons may be emitted by a radionuclide. The idea of beta emission has been extended to include emission of positrons, and beta particles are considered to be positrons and electrons. The rest mass of these particles is the same; they have equal but opposite charges.

Fig. 3.5. Typical Beta Spectrum. (Ralph E. Lapp and Howard L. Andrews, *Nuclear Radiation Physics*, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.).

Unlike alphas, betas are not emitted with discrete energies but show a continuous energy spectrum (see Fig. 3.5). The atom emits electrons of all energies up to some maximum value. This maximum value for the beta spectrum may be found in nuclide tables and is characteristic of that atom.

The average energy of a source is about $1/3 E_{\max}$ in the case of electrons (β^-) and $0.4 E_{\max}$ in the case of positrons (β^+).

Electrons lose energy in a number of ways as they pass through matter. Loss by collisions with orbital electrons, leading to ionization and excitation, is the most frequent mechanism. But interactions with the nucleus do occur and lead to the emission of X rays (bremsstrahlung). The latter process occurs predominantly for the more energetic electrons.

In a given substance, the electron will make a large number of collisions until it eventually loses all of its kinetic energy. The processes leading to ionization in the substance are the same for the electron as those for the alpha. But because of its smaller mass ($\sim 1/7300$ that of an alpha) and lower charge ($1/2$ that of an alpha), the interactions take place

at less frequent intervals. Consequently, electrons do not produce as many ions per centimeter of path as do alphas.

1. Specific Ionization

The W value for electrons in air is 34 eV/ip. The specific ionization values for electrons are much lower than those for alphas.

Roughly, the specific ionization for electrons varies inversely as the square of the speed of the electron for energies up to 10 MeV. Specific ionization values in air for electrons range from 60 to 7000 ip/cm.

At low energy, the specific ionization decreases as energy increases (see Fig. 3.6). It reaches a minimum value above 1 MeV. Then, the specific ionization increases as the energy increases due to relativistic effects.

Fig. 3.6. The General Shape of the Specific Ionization vs. Energy Curve for β Particles. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition (©) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The specific ionization for betas should be higher than the value for monoenergetic electrons of energy equal to the maximum of the beta spectrum. This is expected because most of the betas emitted by a given source will have lower

energies than the maximum value of the spectrum. Since the specific ionization for electrons increases as the energy decreases, the value obtained for the spectrum should be higher than that for monoenergetic electrons.

Similar to the result obtained for alphas, the specific ionization well within the range of an electron is given by

$$\text{SP.I.} = \frac{dE/dx}{W},$$

where dE/dx is the linear stopping power of the substance, usually expressed in eV/cm. Also, the average specific ionization for electrons can be found from the expression

$$\overline{\text{SP.I.}} = \frac{\text{No. of ip formed}}{\text{cm of path}} = \frac{E_e}{WR} \frac{\text{ip}}{\text{cm}},$$

where E_e is the energy of the electron in eV and R , the range in centimeters of the substance.

The specific ionization of betas is given approximately by

$$SP.I \approx 33 + 63 E_{\max}^{-0.9} \quad \text{when } 0.05 < E_{\max} < 2 \text{ MeV}$$

(see Item 3 in Bibliography for this Section).

2. Stopping Power

The linear stopping power for electrons due to ionization and excitation is given by

$$\frac{dE}{dx} = \frac{2\pi e^4 N Z}{m_0 v^2} B^1,$$

where N is the atoms/cc of the medium of atomic number Z . The electron stopping number, B^1 , is a function of the speed of the electron and the mean ionization potential of the substance. This expression gives the rate of energy loss along the actual path of the electron. Since electrons are greatly scattered in passing through a given substance, the actual path traveled is often much greater than the straight-through path (about 1.2-4 times the straight-line path).

3. Radiation Production - Bremsstrahlung

In addition to that energy lost in ionization and excitation by an electron in passing through matter, some energy may be given up in the form of X rays produced in the substance as a result of interactions between high-speed electrons and nuclei of the absorbing substance. The X-ray spectrum which is obtained in these interactions is called bremsstrahlung (literally, braking radiation). The interactions leading to the production of bremsstrahlung are inefficient for electron energies below 1 MeV. As the energy increases above 1 MeV, the production of X rays becomes increasingly greater.

When a charged particle is either accelerated or decelerated in an electric field, electromagnetic radiation may be given off. If an electron passes close to an atom while traversing a substance, the charge Z on the nucleus will exert a force on the electron. This will cause its path to be bent (see Fig. 3.7). During this acceleration the electron may radiate energy of any amount from zero up to its total kinetic energy E_k . The total bremsstrahlung per atom is roughly proportional to $(Z/m)^2$, where Z is the atomic number of the absorbing matter and m is the mass of the charged particle. Because of a $1/m^2$ dependence, the amount of bremsstrahlung is almost completely negligible for all particles except electrons.

Fig. 3.7. Bremsstrahlung

For electrons in thin targets, the maximum intensity of the bremsstrahlung will be at right angles to an incident beam of low-energy electrons. As the beam energy increases to very high values, the maximum shifts to the forward direction.

For an electron beam incident on a thick target, the fraction of its initial energy which is converted to X rays is given approximately by

$$F \approx 7 \times 10^{-4} Z E_k,$$

where Z is the atomic number of the absorbing substance and E_k is the energy in MeV of the beam. Thus, a 1-MeV beam will lose about 6% of its energy as X rays when the beam is absorbed in lead.

In the case of a beta source, the fraction of beta energy converted to X rays is approximately

$$F \approx 3.33 \times 10^{-4} Z E_{\max},$$

where $E_{\max}$ is the maximum energy of the beta spectrum. In this case, a 1-MeV beta will lose about 3% of its energy as bremsstrahlung when absorbed in lead. The spectral distribution will contain X rays of energy from zero up to $E_{\max}$. In most cases, one may assume that the X rays which are emitted correspond to the maximum energy of the beta for purposes of shielding design. This will normally result in a safety factor in the calculations.

4. Range

The stopping power for electrons decreases as the atomic number Z of the absorber increases. This results because substances of high Z have fewer electrons per gram and these are more tightly bound. But as Z increases, the multiple scattering of the electrons increases. The effect of multiple scattering is to increase the actual path of the electron in a substance. These two effects tend to balance each other, so that the density of a substance gives one a good idea of its relative ability to stop electrons. It is common to express the range of electrons in terms of density-thickness, measured in mg/cm^2 , i.e., $t(\text{cm})\rho(\text{mg}/\text{cm}^3) = \text{Range}(\text{mg}/\text{cm}^2)$. The range is then nearly independent of the type of absorbing substance.

Katz and Penfold (Ref. 20) have expressed the relationship between the range (mg/cm^2) and the energy (MeV) of the electron in mathematical form.

$$R (\text{mg}/\text{cm}^2) = 412 E^{1.265-0.0954 \ln E} \quad (0.01 < E < 2.5 \text{ MeV})$$

or

$$\ln E = 6.63 - 3.2376 (10.2146 - \ln R)^{1/2},$$

and

$$R (\text{mg}/\text{cm}^2) = 530 E - 106 \quad (E > 2.5 \text{ MeV}).$$

These expressions can be used to find the range of electrons in any substance in terms of mg/cm^2 . As far as a continuous beta spectrum is concerned, it turns out that the range of the beta is equal to the range of a monoenergetic electron whose energy is the same as the maximum energy of the beta spectrum. Thus, one can use the same relationship to find the range of betas in matter.

What is the range of a 1.5-MeV β in mg/cm^2 of any substance?

$$R = 412 (1.5)^{1.265-0.0954 \ln 1.5} = 412 (1.5)^{1.265-0.0954(0.40547)}$$

$$= 412 (1.5)^{1.265-0.0386} = 412 (1.5)^{1.226};$$

$$\ln R = \ln 412 (1.5)^{1.226} = \ln 412 + 1.226 \ln 1.5$$

$$= \ln 4.12 + 2 \ln 10 + 1.226 \ln 1.5 = 1.41585 + 2 \ln 10 + 0.49711$$

$$= 1.91296 + 2 \ln 10 = 6.51813;$$

$$R \approx 677 \text{ mg}/\text{cm}^2.$$

5. Exponential β -particle Absorption

Suppose beta particles from a source traverse absorbers of various thicknesses. The ionization produced before and after passing through a given thickness t of the substance is measured. It is found that a plot of the ratio of ionization after passage through absorber to the initial ionization versus absorber thickness on semilog paper yields a straight line. This is similar to the plot one gets for radioactive decay. It tells us that beta absorption is an exponential process. The slope of the straight line gives the constant fractional decrease in the ionization per unit thickness of the substance. This constant, denoted by μ , is called the absorption coefficient. The equation describing β absorption is

$$I = I_0 e^{-\mu t},$$

where I_0 is the initial intensity, I is the intensity after passing through a thickness t of absorber (in cm), and μ is the absorption coefficient (in cm^{-1}). Experiments have shown that the mass absorption coefficient μ/ρ (cm^2/gm), where ρ is the density of the substance in gm/cc , is almost independent of the atomic weight of the absorber. Thus, one can express the mass absorption coefficient by

$$\frac{\mu}{\rho} \approx 17 E_{\text{max}}^{-1.43} \text{ cm}^2/\text{gm}$$

for most substances, where E_{max} is in MeV. In terms of the mass absorption coefficient

$$I = I_0 e^{-(\mu/\rho)x},$$

where x is expressed as gm/cm^2 , i.e., ρt . This relationship is valid provided the distance of penetration is well within the range of the beta.

6. Relative Hazard

A beta source may constitute an external hazard, depending upon the energy of the betas. It takes a beta of about 70-keV energy to penetrate the dead layer of skin and do damage to living tissue. Normally, beta radiation is thought of as only a slight external hazard. A beta source can be fully shielded by a small thickness of aluminum.

An external hazard may result from the production of X rays in the shield in stopping the betas from high-energy sources. The amount of bremsstrahlung can be reduced by choosing a shielding substance of low atomic number Z .

The beta particle is not as great an internal hazard as is the alpha particle. A beta will have a much greater range in tissue, but the specific ionization will be less than that of alpha. Not as much energy is given up in a small volume of tissue as by an alpha. Thus, it will not be as effective in causing damage.

C. Wave Properties

Up to this point, we have been dealing with particles. We turn our attention now to waves. A wave may be defined as a disturbance in a medium. The wave moves through matter with a definite velocity and transmits energy. One sees a wave motion moving across a smooth body of water. These waves show the advance of a form of motion which has a definite velocity. From the movement of the water, one can infer that energy is being propagated. In this case, the wave disturbs the surface of the water.

If one fastens a rope to a rigid body and moves the free end up and down, a wave motion will be set up in the rope. The wavelength is the distance between two adjacent particles in the same phase or displacement. The symbol λ is often used to represent wavelength. The frequency of a wave, ν , is the number of waves which pass a point in unit time. The velocity of the wave is given by

$$V \text{ (in cm/sec)} = \nu\lambda,$$

where ν is the number of waves/sec and λ is cm/wave.

D. Electromagnetic Waves

Electromagnetic waves are of concern to the health physicist. When an electromagnetic wave moves through matter, there is set up an electric field E perpendicular to the direction of propagation and a magnetic field H perpendicular to both the electric field and the direction of motion (see Fig. 3.8). Heat, light, X rays, and γ rays are electromagnetic waves which differ only in wave length. Our work concerns X and γ rays.

Fig. 3.8. Electromagnetic Wave. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (©) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The speed of an electromagnetic wave in a vacuum is 3×10^{10} cm/sec. The letter c is used to denote this speed.

X and γ rays have very short wavelengths. (It is common to denote wavelengths in terms of the angstrom unit, $\text{\AA}$; one angstrom unit equals 10^{-8} cm.) X and γ rays have wavelengths in the range from 10^{-6} to 10^{-11} cm.

Up to 1900, the classical concept pictured radiant energy as being emitted and absorbed in a continuous manner. In an attempt to derive the radiation law theoretically, Max Planck abandoned the classical concept.

He assumed that energy was radiated as quanta of energy which have come to be called photons. These small packets (quanta) were given up in integral units by an emitter. He stated that the quantum of energy, E , is directly proportional to the frequency ν of the radiation. These two terms are related by a fundamental constant, h . Planck called h the action constant, but it is now known as the Planck Constant and has a value of 6.625×10^{-27} erg-sec. The energy E of radiation of frequency ν is given by

$$E = h\nu = hc/\lambda.$$

The intensity of a wave gives the time rate at which the wave transmits energy. By definition, the intensity is the average time rate at which energy is transported by the wave per unit area across a surface which is perpendicular to the direction of motion of the wave. The intensity of X and γ rays is often given in units of MeV/cm²-sec. For an isotropic point source of radiation, the intensity at a point varies inversely as the square of the distance of that point from the source. Given a point source of radiation, the intensity at point A distant r_A from the source is given by

$$I_A \propto 1/r_A^2$$

or

$$I_A = k/r_A^2.$$

At point B, the intensity similarly given by

$$I_B = \frac{k}{r_B^2}.$$

From these two equations we get

$$\frac{I_A}{I_B} = \frac{r_B^2}{r_A^2}$$

or

$$I_A r_A^2 = I_B r_B^2.$$

This expression is the mathematical form for the inverse-square law. The law is subject to two conditions:

- 1) The attenuation of the radiation in the intervening space must be negligible.
- 2) The dimensions of the source and the detector at the point must be small compared with the distance between them.

The inverse-square law expresses the general relationship between the source and the point of interest. If one knows the value of the intensity at A, he can find the value at B. The intensity at A is given in terms of the energy emitted by the source and the distance r_A between A and the source. Let S be the power (MeV/sec) emitted by the source. This energy is emitted in all directions about the point source. If the point source is assumed to be at the center of a sphere of radius r_A , all the energy emitted per unit time will pass through the surface of this sphere. The intensity about the point A is then

$$I \text{ (in MeV/cm}^2\text{-sec)} = S / \text{Surface area of Sphere} = S / 4\pi r_A^2.$$

E. X and γ Radiation

Roentgen discovered X rays in 1895, so termed because of their unknown nature. From his experiments the following properties of this new radiation were determined:

- 1) Most substances are transparent to X rays.
- 2) Many substances glow when exposed to X rays.
- 3) X rays produce ionization in gases.
- 4) X rays are produced when energetic electrons strike solids.
- 5) Photographic plates are affected by X rays.
- 6) X rays are not deflected by electric or magnetic fields.

Fig. 3.9. Typical X-ray Spectrum. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The same properties were found for gamma rays by other workers later. Further work showed that X and γ rays were electromagnetic waves. The only distinction made at the present time is one of origin: gamma rays refer to radiation emitted from the nucleus of the atom; X rays refer to radiation which is produced outside of the nucleus of the atom.

When a beam of energetic electrons is stopped in any dense substance, X-rays (bremsstrahlung) are produced. The spectrum of X rays has a continuous distribution from zero up to the energy E of the electrons. Some lines of much greater intensity than the bremsstrahlung also appear (see Fig. 3.9). These are called

characteristic X rays. The wavelengths of these lines are a property of the target substance. Characteristic X rays appear when electrons from the inner shells (K, L, M) undergo transitions.

The characteristic lines may be excited by any method which will remove electrons from the inner shells. If a K-shell electron is removed, all characteristic lines may appear. If the electrons in the outer orbits of the atom undergo transitions, the characteristic radiation may be in the visible range. The visible spectrum given off by a substance will have lines in the range from 4000 to 8000 Å. The only difference between X rays and the optical spectrum is to be found in the amount of energy associated with the lines. X rays have shorter wavelengths, i.e., higher energy, than the lines of the optical spectrum.

As an electron beam strikes a target, a large portion of its energy is lost as heat. Part of the energy is lost in X-ray production, and there is a probability that all the kinetic energy may be converted to a photon. The wavelength of this photon will be given by

$$E = h\nu = hc/\lambda_{\min},$$

where E represents the energy of the electron. The electron energy is also given by

$$E = Ve,$$

where V is the accelerating voltage and e is the charge on the electron. Thus, we obtain

$$\lambda_{\min} \text{ (in cm)} = \frac{hc}{Ve} = \frac{(6.6 \times 10^{-27})(3 \times 10^{10})}{\frac{V}{300} (4.8 \times 10^{-10})} = \frac{12.4 \times 10^{-5}}{V}$$

or

$$\lambda_{\min} \text{ (in Å)} = 1.24 \times 10^4 / V.$$

Here, $\lambda_{\min}$ is the wavelength of the highest-energy X ray produced for a given accelerating voltage V . If V is increased, a higher-energy X ray can be obtained. The spectrum of X rays contains only a small number of these higher-energy X rays. The highest intensity of X rays produced will occur at roughly $1.5 \lambda_{\min}$ for the continuous portion. The intensity of the radiation is a function of both the accelerating voltage and the beam current.

Gamma rays are emitted from the nucleus of a radioactive atom. Most atoms which decay by beta emission give off gamma rays. Many atoms which decay by alpha emission also give off gamma rays. Gamma rays are emitted when the nucleus of an atom has an excess energy above

its lowest energy state after emitting either an alpha or beta. In fact, these rays may be emitted from any nucleus in an excited state.

1. Interactions with Matter

The transfer of energy to matter by alpha and beta radiation occurs mainly by ionization. Each of these particles produces a rather large number of primary ions in passing through a substance. In the case of X and γ rays, the ionization which is produced is almost all secondary, that is, when an X or γ ray interacts with matter, only a small number of primary ions are formed. These ions in turn produce most of the ionization which occurs in the substance.

The three main ways in which X and γ rays interact with matter are by means of: the photoelectric effect, the Compton effect, and pair production. All three processes yield electrons which then ionize or excite other atoms of the substance.

2. Photoelectric Effect

In the Planck concept, each X or γ ray is a photon with energy $E = h\nu$. The photon retains all of this energy as it moves. The photon may strike an electron in an orbit of an atom of the substance. In the photoelectric effect (see Fig. 3.10), all of the photon energy is given up in this

process. Part of the photon energy is used in removing the electron from the atom: this is the work function ϕ . The rest of the photon energy is carried off as the kinetic energy of the electron. This electron will then cause ionization in the medium. The energy relationship between the photon and the electron is

$$E = h\nu = E_k + \phi,$$

where E_k is the kinetic energy of the electron.

The photoelectric effect is important when the energy of the photon is low, i.e., less than 1 MeV. However, the photon energy must be greater than ϕ for the process to occur at all. For small values

Fig. 3.10. The Photoelectric Effect. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

of E , the photons will interact with electrons in the outer shells of the atom. As E increases, more of the inner electrons are released. Also, the photoelectric effect is more likely to occur in materials with a high atomic number Z ; the effect occurs to a greater extent in lead ($Z = 82$) than in copper ($Z = 29$).

When the electron is removed from an inner orbit, the vacancy will be filled by one of the outer electrons. When such a transition occurs, a photon is emitted. If the photoelectron is removed from the K shell, then all of the characteristic X rays may appear. These X rays are often called fluorescent radiation.

Sometimes the photons strike electrons in the outer orbits of an atom and eject these electrons (see Fig. 3.11). These electrons have kinetic energies which are equal to the energy of the characteristic X ray minus the binding energy of the electron. Such electrons are called Auger electrons.

Fig. 3.11. Process Whereby a Photon Ejects an Auger Electron

Many X rays and Auger electrons may result when the photoelectric effect occurs in a heavy atom. These processes take place as the atom tries to return to its original ground state.

Another process which may occur for a radioactive substance which emits γ rays is called internal conversion. At one time this form of decay was thought to be an internal photoelectric effect. The present view is that it simply is an alternative mode of decay, that is, the excited nucleus may emit a γ ray or it may eject an electron from one of the inner shells. In other words, the nucleus interacts with the electron to get rid of excess energy. If the nucleus emits the γ ray, the energy of the photon is given by $E = h\nu$. If an electron is ejected, its kinetic energy will be $E_k = h\nu - \phi$, where ϕ is the binding energy of the electron in the given shell. This process most often occurs with K-, L-, and M-shell electrons and in the higher- Z emitters.

One may define the internal conversion coefficient α_i as the ratio of the number of electrons from the i^{th} shell to the number of unconverted γ rays. Thus, α_K is number of K electrons/number of unconverted γ rays. Thus, the total coefficient α is given by

$$\alpha = \sum_i \alpha_i.$$

3. Compton Effect

In the case of the Compton effect (see Fig. 3.12), a γ ray impinges upon an electron and gives up only part of its energy. The result is that a photon of lesser energy is scattered at an angle θ with the initial direction of the photon. The electron is scattered at an angle ϕ with the initial direction of the γ ray. This process occurs in such a manner that both energy and momentum are conserved. The electron has a kinetic energy equal to the difference in energy between the incident and scattered photon. The electron will lose this energy by ionization of the atoms in the substance.

Fig. 3.12. The Compton Effect. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (©)1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

The change in wavelength of the photon in a Compton process is given by

$$(\lambda_1 - \lambda_0) = \Delta\lambda = 0.0242(1 - \cos \theta)$$

(in Angstroms), where λ_0 and λ_1 are the wavelengths of the initial and scattered photons, respectively. From this one can see that the change in wavelength of the scattered radiation does not depend upon the energy of the incident photon.

The maximum energy which an electron can acquire in a Compton process is attained when the photon is backscattered ($\theta = 180^\circ$). In this case, the energy of the electron is given by

$$E_e = \frac{E_\gamma^2}{E_\gamma + \frac{m_0 c^2}{(1 - \cos \theta)}} = \frac{E_\gamma^2}{E_\gamma + \frac{0.51}{2}} = \frac{E_\gamma^2}{E_\gamma + 0.256} \text{ (in MeV),}$$

where E_γ is the initial energy of the photon in MeV. The energy E_1 of the scattered photon is given by

$$E_1 = \frac{E_\gamma}{1 + 1.96 E_\gamma (1 - \cos \theta)}$$

The Compton effect is important for gamma-ray energies between about 200 keV and 5 MeV in most light elements. The Compton effect decreases with increasing gamma energy but not as quickly as the photoelectric effect. The Compton process depends upon the number of electrons

in the substance. Compton scattering will then be proportional to Z/A . The process is predominant in substances of intermediate Z .

For the gamma rays of interest to health physicists, the predominant interaction is the Compton effect. This presents a problem in shield design since the scattered photons are not truly removed or absorbed from a beam of radiation. In a wide beam of radiation and a thick shield, some γ rays which are at first scattered out of a beam may later be scattered back into the beam. This leads to a greater amount of radiation reaching a point of interest than one would expect. For this reason, a correction must often be made to take into account this buildup effect.

4. Pair Production

The process in which a photon of sufficient energy gives up all its energy and forms two particles, an electron and a positron, is called pair production. This process is pictured (see Fig. 3.13) as occurring near the nucleus of an atom, for it is only in this way that momentum can be conserved. The minimum energy needed for pair production is given by $E = 2m_0c^2$, where m_0 is the rest mass of an electron or positron. Since the rest mass of an electron is equivalent to 0.51 MeV, the photon must have an energy ≥ 1.02 MeV for pair production to occur.

Fig. 3.13. Pair Production and Annihilation. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

When this process does occur, energy of the photon beyond 1.02 MeV is imparted as kinetic energy to the electron-positron pair and a portion to the nucleus in order to conserve momentum. Both electron and positron lose kinetic energy through ionization of atoms in the substance. Eventually the positron interacts with an electron in the substance in a process called annihilation. In this process, the mass of the particles is changed into two

photons of 0.51 MeV each, emitted in nearly opposite directions. They may interact further in the substance through the photoelectric or Compton effects.

Since this process does not occur at all unless the gamma energy is ≥ 1.02 MeV, pair production is important only for gammas of high energy. The process is also proportional to Z^2 of the absorber. Thus, pair production predominates for substances of high Z .

The three processes: photoelectric effect, Compton effect, and pair production, account for the main photon interactions with matter. Both the photoelectric and Compton effects decrease with increase of gamma energy. Pair production increases with gamma energy. The result of these effects is that each substance will have minimum absorption at some gamma energy.

In health physics, we are concerned with the absorption of radiation by matter so that we can protect people and equipment from these rays. The three processes produce electrons which then ionize the absorbing matter. In a dense substance, the range of the electrons will be very short. The more interactions which occur in a given substance will determine how well one is absorbing the radiation.

5. Absorption of X and γ Rays

Suppose a narrow beam of monoenergetic photons is sent through a substance. Photon interactions may occur in any of the three ways already mentioned. Energy will be removed from the beam by

Compton scattering and by absorption, absorption occurring in the photoelectric effect and pair production. Since we assume a narrow beam, a Compton scattering process will be the equivalent of an absorption, that is, a scattered photon will be removed from the beam.

Fig. 3.14. Absorption of X and γ rays. (Ralph E. Lapp and Howard L. Andrews, Nuclear Radiation Physics, second edition. (C) 1954, by permission of Prentice-Hall, Inc., Englewood Cliffs, N. J.)

Let us measure the relative intensity of the beam as it passes through various thicknesses x of the absorber. A plot of the relative intensity versus absorber thickness on semilog paper will give a straight line (see Fig. 3.14). As is the case for radioactive decay, X- and γ -ray absorption is an exponential process: there is a constant fractional decrease in intensity per unit thickness of the substance. The value of this constant, denoted by

μ and called the total linear attenuation coefficient, is the slope of the straight line on the semilog plot.

For a narrow beam of monoenergetic photons, X- and γ -ray attenuation is described by the equation

$$I = I_0 e^{-\mu x},$$

where I_0 is the intensity at zero thickness, I is the intensity after passing through a thickness x (in cm), and μ is the total linear attenuation coefficient (in cm^{-1}).

The total linear attenuation coefficient μ represents the probability of photon interaction per unit path length. The quantity μ is the sum of the probability for each of the three processes, that is,

$$\mu = \tau + \sigma + \pi,$$

where τ is the linear attenuation coefficient for the photoelectric effect, σ is that for the Compton effect, and π is that for pair production. The coefficient μ is a function of the energy of the gamma and the absorbing substance. Thus, μ for a given material is a constant only for a given energy of X and γ radiation. One must consult attenuation coefficient curves in order to find the value of μ for a given energy of X or γ rays in a certain substance. Such curves can be found in the literature (Ref. 11). In these curves, one will more often find the mass attenuation coefficient plotted.

The mass attenuation coefficient μ_m is simply the linear attenuation coefficient divided by the density ρ of the absorber:

$$\mu_m \text{ (in cm}^2\text{/gm)} = \mu/\rho,$$

where ρ is in gm/cc. Thus,

$$\mu = \mu_m \rho \text{ (cm}^{-1}\text{)}.$$

What is the linear attenuation coefficient for a 1-MeV γ in lead? The density of lead is 11.3 gm/cc and $\mu_m = 0.046 \text{ cm}^2\text{/gm}$:

$$\mu = \mu_m \rho = (0.046)(11.3) = 0.52 \text{ cm}^{-1}.$$

In the case of a compound substance, the coefficient for the compound can be found from the equation

$$\mu = \sum_{i=1}^n \left(\frac{\mu}{\rho} \right)_i \rho_i,$$

where $(\mu/\rho)_i$ is the mass attenuation coefficient of the i^{th} element of the compound. The symbol ρ_i represents the density of the i^{th} element as it appears in the mixture. It is given by the product of the weight fraction of the i^{th} element and the density of the compound. The weight fraction of an element in a compound is found from:

$$(\text{Weight Fraction})_i = k_i A_i / A,$$

where k_i is the number of the i^{th} atoms of atomic mass A_i , and A is the molecular weight of the compound.

For example, find the attenuation coefficient for 1-MeV γ rays in H_2O . The mass attenuation coefficients for H and O are 0.126 and $0.0635 \text{ cm}^2/\text{gm}$, respectively. Then,

$$\begin{aligned} \mu &= \sum_{i=1}^2 \left(\frac{\mu}{\rho} \right)_i \rho_i = \left(\frac{\mu}{\rho} \right)_O \rho_O + \left(\frac{\mu}{\rho} \right)_H \rho_H \\ &= 0.0635 \frac{k_O A_O}{A_{\text{H}_2\text{O}}} \rho_{\text{H}_2\text{O}} + 0.126 \frac{k_H A_H}{A_{\text{H}_2\text{O}}} \rho_{\text{H}_2\text{O}} \\ &= 0.0635 \frac{(1)(16)}{18} (1) + 0.126 \frac{(2)(1)}{18} (1) = 0.0704 \text{ cm}^{-1}. \end{aligned}$$

6. Half-value Layer

A quantity of usefulness in quick shielding estimates is the half-value layer. This is the thickness of a substance which reduces the intensity of a beam of radiation to one-half of its initial value. This concept is similar to the half-life in radioactivity. The half-value layer is found from the equation

$$x_{1/2} = 0.693/\mu.$$

7. Mean Free Path

The mean free path λ is the average distance which a photon travels before interacting. The mean free path is equal to the reciprocal of the attenuation coefficient:

$$\lambda = 1/\mu.$$

If the thickness of an absorber is equal to one mean free path, then the intensity of the gamma beam will be reduced by a factor of e . For a narrow beam of γ rays, the mean free path is sometimes called the relaxation length.

8. $\mu - \sigma_s$

Up to this point, all the concepts have been based upon a narrow beam of γ rays. In this case, if a gamma is scattered out of the beam, it is assumed to be lost from the beam. The use of the total coefficient μ is thus valid. This coefficient is often called the narrow beam coefficient.

If one treats a wide beam of radiation (see Fig. 3.15), then along a path well within the beam, some of the rays will be scattered away from this path. On the other hand, rays from other parts of the beam may be scattered into this path. The question arises whether as many are scattered into the path as are scattered away from it. The net affect is difficult to assess.

Fig. 3.15. Wide Beam of γ Rays in an Absorber

One approach to this problem assumes that all scattered radiation remains along the path.

This is done by taking the total coefficient μ and subtracting the probability σ_s that scattering occurs. This new coefficient, $\mu - \sigma_s$, is then used in place of μ . The attenuation of the beam is then given by

$$I = I_0 e^{-(\mu - \sigma_s)x}$$

By using $\mu - \sigma_s$, one assumes that any energy scattered out of the path is replaced by energy scattered into the path. This new coefficient gives the probability of energy being absorbed per unit path length.

In most cases of a wide beam of radiation passing through an absorbing substance to a detector, some of the scattered radiation is lost from the beam and never reaches the detector. The use of the coefficient $\mu - \sigma_s$ in these cases leads to an overestimation of the flux reaching a detector. The use of the coefficient μ leads to an underestimation of the flux reaching a detector. In health physics work, the coefficient $\mu - \sigma_s$ is preferred since it will introduce a safety factor in calculations.

9. Buildup Factor

The buildup factor B is defined as the ratio of the actual gamma flux to that which would be calculated by use of the narrow-beam coefficient:

$$B = \frac{\text{true flux}}{\text{calculated flux}} = \frac{I}{I_0 e^{-\mu x}}$$

or

$$I = I_0 B e^{-\mu x}$$

In this case one can calculate the true flux I if the value of the buildup factor is known. The buildup factor corrects for the underestimation of scattered radiation reaching the detector. Curves of buildup factors are available in the literature. The use of the buildup factor concept will be discussed in the section on shielding.

10. Relative Hazard

X and γ rays are of most concern as potential external hazards. These rays have great ranges in air and other substances, and may present a significant hazard at fairly large distances from a source. In most cases shielding is needed to insure that no hazard exists. The shield which is required is much greater than for either alpha or beta radiation. Also, scattered rays from a source may contribute to the hazard, that is, a shield may provide good protection in one direction, but scattered rays from this shield may cause a hazard to exist in another area. Any cracks or breaks in an otherwise adequate shield may lead to streaming or beams of radiation.

Because X and γ rays do penetrate substances to such a high degree, the damage done to tissue will extend throughout the body. Thus, the more radiosensitive tissues are exposed when one is in an external gamma field. Damage to these organs affects the body more than damage to less critical tissues, such as skin. For this reason, X and γ rays are more of an external hazard than either alpha or beta rays.

On the other hand, X and γ radiation is not as great an internal hazard as is alpha or beta radiation. As is the case with beta radiation, the long range of gamma in tissue means that less energy will be given up in a small volume of tissue. A photon will give up energy only when it interacts. It does not produce a continuous path of ionization as do alpha and beta particles. For this reason, a small organ in the body will receive less damage from a gamma source than an equal alpha or beta source.

F. Neutrons

In previous sections, we have discussed the properties of the three types of natural radiation. These occur as the result of the natural decay of a nucleus. In this section, we will discuss the neutron. The neutron is found mainly as the result of nuclear reactions.

The work of Bothe and Becker showed that a very penetrating radiation was emitted when beryllium was bombarded with alphas from polonium. They assumed that the radiation consisted of gamma rays. Curie and Joliot showed that this radiation ejected protons from a sheet of paraffin. Chadwick applied the concepts of conservation of energy and momentum to show that the gamma-ray assumption would not hold. He assumed that the radiation consisted of particles of zero charge and mass about equal to the proton, which he named neutrons. Thus, neutrons are emitted when beryllium absorbs an alpha according to the reaction

This work by Chadwick in 1932 indicated that the neutron came from the nucleus. This view helped to form the present concept of a nucleus composed of protons and neutrons. Later work revealed that neutrons are emitted from almost any element when bombarded with high-energy particles. Also, neutrons are produced by cosmic-ray bombardment of the earth's atmosphere.

Studies of the properties of the neutron confirmed that it is slightly larger than a proton in mass. The atomic number Z of a neutron is zero, since it has no net charge. Free neutrons are unstable: they decay by beta emission ($E_{\max} = 0.78 \text{ MeV}$) with a half-life $T_{1/2} \approx 12.8 \text{ min.}$

1. Sources of Neutrons

Since the discovery of the neutron, studies have shown that neutrons can be produced in a variety of ways. The use of radioactive sources and certain target substances may yield neutrons by either the (α, n) or (γ, n) reactions. Accelerators produce neutrons when high-energy particles strike suitable targets. Neutrons also result from the fission process in a reactor. Some of the transuranic elements can be used as neutron sources since they undergo spontaneous fission.

Some of the properties of these sources are:

1. (α, n) reactions - These sources are often prepared by making an intimate mixture of an alpha emitter and finely divided powder of the target substance. Ra^{226} , Po^{210} , and Pu^{239} are the common alpha emitters used. B, Be, Li, Na, and F are used as target materials. Be is most commonly used since it gives the highest yield. These sources emit neutrons which have a spectrum of energies. Source strength is often expressed in terms of the amount of the α emitters.

Because of the high toxicity of Ra, Po, Pu, and Be, these sources must be sealed in metal containers. Ra-Be sources provide a strong neutron source but have a high gamma-ray background. Pu-Be sources have low gamma activity, but give a lower yield than Ra-Be. Po-Be sources also have a low gamma activity, but the Po^{210} half-life (138 days) is short. Radium sources have the added danger of radon production; since radon is a radioactive gas, one must be sure that the container is adequate.

The emission of neutrons is often anisotropic. This results from a loss of spherical symmetry in the source. Each source should be checked for anisotropy. This effect can have a magnitude of 10-30%.

The characteristics of some (α, n) sources are given on page 10, NBS Handbook #72.

2. (γ, n) reactions - Some nuclei will emit neutrons when exposed to gamma rays. Most sources do not emit gamma rays with energy greater than 4 MeV. The target substances are thus limited to beryllium and deuterium.

The gamma background is always high in these sources. A further drawback is the short half-life of the gamma emitters which are used. The emission rate for these sources is closely proportional to the amount of the gamma emitter. These sources emit nearly monoenergetic neutrons.

The characteristics of some (γ, n) sources are given on page 11, NBS Handbook #72.

3. Accelerator sources - Nuclear reactions are caused by high-speed charged particles impinging on a suitable target yield neutrons. Types of reactions used are: (α, n), ($\alpha, 2n$), (p, n), (γ, n), and (d, n). Accelerator sources are useful for producing monoenergetic neutrons over a wide energy range. Since these machines produce very high-energy charged particles, many new target substances can be used.

4. Reactors - The fission process in reactors produces neutrons with a spectrum of energy. A reactor provides a significant source of neutrons since, on the average, 2.5 neutrons are emitted per fission.

5. Spontaneous fission sources - The neutron spectra from these sources are similar to that produced in reactors. However, their expense precludes their use as standard sources at the present time. The characteristics of these sources are given on page 12, NBS Handbook #72.

2. Neutron Energy

The types of interactions which neutrons undergo depend quite strongly on the energy of the neutron. As in the case of gamma rays, certain effects predominate depending on the energy of neutron. It is often advantageous to treat neutron interactions in terms of the energy range in which they predominate. The NCRP has classified neutrons according to the following energy scheme:

1. Thermal neutrons - When neutrons are slowed down so that they are in thermal equilibrium with matter, they have speeds comparable to gas molecules at room temperature. In this case, the most probable velocity is about 2.2×10^5 cm/sec (2200 m/sec). The energy of thermal neutrons is then

$$E = \frac{1}{2} mv^2 = \frac{1}{2} (1)(1.66 \times 10^{-24})(2.2 \times 10^5)^2 = 4.02 \times 10^{-14} \text{ ergs}$$

$$= 0.025 \text{ eV.}$$

2. Intermediate neutrons - This energy region of neutrons goes from 0.5 eV to 10 keV. Neutrons in this interval are often called "resonance neutrons."

3. Fast neutrons - Fast neutrons range in energy from 10 keV to 10 MeV.

4. Relativistic neutrons - This range includes neutrons of energy greater than 10 MeV.

3. Interactions with Matter

The study of neutron interactions with matter forms a large part of the present experimental work in the field of physics. There are a number of processes which a neutron can enter into while passing through matter. The particular effect which occurs depends upon the substances and the energy of the neutron.

The binding energy of a neutron in any element is about 8 MeV, except for the lighter nuclei. This is the energy which holds a neutron in the nucleus. To remove the neutron one must supply at least this much energy. For a relativistic neutron, then, nuclear reactions are the most important interaction processes. For neutrons with energy ranging from 9 to 20 MeV, the (n,2n) reaction occurs. As the neutron energy increases, more complex processes, such as spallation, can occur. Spallation is a term used to describe a process in which a number of light fragments are emitted from an excited nucleus. Since the advent of high-energy accelerators, more of these reactions are being found and studied.

Fig. 3.16. Elastic Collision. Energy and momentum are conserved.

For fast neutrons, elastic scattering is the predominant mode of interaction. Because the neutron has no charge, it does not interact with the electric field of a nucleus and can pass very close to charged particles. For this reason, one can view neutron interactions as collisions with nuclei. An elastic collision is pictured as a billiard-ball type of collision in which energy and momentum will be conserved (see Fig. 3.16).

The loss of energy by a neutron in an elastic collision is greatest when the colliding particles have equal mass. Thus, a neutron will lose the most energy in a collision with a proton (hydrogen nucleus). The average fraction of energy lost is about the same for each collision. The energy which is lost appears as the kinetic energy of the recoil nucleus. For a neutron of energy E_n , the average energy lost in a single collision with hydrogen is $\sim 2/3 E_n$. As the mass of the nucleus becomes very large, the neutron will lose hardly any energy. For this reason, substances with high Z are poor for use in slowing down neutrons.

In the energy range above 0.5 MeV, inelastic scattering begins to occur. This process is pictured as a putty-ball type of collision (see Fig. 3.17). In the process, the neutron raises the nucleus to an excited state and comes off with a loss in energy. The excited nucleus often returns to the ground state by emitting the excess energy as a gamma ray. For this process to occur, the neutron must supply the excitation energy to the nucleus. In heavy elements, the first excitation state is about 100 keV above the ground state. For light elements, this state may be at 3 or 4 MeV. This process, then, is more important for heavier nuclei and neutrons of higher energy (> 1 MeV).

Fig. 3.17
Inelastic Collision

Near the upper end of the fast neutron range, inelastic scattering as well as nuclear reactions begin to occur as frequently as elastic scattering.

The process of elastic scattering is still dominant for intermediate neutrons. The phenomenon of resonance absorption also occurs in this region. When a neutron has an energy which is equal to a nuclear energy level of the absorbing substance, the probability of absorption increases greatly. Thus the substance will absorb more neutrons of this energy than of energy either slightly higher or lower. When the neutron energy becomes less than about 100 eV, capture becomes an important process.

As the energy of the neutron is further decreased, it approaches the thermal-neutron range. The dominant process for thermal neutrons is capture, in which the neutron becomes part of the absorbing nucleus. The compound nucleus thus formed must then get rid of excess energy, usually by

emission of gamma rays. This is called radiative capture or an (n,γ) reaction. For some light nuclei, the (n,p) reaction can occur. In this case, the capture of a neutron leads to the emission of a proton. For slow-neutron capture in B and Li, the (n,α) reaction occurs. The capture of thermal neutrons in certain heavy nuclei such as uranium and plutonium leads to fission.

Ionization which is produced as the result of neutron interaction with matter is almost totally secondary in nature. The neutron gives up energy to a recoil nucleus which causes ionization in the medium. In the case of capture or nuclear reactions, ionization occurs due to the emission of charged particles from the excited nucleus. If an (n,γ) capture takes place, the resulting ionization will depend upon the gamma interacting in the medium. The loss of energy by neutrons, as well as by gamma rays, is not a continuous process as it is for alpha and beta particles. The neutron, or the gamma ray, will move through matter with little interference until an interaction takes place. When this occurs, then energy will be lost.

4. Cross Sections

As a beam of neutrons moves through matter, certain interactions occur. From the preceding section we know that the nature of the substance and the energy of the neutron will make certain processes more likely to occur. In discussing neutron interactions, the term cross section, denoted by σ , is used to express the probability that a neutron will interact with a given substance. For any given process, if the probability is high, the cross section will be large. The cross section is expressed by means of a unit called the barn (b), equal to 10^{-24} cm². The barn expresses the probability in terms of an area. In a sense, then, the atom may be viewed as presenting an effective target area to a neutron. If a neutron passes through this area, the reaction occurs.

The cross section σ is called the microscopic cross section since it expresses the probability per atom. Each possible interaction has its own probability. The total microscopic cross section, σ_t , is the sum of the separate cross sections for all processes which may occur:

$$\sigma_t = \sigma_{\text{scatter}} + \sigma_{\text{capture}} + \sigma_{\text{fission}} + \dots$$

One should not construe that the cross section is a measure of the geometrical area of the nucleus. The effective target area which a nucleus presents to a neutron is often much greater than the impact area. For example, the radii of a proton and a neutron are about the same: 1.5×10^{-13} cm. For an impact to just occur, the particle centers cannot be more than 3×10^{-13} cm apart (see Fig. 3.18). The impact area would then be a circle of area $\pi(3 \times 10^{-13})^2 = 28.3 \times 10^{-26}$ cm². But the cross section for capture in hydrogen for 0.01-eV neutrons is 48 b, or 4800×10^{-26} cm².

It is clear that the effective area is much greater than the impact area. One can see from this that σ is much more than simply a geometric area. Curves for σ_t can be found in the literature. (21,22)

Fig. 3.18
Impact Area

In dealing with neutron penetration through a substance, one may wish to express the probability of a reaction in terms of the thickness of the substance. In such a case, the probability per unit path length, called the macroscopic cross section Σ , is desired. The total macroscopic cross section Σ_t is related to the total microscopic cross section σ_t by the equation:

$$\Sigma_t = N\sigma_t \text{ cm}^{-1},$$

where N is the number of atoms per cubic centimeter of the substance.

Let us picture a cubic centimeter of a substance in terms of a cross-sectional area of 1 cm^2 and a thickness 1 cm . Each atom of the substance has a cross section σ_t . The total reaction cross section will be the sum of the contributions from each atom. But N is the total number of atoms. Thus, $N\sigma_t$ is the total reaction cross section in the 1-cm thickness.

The macroscopic cross section Σ for neutrons is similar to the attenuation coefficient μ for gammas, since both quantities express the probability per unit path length. By analogy, the mean free path λ for neutrons is given by

$$\lambda = 1/\Sigma_t.$$

In this case, λ gives the average distance a neutron travels before interacting in a substance.

5. Neutron Absorption

Assume a narrow beam of neutrons is passing through a substance. Let k be the number of neutrons per cubic centimeter in the beam. Assume, further, that each neutron has a speed v . Then the neutron flux ϕ is given by

$$\phi = kv \text{ (in n/cm}^2\text{-sec).}$$

The flux gives the number of neutrons passing through one square centimeter per unit time. In terms of our previous picture of one cubic centimeter of a substance, the probability of a reaction in passing through 1 cm of the substance is given by Σ_t . The number of reactions per unit time in this centimeter of the substance will be

$$\frac{\text{number of reactions}}{\text{cc-sec}} = \phi \Sigma_t.$$

The decrease in the flux as the beam moves through a thickness Δx will be

$$\Delta \phi = -\phi \Sigma_t \Delta x.$$

In this case we assume that if a neutron is scattered, it is removed from the beam. This may be written as a differential equation:

$$\frac{d\phi}{dx} = -\Sigma_t \phi.$$

This form is the well-known expression for an exponential relationship. When integrated, then is obtained

$$\phi = \phi_0 e^{-\Sigma_t x},$$

where ϕ_0 is the initial flux of neutrons. If this equation is plotted on semilog paper, one will get a straight line whose slope will be the value of $-\Sigma_t$.

As is the case with gamma rays, this type of relationship is valid only if one can consider the scattered neutrons as being removed from the beam. For a narrow beam, this condition may be assumed. For a wide beam, some of the neutrons scattered out of the beam will be replaced by neutrons scattered into the beam. The use of the total cross section Σ_t may thus lead to an overestimation of the effectiveness of an absorber.

6. Removal Cross Section Σ_R

Because of the large amount of energy lost by a fast neutron in a collision with hydrogen, this process in effect removes the neutron from the beam. One can then view neutron attenuation in terms of a removal concept. The removal cross section Σ_R is then used to calculate neutron attenuation.

The macroscopic removal cross section is given by

$$\Sigma_R = \frac{0.602 \rho}{A} \sigma_r \text{ (in cm}^{-1}\text{)},$$

where σ_r is the microscopic removal cross section in b/atom. It has been found that σ_r is about equal to $3/4$ of the total cross section, σ_t , at 8 MeV, except for hydrogen.

The concept of removal cross section is based upon the presence of hydrogen in the absorber. It can also be applied to other substances backed up by hydrogen absorbers. The concept is valid for fast neutrons in the range 2-12 MeV. In this energy range, neutrons which suffer inelastic collisions in a heavy substance will lose a large part of their energy. In this case, they then may be captured by hydrogen. If they suffer an elastic collision with a heavy nucleus, they lose hardly any energy. However, an elastic collision with hydrogen causes a large transfer of energy. In this case the neutron will also be effectively removed.

Without the presence of hydrogen in the absorber or backing up the absorber, the removal concept does not apply. In the neutron-energy range treated, elastic or inelastic scattering with heavy nuclei will occur. Again, hardly any energy is lost in an elastic collision. When an inelastic collision occurs, energy will be lost. As the energy of the neutrons drops below about 1 MeV, inelastic scattering will become less probable and the neutrons will still have too much energy to be captured. In this case, the substance is transparent to the neutrons, which stream through. Since these neutrons have not been removed from the beam, the use of the removal cross section will lead to poor results for the calculated attenuation.

If the quantity Σ_R/ρ is plotted against the atomic weight A of an element, a smooth curve results. One can obtain the removal cross section for any element from this plot. Such a curve can be found in NBS Handbook #63 on page 79. The removal cross section for a compound can be calculated if one knows the removal cross sections for each of the constituent elements. The removal cross section of substances mixed together is assumed to be additive, so that

$$\Sigma_R(\text{compound}) = \sum_{i=1}^n (\Sigma_R/\rho)_i \rho_i,$$

where $(\Sigma_R/\rho)_i$ is the value obtained from the curve in Handbook #63 for the i^{th} element, and ρ_i is the density of the i^{th} element as it appears in the compound.

The concept of removal cross section is based on data obtained with thick shields. When thin shields are used, scattered neutrons will not be removed as well. For this type of problem, a buildup factor must be used. The attenuation expression will then be

$$\phi = \phi_0 \text{Be}^{-\Sigma_R x}.$$

The buildup factor for neutrons will be discussed in Section 5.

7. Neutron Activation

One of the processes by which a neutron is attenuated is absorption by a nucleus. For many substances, this process produces a compound nucleus which is unstable. The substance may then become radioactive. Many artificially radioactive substances are produced when other substances are exposed to the neutron flux in a reactor. Absorption occurs mainly for thermal neutrons. For this reason, most artificial radioisotopes have been produced in reactors. When a nucleus absorbs a thermal neutron, the term activation is used to imply that the product becomes radioactive. The thermal activation cross section, σ_{act} , expresses the probability that a certain substance will be activated if it absorbs a thermal neutron.

Assume a sample containing N atoms is exposed to a thermal neutron flux ϕ . If σ_{act} is the probability of activation per atom, the total probability is $N\sigma_{\text{act}}$. The rate of formation of radioactive atoms is then

$$\text{formation rate} = \phi \sigma_{\text{act}} N.$$

During the time that radioactive atoms are being formed, some of the atoms will decay. The decay rate of the radioactive atoms N^1 will be given by

$$\text{decay rate} = \lambda N^1,$$

where N^1 is the number of radioactive atoms present and λ is the decay constant. The net rate of growth is thus

rate of growth = formation rate - decay rate.

The corresponding differential equation is

$$\frac{dN^1}{dt} = \phi\sigma_{\text{act}}N - \lambda N^1.$$

If the flux is constant for the time t that the sample is irradiated, the equation can be integrated to give

$$N_t^1 = \frac{\phi\sigma_{\text{act}}N}{\lambda} (1 - e^{-\lambda t}),$$

where N_t^1 gives the number of radioactive atoms present after an irradiation time t . The activity A_t of the sample at the time t that the irradiation is stopped is expressed by

$$A_t = \lambda N_t^1 = \phi\sigma_{\text{act}}N(1 - e^{-\lambda t}).$$

The activity at time t_1 after a sample is removed, is given by:

$$A = A_t e^{-\lambda t_1} = \phi\sigma_{\text{act}}N(1 - e^{-\lambda t}) e^{-\lambda t_1}.$$

The activity A_t of the sample in the flux will increase until the formation rate is equal to the decay rate. At this point, no further increase in activity will occur. For a given sample, the maximum value that the activity will reach is $\phi\sigma_{\text{act}}N$, which is called the saturation activity A_s . The term $(1 - e^{-\lambda t})$ varies from 0 for $t = 0$ to 1 for $t = \infty$. For any time t , this term gives the fraction of A_s which the irradiated sample reaches.

The equation is valid if all target atoms receive the same flux. In effect, this means that samples must be thin. Also, the relationship assumes that the number of target atoms remains about constant. This means that there should be no significant decrease in the number of target atoms.

Let us take an example of the use of this expression. A one-gram tungsten sample is exposed for 3 days at an average thermal-neutron flux of 10^{11} n/cm²-sec in a reactor. Given that σ_{act} is 40 b and that the half-life of W^{187} is 24 hr. What is the activity of the sample 12 hr after irradiation? Also, determine A_s .

$$A = \phi\sigma_{\text{act}}N(1 - e^{-\lambda t}) e^{-\lambda t_1}.$$

$$N = (m/A) N_A = (1/186) 6.025 \times 10^{23}; \quad t = 3 \text{ d}, \quad t_1 = 0.5 \text{ d}.$$

$$\sigma_{\text{act}} = 40 \text{ b} = 40 \times 10^{-24} \text{ cm}^2.$$

$$\lambda = 0.693/1 \text{ day} = 0.693 \text{ d}^{-1}.$$

$$\phi = 10^{11} \text{ n/cm}^2 \text{ sec.}$$

Therefore,

$$\begin{aligned} A &= 10^{11}(40 \times 10^{-24})\left(\frac{6.025 \times 10^{23}}{186}\right)(1 - e^{-0.693(3)})e^{-0.693(0.5)} \\ &= 8.02 \times 10^9 \text{ dps} = 0.217 \text{ Curie} \end{aligned}$$

and

$$\begin{aligned} A_s &= \phi \sigma_{\text{act}} N = 10^{11}(40 \times 10^{-24})\left(\frac{6.025 \times 10^{23}}{186}\right) \\ &= 1.296 \times 10^{10} \text{ dps} = 0.35 \text{ Curie} \end{aligned}$$

8. Relative Hazard

Along with gamma rays, neutrons are of most concern as potential external hazards. Since neutrons possess no charge, they travel great distances in air and other substances. Thus, it is more difficult to provide shielding for neutrons than for alpha or beta particles. As for gamma rays, scattered neutrons can add to the hazard.

Because the neutron moves through matter rather freely until it interacts, the damage done in tissue extends throughout the body. Just as for gamma rays, the more radiosensitive tissues will receive damage from an external neutron field. The damage done in tissue is a function of neutron energy. Fast and relativistic neutrons (energy < 30 MeV) produce many elastic collisions in tissue. Since almost 2/3 of the body atoms are hydrogen, about 80-95% of the energy given up in tissue by fast neutrons occurs in elastic scattering by hydrogen. The recoil protons which are produced will lose their kinetic energy by ionizing tissue. For this reason, fast neutrons are more hazardous than neutrons of lower energy.

Intermediate and thermal neutrons produce damage in tissue as the result of absorption. In this case, the $\text{H}(n, \gamma)\text{D}$ and $\text{N}^{14}(n, p)\text{C}^{14}$ reactions are important. For a large mass of tissue, the $\text{H}(n, \gamma)\text{D}$ process is more important than the $\text{N}^{14}(n, p)\text{C}^{14}$ process. For a small mass, the reverse is true. Since a gamma photon has a much greater range in tissue than a proton, the damage in a small organ will be greater for a recoil proton than a gamma. For a large organ, the gamma dose will be greater.

Since some of the neutron interactions can produce heavy recoil particles, protons, and gammas, the transfer of energy to tissue will vary. However, the effectiveness of neutrons in producing damage is from 2.5-10 times that of gamma rays. On this basis, the neutron is more hazardous than gamma rays. However, each radiation is significant as an external hazard; one cannot ignore the gamma field if a neutron field is also present.

The neutron is not treated as an internal hazard. Since there are no natural neutron-emitting sources, there is little chance of a neutron source being fixed in the body.

BIBLIOGRAPHY

1. Evans, R. D., THE ATOMIC NUCLEUS, McGraw-Hill Book Co., Inc., New York, N. Y. (1955).
2. Kaplan, I., NUCLEAR PHYSICS, Addison-Wesley Publishing Co., Inc., Reading, Pa. (1958).
3. Blatz, H., Editor, RADIATION HYGIENE HANDBOOK, McGraw-Hill Book Co., Inc., New York, N. Y. (1959).
4. Halliday, D., INTRODUCTORY NUCLEAR PHYSICS, John Wiley & Sons, Inc., New York, N. Y. (1955).
5. Friedlander, G., and Kennedy, J. W., NUCLEAR AND RADIOCHEMISTRY, John Wiley & Sons, Inc., New York, N. Y. (1960).
6. Glasstone, S., SOURCEBOOK ON ATOMIC ENERGY, D. Van Nostrand Co., Inc., Princeton, N. J. (1958).
7. Stranathan, J. D., THE "PARTICLES" OF MODERN PHYSICS, The Blakiston Co., Philadelphia, Pa. (1948).
8. Lapp, R. E., and Andrews, H. L., NUCLEAR RADIATION PHYSICS, Prentice-Hall, Inc., New York, N. Y. (1954).
9. Ellis, Jr., and Hobart, R., NUCLEAR TECHNOLOGY FOR ENGINEERS, McGraw-Hill Book Co., Inc., New York, N. Y. (1959).
10. Murray, R. L., INTRODUCTION TO NUCLEAR ENGINEERING, Prentice-Hall, Inc., Englewood Cliffs, N. J. (1961).
11. Rockwell, T., REACTOR SHIELDING DESIGN MANUAL, D. Van Nostrand Co., Inc., Princeton, N. J. (1956).

12. Semat, H., INTRODUCTION TO ATOMIC AND NUCLEAR PHYSICS, Rinehart and Co., Inc., New York, N. Y. (1955).
13. Bleuler, E., and Goldsmith, G. J., EXPERIMENTAL NUCLEONICS, Rinehart and Co., Inc., New York, N. Y. (1952).
14. Lemon, H. B., and Ference, Jr., M., ANALYTICAL EXPERIMENTAL PHYSICS, The University of Chicago Press, Chicago, Illinois (1947).
15. Stephenson, R., INTRODUCTION TO NUCLEAR ENGINEERING, McGraw-Hill Book Co., Inc., New York, N. Y. (1958).
16. Hine, G. J., and Brownell, G. L., RADIATION DOSIMETRY, Academic Press, Inc., New York, N. Y. (1956).
17. Curtiss, L. F., INTRODUCTION TO NEUTRON PHYSICS, D. Van Nostrand Co., Inc., Princeton, N. J. (1959).
18. Fossati, F., Editor, QUANTITIES, UNITS AND MEASURING METHODS OF IONIZING RADIATION, Ulrico Hoepli, Milan, Italy (1959).
19. Young, M. E. J., RADIOLOGICAL PHYSICS, H. K. Lewis and Co., Ltd., London, England (1957).
20. Katz, L., and Penfold, A. S., REV. MOD. PHYS., 24, 28 (1952).
21. Division of Radiological Health, U. S. Department of Health, Education, and Welfare, RADIOLOGICAL HEALTH HANDBOOK, U. S. Government Printing Office, Washington, D. C. (1960).
22. Hughes, D. J., Maguro, B. A., and Brussel, M. K., NEUTRON CROSS SECTIONS, Supplement No. 1 to BNL-325, 2nd Edition (Jan 1, 1960), U. S. Government Printing Office, Washington, D. C.
23. Goldstein, H., FUNDAMENTAL ASPECTS OF REACTOR SHIELDING, Addison-Wesley Publishing Co., Inc., Reading, Mass. (1959).
24. Price, B. T., Horton, C. C., and Spinney, K. T., RADIATION SHIELDING, Pergamon Press, New York, N. Y. (1957).
25. Danforth, J. P., and Stapp, R. P., Editors, RADIOISOTOPES IN INDUSTRY TRAINING PROGRAM, Vol. 1, General Motors Institute, Flint, Mich. (1959).

National Bureau of Standards Handbooks and Circulars (Obtainable from U. S. Government Printing Office - Washington, D. C.):

26. N.B.S. Circular 577, ENERGY LOSS AND RANGE OF ELECTRONS AND POSITRONS (1956).

27. N.B.S. H55, PROTECTION AGAINST BETATRON - SYNCHROTRON RADIATIONS UP TO 100 MeV (1954).
28. N.B.S. H63, PROTECTION AGAINST NEUTRON RADIATION UP TO 30 MeV (1957).
29. N.B.S. H72, MEASUREMENT OF NEUTRON FLUX AND SPECTRA FOR PHYSICAL AND BIOLOGICAL APPLICATIONS (1960).
30. N.B.S. H78, REPORT OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL UNITS AND MEASUREMENTS (ICRU) (1959).
31. N.B.S. H79, STOPPING POWERS FOR USE WITH CAVITY CHAMBERS (1961).

SECTION 4 - CONCEPTS OF RADIATION UNITS AND DOSE DETERMINATIONS

A. Definition of Terms

The organization which selects and defines the units and quantities of radiation is the International Commission on Radiological Units and Measurements (ICRU). This group, formed in 1925, gathers the most recent data, evaluates these data, and then issues its recommendations. Early reports of this group giving a history of the development of the units can be found in the literature (Ref. 1-9). The current reports of the ICRU can be found in NBS Handbooks 84-89. Much of the information used in this section is taken from Handbook 84.

It is the desire of the ICRU to restrict the use of the term dose. In previous reports, three types of doses were discussed, namely, exposure dose, absorbed dose, and RBE dose. The term exposure is now used in place of exposure dose. The term absorbed dose is retained. Dose equivalent replaces the term RBE dose. The definitions of these terms are:

1. Exposure - The exposure X is the quotient of ΔQ by Δm , where ΔQ is the sum of the electrical charges on all ions of one sign produced in air when all the electrons, liberated by photons in a volume element of air whose mass is Δm , are completely stopped in air:

$$X = \Delta Q / \Delta m.$$

The special unit of exposure is the Roentgen (symbolized as R): $1 \text{ R} = 2.58 \times 10^{-4} \text{ C/kg}$, where C stands for Coulomb. This unit is numerically equal to the old value for R, 1 esu/cc of air. The use of the Roentgen is restricted solely to exposure.

2. Absorbed Dose - The absorbed dose D is the quotient of ΔE_D by Δm , where ΔE_D is the energy imparted by ionizing radiation to the matter in a volume element containing mass Δm

$$D = \Delta E_D / \Delta m.$$

The special unit of absorbed dose is the rad. Its use is reserved solely for absorbed dose. By definition, $1 \text{ rad} = 100 \text{ ergs/gm} = 10^{-2} \text{ Joule/kg}$.

3. Dose Equivalent - The dose equivalent (denoted as DE) is defined as the product of absorbed dose and other necessary modifying factors. The ICRU wishes to reserve the term RBE for use in radiobiology only. The term quality factor (symbolized as QF) should be used for protection purposes. The product of an absorbed dose and a suitable quality factor expresses the irradiation in terms of a common scale for all ionizing

radiations. The distribution factor (denoted as DF) is used to correct for nonuniform distribution in the case of internally deposited radionuclides. Thus, the dose equivalent is given by

$$DE = D(QF)(DF)....$$

DE is numerically equal to the dose in rads times the appropriate factors. The unit of dose equivalent is the Rem.

B. Quality Factor

For the present, the values of the quality factors which are to be used correspond to the previous values of RBE used for protection work (see Ref. 22). However, there is a need to distinguish these values from the RBE values used in experimental work. Hence the new name quality factor.

In the future, these values may be revised. One limitation already present is that the QF values are not applicable in the case of acute exposures at high dose rates, such as in an accident. These values apply only for the case of normal protection work. The values in present use are:

<u>Type of Radiation</u>	<u>Quality Factor (QF)</u>
X ray, γ ray, β^- , β^+	1
β^- , β^+ for $E_{\max} \leq 0.03$ MeV	1.7
α particles	10
heavy recoil atoms	20
neutrons (See page 79 below)	

C. Exposure

As a beam of photons passes through air, the interactions which take place produce electrons, which then lose energy by creating ion pairs. The exposure is measured by collecting these ion pairs. Thus, the exposure concept is based on the ability of photons to produce ionization.

There is very little attenuation of photons in air. What does occur is due to the photoelectric effect, the Compton effect, and pair production. If a wide beam of photons is used, then well within this beam (that is, not near the edge) the field will be uniform. In this case, one may say that in each small volume of air, the effects should be the same, that is, about the same number and type of interactions should occur. Then, the number of electrons which are produced in these processes will be about equal.

In the consideration of a given volume of air then, the electrons which are produced will give up part of their energy outside this volume before being stopped. However, this loss of electrons may be compensated by electrons entering the given volume from other parts of the field. If the field is uniform over a large enough volume, the compensation will be complete. In this case, a state of electronic equilibrium is reached, that is, the energy imparted to a given small volume of air by all the electrons which traverse this volume equals the energy imparted to the medium along the total path of all electrons produced in the volume by the photons.

The collection of the ions which are produced in a given small volume of air under equilibrium conditions is then a measure of the exposure. The exposure may be expressed in units of Coulombs/kg. Since one electron has a charge of 1.6×10^{-19} C, one Coulomb represents the collection of 6.25×10^{18} electrons. Because of historical reasons, the Roentgen is retained as a special unit of exposure. In the past, one Roentgen corresponded to 1 esu/cc of air. In terms of the new units for exposure,

$$1 \text{ R} = \frac{1 \text{ esu}}{1 \text{ cm}^3 \text{ of air}} \left(\frac{1 \text{ C}}{3 \times 10^9 \text{ esu}} \right) \left(\frac{1 \text{ cm}^3 \text{ of air}}{1.293 \times 10^{-6} \text{ kg}} \right) = 2.58 \times 10^{-4} \frac{\text{C}}{\text{kg}},$$

in which 1 cc of air at 0°C and 76 cm Hg = 1.293×10^{-3} gm.

D. Exposure Rate - Isotropic Point Source

Assume an isotropic point source of C Curies of a γ emitter of energy E_γ at the point O (see Fig. 4.1). Our interest is to obtain an expression for the exposure rate at the point A , distant r from the source, in terms of Roentgens per unit time. Since the source is isotropic, the intensity I_A at the point A will be given by

$$I_A = S/4\pi r^2,$$

where S is the total energy per second emitted by the source. If S is expressed in MeV/sec,

$$S = 3.7 \times 10^{10} \left(\frac{\text{dis}}{\text{sec-Ci}} \right) n_\gamma \left(\frac{\text{fraction of } \gamma}{\text{dis}} \right)$$

$$C(\text{Ci}) E_\gamma \left(\frac{\text{MeV}}{\gamma} \right) = 3.7 \times 10^{10} n_\gamma C E_\gamma.$$

Then

$$I_A = 3.7 \times 10^{10} n_\gamma C E_\gamma / 4\pi r^2 = 2.944 \times 10^9 n_\gamma C E_\gamma / r^2 \text{ in (MeV/cm}^2\text{-sec)}.$$

The exposure X is defined in terms of the ionization produced in a volume of air. The exposure rate is given by

$$\text{Exposure rate} = \Delta X / \Delta t.$$

A special unit of exposure rate is the R/hr.

To find the ions produced in air, the energy given up by the beam must be known. The term I_A gives the energy delivered per unit time about the point A. Some of this energy will be given up in interactions which produce electrons. These electrons in turn produce many ion pairs. The mass energy-absorption coefficient (μ_{en}/ρ) expresses the probability of energy being transferred to the substance. The product $I_A \mu_{en}/\rho$ gives the amount of energy transferred per unit time per unit mass. If one assumes that all this energy is used to produce ion pairs, then

$$\frac{\text{ion pairs}}{\text{gm-sec}} = \frac{I_A}{W} \left(\frac{\mu_{en}}{\rho} \right) 10^6,$$

where $W = 34$ eV/ip for electrons in air, μ_{en}/ρ is equivalent to $(\mu - \sigma_s)/\rho$ (cm^2/gm) as given in Section 3, and the factor 10^6 converts MeV to eV. This leads to

$$\frac{\text{ion pairs}}{\text{gm-sec}} = 2.944 \times 10^{15} \frac{n_\gamma C E_\gamma}{34 r^2} \left(\frac{\mu_{en}}{\rho} \right) = 8.66 \times 10^{13} \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho} \right).$$

Conversion to exposure rate units gives

$$8.66 \times 10^{13} \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho} \right) \left(\frac{\text{ip}}{\text{gm-sec}} \right) 3.6 \times 10^3 \left(\frac{\text{sec}}{\text{hr}} \right) \\ 1.6 \times 10^{-19} \left(\frac{\text{C}}{\text{ip}} \right) \left(\frac{10^3 \text{ gm}}{\text{kg}} \right) = 49.882 \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho} \right),$$

in C/kg-hr. Then,

$$\text{Exposure rate in R/hr} = 49.882 \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho} \right) \frac{\text{C}}{\text{kg-hr}} \frac{1}{2.58 \times 10^{-4}} \left(\frac{\text{kg-R}}{\text{C}} \right) = \\ 1.93 \times 10^5 \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho} \right)$$

from an isotropic point source of a gamma emitter of activity $n_\gamma C$ (in Ci), energy E_γ (in MeV), and mass energy-absorption coefficient μ_{en}/ρ , at a point a distance r cm from the source.

For air, the attenuation is very small for short distances, so that one can neglect the attenuation factor. For large distances in air, or for the case of an absorber between the source and the point A, an attenuation factor of the form $e^{-(\mu_{en}/\rho)x}$ must be included. Such a factor is needed for each such absorber and in the general case is

$$e^{-\sum_{i=1}^j (\mu_{en}/\rho)_i x_i} = e^{-(\mu_{en}/\rho)_1 x_1} e^{-(\mu_{en}/\rho)_2 x_2} \dots e^{-(\mu_{en}/\rho)_j x_j},$$

where x_i is expressed in gm/cm^2 .

The general expression becomes

$$\text{Exposure rate (in R/hr)} = 1.93 \times 10^5 \frac{n_\gamma C E_\gamma}{r^2} \left(\frac{\mu_{en}}{\rho} \right) e^{-\sum_{i=1}^j (\mu_{en}/\rho)_i x_i}$$

If the only substance of concern is air, then certain simple relations can be deduced. For example, the attenuation in a few feet of air is negligible and no attenuation factor need be included. Also, μ_{en}/ρ can be taken as $0.0275 \text{ cm}^2/\text{gm}$ in the range $0.2 < E_\gamma < 2 \text{ MeV}$, so that

$$\text{Exposure rate (in R/hr)} \cong 6 n_\gamma C E_\gamma \text{ at 1 ft and } 0.2 < E_\gamma < 2.0 \text{ MeV}$$

$$\text{Exposure rate (in R/hr)} \cong 0.53 n_\gamma C E_\gamma \text{ at 1 m and } 0.2 < E_\gamma < 2.0 \text{ MeV.}$$

These simple formulae are often used to advantage in Health Physics applications to estimate quickly the exposure rate. In cases for which more accuracy is desired, the general formula should be used and the actual value of μ_{en}/ρ for the gamma obtained.

E. Exposure Rate - Flux Density

Along the same line, one may wish to express the exposure rate in terms of the flux density of photons in a beam of radiation. If one assumes ϕ photons/ cm^2 -sec of energy E_γ in the beam, the resulting expression is

$$\text{Exposure rate (in R/hr)} = 6.57 \times 10^{-5} \phi E_\gamma (\mu_{en}/\rho),$$

where E_γ is in MeV and μ_{en}/ρ is in cm^2/gm . In this case, no attenuating substance is included, and the exposure rate as given would be well within the beam dimensions to avoid edge effects.

F. Absorbed Dose

The exposure concept is concerned with the production of ions. The concept of absorbed dose focuses attention on the energy imparted to matter in a volume element. Since one can presume that the biological effects result only from the energy imparted to matter, the concept of absorbed dose is more closely associated with these effects than is the exposure. Also, the absorbed dose is not specified for any given substance, so that the unit applies for any medium. The ICRU defines energy imparted as the difference between the sum of all energy entering the volume and the sum of all energy leaving the volume, minus the energy equivalent of increased rest mass from nuclear reactions within the volume (see Ref. 10).

Whereas the exposure refers to photons, the absorbed dose may be used for any ionizing radiation. Thus the concept can be used for α , β , γ , and neutrons.

G. Absorbed Dose Rate - Isotropic Point Source of Photons

Consider again an isotropic point gamma-emitting source. As we saw in Section D above, $I_A (\mu_{en}/\rho)$ gives the amount of energy transferred per unit time per unit mass. If this absorbed dose rate in air is given in MeV/gm-sec, we may convert it as follows:

$$\frac{\text{ergs}}{\text{gm-hr}} = I_A \left(\frac{\mu_{en}}{\rho} \right) \left(\frac{\text{MeV}}{\text{gm-sec}} \right) 1.6 \times 10^{-6} \left(\frac{\text{ergs}}{\text{MeV}} \right) 3.6 \times 10^3 \left(\frac{\text{sec}}{\text{hr}} \right).$$

Since the rad is a special unit of absorbed dose, we also have

$$\begin{aligned} \frac{\text{rad}}{\text{hr}} &= 5.76 \times 10^{-3} I_A \left(\frac{\mu_{en}}{\rho} \right) \left(\frac{\text{ergs}}{\text{gm-hr}} \right) \frac{1}{100} \left(\frac{\text{gm-rad}}{\text{ergs}} \right) \\ &= 5.76 \times 10^{-5} \frac{2.944 \times 10^9 n_{\gamma} C E_{\gamma}}{r^2} \left(\frac{\mu_{en}}{\rho} \right) \\ &= 1.7 \times 10^5 \frac{n_{\gamma} C E_{\gamma}}{r^2} \left(\frac{\mu_{en}}{\rho} \right). \end{aligned}$$

This gives the absorbed dose rate from an isotropic point source of a gamma emitter of activity $n_{\gamma}C$ Curies, energy E_{γ} , and mass energy-absorption coefficient μ_{en}/ρ , at a point a distance r from the source. The general expression for the absorbed dose rate in rad/hr is then

$$\frac{\text{rad}}{\text{hr}} = 1.7 \times 10^5 \frac{n_{\gamma} C E_{\gamma}}{r^2} \left(\frac{\mu_{en}}{\rho} \right) e^{-\sum_{i=1}^j (\mu_{en}/\rho)_i x_i}.$$

The relationship between the absorbed dose rate and the exposure rate at any point in a medium irradiated by photons is

$$\frac{\text{rad}}{\text{hr}} = (0.869) \left[\frac{(\mu_{\text{en}}/\rho)_{\text{medium}}}{(\mu_{\text{en}}/\rho)_{\text{air}}} \right] \left(\frac{\text{R}}{\text{hr}} \right).$$

From this, if the only substance is air, one gets

$$\text{rad/hr} = 0.869 (\text{R/hr}).$$

H. Absorbed Dose Rate - Flux Density of Photons

The absorbed dose rate in terms of the flux density of photons in a beam of radiation is given by

$$\text{rad/hr} = 5.76 \times 10^{-5} \phi E_{\gamma} \mu_{\text{en}}/\rho,$$

where ϕ is the flux density in photons/cm²-sec, E_{γ} is the energy of the photons in MeV; and μ_{en}/ρ is in cm²/gm. This expression is valid well within the beam dimensions.

I. Absorbed Dose Rate - Isotropic Point Source - Alpha and Beta Radiation

If an isotropic point source of C Curies is an alpha or a beta emitter, then the absorbed dose rate at the point A (see Fig. 4.1) is given by

$$\frac{\text{rad}}{\text{hr}} = \frac{0.17 n_p C}{r^2} \left(\frac{S}{\rho} \right),$$

where n_p is the fraction of particles of energy E (in MeV) and S/ρ is the mass stopping power for a particle of energy E in eV/(gm/cm²). Since neither alpha nor beta radiation has a very long range in any substance, this expression is valid only when the distance r to the point A is well within the range of the particle.

J. Mass Stopping Power

For purposes of calculation, the mass stopping power S/ρ for use in the previous formula is given by K. Z. Morgan (Ref. 23). For alpha particles,

$$\left(\frac{S}{\rho} \right)_{\alpha} = \frac{(\text{SP.I.})_{\text{air}} W}{\rho_{\text{air}}} \left(\frac{S}{\rho} \right)_{\text{m}} \left(\text{in } \frac{\text{eV}}{\text{gm/cm}^2} \right),$$

where $(SP.I.)_{air}$ is given in ip/cm, ρ_{air} is the density of air at which the specific ionization was measured, W is 35 eV/ip for alpha particles in air, and $(S/\rho)_m$ is the relative mass stopping power of the medium with respect to air.

Figure 4.2, based on data from Ref. 23, gives the mass stopping power for alpha radiation in air in terms of MeV/(gm/cm²) as well as the relative mass stopping power $(S/\rho)_t$ of tissue with respect to air. For substances other than tissue,

$$(S/\rho)_m \approx \sqrt{A_{air}}/\sqrt{A_{medium}}$$

(see Section 3).

Fig. 4.2. Mass Stopping Power of Alpha Radiation as a Function of Energy (Based on information in Radiation Hygiene Handbook, section 14, McGraw-Hill Book Co., Inc., New York, 1959)

This information allows one to calculate roughly the absorbed dose rate from an alpha source, but only well within the range of the particle in the medium. The reason for this is that the specific ionization of the particle does not change too much until near the end of its range, and also the scattering can be neglected. Near the end of the range of an alpha particle, the specific ionization, as well as the stopping power, changes very rapidly, and the above information is no longer of value.

In the case of beta radiation, a further difficulty is encountered, namely, the existence of a spectrum of energy. Also, the scattering of beta particles is quite extensive. In Section 3, we saw that, well within the range of the beta, the absorption could be taken as exponential. The mass absorption coefficient for most substances was given as

$$\mu/\rho \approx 17 E_{max}^{-1.43} \text{ (cm}^2\text{/gm)}$$

for $0.05 < E_{max} < 5$ MeV. The mass stopping power for beta radiation may then be roughly expressed, in eV/gm-cm², as

$$(S/\rho)_\beta \approx 5.67 \times 10^6 E_{\max}^{-0.43} e^{-17(E_{\max}^{-1.43})x},$$

where $E_{\max}$ is in MeV and x is in gm/cm^2 of the medium. Again, these expressions are of value only well within the range of the beta.

Although these expressions for the stopping power of alpha and beta radiations are only rough approximations, the need for more accurate expressions is not grave. Since the range of charged particles from radioactive substances in most materials is quite short, neither alpha nor beta radiation presents much concern as external sources of radiation. For this reason, as well as the fact that these radiations are easily shielded, it is not expedient that one have exact expressions.

K. Absorbed Dose Rate - Flux Density of Particles

The absorbed dose rate for a flux density of alpha or beta particles in a beam of radiation is given by

$$\text{rad/hr} = 5.76 \times 10^{-11} \phi_p (S/\rho),$$

where ϕ_p is the particles/ cm^2 -sec and S/ρ is the mass stopping power for a particle of energy E (in MeV). In this case, the values of S/ρ for alpha particles can be found by the same method as used for a point isotropic source. Figure 4.3 gives the mass stopping power for beta radiation in air in terms of $\text{MeV}/(\text{gm}/\text{cm}^2)$. One may assume the value 1.14 for the relative mass stopping power $(S/\rho)_t$ of tissue with respect to air for use in the above expression. This figure is based on information found in Ref. 23.

Fig. 4.3

Mass Stopping Power of Beta Radiation as a Function of Energy (Based on information in Radiation Hygiene Handbook, section 14, McGraw-Hill Book Co., Inc., New York, 1959)

L. Absorbed Dose for Neutrons

The measurement of the absorbed dose from a neutron source is very difficult, because the energy transfer depends so much on the neutron energy as well as the properties of the struck nucleus. In Section 3, the neutron interactions were pointed out. For fast neutrons (energies from 10 keV to 10 MeV), elastic scattering is the important process, especially for the light elements. For thermal neutrons (energy of 0.025 eV), capture is the important process. In these processes, it is the secondary products which directly impart the energy to the substance. The problem increases in magnitude when one deals with a spectrum of energy rather than a mono-energetic neutron source. Also, if one is dealing with a compound, the presence of many elements complicates the picture.

In NBS Handbook #88, an expression is given for the absorbed dose in the case of fast neutrons. The results are applicable to a pure material when the scattered neutrons can be neglected. A curve has been drawn for the dose in pure hydrogen, carbon, nitrogen, and oxygen. This curve has been reproduced as Fig. 4.4. The absorbed dose indicated for the Standard Man is based on the percentage composition as agreed upon by the ICRP.

Fig. 4.4. Absorbed Dose for Neutrons as a Function of Energy (From NBS Handbook 88, U. S. Govt. Printing Office, Washington, D. C., 1963)

The neutron dose was calculated by multiplying the value of absorbed dose for each member at the given neutron energy by the percentage composition by weight of that member and summing the contributions:

$$D = \sum_{i=1}^n (\text{wt } \%)_i [\text{rads for pure element}]_i$$

The absorbed dose is given in terms of the particle fluence 10^9 n/cm^2 . The influence Φ represents the time integrated flux density, so that

$$\phi = \Phi/t,$$

where t is in seconds, Φ is in n/cm^2 , and ϕ is in $\text{n/cm}^2\text{-sec}$.

No similar expressions are available for intermediate or thermal neutrons. Resonance absorption plays a key role in these energy regions, so that the presence of certain elements can greatly alter the absorbed dose.

There is a lack of useful formulae for calculating the absorbed dose rate from neutrons. In terms of instrument measurements, the neutron flux density can be measured rather readily. The Monte Carlo calculations of Snyder which relate the neutron flux density for a given energy to the dose equivalent are thus more useful in terms of protection work. These calculations are the basis of the following table, which is taken from NBS Handbook #63, p. 15, and which has been revised according to the new concepts of the ICRU:

MAXIMUM PERMISSIBLE NEUTRON FLUX DENSITY

Time-average flux density for 40-hr week
to deliver either 100 or 300 mrem

Neutron Energy, MeV	Quality Factor	$\text{n/cm}^2\text{-sec}$ for 100 mrem	$\text{n/cm}^2\text{-sec}$ for 300 mrem
Thermal	3	670	2000
0.0001	2	500	1500
0.005	2.5	570	1700
0.02	5	280	850
0.1	8	80	250
0.5	10	30	90
1.0	10.5	18	55
2.5	8	20	60
5.0	7	18	55
7.5	7	17	50
10	6.5	17	50
10-30	-	10^a	30^a

a = suggested limit.

If one has a measure of the neutron flux density, then this table can be used to find the dose equivalent which this flux density would represent over a 40-hr week. When little information on the neutron energy is available, one should assume a quality factor of 10. The dose equivalent rate from a flux density of ϕ n/cm²-sec of a given energy can be found by use of the table. In this case, a simple proportion would exist,

$$\frac{\text{DE}}{100 \text{ mrem}} = \frac{\phi}{\phi_{\text{MPC-100 mrem}}}$$

This gives the dose equivalent of the flux density ϕ if the time in the neutron field were 40 hr. The dose equivalent rate would then be

$$\frac{\text{mrem}}{\text{hr}} = \frac{\text{DE}}{40 \text{ hr}} = \frac{\phi}{\phi_{\text{MPC-100 mrem}}} 2.5 \frac{\text{mrem}}{\text{hr}}$$

In general, this procedure will overestimate the dose equivalent when the energy of the neutron flux density is not known, but for protection work the added safety factor is useful.

BIBLIOGRAPHY

1. ICRU REPORT NUMBER 1 - Discussion on International Units and Standards for X-ray Work, Brit. J. Radiol., 23, 64 (1927).
2. ICRU REPORT NUMBER 2 - International X-ray Unit of Intensity, Brit. J. Radiol. (new series) 1, 363 (1928).
3. ICRU REPORT NUMBER 3 - Report of Committee on Standardization of X-ray Measurements, Radiology 22, 289 (1934).
4. ICRU REPORT NUMBER 4 - Recommendations of the International Committee for Radiological Units, Radiology 23, 580 (1934).
5. ICRU REPORT NUMBER 5 - Recommendations of the International Committee for Radiological Units, Radiology 29, 634 (1937).
6. ICRU REPORT NUMBER 6 - Report of International Commission on Radiological Protection and International Commission on Radiological Units, NBS HANDBOOK 47, Washington, D. C., (1951).
7. ICRU REPORT NUMBER 7 - Recommendations of the International Commission for Radiological Units, Radiology 62, 106 (1954).

8. ICRU REPORT NUMBER 8 - Report of International Commission on Radiological Units and Measurements (ICRU) 1956, NBS HANDBOOK 62, Washington, D. C., (1957).
9. ICRU REPORT NUMBER 9 - Report of International Commission on Radiological Units and Measurements (ICRU) 1959, NBS HANDBOOK 78, Washington, D. C., (1961).
10. ICRU REPORT NUMBER 10a - Radiation Quantities and Units, NBS HANDBOOK 84, Washington, D. C., (1962).
11. ICRU REPORT NUMBER 10b - Physical Aspects of Irradiation, NBS HANDBOOK 85, Washington, D. C., (1964).
12. ICRU REPORT NUMBER 10c - Radioactivity, NBS HANDBOOK 86, Washington, D. C., (1963).
13. ICRU REPORT NUMBER 10d - Clinical Dosimetry, NBS HANDBOOK 87, Washington, D. C., (1963).
14. ICRU REPORT NUMBER 10e - Radiobiological Dosimetry, NBS HANDBOOK 88, Washington, D. C., (1963).
15. ICRU REPORT NUMBER 10f - Methods of Evaluating Radiological Equipment and Materials, NBS HANDBOOK 89, Washington, D. C., (1963).
16. Stephenson, R., INTRODUCTION TO NUCLEAR ENGINEERING, McGraw-Hill Book Co., Inc., New York, N. Y. (1958).
17. Hine, G. J., and Brownell, G. L., RADIATION DOSIMETRY, Academic Press Inc., New York, N. Y. (1956).
18. Evans, R. D., THE ATOMIC NUCLEUS, McGraw-Hill Book Co., Inc., New York, N. Y. (1955).
19. Fränz, H., and Hübner, W., CONCEPTS AND MEASUREMENT OF DOSE, Proceedings of Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, Switzerland, 21, 101 (1958).
20. Rossi, H. H., and Roesch, W. C., FIELD EQUATIONS IN DOSIMETRY, Radiation Research 16, 783 (1962).
21. Fossati, F., Editor, QUANTITIES, UNITS AND MEASURING METHODS OF IONIZING RADIATION, Ulrico Hoepli, Milan, Italy (1959).

22. Report of the RBE Committee to the International Commission on Radiological Protection; Health Phys., 9(4), 357 (1963).
23. Morgan, K. Z., Determination of Exposures, RADIATION HYGIENE HANDBOOK, edited by H. Blatz; McGraw-Hill Book Co., Inc., New York, N. Y., (1959).
24. Stopping Powers for Use with Cavity Chambers; NBS HANDBOOK 79, Washington, D. C., (1961).
25. Measurement of Absorbed Dose of Neutrons and of Mixtures of Neutrons and Gamma Rays, NBS HANDBOOK 75, Washington, D. C., (1961).
26. Measurement of Neutron Flux and Spectra for Physical and Biological Applications; NBS HANDBOOK 72; Washington, D. C., (1960).
27. Protection Against Neutron Radiation Up to 30 Million Electron Volts, NBS HANDBOOK 63, Washington, D. C., (1957).

SECTION 5 - SHIELDING

A. Factors Affecting Exposure in a Radiation Field

The philosophy inherent in any program of radiation safety is to reduce exposure, whether internal or external, to a minimum. If it is impossible or impractical to remove a source of radiation, other means must be considered for purposes of personnel protection.

Three factors which determine the total exposure one receives in a given radiation field are:

- (1) time of exposure;
- (2) distance from the source;
- (3) amount of shielding present.

The time factor simply means that the longer one remains in a radiation field, the greater will be the exposure received. At times, especially during emergencies, work must be performed in a strong radiation field. In this case, the work procedure should be carefully planned outside the work area so that a minimum amount of time is used to complete the job. If the time required for one man to complete the job would result in an exposure beyond prescribed limits, then a team of workers should be employed. This would mean a small exposure for several people instead of a large exposure for one individual.

Assume that work must be done in a field of 1 R/hr (= 1,000 mR/hr). If we are allowed 60 mR/day, how long can a person work in this area so as not to exceed this limit?

$$t(\text{hr}) \times 1,000(\text{mR/hr}) = 60(\text{mR}).$$

Then

$$\begin{aligned} t &= 60/1,000 \text{ hr} = 0.06 \text{ hr} = 0.06 \text{ hr} \times 60 \text{ min/hr} \\ &= 3.6 \text{ min.} \end{aligned}$$

This means that an individual would receive a daily permissible exposure in 3.6 min while working in this radiation field.

The intensity of a radiation field decreases with distance from the source. If we consider a point source of penetrating radiation, the decrease in intensity will be inversely proportional to the square of the distance between any two radial points from the source. This "inverse square law" can

be utilized only when the dimensions of source and measuring device are small compared with the distance between them and the distances are measurements in air or vacuum.

For other than point sources, the intensity will decrease inversely with distance, but not necessarily as the square of the distance.

However, we are often faced with a space problem or a job in which the worker has to be in close proximity to the source. This essentially places a restriction on the distance factor. We can eliminate the radiation hazard by placing a suitable attenuating material, or combination of materials, between the source and worker. The choice and/or thickness of attenuating material (shield) will depend upon the type and energy of the radiation.

The time, distance, and shielding factors are used individually or in combination to minimize external radiation hazards.

In choosing a shielding material, our first consideration must be personnel protection. However, other factors may influence our choice of material - such as: is it economical; is it too heavy; how much space are we allowed for the shield; does it have proper structural strength? Also, will it create a toxicity or contamination problem due to radiation damage; to what levels must the radiation be reduced in order to obtain accurate measurements with various instruments in the area; etc.?

The answer to some of these questions may be determined from the discussion and examples which follow.

A shield is a medium of some thickness which will stop or effectively attenuate radiation to nonhazardous levels. The effectiveness of the shield is determined by the interactions between the incident radiation and the atoms of the absorbing medium. The interactions which take place depend mainly upon the type of radiation (α , β , X- γ photons, neutrons, etc.), the energy of the radiation, and the atomic number of the absorbing medium.

B. Alpha Radiation

Alpha particles, which lose energy rapidly in any medium because of their relatively high specific ionization, can be stopped by very thin absorbing materials. A few sheets of paper or thin (less than 1/64 in.) aluminum foils will absorb alphas from α -emitting sources. The most energetic alphas will travel only a few centimeters in air. The outer layer of skin ($\sim 7 \text{ mg/cm}^2$) on our body will absorb α particles up to $\sim 7.5 \text{ MeV}$. Since this is dead tissue, no harmful effect is produced upon the body. Therefore, α radiation is not considered an external exposure problem.

C. Beta Radiation

The processes by which β particles lose energy in absorbers are similar to those for alphas. However, an additional process must be considered in dealing with β absorption. This is the process whereby electromagnetic radiation, called bremsstrahlung, is produced.

A β particle has a very small mass and one-half the value of the magnitude of charge on an α particle. So for a given energy, a β particle will have a much greater velocity than an α particle. As a result of these and other factors, the β particle has a lower specific ionization, which means that its penetration in any absorber will be much greater than that of an α particle.

A 1-MeV β particle will travel about 10 ft in air. In order to penetrate the dead layer of skin, a β particle must have an energy of about 70 keV. Beta radiation is considered a slight external hazard:

The thickness and choice of material for shielding from beta radiation depends upon: (1) stopping the most energetic β , and (2) bremsstrahlung.

The shielding thickness which is necessary to stop β particles of a given energy will decrease with increasing density. For example, ~ 0.1 in. of aluminum ($\rho = 2.7$ gm/cc) will stop 1.5-MeV betas, whereas only ~ 0.021 in. of lead ($\rho = 11.3$ gm/cc) is needed for the same purpose.

However, the production of bremsstrahlung increases with increasing atomic number of absorber. A β source with $E_{\max} = 1$ MeV will lose about 3% of its energy as bremsstrahlung when lead ($Z = 82$) is the absorber. If aluminum ($Z = 13$) is the absorber, the fraction is about 0.4%. Therefore, low- Z absorbers such as aluminum, plastics (Lucite), or even glass are effective β shields.

If cost and/or weight must be considered, a combination shield may be used. In this case, a low- Z absorber is used to stop the betas, followed by a high- Z absorber to attenuate the bremsstrahlung. This type of shield (as a shipping pot) is further discussed in Section D.

D. Gamma Radiation

Gamma rays do not lose energy continuously, as will α or β particles, in traversing a medium. As a result, γ rays are much more penetrating than α or β radiation. Gamma radiation is attenuated in matter by: (1) the photoelectric effect, (2) Compton effect, and (3) pair production.

Materials of higher density, such as Pb, U, Th, Au, and W, are best suited for γ shields. However, the use of these metals is limited by their high cost and/or weight. Therefore, metals such as Fe, Pb, Cr, and Ni are used for γ shielding.

1. Calculations of Shield Thickness

Gamma-ray absorption is an exponential process. Theoretically, this means that γ rays are never completely absorbed no matter how thick the absorber. However, we can choose a shield thickness which will reduce the intensity to nondetectable or nonhazardous levels.

Three relationships (see Fig. 5.1) can be used for calculating shield thickness:

$$I = I_0 e^{-\mu x}; \quad (A)$$

$$I = I_0 e^{-(\mu - \sigma_s)x}; \quad (B)$$

$$I = I_0 b e^{-\mu x}. \quad (C)$$

Fig. 5.1. Relationships for Calculating Shield Thicknesses

In radiation protection work we prefer equation (B), since this equation incorporates a safety factor in our calculations (see Fig. 5.1). The attenuation coefficient $(\mu - \sigma_s)$ assumes that all scattered gammas reach our point of interest. This means that we will overestimate the intensity I when radiation I_0 moves through a shield thickness x . As a result, the calculated thickness will be greater than is necessary to reduce the intensity to prescribed levels.

Equation (A) is used for narrow (collimated) beam conditions. Here, we assume that all γ photons which deviate (scatter) from the beam never reach our detector. If this equation is used in treating a wide beam of radiation, we would underestimate the required shield thickness.

The third equation (C), is used when cost, weight, or space are factors which must be considered. If we know the buildup factor b , we calculate the true γ flux I ; hence, we get a more accurate value for the needed shield thickness.

For quick shielding estimates, we can use multiples of tenth-value or half-value layers. A half-value layer ($x_{1/2}$) is defined as that thickness of a material which reduces the radiation intensity to one-half its initial value. A tenth-value layer, or any other thickness which reduces the intensity to some desired fraction of the initial intensity, is similarly defined.

A half-value layer is determined as follows:

$$I = I_0 e^{-(\mu - \sigma_s)x}$$

Then,

$$I/I_0 = 1/2 = e^{-(\mu - \sigma_s)x_{1/2}}$$

or

$$\ln \frac{1}{2} = -0.693 = -(\mu - \sigma_s) x_{1/2}.$$

From this

$$x_{1/2} = 0.693/(\mu - \sigma_s).$$

The above expression can be used to determine the number of half-value layers needed to reduce the initial intensity to any other desired level.

If we substitute $0.693/x_{1/2}$ for $\mu - \sigma_s$ in the exponential expression, we obtain

$$I/I_0 = e^{-0.693(x/x_{1/2})}$$

From this

$$\ln(I_0/I) = (x/x_{1/2})(0.693) = (x/x_{1/2}) \ln 2 = \ln 2^{x/x_{1/2}},$$

where $x/x_{1/2}$ is the number of half-value layers. If we let $n = x/x_{1/2}$, then $I_0/I = 2^n$.

Example 1. It is desired to reduce a beam of γ rays to $1/16$ of its initial intensity. The gammas have an energy of 1 MeV and lead will be used as the shielding material. How many half-value layers are required? How many cms of lead are required?

(a)

$$I_0/I = 16 = 2^n.$$

Now,

$$\ln 16 = n \ln 2$$

or

$$n = \frac{\ln 16}{\ln 2} = \frac{2.773}{0.693} = 4.$$

Therefore, 4 half-value layers are required.

(b) The value of $\left(\frac{\mu - \sigma_s}{\rho}\right)_{Pb}$ is obtained from the absorption coefficient versus energy curve:

$$\left(\frac{\mu - \sigma_s}{\rho}\right)_{Pb} = 0.045 \frac{\text{cm}^2}{\text{gm}}; 0.045 \frac{\text{cm}^2}{\text{gm}} \left(11.3 \frac{\text{gm}}{\text{cc}}\right) = 0.509 \text{ cm}^{-1} = (\mu - \sigma_s)_{Pb}$$

Then

$$1 \text{ half-value layer} = 0.693/(\mu - \sigma_s)_{Pb} = 0.693/0.509 \text{ cm},$$

and

$$4 \text{ half-value layers} = 4 \times (0.693/0.509) = 5.45 \text{ cm of lead.}$$

Example 2. A radium source will be contained at the center of a wooden box. The γ reading through the surface of this box is 1 R/hr. What thickness of lead will be required, as a lining inside the box, to reduce the reading at the surface to 2 mR/hr? Assume an effective γ energy of 0.8 MeV.

$$I_0/I = 1,000/2 = 500 = 2^n.$$

Then

$$n = \frac{\ln 500}{\ln 2} = \frac{6.215}{0.693} \approx 9.$$

Therefore, 9 half-value layers are required.

$$\left(\frac{\mu - \sigma_s}{\rho}\right)_{\text{Pb}} = 0.057 \frac{\text{cm}^2}{\text{gm}}; 0.057 \frac{\text{cm}^2}{\text{gm}} \left(11.3 \frac{\text{gm}}{\text{cc}}\right) = 0.644 \text{ cm}^{-1} = (\mu - \sigma_s)_{\text{Pb}}$$

Hence,

$$9 \times 0.693/0.644 \text{ cm} = 9.68 \text{ cm} \approx 4 \text{ in. of lead.}$$

2. Buildup Factor

In calculations which include the buildup factor b , we take into account the scattered photons resulting from the Compton effect and the "uncollided γ flux" which reach our detector. The value of b will vary with radiation energy and shield material, and also with source geometry and depth of shield penetration.

Curves for b in a number of materials can be found in the Radiological Health Handbook (Ref. 14) and other references. Frequently, the value of b is plotted on semilog paper against a quantity called the relaxation length. The relaxation length is that thickness of absorber which will result in a reduction of $1/e$ in the initial beam intensity I_0 , i.e., $I = 0.368 I_0$.

We use the narrow-beam relationship, $I = I_0 e^{-\mu x}$, to determine the value of b . The number of relaxation lengths is given by $\ln(I_0/I)$ to a first approximation. One may iterate to obtain a better approximation.

Example 3. A 1-MeV γ point source is surrounded by a lead shield which reduces the intensity at a point outside the shield by a factor of 20 from what it would be without the shielding. What is the value of b to a first approximation?

$$I_0/I = e^{\mu x}$$

or

$$\ln I_0/I = \mu x.$$

Then

$$\ln 20 = 2.99573 \approx 3 \text{ relaxation lengths.}$$

Now consult the curve for a point isotropic source in lead. From this we see that $b = 2$.

To illustrate the use of both b and calculations involving bremsstrahlung, we have the following example.

Example 4. A 100-Ci P^{32} source will be shipped in a cylindrical container 3 in. in diameter and 4 in. deep. The I.C.C. regulations require a reading of ≤ 10 mR/hr at 1 meter. Calculate the container wall thickness using equations (A), (B), and (C) on page 86.

We first consider an all lead container; we will then work out the problem for a combination wall material of Lucite and lead.

The decay scheme for P^{32} is as follows:

The fraction of beta energy converted to bremsstrahlung is given by $F = 3.33 \times 10^{-4} ZE$. For the all lead container:

$$F = 3.33 \times 10^{-4} (82)(1.71) = 0.047.$$

Now

$$100 \text{ Ci } (0.047) = 4.7 \text{ Ci of bremsstrahlung.}$$

The R/hr at 1 meter, unshielded is $0.53 \text{ CE} = 0.53 (4.7)(1.71) = 4.26$. For $E = 1.71 \text{ MeV}$, $(\mu/\rho)_{Pb} = 0.049 \text{ (cm}^2/\text{gm)}$, $(\mu - \sigma_s/\rho)_{Pb} = 0.03 \text{ (cm}^2/\text{gm)}$, and b is found from the curve. Number of relaxation lengths $= \ln (I_0/I) = \ln (4,260/10) = 6.05444$; therefore, $b \approx 3$. The density of lead is 11.3 (gm/cc) .

$$I = I_0 e^{-\mu x}; \quad I_0/I = e^{\mu x}; \quad 4,260/10 = e^{0.049(11.3)x};$$

$$\ln 426 = 0.554 x; \quad x = 6.054/0.554 = 10.93 \text{ cm} = 4.3 \text{ in.} \quad (\text{A})$$

$$I = I_0 e^{-(\mu - \sigma_s)x}; \quad I_0/I = e^{(\mu - \sigma_s)x}; \quad 4,260/10 = e^{0.03(11.3)x};$$

$$\ln 426 = 0.339 x; \quad x = 6.054/0.339 = 17.86 \text{ cm} \approx 7 \text{ in.} \quad (\text{B})$$

$$I = I_0 b e^{-\mu x}; \quad I_0 b/I = e^{\mu x}; \quad 4,260 (3)/10 = e^{0.049(11.3)x};$$

$$\ln 1,278 = 0.554 x; \quad x = 7.153/0.554 = 12.91 \text{ cm} \approx 5.1 \text{ in.} \quad (\text{C})$$

Using the above value for the wall thickness, the weight of the container would be about 781 lb. In using $(\mu - \sigma_s)$ to calculate the amount of lead required, we get a container with a weight of 1,655 lb. This is more than twice the amount of lead that is actually needed!

3. Effective Atomic Number

We will now consider the problem if Lucite is used to shield the betas and lead to shield the bremsstrahlung.

According to Evans (Ref. 4), the effective atomic number Z_{eff} for a compound, in reference to the bremsstrahlung formula, is given by

$$Z_{\text{eff}} = \frac{N_1 Z_1^2 + N_2 Z_2^2 + N_3 Z_3^2 + \dots}{N_1 Z_1 + N_2 Z_2 + N_3 Z_3 + \dots},$$

where N_i is the number of i^{th} atoms per cc of atomic number Z_i . For Lucite, $\text{C}_5\text{H}_8\text{O}_2$, $Z_{\text{eff}} = 5.85$.

The range of P^{32} betas in Lucite is approximately $1/4$ in. The fraction of beta energy converted to bremsstrahlung is

$$F = 3.33 \times 10^{-4} (5.85)(1.71) = 0.0033.$$

Since the source activity is 100 Ci, the amount of bremsstrahlung is

$$100 \text{ Ci} (0.0033) = 0.33 \text{ Ci}.$$

It is difficult to derive a function which describes the spectrum which is emitted. However, one can assume that the bremsstrahlung consists of monoenergetic X rays equal to the maximum beta energy.

In this case, the exposure rate is

$$\begin{aligned} R/\text{hr at 1 meter} &= 0.53 \text{ CE} = 0.53 (0.33)(1.71) = 0.299 \text{ R/hr} \\ &\approx 300 \text{ mR/hr.} \end{aligned}$$

Once again b is found from the curve:

$$\ln 300/10 = 3.4 \text{ relaxation lengths.}$$

From the curve, $b = 2.2$.

$$I = I_0 e^{-\mu x}; \quad I_0/I = e^{\mu x}; \quad 300/10 = e^{0.049(11.3)x};$$

$$\ln 30 = 0.554 x; \quad x = 3.401/0.554 = 6.14 \text{ cm} \approx 2.4 \text{ in.} \quad (\text{A})$$

$$I = I_0 e^{-(\mu - \sigma_s)x}; \quad I_0/I = e^{(\mu - \sigma_s)x}; \quad 300/10 = e^{0.03(11.3)x};$$

$$\ln 30 = 0.339 x; \quad x = 3.401/0.339 = 10.03 \text{ cm} \approx 3.9 \text{ in.} \quad (\text{B})$$

$$I = I_0 b e^{-\mu x}; \quad I_0 b/I = e^{\mu x}; \quad 300(2.2)/10 = e^{0.049(11.3)x};$$

$$\ln 66 = 0.554 x; \quad x = 4.19/0.554 = 7.56 \text{ cm} \approx 3 \text{ in.} \quad (\text{C})$$

Hence, a small amount of Lucite will appreciably reduce the lead requirement. This container will have about $1/3$ as much lead as is necessary for the "all lead" container. Again, if $\mu - \sigma_s$ is used in the calculations, the amount of lead is almost twice as much as is actually needed.

It should be remembered that the use of the buildup factor is important where economy, space, and weight must be considered, such as in the design of a reactor. If these considerations are unimportant, the simpler expression involving $\mu - \sigma_s$ can be used. For calculations in air, $\mu - \sigma_s$ can always be used. The buildup factor is important when absorbers are used in which the scattering can be quite extensive.

E. Neutrons

Neutrons, like gammas, are a highly penetrating form of radiation. They possess no charge and, therefore, are unaffected by the electric fields of atoms in the traversed medium.

Neutron attenuation is accomplished mainly through elastic and inelastic collisions which reduce the energy of the neutrons until they are absorbed in the medium. In many cases, penetrating γ rays are produced as a result of neutron absorption. The cross sections for these interactions

are not as well known as for γ rays. The data is especially lacking for those energies where the penetration is highest and cross section lowest. This contributes to the difficulties in calculating neutron attenuation.

The problem of neutron shielding is further complicated by the fact that substances which effectively attenuate neutrons are generally poor gamma shields. Neutrons are more hazardous to a biological system than gammas; therefore, the neutron flux emerging from a shield should be smaller than the γ flux. Fast neutrons are most difficult to attenuate and therefore are the main concern in the shielding problem. Almost all neutrons are fast upon release from a source.

Elements of low mass number are ideal neutron moderators. Therefore, hydrogen (in the form of water, plastic, or paraffin), beryllium, and carbon (in graphite) are popular shield constituents. A shield composed entirely of a hydrogenous material, such as water, is satisfactory for neutron attenuation; however, intermediate or heavy elements alone are not suitable neutron shields. This is due to the fact that in a nonhydrogenous material, a larger number of collisions are necessary before absorption can take place. As a result, neutrons will penetrate greater depths in such a shield. If iron is a shield component, additional precautions must be taken because thermal-neutron capture in iron produces high-energy gammas. A 2.2-MeV γ is produced upon neutron capture by hydrogen.

Concrete is an effective shielding material because of its water content. In addition, it is cheap, strong, and can be formed in almost any desired shape and size. Also, concrete is a better γ attenuator than water because of its higher-density additives.

For most cases, a single shielding material will not provide adequate protection. Therefore, shields are generally composed of both neutron-attenuating material and γ absorbers.

The attenuation of a narrow beam of neutrons passing through a substance is determined from the relationship

$$\phi = \phi_0 e^{-\Sigma_t x},$$

where ϕ_0 is the incident flux density, Σ_t is the total macroscopic cross section, and x is the absorber thickness. In using Σ_t , we assume all scattered neutrons to be removed from the beam.

For a wide beam, some of the neutrons which are scattered away from the point of interest are replaced by neutrons scattered toward the point of interest. Therefore, the calculated value for ϕ will be smaller than the measured value; hence, we overestimate the effectiveness of the attenuating substance.

In thick shields which contain hydrogen, the removal cross section Σ_R is used in the equation instead of the total cross section. It may also be used for a shield composed of one substance followed by a hydrogenous shield. In these cases, the removal cross section itself accounts for buildup.

In the case of thin shields, information concerning buildup is very sparse at this time. A conservative approach is suggested in NBS Handbook 63, in which a buildup factor of ~ 5 for water or paraffin is given for shields at least 20 cm thick.

When the shield is thin, even one which contains hydrogen, the scattered neutrons may contribute significantly to the flux at a point of interest. Since the removal cross section is based on measurements through thick shields, the concept does not adequately apply to thin shields, and a buildup factor must be included.

The flux from an isotropic neutron point source can be approximated by the formula

$$\phi = \frac{\phi_0 b e^{-\Sigma_R t}}{r^2},$$

where

ϕ_0 = flux at 1 cm;

r = distance from source to point of interest (cm);

b = buildup factor;

Σ_R = macroscopic removal cross section for the given source and shield material;

t = shield thickness (cm).

A list of macroscopic removal cross sections can be found in NBS Handbook 63.

Example 5. What is the flux at a point 1 meter away from a neutron source which is surrounded by a 20-cm sphere of water? The source is a Po-Be source with a flux of 10^6 n/cm²/sec at 1 cm. The value of Σ_R for water is 0.103 cm⁻¹ (from NBS Handbook 63).

$$\phi = \frac{10^6(5) e^{-0.103(20)}}{10^4} = 64 \text{ n/cm}^2/\text{sec.}$$

If we ignored the buildup factor, $\phi = 13 \text{ n/cm}^2/\text{sec.}$

Shield design will eventually become possible through purely calculational methods; however, at present, it is partly an "art." Many of the currently designed shields make use of data from experimental studies.

F. Shielding Materials

Water, concrete, steel, and lead are the more common materials used for shielding purposes.

Some of the things which must be considered when choosing materials for a neutron shield are: (1) will it effectively decrease the neutron energy; (2) does it have a high capture cross section for thermal neutrons; (3) will it effectively attenuate the accompanying γ radiation? In addition, radiation absorption in the shield will release energy in the form of heat. Therefore, to insure shield integrity, it may be necessary to know the temperature distribution.

Water, because of its high hydrogen content per unit volume and low cost, is widely used as a neutron shield; however, it is a poor γ absorber. In cases where a liquid is undesirable, plastics, wood, or paraffin can be used. Polyethylene is probably the most commonly used plastic.

Concrete is an adequate shield material if the water content is at least 7% by weight. High-density concretes are recommended where space considerations are important. A certain amount of both neutron and γ attenuation can be obtained by use of a thinner shield of high-density concrete than is possible with ordinary concrete.

Boral, a mixture of aluminum and boron carbide (B_4C), is used for neutron shielding. Boron carbide, containing 80% boron, is a good neutron absorber, and the capture reaction in boron yields a relatively low-energy (0.5 MeV) gamma.

High-density materials are necessary for shielding the γ radiation. Lead is a valuable gamma shield; however, uranium, tin, and iron (in steel) are also used. For γ rays of about 2 MeV, a given mass of lead will achieve approximately the same attenuation as an equal mass of iron. Above or below this energy, lead is superior to iron. However, lead has a low melting point and temperatures must be considered if this element is used for shielding. Lead also presents a toxicity problem if improperly handled.

BIBLIOGRAPHY

1. Lapp, R. E., and Andrews, H. L., NUCLEAR RADIATION PHYSICS, Prentice-Hall, Inc., New York, N. Y. (1955).
2. Friedlander, G., and Kennedy, J. W., NUCLEAR AND RADIOCHEMISTRY, John Wiley and Sons, Inc., New York, N. Y. (1960).

3. Blatz, H., Editor, RADIATION HYGIENE HANDBOOK, McGraw-Hill Book Co., Inc., New York, N. Y. (1959).
4. Evans, R. D., THE ATOMIC NUCLEUS, McGraw-Hill Book Co., Inc., New York, N. Y. (1955).
5. Glasstone, S., SOURCEBOOK ON ATOMIC ENERGY, D. Van Nostrand Co., Inc., Princeton, N. J. (1958).
6. Price, B. T., Horton, C. C., and Spinney, K. T., RADIATION SHIELDING, Pergamon Press, New York, N. Y. (1957).
7. Rockwell, T., REACTOR SHIELDING DESIGN MANUAL, D. Van Nostrand Co., Inc., Princeton, N. J. (1956).
8. Glasstone, S., and Sesonske, A., NUCLEAR REACTOR ENGINEERING, D. Van Nostrand Co., Inc., Princeton, N. J. (1963).
9. Hine, G. J., and Brownell, G. L., RADIATION DOSIMETRY, Academic Press Inc., New York, N. Y. (1956).
10. Blizard, E. P., Nuclear Radiation Shielding, in ANNUAL REVIEW OF NUCLEAR SCIENCE, Vol. 5, Annual Reviews, Inc., Stanford, Calif. (1955).
11. Blizard, E. P., Editor, REACTOR HANDBOOK, Vol. III, Part B, Shielding, Interscience Publishers, New York, N. Y. (1962).
12. U. S. Department of Health, Education, and Welfare, OCCUPATIONAL RADIATION PROTECTION, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio (1957).
13. U. S. Department of Health, Education, and Welfare, ENVIRONMENTAL HEALTH ASPECTS OF NUCLEAR REACTOR OPERATION, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio (1958).
14. Division of Radiological Health, U. S. Department of Health, Education, and Welfare, RADIOLOGICAL HEALTH HANDBOOK, U. S. Government Printing Office, Washington 25, D. C. (1960).
15. Barbieri, R. E., Sweeton, R. F., Sakalosky, G. P., and Klatt, P. E., A RADIOBIOLOGY GUIDE, RCA Service Co., Inc., Camden, N. J. (1958).
16. PROTECTION AGAINST NEUTRON RADIATION UP TO 30 MILLION ELECTRON VOLTS, National Bureau of Standards Handbook 63, U. S. Government Printing Office, Washington 25, D. C. (1957).

SECTION 6 - BIOLOGICAL EFFECTS OF RADIATION

A. The Cell - Basic Unit of Structure

This section will be devoted to the effects of ionizing radiation on the human body. The structure of the body is quite complex, and it is often of value to deal with effects at certain levels of organization within the body. Thus, the human body contains many organs, each of which is composed of two or more types of tissue. In turn, a tissue is composed of similar cells, there being four types of tissues in the body. The job of a tissue is to perform a special body function. A cell is composed of many elements; hydrogen, oxygen, carbon, and nitrogen are the main components. For the present, our interest will be directed to the basic unit of structure in the body, namely, the cell.

A cell is normally composed of a nucleus surrounded by cytoplasm, both encased in membranes. Although highly complex in structure, the nucleus and cytoplasm contain about 70 per cent water. The vital part of the cell is the nucleus, usually an oval body near the center of the cell. From a chemical standpoint, the nucleus is quite active. The normal growth of the cell is controlled by the nucleus. Also, it initiates cell division and controls the repair of injured cells. The cytoplasm is a more or less colorless liquid substance. It secretes enzymes, and controls absorption and excretion in the cell.

Many body cells have only a limited life span. In order that their functions be carried on, cells divide at a certain stage in their life. The daughter cell then takes over the functions of the parent cell. As a cell divides, there appear in the nucleus threadlike structures called chromosomes. The chromosome number is fixed for a given species. Arranged linearly along the chromosomes are the genes which determine hereditary characteristics. When a cell divides, the daughter cell receives a duplicate set of chromosomes from the parent as well as identical genes. If the process is normal, no alterations or changes occur. However, any changes which do occur in the chromosomes and/or genes are called mutations. These changes or mutations can then affect either the daughter cell or future cells.

The development of an organ then proceeds from mitosis, the term used to designate cell division. In a bisexual species, the union of two cells (gametes) - one from the male and the other from the female - produces an original cell from which the species will be reproduced.

This cell then undergoes a number of divisions which increase its number. In this embryonic state, all the cells look alike. However, changes in the structure of the cells begin to take place. The changes enable the

cells to perform specialized functions. This process of change is referred to as differentiation. The result of this process is the development of different cell types or lines. One of these lines is the germ line, the rest are called somatic. The germ line gives rise to either male or female gametes. The somatic lines develop into the tissues of the individual.

Since only the gametes can be transmitted to future generations of the species, damage to somatic cells is limited to the individual. Damage to the offspring of an individual may occur when there is damage to the cells of the germ line.

A great many agents can cause injuries to the cells. When such injury occurs, the effects are the same regardless of the agent which caused the damage. Ionizing radiation produces damage to cells, but in a mostly nonspecific way, that is, other physical and chemical substances cause the same effects because the body responds the same to certain cell damage regardless of the cause.

Radiation passing through living cells will ionize or excite atoms and molecules in the cell structure. These changes affect the forces which bind the atoms together into molecules. If the molecule breaks up, some of the parts will be charged. These fragments are called radicals and ions, and are not chemically stable. Because the cell has a high water content, the most important radicals are those formed from water molecules. Further effects are produced when the radicals and ions interact with other cell material. In this way, damage is caused in a direct and indirect manner. The role that each type of action plays in the total damage to the cell is still an unsolved problem. Of the damage which is done, the effects are greatest in the nucleus of the cell, but injury to the cytoplasm can also cause serious effects in the cell.

The total effect on cell processes is a function of the dose of radiation. The cell processes will be affected in varying degrees up to the ultimate result - cell death. Some damage to the cell may be repaired. This can be accomplished by action of the cell itself or by replacement of badly injured cells in a given tissue through mitosis of healthy cells. On the other hand, if the extent of the damage to an organ is quite large, the organ may not be able to repair itself. Although many factors are important in assessing the total damage, it seems likely that most cell functions and structures are somewhat impaired by radiation.

Much work of a theoretical nature has been done concerning the threshold question. The threshold dose would be that dose below which no effects are observed. If such a concept is valid, then there would exist a dose of radiation such that no damage would be done. At the present time there is not enough empirical data to prove or disprove the existence of a threshold dose. For this reason, one should always assume that, however small the amount of radiation, it will produce biological effects.

B. Radiosensitivity

Since the cells which make up the tissues of the body differ both in appearance and function, one might suspect that their response to radiation would also differ. Such is the case, and this property is known as the radiosensitivity of the cell. The first statement about this property was given by Bergonié and Tribondeau. They found that the radiosensitivity of a tissue is directly proportional to the reproductive capacity and varies inversely with the degree of differentiation. Since then, other factors have been found which affect the radiosensitivity. Among these are the metabolic state of the cell, the state of cell division, and the state of nourishment. It turns out that to produce a given effect, the necessary radiation dose varies inversely as the relative sensitivity of the given tissue.

Thus, cells which are most active in reproducing themselves, cells which have a high metabolic rate (rate of chemical changes in the cell), and those cells which are more nourished than others are more sensitive to radiation. Also, there is evidence that cells are more susceptible to radiation at certain stages of division than at others. Moreover, cells not fully mature will also be more harmed by radiation than mature cells. In the body, bone marrow, lymphoid tissues, and the reproductive organs rank among the most radiosensitive. Muscle and bone cells are the least radiosensitive.

C. Radiation Damage

As has been pointed out, damage to somatic cells is limited to the individual whereas damage to germ cells may result in damage to the offspring of an individual. One may broadly classify biological effects in man as somatic or hereditary. Somatic effects include any and all types of damage which affect only the individual; hereditary effects are those effects which can be transmitted to a future generation (see Ref. 1). Thus, damage to the genes of a somatic cell may produce damage to a daughter cell, but this would be a somatic effect, not hereditary. The term genetic damage refers to effects caused by chromosome and/or gene mutations. This may lead to hereditary effects only when the damage affects the germ line since only then can these effects be transmitted to a future generation.

D. Somatic Effects

The somatic effects in man and animals are usually discussed in terms of total body and partial body irradiation, and with reference to damage to an organ. Because of the importance of some organs in the body, certain damage to these can induce effects in other organs. A number of physical factors are important in the determination of somatic effects:

- 1) nature or type of radiation - some types of radiation are more effective in producing damage;
- 2) the absorbed dose - this is a function of the energy absorbed per gram of tissue;
- 3) time distribution - a lethal dose given in a short time may not be lethal if protracted over a long time;
- 4) dose distribution - is the total body involved or only a specific organ?

All these factors combine to make the effects on different organs differ for changes in any of the parameters. Indeed, even the age of the individual enters the picture, since children are more affected by radiation than are adults.

The effects which are observed can be loosely divided into early and late effects. Although quite an arbitrary grouping, early effects are classed as those which appear within a few weeks after the exposure. The range of these effects as well as their duration depend upon the dose. For very high doses, there are three basic forms of early or acute damage. In the range above a few thousand rad, the dose is fatal within minutes to hours. The same symptoms appear when the head suffers severe irradiation, which points to a breakdown of the central nervous system. This type of acute radiation syndrome is thus referred to as central nervous system death (CNS death).

For the range 500-2000 rad, symptoms may appear within hours. Death often occurs within a week or so. In this mode, the damage to the lining of the intestinal tract is the most severe. This form is called gastro-intestinal tract death (GI death). At the lower end of this dose range, it is possible for one to survive this mode of death only to fall victim to the effects which prevail at lower doses.

At doses <500 rad, the most important effect is damage to the blood-forming organs. Since these centers are located in the bone marrow, this mode of death is often called bone-marrow death. The first signs may appear within a few days, depending upon the dose, and the total effect may not develop for a few weeks. Severe changes occur when the dose is >200 rad. In the range above 300 rad, the damage is severe enough so that death becomes more and more probable.

In the preceding sections, the results are based on X- and gamma-ray data. As such, the conversion to rem for other types of radiation is not justified in this instance.

The clinical effects which follow acute exposure to total body doses of ionizing radiation are given in summary in Table 6.1. This table is taken from Ref. 2 and may be of use as a guide. Note the virtual absence of any symptoms in the range below a dose of 100 rems. Some people would be expected to have mild symptoms in the range 50-100 rems because of differences among individuals. Below 50 rems, however, no symptoms at all are expected.

Death occurs in a larger fraction of cases as the dose increases. If the dose becomes large enough, all cases of exposure result in death. In the range where survival is possible, the concept of the median lethal dose (LD_{50}) is used: this expresses the dose at which 50 per cent of those exposed would die. For man, the best estimate places the LD_{50} at 300 to 500 rad. Of course, in this range all would have severe symptoms. Note that this dose refers to short-term total body radiation.

The previous paragraphs dealt with the overall effect of very high doses. In each case, the site of major damage leading to death was discussed. Other organs and systems will also be involved along with those whose damage may be conducive to death. The following breakdown attempts to point out some of the somatic effects caused in specific tissues and organs of the body:

1. Blood and Bone Marrow

The blood is composed of three major types of cells, namely, the erythrocytes (red cells), the leukocytes (white cells), and platelets, suspended in a fluid called plasma. The red cells supply other body cells with food and oxygen, and remove waste products. White cells help to combat infections, and the platelets aid in blood-clotting action. Plasma is a viscid liquid which contains water, proteins, salts, and free ions. Blood contains about 45% red cells, ~1% white cells and platelets, and 54% plasma.

Bone centers are filled with marrow, either red or yellow. Red marrow is found in the skull, breastbone, ribs, pelvis, and spine of adults. The red marrow provides the blood-forming function. Yellow marrow provides fat storage.

White cells are the first to be affected by radiation. Although there are subtypes of white cells which differ in their sensitivity, the net effect of irradiation is to reduce the number of white cells. This lack of white cells is known as leukopenia. In cases of severe irradiation, the platelets drop in number within about a week. A few weeks later, a loss of red cells (anemia) occurs.

The loss of white cells affects resistance to infection. The drop in the platelet number affects clotting action, so that open wounds

Table 6.1*
SUMMARY OF CLINICAL EFFECTS OF ACUTE IONIZING RADIATION DOSES

Range	Subclinical Range, 0 to 100 rems	Therapeutic Range, 100 to 1,000 rems			Lethal Range, over 1,000 rems	
		100 to 200 rems	200 to 600 rems	600 to 1,000 rems	1,000 to 5,000 rems	Over 5,000 rems
		Clinical surveillance	Therapy effective	Therapy promising	Therapy palliative	
Incidence of vomiting	None	100 rems: 5% 200 rems: 50%	300 rems: 100%	100%	100%	
Delay time	-	3 hr	2 hr	1 hr	30 min	
Leading organ	None	Hematopoietic tissue			Gastrointestinal tract	Central nervous system
Characteristic signs	None	Moderate leukopenia	Severe leukopenia; hemorrhage; infection; purpura Ecchymation above 300 rems		Diarrhea; fever; disturbance of electrolyte balance	Convulsions; tremor; ataxia; lethargy
Critical-period post-exposure	-	-	4 to 6 wk		5 to 14 d	1 to 48 hr
Therapy	Reassurance	Reassurance; hema- tologic surveillance	Blood transfusion; antibiotics	Consider bone-marrow transplantation	Maintenance of electrolyte balance	Sedatives
Prognosis	Excellent	Excellent	Good	Guarded	Hopeless	
Convalescent period Incidence of death	None None	Several weeks None	1 to 12 mo 0 to 80% (variable)	Long 80 to 100% (variable)	90 to 100%	
Death occurs within	-	-	2 mo		2 wk	2 d
Cause of death	-	-	Hemorrhage; infection		Circulatory collapse	Respiratory failure; brain edema

*The Effects of Nuclear Weapons, U. S. Government Printing Office, Washington 25, D. C. (April 1962).

may not heal. Anemia causes a general weakness in the individual. Recovery will take place if the damage to the bone marrow is not too great. The marrow will regenerate and produce new blood cells to replace the cells which were lost. When the damage to the red marrow is too great, the effects are likely to be permanent.

2. Lymphatic System

The lymphatic system is a network of small tubes which permeate the body tissues. A fluid called lymph, somewhat like plasma but with less proteins, drains from tissues into the lymphatic system. The lymph picks up waste products from the tissues. Along the course of a lymph vessel are oval-shaped glands (lymph nodes) which filter out foreign substances from the lymph. Thus purified, the lymph is passed back into the blood stream.

The spleen contains the largest mass of lymphatic tissue in the body. The spleen filters dead blood cells from the blood and is a source of white blood cells. Also, the spleen stores red blood cells.

The lymph nodes show the first signs of hemorrhaging and infection after acute irradiation. The spleen may exhibit weight loss and damage to lymphocytes (a subtype of white blood cells).

3. Digestive Tract

The digestive tract, or alimentary canal, consists of the mouth, pharynx, esophagus, stomach, and the small and large intestines. In an adult man, the canal may be as much as 30 ft in length. The cells which line the walls of the intestines secrete substances which act on food to make absorption into the blood stream possible. The stomach is the reservoir in which the chemical phases of digestion begin. The radiosensitivities of the many sections of the canal vary greatly. The small intestine is quite radiosensitive, whereas the stomach and esophagus are much less radiosensitive.

The symptoms of damage to the canal are nausea and vomiting. The initial effects are impaired secretion and discontinued cell production. When cell breakdown occurs, the dead cells are released from the walls of the tract. This debris clutters up the intestine. The exposure of tissues under the surface layer may lead to ulcers. In fatal cases, infection, failure of food absorption, and dehydration from diarrhea are the causes.

4. Reproductive Organs

Since the reproductive organs are the source of germ cells, damage to these cells can result both in somatic and hereditary effects. For the present, our concern is only with the somatic effects. The response of

germ cells to radiation differs slightly in the male and female. If the dose is high enough, the effect will be sterility. It requires a larger dose to produce permanent sterility in the male than in the female. In most cases, the dose required to produce permanent sterility is in the region of fatal doses. In man, partial sterility can be induced at doses around 150 rad.

Germ cells which survive damage can transmit any genetic changes caused by the radiation. For this reason, the total effect of radiation on the gonads may not be seen for several generations.

5. Nervous System

The nervous system, composed of the brain, spinal cord, and the peripheral nerves, acts to coordinate body activity. The spinal cord and peripheral nerves are highly radioresistant, but the brain is more sensitive than often supposed.

The effect of radiation on the brain is to alter function rather than structure for doses up to LD₅₀. At higher doses, brain damage may occur directly or through lack of blood supply due to damage of blood vessels.

6. Thyroid Gland

The thyroid, a gland located at the base of the throat, secretes a hormone known as thyroxine, which helps to control basal metabolism. The action of the thyroid seems to be closely connected with the functions of the pituitary and adrenal glands. Thyroxine contains about 65 per cent iodine, and is essential for growth and development. Damage to the thyroid, or to the other two glands, have marked affects in the body.

The thyroid is radioresistant from the standpoint of external radiation. It can be severely damaged if radio-iodine is inhaled, since iodine will concentrate in the thyroid. Damage causes a decrease in production of thyroxine, which produces a lower metabolism rate. Muscle tissue may then fail to absorb enough oxygen and health can be badly impaired.

7. Eyes

The lens of the eye is highly susceptible to irreversible damage by radiation. The lens cells of the eye are not replaced by regrowth. The retina is much less sensitive than the lens.

When the cells of the lens become damaged, the cells lose their transparency. The opacities which may occur will appear after a latent period. The term cataract is applied to lens opacities. The formation of cataracts is thus a late somatic effect. Acute effects in other eye structures occur only after high doses.

Neutrons are more effective than X rays in producing cataracts. Doses of X rays greater than 500 rad will produce significant cataract, but after only 200 rad of mixed gamma and neutrons, lens opacities have been reported.⁽¹⁾

The susceptibility to cataract formation depends somewhat on age. Radiation is more likely to cause cataracts in younger people.

8. Lungs

The lungs are cone-shaped organs made up of very small air sacs called alveoli. When a person breathes, the air is directed down the trachea (wind pipe). From there, two large tubes (bronchi) direct the air toward each lung section. Many small tubes (bronchioles) branch out from the bronchi to connect with the alveoli in the lungs.

During breathing, each air sac is expanded and compressed by lung muscles, and is thus filled and emptied. Air passes through the walls of the alveoli into tiny blood vessels (capillaries).

The effects produced in the lung by radiation are the result of damage to the air sacs. The lungs are not normally affected by external radiation. As in the case of the thyroid, the greater hazard occurs from internal radiation from inhaled dust and vapors.

9. Liver and Gall Bladder

The liver is radioresistant as compared with other organs. The liver, the largest gland in the body, secretes bile for digestion. The gall bladder stores and concentrates the bile secreted by the liver. When bile is needed, it passes from the gall bladder to the stomach.

External radiation is not too effective in causing damage to these organs. Most damage is caused by internal exposure from radioisotopes which concentrate in the liver.

10. Kidneys

The kidneys help to control the concentration and content of the blood by excreting water and waste products. The waste products pass from the kidneys through small tubes (ureters) into the bladder. Impairment of renal functions does not add to mortality in the case of total body radiation. Damage to the kidney is indicated by an increase in amino acids in the urine. These effects occur mostly from internal radiation. The appearance of blood in the urine is an indication of severe renal damage.

11. Circulatory System

The heart and blood vessel system are damaged seriously only for very high doses of radiation.

12. Skin

The degree of skin damage varies with the dose and the species of animal. Skin is easily damaged but has a tremendous capacity for repair. Various structures of the skin show quite different sensitivities. The damage seems to be greater for less penetrating radiations.

Slight damage to the skin may result in an erythema (reddening). Skin cancer is a late effect of chronic irradiation at high dose rates.

13. Hair

Irradiation can lead to temporary baldness (epilation). This condition may last for a few weeks. The hair begins to return, but the new hair may have different characteristics, such as a new color.

14. Bones

Bone is composed of living cells which are distributed in a matrix of fibers and bone salts. Although the marrow of the bone is very radiosensitive, the bone cells, fibers, and salts are relatively radioresistant.

When radioisotopes, such as strontium or plutonium, are internally deposited in the bone marrow or bone tissue, then great damage can be done. These effects again are late effects since the damage may take years to show up.

E. Late Effects

The problem in the study of late effects resulting from exposure to radiation is that the elapsed time may be rather long, thus making it hard to relate the cause to the effect. Since the late effects may be caused by many other agents besides radiation, there can be no positive assignment of the cause in most cases. At best, it can be shown that radiation increases the incidence of these nonspecific injuries.

The main late effects are discussed in the following.

1. Neoplasms

Animal studies indicate that the incidence of cancer is a function of the dose and the time duration of the dose. Many types of cancer are

produced in man, most likely in the skin, the lungs, the bones, and the bone marrow (leukemia). The latent period is long for radiation-induced cancers, of the order of years.

2. Tissue Effects

Among the more prominent late effects in tissues are cataracts and sterility. Radiation-induced cataracts are slowly progressive over a period of time, but may stop or even regress. Sterility is a late effect which may be either permanent or temporary. In some cases, fertility will return in a few years.

3. Life-span

Information on life-shortening effects in man is still inadequate. The effects of long-term, low-level irradiation on longevity cannot be predicted. Much more data are needed before any conclusions can be drawn.

4. Growth and Development

Effects on the embryo depend upon the dose as well as the age of the fetus. The younger the fetus, the more it is affected. Here, as in the case of other late effects, the results of damage are the same as those caused by other agents. The effects on the fetus are so much more important since minor damage may be amplified during growth into a major anomaly.

The incomplete status of these problems at the present time reflects the lack of needed information. Such data are very hard to obtain, and require careful work and analysis. In each case of late effects, extensive data are needed before any real firm conclusions may be drawn. At present, all that is really known is that radiation does increase the incidence of these effects.

F. Hereditary Effects

The study of hereditary effects attempts to discover the traits which can be transmitted from generation to generation in a given species. The genes are the determinants of the inherited traits. Any change or mutation of a gene, which is usually quite stable, can result in an altered trait. Such changes can be produced by radiation, as well as other agents. The study of radiation-induced mutations is thus hampered by the fact that other substances also act to produce the same effects. Since the change is not unique, radiation only serves to increase the frequency of the effect. Increases in the rate are small even for high doses. Thus, the study requires the use of large numbers of subjects studied over many generations. In the case

of man, the study is very difficult, since large numbers are seldom available and the time between generations is so long.

For this reason, much of the present knowledge is based on work with animals. At all doses and dose rates used up to the present, radiation is known to induce mutations in all species studied. Because of this, any increase of radiation to humans should bring about an increase in the mutation rate. Sometimes the application of animal data to man can result in error. However, in this case, the effects on some other species are similar enough to those in man.

Recent studies have shown that some hereditary effects in man are caused by chromosome damage. Up to the present time, the lack of knowledge about the harmful traits has made estimates of the magnitude of these effects quite doubtful. When more diseases and effects can be related to certain types of mutations, then better estimates can be made.

Genes may be dominant or recessive. When genes differ, the trait which results may be determined by either gene, or some intermediate trait may occur. If the trait is determined by one or the other of the genes, this gene is then a dominant gene. The other is a recessive gene. Mutations in dominant genes give rise to damage in the offspring of the first generation. Damage to offspring caused by recessive genes occurs only if the same altered gene is received from each parent. Unless these changes occur frequently, recessive damage won't show up for many generations.

As in the study of late somatic effects, further progress in the problems which concern hereditary effects await much needed data from careful experimentation. Perhaps the study of those groups which have been exposed to higher doses may yield useful data. In any case, there remain many problems of real concern which must be overcome.

There are many areas in which the surface has only been scratched. It is only through further study that many of the factors which play a role in hereditary effects will be ascertained.

BIBLIOGRAPHY

1. REPORT OF THE UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Seventeenth Session, Supplement No. 16 (A/5216), United Nations, New York (1962).
2. Glasstone, S., editor, Effects on Personnel, THE EFFECTS OF NUCLEAR WEAPONS, U. S. Government Printing Office, Washington 25, D. C., (April 1962).

3. Nickson, J. J., and Bane, H. N., "Physiological Effects of Radiation," RADIATION HYGIENE HANDBOOK, edited by H. Blatz, McGraw-Hill Book Co., Inc., New York, N. Y. (1959).
4. A RADIOBIOLOGY GUIDE, WADC TR 57-118 (June 1958). Available from the Armed Services Technical Information Agency, Arlington Hall Station, Arlington 12, Va.
5. Millard, N. D., and King, B. G., HUMAN ANATOMY AND PHYSIOLOGY, THIRD EDITION, W. D. Saunders Company, Philadelphia, Pa. (1953).
6. Hercik, F., and Jammet, H., SAFE HANDLING OF RADIOISOTOPES: MEDICAL ADDENDUM, IAEA, Vienna (1960).
7. THE BIOLOGICAL EFFECTS OF ATOMIC RADIATION, SUMMARY REPORTS, Washington: National Academy of Sciences - National Research Council (1960).
8. Hine, G. J., and Brownell, G. L., RADIATION DOSIMETRY, Academic Press Inc., New York, N. Y. (1956).

SECTION 7 - BACKGROUND RADIATION

Apart from the amount of radiation which a worker may receive while he performs his work, he will also be exposed to radiation because of the very nature of his environment. All individuals are subject to some irradiation even though they may not work with radioactive substances. This natural source of exposure is often referred to as "background radiation."

Studies of the nature and origin of this source of exposure to man have revealed five main components: cosmic radiation, the radioactivities of the earth's surface, of air, of water, and of the human body. One might add that a man-made source (that is, fallout from nuclear weapons testing) influences the contribution from some of these sources. Although the amount which each of these factors contributes varies with the locale, each locale will have some background levels.

The study of these factors throughout the world is of value for a number of reasons. Foremost among these is that the use of such data provides a basis or standard from which allowable exposure limits for radiation workers may be developed. In areas where the levels are much higher because of larger concentrations, knowledge may be gained about human hereditary effects at these increased levels. Such data are also needed in assessing the impact on or contribution of a nuclear facility to the existing concentrations in a given area. In the design of buildings and/or shielding for low-level work, it is of value to know the radioactive contents of the substances used. Often the levels inside a building are higher than the outside of the building because this factor has been neglected.

Because of these needs, much data about background levels in many areas have been acquired. The rest of this section will be devoted to a discussion of these background factors.

A. Cosmic Radiation

Much work has been carried out in the study of cosmic radiation. This factor in background levels was discovered during attempts to reduce background. Though detection devices showed a response even in the absence of any known sources, it was assumed this background was due entirely to traces of radioactive substances in the air and ground. Thus, if a detector were elevated to a greater height above the earth's surface, the background should be greatly reduced. The use of balloons carrying ion chambers to great heights yielded data which showed the effect increased, rather than decreased. These and other data showed that radiation was really coming from outer space. The name "cosmic rays" was given to this high-energy, extraterrestrial radiation.

Further study has shown that cosmic radiation consists of two parts: primary and secondary. The primary component may be further divided into galactic, geomagnetically trapped radiation, and solar.

The galactic cosmic rays come from outside the solar system and are composed mostly of positively charged particles. Studies have shown⁽¹⁾ that outside the earth's atmosphere, cosmic rays consist of 87% protons, of 12% alpha particles, and of 1% other heavier nuclei at latitudes above 55°. These particles may have energies in the range from a few BeV to greater than 10^{17} eV.

As a charged particle approaches the earth, it is acted upon by the earth's magnetic field. In order to pass on through to the earth, the particle must have a certain momentum. Otherwise, it may be trapped by the earth's magnetic field. This gives rise to the second type of primary cosmic rays, the geomagnetically trapped radiation. This consists of two belts of radiation (electrons and protons) which are observed at high altitudes. These belts are symmetrical with respect to the magnetic equator.

The first band occurs at about 1000 km and extends from 30° N to 30° S. The intensity in the belt increases with altitude until about 3000 km and then drops off. The energy spectrum is not very well known at the present time. These belts of radiation are being studied because of the dose rates they would produce during space travel.

The second band starts at about 12,000 km and reaches its maximum at 15,000 km. It extends from 60° N to 60° S. It is thought that dose rates in this band are higher than those in the inner belt, but that shielding would reduce the dose rate considerably.

Solar cosmic rays are produced following severe solar flares on the surface of the sun. These rays consist of protons. The events are classed as high energy or low energy. The high-energy events can be observed by ground-level neutron devices. The low-energy events are more frequent but must be detected at high altitude. Since these events produce radiation throughout the solar system, they are of great importance in shielding design for manned space missions.

Secondary cosmic rays result from interactions which occur when the primary rays reach the earth's atmosphere. When the high-energy particles collide with atoms of the atmosphere, many products are emitted: mesons, electrons, photons, protons, and neutrons. These, in turn, produce other secondaries as they collide with elements or decay on the way toward the earth's surface. Thus, a multiplication or shower occurs in which as many as 10^8 secondaries may result from a single primary.

Most of the primary rays are absorbed in the upper $1/10$ of the atmosphere. At about 20 km and below, cosmic rays are almost wholly secondary in nature. The total intensity of cosmic rays shows an increase from the top of the atmosphere down to a height of 20 km. Although the primary intensity decreases, the total effect increases because of the rapid rise in the number of the secondaries. Below 20 km, the total intensity shows a decrease with height because of attenuation of the secondaries without further increase in their number due to primaries.

At the earth's surface, the secondary cosmic rays consist of mesons mainly (hard component), electrons and photons (soft component), and neutrons and protons (nucleonic component). At sea level, about $3/4$ of the cosmic ray intensity is due to the hard component.

Because of the earth's magnetic field, cosmic ray intensity also varies with latitude. The energy which is needed for a charged particle to reach the earth's atmosphere at the geomagnetic equator is larger than that needed at other latitudes. The effect is greatest for latitudes between 15 and 50° . Above 50° , the intensity remains almost constant. Thus, the lowest value of the intensity occurs at the geomagnetic equator, and the effect is expressed as the percentage increase at 50° over that at the equator. At sea level, the effect is small for the ionizing component (10%) but large for the neutron component (150%).

The dose rate produced by this source of background may be divided into two parts. The portion caused by the ionizing component is estimated from ion chamber readings. The portion caused by the neutron component is hard to measure because the dose rate depends so much on the energy spectrum of the neutrons.

At sea level and high latitudes, the ionization is about $1.96 \text{ ip/cm}^3\text{-sec}$, giving a dose rate of about 28 mrad/yr for soft tissue. Patterson *et al.*⁽²⁾ have given an estimate of 25 mrem/yr as the dose rate from neutrons. The total dose rate at sea level would then be on the order of 50 mrem/yr at middle latitudes.⁽¹⁾ This dose rate would be expected to increase with altitude and decrease with latitude.

B. Radioactivity of the Earth

As was pointed out before, the presence of certain small amounts of radioactivity in the soil adds to the background levels to which man is exposed. The amount of radioactive materials found in soil and rocks varies widely with the locale. The main contribution to the background is the gamma-ray dose from radioactive elements of the uranium and thorium series and from radioactive K^{40} .

In certain areas of the world, such as Brazil, France, India, Niue Island, and the United Arab Republic, much higher amounts of these substances have been found in soil and rock. In some of these areas, the average gamma dose rate for the year exceeds one rad.

For other areas, studies have been made in an attempt to measure the average gamma dose rate to the world population. Not enough data have been collected as yet to make a firm estimate. The studies which have been made include values for dose rates both indoors and outdoors. By means of these data and the assumption that most of a person's time is spent inside buildings, the estimated mean dose rate turns out to be about 50 mrad/yr.

C. Radioactivity of Air

The background which is found in air is due mainly to the presence of radon and thoron gas, formed as daughter products of elements of the uranium and thorium series. The decay of U^{238} proceeds to Ra^{226} . When Ra^{226} emits an alpha as it decays, the gas Rn^{222} is formed. In the thorium chain, the decay of Ra^{224} results in the product Rn^{220} , which is called thoron.

Since uranium and thorium are present to some extent throughout the crust of the earth, these products are being formed all the time. Since they are gases, they tend to diffuse up through the earth's surface to become airborne. In turn, the decay products of these gases attach themselves to dust in the air.

The amount of these gases in the air depends upon the uranium and thorium content of a given area. In any given area, the weather conditions will greatly affect the concentrations of these gases. It is also common to find that the levels indoors are higher than those outdoors. This is a function of the material of the building and the ventilation rate. In mines and other underground caverns, the concentrations have been found to be quite high.

The external gamma dose rate from Rn^{222} and Rn^{220} is based upon an average concentration of $0.5 \mu\mu\text{Ci}/\ell$ of Rn^{222} and $0.02 \mu\mu\text{Ci}/\ell$ of Rn^{220} . The tissue dose rate turns out to be about 4.5 mrem/yr. Many attempts have been made to try to assess the dose rate to the lungs from these products. This estimate depends quite a bit on the assumptions made in regard to the many factors which enter into the respiration process. Data from Sweden seem to indicate that the lungs may receive a higher dose from natural sources than any other body tissue!

Among other radioactive products which are found in air are C^{14} , H^3 , and Be^7 . None of these products add a significant amount to the background dose rate.

D. Radioactivity of Water

Depending upon the type of water supply one is talking about, a number of products may turn up. For example, sea water contains large amounts of K^{40} . On the other hand, many natural springs show amounts of uranium, thorium, and radium. Almost all water should be expected to contain certain amounts of radioactivity. Since rain water will pick up radioactive substances from air, and ground water will pick up activity present in rocks or soil, one would expect to find some radioactivity in water throughout the world.

The chief source of dose rate from this background factor occurs as the result of uptake of these waters by ingestion. This leads to an internal exposure. Any estimate of the dose rate from this source is thus included in the estimate of the dose rate from radioactivity in the human body. The transfer of radioactive substances to the body seems to be mainly by food intake though, except in cases of very high water concentrations. Tables of Ra^{226} and Ru^{222} concentration, as well as other data, can be found in Ref. 1.

E. Radioactivity of the Human Body

Since small amounts of radioactive substances are found throughout the world in soil and water, some of this activity is transferred to man by way of the food-chain cycle. A number of studies have been made to try to find a correlation between the amounts in soil and that in man. Results have not shown a clear-cut relationship as yet.

In the human body, K^{40} , Ra^{226} , Ra^{228} , and C^{14} are the main radioisotopes of concern. Of these, K^{40} is the most abundant substance in man. The amount in food varies greatly, so that intake is quite dependent on diet. The content in the body organs of man varies widely. Based on an average content of 0.2% by weight in man, the dose rate to the gonads turns out to be about 20 mrem/yr.

Most of the Ra^{226} which is taken into the body will be found in the skeleton. Much data has been gathered on the concentration in humans, and the present assumed average concentration is taken as 75 $\mu\mu\text{Ci}$. Of this amount, 80% is assumed to be in the skeleton. The content of Ra^{228} is taken as 50 $\mu\mu\text{Ci}$. The dose rate produced by these products is shown in Table 7.1.

The average whole-body content of carbon is taken as 18%. However, C^{14} is present in normal carbon only to a very small extent ($C^{14}/C^{12} \sim 10^{-12}$), so that only a small amount of C^{14} is present. The average dose rate turns out to be about 1 mrem/yr total body.

Table 7.1 gives a summary of the present estimates for the dose rate from all background sources.

Table 7.1*
 BODY TISSUE DOSE RATES DUE TO EXTERNAL AND INTERNAL
 IRRADIATION FROM NATURAL SOURCES OF RADIATION IN
 "NORMAL REGIONS"

Source of Irradiation	Dose Rate, mrem/yr		Bone Marrow
	Gonad	Haversian Canal	
External Irradiation:			
Cosmic Rays (Including Neutrons)	50	50	50
Terrestrial Radiation (Including Air)	50	50	50
Internal Irradiation:			
K ⁴⁰	20	15	15
Ra ²²⁶ and decay products (35 per cent equilibrium)	0.5	5.4	0.6
Ra ²²⁸ and decay products (equilibrium)	0.8	8.6	1.0
Pb ²¹⁰ and decay products (50 per cent equilibrium)	0.3	3.6	0.4
C ¹⁴	0.7	1.6	1.6
Rn ²²² (absorbed into bloodstream)	3	3	3
Total	125	137	122

*This table taken from Report of the United Nations Scientific Committee on the Effects of Atomic Radiation (1962).

F. Fallout

The term "fallout" has been applied to debris which settles to the earth as the result of a nuclear blast. This debris is radioactive and thus a source of potential radiation exposure to man. Since fallout is a man-made source of irradiation, it does not properly come under the title of natural background. However, since fallout can alter the contribution due to other background sources, it will be treated in this section.

Because of the intense heat produced in a nuclear explosion during a very short time, matter which is in the vicinity of the bomb is quickly vaporized. This includes fission products formed in the fission process, unused bomb fuel, the bomb casing and parts, and, in short, any and all substances which happen to be around. These are caught in the fireball which expands and rises very quickly. As the fireball cools and condensation occurs, a mushroom-shaped cloud is formed, containing small solid particles of debris as well as small drops of water. The cloud continues to rise to a height which is a function of the bomb yield and the meteorological factors of the area. For yields in the megaton range (1 megaton equals an energy release equivalent to 10⁶ tons of TNT), the cloud top may reach a height of 25 miles.

The fallout which occurs may be described as local or world-wide. The portion of debris which becomes local fallout varies from none

(in the case of a high-altitude air burst) to about half (in the case of a contact surface burst). The height at which the bomb goes off is thus quite important in the case of local fallout. If the fireball touches the surface of the earth, it will carry aloft large amounts of surface matter. Also, because of the suction effect created by the rapid rise of the fireball, other matter may be taken up into the rising fireball. This leads to the formation of larger particles in the cloud that tend to settle out quickly. If the wind is not too great, the fallout pattern will be roughly a circle about ground zero. Ground zero is the point on the surface directly under, at, or above the burst.

Other bits of matter will fall out at various stages. The distance from ground zero at which they strike the surface and the time it takes depend upon the height from which they fall, their size, and the wind patterns at all altitudes. This results in a cigar-shaped pattern downwind of the burst point. Local fallout usually occurs within the first 24 hr after the blast.

If the height of the burst is such that the fireball does not touch the surface, then the debris is carried aloft and dispersed into the atmosphere. This matter then descends to earth at a later time and is called world-wide fallout.

The residence time of this debris is a function of the bomb yield. For yields in the kiloton range, the debris is not projected into the stratosphere. It is limited to a region called the troposphere, between about 30,000 to 55,000 ft. In this region, there is quite a bit of turbulence as well as precipitation. The debris is removed rather quickly, from about one day to one month.

If the burst is in the megaton range, the debris is carried into the stratosphere. In this region little mixing will occur, and the absence of rain or snow prevents this matter from being washed down. The time that it takes for this debris to return to the troposphere and be washed down varies. It is a function of both the height in the stratosphere to which the debris is lifted and the locale at which the burst occurs. It may take up to 5 yr or more for this debris to return to earth. On the other hand, for bursts in the northern hemisphere in which the debris is confined to only the lower part of the stratosphere, the half-residence time is thought to be less than one year. The concept half-residence time is the time for one-half of the debris to be removed from the stratosphere.

In all, there are more than 200 fission products which result from a nuclear blast. The half-life of each of these products covers the range from a fraction of a second to millions of years. Local fallout will contain most of these products. Because of the time delay in the appearance of world-wide fallout, only a few of these products are important from that standpoint. Since local fallout is confined to a relatively small area, its

effect on the human population can be negated by proper choice of test sites, weather conditions, and type of burst. The fallout of interest from the standpoint of possible effects on man due to testing is the world-wide fallout. It should be noted that in a nuclear war, local fallout would by far present the most danger.

G. Effect of World-wide Fallout

A number of factors must be considered when one attempts to assess the hazard from world-wide fallout. Because of the associated time delay before world-wide fallout shows up, many fission products decay out in transit. Others, because they are produced in such small amounts, are diluted so that they do not produce much of an effect. Also, once the fallout does arrive, there must be a transfer to the body and absorption into the body organs. All these factors combine to limit the number of fission products which may have an effect on man. These are: I^{131} , Sr^{89} , Sr^{90} , and Cs^{137} . Another product, C^{14} , which is produced by neutron absorption by nitrogen atoms in air, is also of interest.

Both I^{131} and Sr^{89} , because of their short half-lives are of concern only as short-term hazards. On the other hand, Sr^{90} , Cs^{137} , and C^{14} are important from the standpoint of long-term hazards. At the present time, the impact of world-wide fallout on natural background levels has been quite small. The estimates of the average dose to a person in the United States during the next 70 yr are placed between 0.4 and 0.9 rem to the bone. Values for bone marrow and gonads are even less than this. Thus, the contribution to the background dose rate from the testing of nuclear weapons does not seem to be very serious. Danger would come from a large increase in the fission product inventory in the stratosphere. This condition could result from large-scale, long-term atmospheric testing of weapons.

BIBLIOGRAPHY

1. REPORT OF THE UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Seventeenth Session, Supplement No. 16(A/5216), United Nations, New York (1962).
2. Patterson, H. W., et al., The flux and spectrum of cosmic-ray produced neutrons as a function of altitude, HEALTH PHYSICS, 2, 69-72 (1959).
3. Langham, W. H., Some radiation problems of space conquest, Paper presented at the XIth INTERNATIONAL ASTRONAUTICAL CONGRESS, Stockholm, Aug. 15-20, 1960.

4. Cowan, F. P., "Natural Radioactive Background," in RADIATION HYGIENE HANDBOOK, edited by H. Blatz, McGraw-Hill Book Co., Inc., New York (1959).
5. Greisen, K., "Cosmic Ray Showers," in ANNUAL REVIEW OF NUCLEAR SCIENCE, E. Segre, editor, Annual Reviews, Inc., Palo Alto (1960).
6. Waddington, C. G., "The Composition of the Primary Cosmic Radiation," in PROGRESS IN NUCLEAR PHYSICS, O. R. Frisch, editor, Pergamon Press, London (1960).
7. Singer, S. F., "The Primary Cosmic Radiation and its Time Variations," in PROGRESS IN ELEMENTARY PARTICLE AND COSMIC RAY PHYSICS, 4, 205-335 (1958).
8. Eisenbud, M., ENVIRONMENTAL RADIOACTIVITY, McGraw-Hill Book Co., Inc., New York (1963).
9. Lucas, H. F., Jr., Correlation of the natural radioactivity of the human body to that of its environment: Uptake and retention of Ra^{226} from food and water, ANL-6297, pp. 55-56 (1961).
10. Scott, R. C., and Barber, F. B., Radium and uranium in ground water of the United States, PROCEEDINGS 2ND INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY, Geneva, Switzerland, 2, 153-157 (1958).
11. REPORT OF THE UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Report to the General Assembly, Thirteenth Session, Supplement No. 17(A/3838) 1958.
12. Comar, C. L., FALLOUT FROM NUCLEAR TESTS, U. S. Atomic Energy Commission, Division of Technical Information.
13. Glasstone, S., editor, Chapters IX and XI, THE EFFECTS OF NUCLEAR WEAPONS, Washington: U. S. Government Printing Office, April 1962.

SECTION 8 - RADIATION PROTECTION STANDARDS

A. History of Protective Standards

The task of setting exposure limits is both a vital and yet a very difficult undertaking. It is vital because workers must be protected from the harmful effects of ionizing radiation. It is difficult because of the many factors which enter into the effects which radiation produces (see Section 6). Even though a vast amount of data has been gathered and studied, there are still many areas where much work is needed before firm conclusions can be drawn. Nevertheless, in order to advance in the field of nuclear energy, men must work with radiation. Thus, certain levels must be set which will protect workers from undue exposure.

Because there are still some unknowns which must be evaluated, the setting of limits involves judgments which cannot be wholly based upon the present body of scientific knowledge. For this reason, the concept of an "acceptable risk"⁽¹⁾ is used. In other words, the benefits are weighed against the potential damage and then limits are set at some level at which the most benefit to mankind will accrue.

From time to time, these limits will be revised as new knowledge is gained. When some of the assumptions can be replaced by facts, then it becomes prudent to review the limits and perhaps make firmer recommendations. The whole history of the development of exposure limits points out this feature of re-evaluation in the light of present knowledge.

With the discovery of radioactivity and a consequent intensive investigation of the phenomenon, many people were subjected to very high dose rates, and it did not take long for deleterious effects to become manifest. As early as 1897, cases of skin damage began to appear.

1. Erythema Dose

Early efforts at control were hampered by a lack of quantitative methods. There were no units by which one could assess the amount of radiation. No one even knew what was how much, let alone too much radiation! As a result of the use of radiation by doctors in treating patients, a unit called the erythema dose came into use. This was a highly qualitative unit since it was defined in terms of the amount of radiation which would produce a well-defined reddening of the skin. It soon became apparent that this dose unit was not at all satisfactory. It was found that it varied not only with the type of radiation and the dose rate, but also with the response of different parts of the body. Thus, two people could receive the same supposed fraction of an erythema dose, yet one might show skin effects and the other none. This lack of a certain value for this unit made protection work more or less of a trial-and-error process. What may have been safe with one type of source in a given type of treatment might then turn out to be not so safe for some other source.

During this early period, much of the use of radiation was centered about the medical field. Around 1914, radiation began to be used in industry. The radium dial-painting process came into being, and X rays were found useful for showing up flaws in materials. With the advent of radiation uses in industry, the need for standards and exposure limits became more acute. Larger numbers of people were exposed. No longer could the vague notion of erythema dose serve the purpose of a protection standard. Yet progress toward better standards still lagged because of lack of knowledge of the complex factors which enter into radiation effects.

2. Tolerance Dose

In the early 1920's, a movement for better standards and stricter control of radiation began.⁽²⁾ By this time, some of the long-term effects of exposure to high dose rates were known to radiologists and technicians. A great deal of effort was expended in many countries in an attempt to improve protection standards. The concept of a tolerance dose arose out of these studies as well as a search for a better unit of dose.

3. ICRU, ICRP, and NCRP

In 1925 at the First International Congress of Radiology, the International Commission on Radiological Units and Measurements (ICRU) was formed, mainly because of the lack of a suitable dosage unit of international acceptance.⁽³⁾ In 1928, this group adopted the definition of an international unit, the Roentgen. For the first time measurements throughout the world could be made in terms of the same unit. Over the years the ICRU has been the main force in defining and adopting units for use on an international basis (see Section 4).

At the Second International Congress of Radiology in 1928, the first international body concerned with protection standards was formed.⁽⁴⁾ At first known as the International X-ray and Radium Protection Commission, this group is now called the International Commission on Radiological Protection (ICRP). The task of this group has been to discuss and review basic protection principles, and to then prepare recommendations of safe practices based on their findings. These recommendations then serve as a guide from which regulations can be drawn up by each country to suit its needs. Although this group acts only as an advisory board, it has had a tremendous impact on the field of radiation protection.

In its early days, the ICRP did not have a wealth of data from which to draw. Because the need for exposure limits was a pressing matter, attention was directed to this aspect. In 1934, the ICRP made its first recommendation of a tolerance level of exposure: 0.2 R/day. This limit remained in force until 1950. However, because of World War II, the ICRP did not meet between 1937-50. This left much of the study of protection standards during this time to the national committees.

In this regard, one cannot help but mention the work done by the National Committee on Radiation Protection and Measurements (NCRP). Acting on the suggestion of the chairman of the ICRP, this group was formed in the United States in 1929.⁽⁵⁾ The work of this body is coordinated by the National Bureau of Standards. The recommendations of the Committee can be found in the National Bureau of Standards Handbooks. The first of these reports, dealing with X-ray protection, was published in 1931 as NBS Handbook 15.⁽⁶⁾ In 1935, this handbook was revised and issued in 1936 as NBS Handbook 20.⁽⁷⁾ This report contained a recommendation of a tolerance exposure level of 0.1 R/day. Handbook 23,⁽⁸⁾ dealing with radium protection, was issued in 1938. The work of the NCRP was disrupted by the advent of the war. However, its recommendations as outlined in Handbooks 20 and 23 served as the basis for protection practices during the days of the Manhattan project. Many members of the NCRP were engaged in this program and were helpful in seeing that protection standards prevailed.

From the standpoint of protection problems, it is hard to believe the dramatic impact that the war years produced. Of course, most of this effect can be traced to the development of the atomic bomb. Before the war, most of the problems concerned rather low-energy X rays. Now, not only were there these to treat, but also other types of radiation with a wide range of energies. Added to this was the large increase of workers in the radiation field. Also, many new techniques and operations were born which could lead to serious problems. Internal exposure became a topic of real concern. New units would be needed to define the dose contributed by radiation other than X rays. Large amounts of waste were now produced and methods of disposal would have to be worked out. With reactors in use, not only the workers but also others not connected with the work would have to be considered. The scope of the radiation field had enlarged to an undreamed of extent!

The NCRP met in 1946 to reorganize. At this time a number of subcommittees were formed to deal with the problems more effectively. A review of new data led to certain changes and additions to the old recommendations. By 1948, the reports of the subcommittees were ready. These recommendations formed the bulk of subject matter for NBS Handbook 59.⁽⁹⁾

4. Permissible Dose

Of prime importance was the adoption of the term permissible dose. Data from biological studies showed that one could not assume that all effects had a threshold dose. In the case of gene damage, effects could be expected at very low doses. The term tolerance dose seemed to suggest that this was a dose below which no effects would occur. The NCRP felt that a change in terminology was warranted to get away from this idea. Since some effects could be expected even from very low doses, the NCRP felt it was prudent to point this out and suggest lower limits of exposure. The figure of 0.3 R/week was recommended.

The ICRP then met in 1950 and based its new recommendations on the large body of data which had been collected by the NCRP. The ICRP also lowered the exposure level to 0.3 R/week. Their report⁽¹⁰⁾ gave recommendations for maximum permissible concentrations for a few radio-isotopes. In addition, the Commission set up a number of committees to deal with specific areas of protection problems.

At both of these meetings, the subject of genetic effects was discussed. Since data was lacking on key points no firm decisions could be reached.

5. Total Accumulated Dose

The report⁽¹¹⁾ issued by the ICRP after its 1953 meeting contained much more matter than any previous report. This was due to the fact that a number of the committees made very detailed recommendations. The Commission met again in 1956. At this time the recommendation was made that the maximum permissible exposure level be lowered to 0.1 R/week. At this meeting, concern was expressed over long-term exposure at low dose rates, as well as genetic effects. Again, these were areas for which data could not be gathered except through long-term studies. Also, large numbers of subjects would be required, since the damage would be nonspecific. Thus, the damage due to radiation would show up only on a statistical basis. For these reasons, the level was lowered and also the total accumulated dose was restricted. The latter results if one accepts the recommendation that the total accumulated dose D be governed by $D = 5(N-18)$ rem, where N is the age in years.

Because of the growth of the field and the uses of radiation which might involve exposure to large populations, it was felt that this matter should be treated. On the basis of recommendations made by the Genetics Committee of the National Academy of Sciences,⁽¹²⁾ the ICRP has suggested limits for population groups also.

The recommendations which were worked out at this meeting have been published⁽¹³⁻¹⁶⁾ and have served as a basis for many of the present limits. However, work continues in the field, and from time to time new reviews will be made and perhaps the limits will be changed again. At any rate, one should not be disturbed by the fact that the limits always seem to be reduced. The basic premise which is the guideline in all protection work is to see that exposures are kept at the lowest practical levels. Regardless of what has been the permitted levels, protection efforts to date have resulted in an average exposure well below these levels.⁽²⁾

Down through the years since the discovery of radiation, one can see the care and concern with which the problem of radiation protection has been approached. Back in the early days, the main problem was the gross somatic effects. Now, the main concern has switched from these blatant

effects to the more subtle effects of radiation. As knowledge has been gained, it has become quite evident that more knowledge is needed.

In any case, the quest for knowledge in this field has not suffered and more and more groups have joined in the search. In addition to the work of the ICRP, NCRP, and ICRU, in recent years the National Academy of Sciences-National Research Council has undertaken the study of biological effects.^(12, 18) This group consists of about 140 scientists throughout the country. The reports issued by this body are in summary form and the group functions as an advisory body. Its purpose is to supply technical information from which regulations can be developed. On a world-wide basis, the United Nations has established a Scientific Committee. Their reports^(19, 20) on the effects of atomic radiation have helped greatly to supply much needed background knowledge.

6. Federal Radiation Council

Because of the scope of the nuclear energy field in this country, the Federal Radiation Council (FRC) was formed in 1959 (Public Law 86-373). This body acts to advise the President concerning radiation matters and to provide guidance for all Federal agencies in setting standards and in working with the States.⁽²¹⁾ The Secretary of Health, Education, and Welfare is the chairman of this group. Up to this time, the Council has issued three staff reports.⁽²²⁻²⁴⁾

7. Radiation Protection Guides

The concepts of "Radiation Protection Guide (RPG)" and "Radioactivity Concentration Guide (RCG)" have been adopted by the Council for Federal use. The Radiation Protection Guide is defined as "the radiation dose which should not be exceeded without careful consideration of the reasons for doing so; every effort should be made to encourage the maintenance of radiation doses as far below this guide as practicable."⁽²²⁾ The import of this definition is to focus on the concept of an acceptable risk. The FRC feels that the term "maximum permissible dose" is often misunderstood. In order to convey the meaning that there can be no level of exposure without proper thought about the reasons for permitting the exposure, the Council decided on the new term.

The RCG is the concentration of radioactivity in the environment which is determined to result in whole body or organ doses equal to the Radiation Protection Guide. This term is advocated instead of the ICRP term "maximum permissible concentration."

The RPG values recommended by the FRC appear in the Council's first report.⁽²²⁾ These values are intended as interim levels for normal peacetime operations:

RADIATION PROTECTION GUIDES*

<u>Occupational Exposure</u>	<u>Condition</u>	<u>Dose (rem)</u>
1. Whole body, head and trunk, blood-forming organs, gonads, lens of the eye	Accumulated dose 13 weeks	5(N-18); N > 18 years 3
2. Skin and thyroid	Year 13 weeks	30 10
3. Hands and forearms, feet, and ankles	Year 13 weeks	75 25
4. Bone	Body burden	0.1 $\mu\text{gm Ra}^{226}$ or its equivalent
5. Other organs	Year 13 weeks	15 5
<u>Population</u>		
1. Individual	Year	0.5 (whole body)
2. Average	30 years	5 (gonads)

*Adapted from Ref. 22.

In the second report,⁽²³⁾ the FRC provides guidance concerning radioactive substances which may be deposited in the body. The recommendations are designed to limit exposure of members of population groups from radioactivity in the environment. Although most of the recommendations are quite general, specific guidance is provided for exposure of population groups to Ra^{226} , I^{131} , Sr^{89} , and Sr^{90} . However, no recommendations are made concerning Radioactivity Concentration Guides (RCG). The matter is left to the Federal agencies to develop the concentration values needed for control actions.

8. General Philosophy of the Federal Policy on Radiation Matters

These two reports form the basic general philosophy of the Federal policy on radiation matters. Each Federal agency has the responsibility to determine specific regulations in its area of jurisdiction. In some cases, these guides may be exceeded but "... only after the Federal agency having jurisdiction over the matter has carefully considered the reason for doing so in light of the recommendations in this staff report." (22-23)

B. Regulating Agencies

So far, our attention has been directed to those groups which supply recommendations for exposure levels and safe practices. The rest of this section will be concerned with the organizations which are charged

with developing regulations. Of prime interest will be those groups which regulate radiation matters in this country.

Under the Atomic Energy Act of 1954, the United States Atomic Energy Commission (AEC) was given the responsibility of regulating the atomic-energy industry.⁽¹⁷⁾ In this respect, the Act authorizes the AEC to set up a licensing program to be augmented by whatever rules or regulations are deemed appropriate. The bases for these rules are: to protect the public health and safety, and provide for national defense and security. Under this mandate, the AEC is concerned with the development of regulatory safety standards.

In regard to the licensing program, the AEC has set forth its regulations in the FEDERAL REGISTER. These are contained under Title 10, Code of Federal Regulations. The parts of Title 10 which deal with health and safety are: 2, 7, 8, 20, 25, 30, 40, 50, 55, and 70. Part 20, Standards for Protection against Radiation, deals specifically with the regulations for control of radiation hazards by the licensee.

In addition to this, the AEC has issued regulations which pertain to its own activities as well as to those of contractors not subject to licensing. These appear in the AEC MANUAL, Chapter 0524 - Standards for Radiation Protection. The standards are based upon those which apply to licensees and those recommended by the FRC. The AEC defines contractor as "an organization whose activities are conducted under a contract containing the standard safety, health and fire protection clause AECPR 9-7.5004-8."

As part of its duties, the AEC is charged with the task of seeing that these measures prevail. This aspect requires inspection and review in order to assure this. This function is carried out by AEC personnel at regular intervals. Their job is to make the inspections and report their findings. In the event that a failure to comply is noted, the licensee or contractor is required to correct this.

Many of the states have taken up the task of setting up their own safety standards. In this respect, the AEC has been directed to assist the states to assure that the state and Commission programs are compatible. An outline of programs which have been set up can be found in Ref. 17.

Safety in the shipment of radioactive substances is regulated by five agencies. These are the Interstate Commerce Commission, Coast Guard, Federal Aviation Agency, Post Office Department, and the AEC. Because of the large increase in shipments in recent years, a move to coordinate their activities is under way. In 1957, the Interagency Committee on Transportation of Radioactive Materials was formed.⁽²⁵⁾

This group has members from each of the five agencies and also from the Bureau of Explosives of the Association of American Railroads. The purpose of this body is to assure that consistent and uniform regulations are issued. At present, this group is engaged in a review and revision of the present transportation rules. Reference 17 contains excerpts which cover the salient points of these rules.

This concludes the brief outline of those groups whose function it is to regulate. From the dynamic nature of the field of atomic energy, one can expect that many new problems will arise. For this reason, no attempt has been made to discuss any of the present regulations in detail. As new problems arise, new rules must be worked out. Thus, as in the case of exposure limits, changes will occur. However, these changes will reflect a conscious effort to attain added safety in the field.

BIBLIOGRAPHY

1. SELECTED MATERIALS ON RADIATION PROTECTION CRITERIA AND STANDARDS: THEIR BASIS AND USE, Joint Committee on Atomic Energy - Congress of the United States, United States Government Printing Office, Washington 25, D. C. (May 1960).
2. Failla, G., Discussion, SELECTED MATERIALS ON RADIATION PROTECTION CRITERIA AND STANDARDS: THEIR BASIS AND USE, Joint Committee on Atomic Energy - Congress of the United States, United States Government Printing Office, Washington, D. C., 201-220 (May 1960).
3. Taylor, L. S., History of the International Commission on Radiological Units and Measurements (ICRU), HEALTH PHYSICS, 1:306-314 (1958).
4. Taylor, L. S., History of the International Commission on Radiological Protection (ICRP), HEALTH PHYSICS, 1:97-104 (1958).
5. Taylor, L. S., Brief History of the National Committee on Radiation Protection and Measurements (NCRP) Covering the Period 1929-1946, HEALTH PHYSICS, 1:3-10 (1958).
6. X-ray Protection, NATIONAL BUREAU OF STANDARDS HANDBOOK NO. 15 (Superseded by H 20).
7. X-ray Protection, NATIONAL BUREAU OF STANDARDS HANDBOOK NO. 20 (Superseded by H 41).
8. Radium Protection, NATIONAL BUREAU OF STANDARDS HANDBOOK NO. 23 (Superseded by H 54).

9. Permissible Dose from External Sources of Ionizing Radiation, NATIONAL BUREAU OF STANDARDS HANDBOOK NO. 59, United States Government Printing Office, Washington, D. C., September 1954.
10. Recommendations of the ICRP, NATIONAL BUREAU OF STANDARDS HANDBOOK NO. 47 (1951).
11. Recommendations of the ICRP, BRITISH JOURNAL OF RADIOLOGY, Suppl. 6 (1955).
12. THE BIOLOGICAL EFFECTS OF ATOMIC RADIATION, SUMMARY REPORTS, National Academy of Sciences - National Research Council, Washington (1956).
13. RECOMMENDATIONS OF THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION (Adopted September 9, 1958), London: Pergamon Press, 1959.
14. REPORT OF COMMITTEE II ON PERMISSIBLE DOSE FROM INTERNAL RADIATION (1959), London: Pergamon Press, 1959.
15. REPORT OF COMMITTEE III ON PROTECTION AGAINST X-RAYS UP TO ENERGIES OF 3 MeV AND BETA AND GAMMA RAYS FROM SEALED SOURCES (1960), London: Pergamon Press, 1960.
16. Recommendations of the ICRP, HEALTH PHYSICS, 2:1-20 (1959).
17. Taylor, L. S., and Harris, S. J., Exposure Standards and Radiation Protection Regulations; RADIATION HYGIENE HANDBOOK, edited by H. Blatz, New York: McGraw-Hill Book Company, Inc., 1959.
18. THE BIOLOGICAL EFFECTS OF ATOMIC RADIATION, SUMMARY REPORTS, National Academy of Sciences - National Research Council, Washington (1960).
19. REPORT OF THE UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Report to the General Assembly, Thirteenth Session, Supplement No. 17 (A/3838) 1958.
20. REPORT OF THE UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Seventeenth Session, Supplement No. 16 (A/5216), United Nations, New York 1962.
21. Chadwick, D. R., The Federal Radiation Council, HEALTH PHYSICS, 4:223-227 (1961).

22. BACKGROUND MATERIAL FOR THE DEVELOPMENT OF RADIATION PROTECTION STANDARDS, REPORT NO. 1, Staff Report of the Federal Radiation Council, United States Government Printing Office, Washington, D. C. May 1960.
23. BACKGROUND MATERIAL FOR THE DEVELOPMENT OF RADIATION PROTECTION STANDARDS, REPORT NO. 2, Staff Report of the Federal Radiation Council, United States Government Printing Office, Washington, D. C., September 1961.
24. HEALTH IMPLICATIONS OF FALLOUT FROM NUCLEAR WEAPONS TESTING THROUGH 1961, REPORT NO. 3, Report of the Federal Radiation Council, United States Government Printing Office, Washington, D. C., May 1962.
25. Rogers, L. R., The Development of Regulatory Standards, HEALTH PHYSICS, 9:1-5 (1963).

SECTION 9 - INTERNAL DOSE CALCULATIONS

A. Main Factors Affecting Dose Calculations

Much effort has been directed toward the problem of calculating the dose which a person receives as a result of internally deposited radionuclides. Among the many factors of main concern that enter into such a calculation are: the shape of the organ, the type of radiation, and the distribution of the deposit.

When the organ is in the shape of some standard geometric form, such as a sphere, the problem is somewhat lessened. If one can assume a uniform distribution of the deposit, the problem is further reduced. Then, if the range of the radiation is small compared with the size of the organ, the calculation of the absorbed dose becomes rather simple. One can then say that the energy emitted per unit time is equal to the energy absorbed per unit time. A knowledge of the concentration ($\mu\text{Ci/gm}$) and the energy of the emitted radiation is all that is needed to complete the calculation.

It is rare when one is able to apply this ideal model to an actual case. For this reason, one cannot make a truly precise calculation of the organ dose. The most vital factor which affects the accuracy of the result is the nonuniform distribution of the deposit. This is clearly seen in the case of alpha particles, whose range in tissue is so very short. In this respect, local "hot" spots may be the item of chief concern.

Of further interest is the problem of stating the dose in terms of a common scale. One can make a reasonable estimate of the absorbed dose in an organ. However, this merely specifies the energy absorbed and does not supply information about the effect of this dose on the organ. As it turns out, the absorbed dose which will give the same biologically equivalent dose is not the same for all types of radiation. Also, the lack of knowledge about the effects from low dose rates over long exposure times adds to the complex nature of the problem. This is pointed out in the recent report of the RBE Committee of the ICRP and ICRU.⁽⁴⁾

For the purposes of protection work, the present method of dose determination, proposed by the ICRP,⁽²⁾ provides an adequate estimate. One must keep in mind that as knowledge is gained, changes may be warranted. But the changes which may occur will only serve to reflect an attempt to obtain more precise results.

Given an amount of radionuclide in an organ of the body, a reasonable estimate of the organ dose can be made. A much greater problem is to estimate the amount of a radionuclide in the organ which results from intake into the body. This involves many biological factors, most of which

are not quantitatively known. Much work is needed in this area to improve the sparse body of knowledge which now exists. At present, many assumptions must be made in order to relate an environmental concentration to which a person is exposed to the amount of the radionuclide which he receives as a result. In view of this fact, the ICRP has always taken the stand that review and modification are needed on a continuing basis.

The method of calculation suggested by the ICRP⁽²⁾ is not very precise. It can only yield a good estimate in cases where there are sufficient data to back up the assumptions made. But the value of these calculations lies in the fact that they provide a guide for evaluation of a given environmental concentration. For the purposes of protection work, there is a very special need for these guidelines. For this reason, the methods of estimating maximum permissible concentration (MPC) values are very useful. On the other hand, one must be careful to note the conditions under which these values apply. In many cases, the use of these values without due regard to the assumptions which underlie them can lead to many pitfalls.

At the present time, the FRC is in the process of studying the problem of setting concentration limits.⁽³⁾ When completed, their recommendations will provide guidance for the Federal agencies. Until these guides are ready, it is useful to follow the recommendations of the ICRP. The rest of this section will be devoted to the methods used by the ICRP to calculate body burdens and MPC values. The current report of the ICRP which deals with internal dose can be found in Ref. 2. Much of the information which follows has been taken from this.

Let us focus on the nature of the problem. One is given a maximum permissible exposure level (MPL) for a certain organ. A knowledge of the mass and size of the organ is needed. For a certain radionuclide of known decay scheme and energy of emissions, one can calculate an organ burden. This will be the amount of the radionuclide in the organ which delivers the stated dose equivalent rate.

There are many ways in which the radionuclide may reach this organ. For most purposes, only two are of concern: inhalation and ingestion. The rest of the problem is then to relate the concentration in air and/or in water to the amount of the radionuclide which ends up in the organ. Many factors enter into this part of the problem.

From the standpoint of absorption into the body, the solubility of the radioactive substance in body fluids plays a major role. Whether a given substance will be soluble in body fluids or not is not easy to tell. However, as a mode of entry, ingestion is of concern only for soluble isotopes. Also, in the case of elements not required by the body, absorption by ingestion is poor.

In those cases where absorption does occur, one is concerned with the transfer of the substance to the organ, that is, only a fraction of the ingested material will go into the blood stream. Of this, only a fraction will go into the organ of concern. These fractions are often poorly known. To add to the problem, a portion of the substance may end up in a large number of organs. In this case, more than just a single organ will have to be considered.

In regard to inhalation, both soluble and insoluble matter must be treated. Another factor is particle size. The movement and retention of matter in the respiratory tract is a function of the particle size. Because of the complex relationship which exists, estimates of the transfer of radioactive matter by this mode of entry are poor. More data are needed before firm estimates can be made.

At present, the ICRP assumes a general model for use in the calculations. In this model, one-quarter of that which is inhaled is assumed to be immediately exhaled. One-half of the material is trapped by the upper respiratory tract and swallowed. As in the case of ingestion, a portion of the soluble matter will reach the organ of interest. The final one-quarter of the inhaled matter reaches the lungs. For a soluble substance, this will be taken up by the blood stream. A fraction of this matter is then transferred to the organ of concern.

For insoluble matter, the model is somewhat the same. Of that amount which reaches the lungs, it is assumed that half is worked up and swallowed within 24 hr. In this case then, five-eighths of the material is swallowed with no absorption. The one-eighth which remains in the lungs is slowly taken up by body fluids.

One can see that many factors are already present in an attempt to state intake parameters. But one must also treat the matter of elimination; although a substance is deposited in an organ, the time of stay in that organ is not indefinite. A fraction of this substance will be eliminated from the organ as time passes. Because the substance is radioactive, a certain fraction will decay per unit time, but also some of the material will be lost from the organ because of biological processes. Therefore, one must have knowledge of the net growth of a material in a given organ in order to proceed with the calculation. This means that one must have an elimination model for substances which are taken up by a given organ. The ICRP assumes that each organ can be treated as a separate compartment in which the elimination of a substance proceeds at a constant rate. This exponential model for elimination means that each organ has a half-life for biological elimination of a given substance. This is only very roughly true and in some cases a power series may be a better model.⁽²⁾ However, it provides a simple means of obtaining a rough estimate of the retention in a certain organ.

Given the intake rates for air and water, one can then solve for the MPC values. The calculation consists of finding the daily intake which would not result in an accumulation greater than the organ burden. This daily intake is assumed to continue over a period of 50 yr. This means that the dose equivalent rate in the organ is not greater than the allowed limit at the end of this period of exposure. In the ICRP method, then, the net amount of the radionuclide in the organ during the 50-yr exposure time approaches a limit which is equal to the organ burden. For many radioisotopes, the limit will be reached long before the end of the exposure time. When this occurs, an equilibrium state is reached in which there is no further increase in accumulation in the organ. For other radioisotopes, this equilibrium is not reached in the 50-yr period. In this case, continued exposure will result in further accumulation. The ICRP uses 50 yr since the working period for most people extends from age 18 to about age 65.

B. Standard Man

Throughout the above outline of the calculation of MPC values, and its basis, many factors which concern biological processes in man were mentioned. Because one individual varies from the next, these factors differ from one person to the next. In order to obtain agreement throughout the world in regard to the calculations of internal dose, certain values have been agreed upon for these factors. The ICRP achieves this goal by the use of the "standard man" concept. The standard man represents a set of internationally agreed-upon values for the many characteristics of man which are needed for internal dose calculations. These values can be found in Tables 6 through 11 of the ICRP report.⁽²⁾ Since the values are intended to represent an average adult, no account is taken of the differences which occur among individuals. The published values of MPC's^(2,5,6) have been calculated using the standard man data.

C. Critical Organ

One of the problems of setting MPC values is to pick out the organ which will be most affected by the given radionuclide. In many cases, a substance will be taken up by a number of body organs. The MPC value should be such that none of these organs accumulate an amount in excess of the organ burden. In dealing with this facet of the problem, the ICRP uses the concept of "critical organ:" the critical organ is taken to be that organ of the body whose damage by the radiation results in the greatest damage to the body.

A number of factors affect the choice of a critical organ. Some organs are more essential to the body than are others, and slight damage to these may affect the body more than heavy damage to some of the others. Since there is a difference in radiosensitivity among the organs, this must

also be taken into account. In cases such that little absorption into the body occurs, the greatest harm may be produced in the organ of entry. In most cases studied by the ICRP, the critical organ turns out to be that organ which receives the largest accumulation of the radioactive substance. However, the ICRP has included the MPC values for organs other than the critical organ in its report.⁽²⁾ The critical organ and MPC values for a given radionuclide appear in boldface type. These are the values which the ICRP recommends, and they apply primarily to occupational exposure. Occupational exposure refers to the exposure of a person who normally works with radiation as part of his job.

D. Effective Energy per Disintegration

In order to find the organ burden for a given radioisotope, a knowledge of the energy absorbed by the tissue is needed. In an assumed radioactive process, the energy which is released may be divided among a number of products. Some or all of these products will then lose energy in passing through tissue. The total energy ΣE which is absorbed per disintegration will be equal to the sum of the energy lost by each of the products. These products may include X , γ , α , β^- , β^+ , e^- , and recoil atoms. The actual purpose which are involved in a given process depend upon the decay scheme of the radioactive substance. Also, one may have to take into account decay chains, that is, if the parent decays to a radioactive daughter, then one must deal with the energy absorbed in tissue as a result of the radioactive process of the daughter product also.

Since one is more concerned with the effect of the absorbed energy on tissue, it is not enough merely to find the total energy absorbed. For this reason, the ICRP uses the concept of the "effective energy per disintegration." This includes the total energy absorbed modified by certain factors which take into account the effect on tissue. One can then express on a common scale, namely, the dose equivalent, the effectiveness of the energy absorbed in tissue.

The general form used by the ICRP to indicate the effective energy per disintegration is $\Sigma EF(RBE)n$. In any given calculation, some of the factors may be unity or may not even apply. Thus, simpler forms of the above may appear, i.e., ΣE .

The symbols under the summation sign are used to denote:

1. The Energy Absorbed per Disintegration, E

This quantity must be found for each of the products that result from the radioactive process. The total energy absorbed will then be the sum of the contributions from each of these products. One can find the amount contributed by each of the products with a knowledge of the decay scheme and the use of these equations:

a. β^- Radiation

$$E = 0.33 f E_{\max} \left(1 - \frac{Z^{1/2}}{50} \right) \left(1 + \frac{E_{\max}^{1/2}}{4} \right) \frac{\text{MeV}}{\text{dis}}.$$

In this equation, Z is the atomic number of the substance which emits the β^- and f is the fraction of the disintegrations in which a beta of maximum energy $E_{\max}$ (MeV) is emitted. In this case, it is assumed that all the beta energy is given up in the tissue.

b. β^+ Radiation

$$E = 0.33 f E_{\max} \left(1 + \frac{E_{\max}^{1/2}}{4} \right) + 2f(0.51)(1 - e^{-\sigma x}) \frac{\text{MeV}}{\text{dis}}.$$

The symbols f and $E_{\max}$ have the same meaning as for β^- . The positron gives up all its energy in the organ when it combines with an electron to form two 0.51-MeV gamma rays. The γ rays may then be absorbed in the organ and add to the total energy. The symbol x stands for the radius of the body organ (cm). Values for each organ can be found in Table 8 of Ref. 2. The symbol σ stands for $(\mu - \sigma_s)$ in tissue. Values for $\left(\frac{\mu - \sigma_s}{\rho} \right)$ versus the photon energy can be found in Ref. 8. Since ρ for tissue is about 1 gm/cc, $\left(\frac{\mu - \sigma_s}{\rho} \right)$ for tissue equals $(\mu - \sigma_s)$.

c. γ Rays

$$E = f E_{\gamma} (1 - e^{-\sigma x}) \frac{\text{MeV}}{\text{dis}},$$

where E_{γ} is the energy of the gamma and the other symbols have been discussed before. This expression assumes that no internal conversion is taking place.

d. Internal Conversion, e^-

$$E = f \left[(E_{\gamma} - \eta) \left(\frac{\alpha_k}{1 + \alpha_k} \right) + \eta \left(\frac{\alpha_k}{1 + \alpha_k} \right) (1 - e^{-\sigma x}) + E_{\gamma} \left(\frac{1 - e^{-\sigma x}}{1 + \alpha_k} \right) \right] \frac{\text{MeV}}{\text{dis}}.$$

When internal conversion must be taken into account, the above form, which gives the total absorbed energy for the entire process, should be used. Again, E_{γ} is the energy of the gammas which are emitted, in MeV. The term $(E_{\gamma} - \eta)$ gives the energy of the conversion electron, where η is the binding energy in MeV of the daughter product. The symbol α_k is the

conversion coefficient for the K shell. This is the ratio of K electrons to the "unconverted" gamma rays. If the process also occurs in the L and M shells, one must take this into account also. In cases for which α_L and α_M are unknown, the above can be used to yield a conservative value for the energy absorbed.

e. α Radiation

$$E = fE_{\alpha} \frac{\text{MeV}}{\text{dis}},$$

where E_{α} is the energy of the alpha in MeV.

f. Recoil Atoms

$$E = \frac{fE_{\alpha}m_{\alpha}}{m_r} \frac{\text{MeV}}{\text{dis}}.$$

In the process of alpha emission, the atom recoils and its energy must be included in the total for alpha decay. In the expression, m_{α} and m_r are the mass of the alpha and the recoil atom, respectively.

g. K and L Capture X Rays

$$E = f\eta(1 - e^{-\sigma x}) \frac{\text{MeV}}{\text{dis}}.$$

The above form assumes that only one X ray is emitted and that it has an energy equal to η . Actually, a number of X rays and/or Auger electrons may be emitted, but the total energy of all these products will equal to η . This simple form gives an answer which is conservative and saves undue effort.

2. Radioactive Chain Factor, F

The radioactive chain factor appears in the calculations when one must deal with a decay chain. The ICRP assumes that only the parent is present when the radionuclide is taken into the body. However, the effect of daughter products which are formed by decay in the organ are taken into account through the use of the factor F, which is defined as the ratio at time t of the number of disintegrations per unit time of the daughter atoms to the number of disintegrations per unit time of parent atoms in the critical organ. The product of F_i and the absorbed energy of the i^{th} daughter gives the weighted contribution of the i^{th} daughter. The sum of the contributions from the parent and all the daughters will then give the equivalent energy absorbed if the decay chain were to act as a single radionuclide.

The factor F_0 , where the subscript zero refers to the parent, equals one. The factor for the i^{th} daughter can be found from:

$$F_i = \frac{\prod_{j=1}^i T_j / T_j^r}{1 - e^{-\lambda_0 t}} \sum_{n=0}^i \frac{T_n^i (1 - e^{-\lambda_n t})}{\prod_{\substack{p=0 \\ p \neq n}}^i (T_n - T_p)}$$

In this equation, λ_0 is the total decay constant of the parent (see page 137) and λ_i is the total decay constant of the i^{th} daughter. The time t is taken as 50 yr, and T_i is the total half-life of the i^{th} daughter. The symbol T_i^r stands for the radioactive half-life of the i^{th} daughter.

Table 5a of the ICRP report⁽²⁾ contains a listing of the effective energies for many decay chains. Values for the factor F_i are also contained in this table for each daughter nuclide in the chain.

3. Relative Biological Effectiveness, RBE

It was pointed out in Section 4 that the ICRU wishes to restrict the use of the term RBE to radiobiology only. In regard to protection purposes, the term "quality factor (QF)" is suggested to replace the former term RBE.

As used in experimental work, the RBE is often defined as the inverse ratio of the absorbed doses of two kinds of radiation that produce the same effect. Other factors which affect the response must be the same or the results will not give a measure of the RBE. Thus, in order to talk about an RBE one must define exposure conditions as well as the effect studied.

For protection work, the dose levels which are used are well below those employed in experiments. Also, more than one effect must be taken into account for different types of radiation. Over and above this, one cannot control the many factors which may affect the exposure. For this reason, a distinction should be made between RBE and the factor used in protection work.

The ICRP endorses the use of the term quality factor for protection purposes.⁽⁴⁾ This factor is applied without regard to exposure factors, and for all biological effects. The values of QF which are to be used are the same as those formerly termed RBE values. They are not

intended to imply what the concept RBE conveys. They do provide a gross quality correction for the type of radiation so that one may use a common scale for all radiation. For internal dose considerations, the QF values are:

$$\begin{aligned} \text{QF} &= 1 && \text{for } \beta^-, \beta^+, e^-, \text{X and } \gamma \text{ rays;} \\ &= 1.7 && \text{for } \beta^-, \beta^+, e^- \text{ with } E_{\text{max}} \leq 0.03 \text{ MeV;} \\ &= 10 && \text{for } \alpha; \\ &= 20 && \text{for recoil atoms.} \end{aligned}$$

4. Relative Damage Factor, n

In the case of bone-seeking radionuclides, the amount of the nuclide permitted in the bone is based upon a direct comparison with Ra^{226} . However, some radionuclides appear to produce more damage to the bone than an equal dose of Ra^{226} . To account somewhat for this fact, the ICRP recommends the use of the relative damage factor, n . This factor applies only in the case of bone-seeking nuclides. Subject to the stated conditions, the value of n is:

$$\begin{aligned} n &= 1, && \text{if (a) the parent of the chain is an isotope of radium, or} \\ &&& \text{(b) the energy component originates as X or } \gamma \text{ radiation;} \\ &= 5, && \text{for all other cases.} \end{aligned}$$

Table 5 of the ICRP report⁽²⁾ contains a listing of the value of $\Sigma \text{EF(RBE)}n$ for many of the radionuclides. The sample calculations at the end of this section contain examples of the use of the factors to find $\Sigma \text{EF(RBE)}n$.

E. Other Factors

Table 12 of the ICRP report⁽²⁾ contains much other information needed to make the calculations. Although many sources were reviewed for needed data, little was learned concerning these factors. For this reason, many approximations have been made for the distribution fractions and biological half-lives found in this table.

Values of the effective or total half-life are also listed in Table 12. Since the process of decay and elimination are both exponential, each will have a certain value for the decay constant. The total fraction lost by the organ per unit time, λ , will be the sum of the two individual decay constants:

$$\lambda = \lambda_r + \lambda_b.$$

From the definition of half-life,

$$\frac{0.693}{T} = \frac{0.693}{T_r} + \frac{0.693}{T_b}$$

From this, we derive

$$T = \frac{T_b T_r}{T_b + T_r},$$

where the subscripts b and r refer to biological and radioactive, respectively. The total or effective half-life, T, is then the time for one-half of the nuclide to be removed from the organ. If the half-life for either process is very large compared with that of the other, T approaches the value of the smaller of the two as an upper limit. In most cases, the radioactive half-life is well known, but the biological half-life is often poorly known. This can be traced to the lack of continuous exposure data. Thus, many of the values have been extrapolated from single dose studies and may not represent too well the situation for the continuous case.

F. Maximum Permissible Body Burden

Early in this section, the term organ burden was mentioned as the amount in the organ which would deliver the stated dose equivalent rate. Since a nuclide may be deposited in a number of organs, the amount in the total body becomes important. The ICRP uses the term "Maximum Permissible Body Burden" (MPBB) and the letter q to denote the amount of a radionuclide, in μCi , allowed in the total body. The body burden q is the amount in the total body such that the allowed amount in the critical organ is not exceeded. In all cases, the organ burden for a given nuclide will be some fraction of the body burden q. The value of this fraction will depend upon how the nuclide is distributed in the body. If the substance is taken up by a number of organs, each will have its own fraction (f_2) of the body burden. For the organ which is taken as the critical organ for the assumed nuclide, the product qf_2 gives the organ burden. For the other organs which receive the radionuclide, the product qf_2 for these organs results in less than the organ burden.

There are two criteria which the ICRP uses to determine body burdens:

- (1) for bone-seeking radionuclides - the body burden is estimated by a direct comparison with Ra^{226} and its daughter products. However, the radionuclide in question must emit significant amounts of particulate radiation. In the event it does not, then it falls under the second criterion.

- (2) for all other radionuclides - the body burden is set to limit the weekly dose equivalent received by a given organ. These limits are:

- (a) 0.1 rem/wk to the gonads and total body;
- (b) 0.6 rem/wk to the skin and thyroid;
- (c) 0.3 rem/wk to all other soft tissue.

One should note from the first criterion that a bone-seeker which emits only X and γ rays is not compared with Ra^{226} . Inasmuch as X and γ rays have a much greater range in tissue than do particles, the nearby tissue will be exposed almost to the same extent as bone. Consequently, the body burden should be limited by part c above.

G. Body Burden Based on Radium

The body burden q for Ra^{226} has been set at $0.1 \mu\text{g}$ ($\sim 0.1 \mu\text{Ci}$),⁽¹¹⁾ as based upon direct observations of humans.⁽¹²⁾ Down through the years, the best source of data about internally deposited radionuclides has been the studies concerning radium deposition in humans. This wide experience has led to the adoption of radium as the standard for the chronic effects of bone-seekers on humans.

Chronic exposure data furnishes the value of q directly. However, these data are lacking for most radionuclides, so that a comparison method is used in many cases to find q . In this method, the energy given up in bone by the radionuclide is set equal to the energy given up by $0.1 \mu\text{Ci}$ of Ra^{226} and its daughter products. The relative damage factor n is used to modify the energy term of the given radionuclide. This allows for the greater effectiveness of some radionuclides as compared with radium.

For a given radionuclide X, one can write

$$[q f_2 \Sigma \text{EF}(\text{RBE})n]_X = [q f_2 \Sigma \text{EF}(\text{RBE})n]_{\text{Ra}}$$

In this equation, $q = 0.1 \mu\text{Ci}$, $f_2 = 0.99$, and $\Sigma \text{EF}(\text{RBE})n = 110 \text{ MeV/dis}$ for radium. When the symbols are replaced by these values, q is given in μCi by

$$q_X = \frac{11}{[f_2 \Sigma \text{EF}(\text{RBE})n]_X}$$

The average absorbed dose rate to the bone from $0.1 \mu\text{Ci}$ of Ra^{226} and its daughters turns out to be 0.06 rad/wk . In terms of the dose equivalent rate, the figure is 0.56 rem/wk . Although this dose equivalent rate is higher than that allowed for most other organs of the body, medical data have shown that $0.1 \mu\text{Ci}$ of radium is a safe burden.⁽¹²⁾

H. Body Burden Based on Dose Equivalent Rate

The lack of data for other than bone-seeking radionuclides precludes the use of a standard for these cases. One must calculate the body burden based upon the equivalent dose rate in the critical organ.

For a given organ of mass m (in gm), the permissible dose equivalent rate will be R rem/wk. If one permits a body burden of q (in μCi) in the total body, then qf_2 is the amount in the assumed organ. The amount allowed in the organ will be that which gives R rem/wk, or

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{qf_2(\mu\text{Ci}) 2.22 \times 10^6}{m(\text{gm})} \left(\frac{\text{dis}}{m\mu\text{Ci}}\right) \Sigma E(\text{RBE}) \left(\frac{\text{MeV}}{\text{dis}}\right) 1.6 \times 10^{-6} \left(\frac{\text{ergs}}{\text{MeV}}\right)$$

$$1.008 \times 10^4 \left(\frac{m}{\text{wk}}\right) \frac{1 \text{ rad}}{100(\text{ergs/gm})} = \frac{358 qf_2 \Sigma E(\text{RBE})}{m}$$

One then solves for q , which gives

$$q = \frac{2.8 \times 10^{-3} \text{ mR}}{f_2 \Sigma E(\text{RBE})} \quad (\text{in } \mu\text{Ci}).$$

The value of R for a given organ was discussed on page 139. The fraction f_2 and the mass m are given in Table 12 of the ICRP report.⁽²⁾

I. MPC in Air and Water - Single Radionuclide

Let us consider for the moment a concentration C_i ($\mu\text{Ci/cc}$) of a radionuclide in air or water. In this case, the subscript i denotes either method of intake. Assume that C_i remains constant during the time a person is exposed to the concentration. The average breathing and water intake rates (cc/day) of the standard man are also assumed to be constant. This means that the rate of uptake P_i ($\mu\text{Ci/day}$) of the radionuclide in a given organ will be constant.

If there were no elimination from the organ, the amount in the organ would increase at a constant rate. However, elimination does occur and the fraction eliminated per unit time is given by $\lambda (= 0.693/T)$ for the given organ. The rate of elimination will not be a constant. It will depend upon the amount of the radionuclide which is in the organ at any given time.

The rate of change of the amount in the organ $\left[\frac{d(q'f_2)}{dt}\right]$ will be given by

$$\frac{d(q'f_2)}{dt} = P_i - \lambda(q'f_2),$$

where $q'f_2$ stands for the amount in the organ at any time t . This differential equation has the solution

$$q'f_2 = \frac{P_i}{\lambda} (1 - e^{-\lambda t}),$$

for the condition that $q'f_2 = 0$ when $t = 0$. This equation relates the amount in the organ, $q'f_2$, to the rate of uptake, P_i , at any time t . If q' is set equal to the body burden q for the given organ, then qf_2 will be the allowed amount in the organ. One can then solve for the allowed rate of uptake, P_i , which would result in an organ burden qf_2 at any time t :

$$P_i = \frac{\lambda qf_2}{1 - e^{-\lambda t}} = \frac{0.693 qf_2}{T(1 - e^{-0.693t/T})}$$

Let S_i stand for the rate of intake (cc/day) of either air or water. Also, let f_i be the fraction of that inhaled or ingested which gets to the organ of reference. Then, the rate of uptake P_i ($\mu\text{Ci/day}$) is given by

$$P_i = C_i(\mu\text{Ci/cc}) S_i(\text{cc/day}) f_i(\text{fraction arriving at organ}).$$

The allowed rate of uptake for the given method of intake results when C_i is replaced by $(\text{MPC})_i$. From the two expressions for P_i , one arrives at

$$(\text{MPC})_i = \frac{0.693 qf_2}{S_i f_i T(1 - e^{-0.693t/T})} \frac{\mu\text{Ci}}{\text{cc}},$$

where t and T are in days.

One can then solve this equation for the Maximum Permissible Concentration in air and/or in water. The value of the factor S_i depends on the method of intake and the type of exposure. For the case of continuous exposure, $S_a = 2 \times 10^7 \text{ cc/day}$ and $S_w = 2.2 \times 10^3 \text{ cc/day}$, where the subscripts a and w denote inhalation and ingestion, respectively. For occupational exposure (40 hr/wk), $S_a = 6.9 \times 10^6 \text{ cc/day}$ and $S_w = 750 \text{ cc/day}$. The MPC values based on a 40-hr/wk exposure become

$$(\text{MPC})_a = \frac{10^{-7} qf_2}{f_a T(1 - e^{-0.693t/T})} \mu\text{Ci/cc};$$

$$(\text{MPC})_w = \frac{9.2 \times 10^{-4} qf_2}{f_w T(1 - e^{-0.693t/T})} \mu\text{Ci/cc},$$

in which

f_a = the fraction of inhaled radionuclide reaching the organ of reference and

f_w = fraction of that taken into the body by ingestion which is retained in the critical organ.

Values for both the fractions, f_a and f_w , can be found in Table 12 of the ICRP report.(2)

In the above expressions, the ICRP has set $t = 50$ yr in order to obtain the recommended values of $(MPC)_a$ and $(MPC)_w$. It should be noted that this assumes that the worker is exposed to a constant concentration of the radionuclide throughout the 50-yr period in order to reach the organ burden. For short-term exposures, the concentrations can be many times the MPC values without creating a serious hazard.

J. MPC Formulae for a Radioactive Series

If the radionuclide is the parent of a decay chain, one must consider the contribution of the daughter products. In this case, the dose equivalent rate R rem/wk will be given by

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{358}{m} \sum_{i=0}^k (qf_2)_i E_i (\text{RBE})_i n_i,$$

where the sum is taken over the k members of the chain. This may be written as

$$R = \frac{358}{m} (qf_2)_0 \sum_{i=0}^k \frac{(qf_2)_i}{(qf_2)_0} E_i (\text{RBE})_i n_i.$$

Since

$$F_i = \frac{(qf_2)_i}{(qf_2)_0},$$

we may rewrite this expression as

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{358}{m} (qf_2)_0 \sum_{i=0}^k E_i F_i (\text{RBE})_i n_i.$$

The body burden q_0 for the parent is then given by

$$q_0 = \frac{2.8 \times 10^{-3} \text{ mR}}{(f_2)_0 \sum_{i=0}^k E_i F_i (\text{RBE})_i n_i} \mu\text{Ci.}$$

The weighted energy sum in this expression allows one to treat the whole chain as a single radionuclide. The product $(qf_2)_0$ may then be replaced in the general MPC formula by the above value. The MPC values for occupational exposure are then

$$(\text{MPC})_a = \frac{10^{-7} qf_2}{f_a T(1 - e^{-0.693t/T})} = \frac{2.8 \times 10^{-10} \text{ mR}}{f_a T_0(1 - e^{-0.693t/T_0}) \sum_{i=0}^k E_i F_i (\text{RBE})_i n_i} \frac{\mu\text{Ci}}{\text{cc}}$$

and

$$(\text{MPC})_w = \frac{9.2 \times 10^{-4} qf_2}{f_w T(1 - e^{-0.693t/T})} = \frac{2.58 \times 10^{-6} \text{ mR}}{f_w T_0(1 - e^{-0.693t/T_0}) \sum_{i=0}^k E_i F_i (\text{RBE})_i n_i} \frac{\mu\text{Ci}}{\text{cc}}.$$

In these relationships, T_0 is the effective half-life for the parent of the decay chain. If bone is the critical organ, $R = 0.56 \text{ rem/wk}$ when particulate radiation is involved. For other organs, R has the values discussed on page 139. For continuous exposure, replace 2.8×10^{-10} and 2.58×10^{-6} by 9.6×10^{-11} and 8.8×10^{-7} , respectively. An example of the use of the above formulae can be found at the end of this section.

K. MPC of Noble Gases - Submersion Exposure

The problem of setting MPC values for inert gases involves external rather than internal exposure. These gases are poorly absorbed by the body, so that only a small portion of the gas in a cloud of large volume is held in the body. One is then concerned with the exposure of a person surrounded by a semispherical infinite cloud of radioactive gas. The radius of the cloud is then so large that one can assume that the energy emitted per unit volume equals the energy absorbed per unit volume. One can further assume that only one-half of the total solid angle contributes to the irradiation. For occupational exposure, the time that the worker is exposed is taken as 40 hr/wk for 50 out of 52 wk. The dose equivalent rate $R(\text{rem/wk})$ in tissue will be given by the expression

$$R\left(\frac{\text{rem}}{\text{wk}}\right) = \frac{1}{2}\left(\frac{S}{\rho}\right)_t C\left(\frac{\mu\text{Ci}}{\text{cc}}\right) \frac{3.7 \times 10^4}{\rho_a(\text{gm/cc})} \left(\frac{\text{dps}}{\mu\text{Ci}}\right) \Sigma E(\text{RBE}) \left(\frac{\text{MeV}}{\text{dis}}\right) 1.6 \times 10^{-6} \left(\frac{\text{ergs}}{\text{MeV}}\right)$$

$$1.385 \times 10^5 \left(\frac{\text{sec}}{\text{wk}}\right) \frac{1 \text{ rad}}{100(\text{ergs/gm})} = 41 \left(\frac{S}{\rho}\right)_t \frac{C \Sigma E(\text{RBE})}{\rho_a},$$

where $C(\mu\text{Ci/cc})$ is the concentration of the radionuclide, $\rho_a(\text{gm/cc})$ is the density of air, and $(S/\rho)_t$ is the relative mass stopping power of tissue with respect to air. If R is set equal to the allowed dose equivalent rate, then C can be replaced by $(\text{MPC})_a$:

$$(\text{MPC})_a = \frac{\rho_a R}{41(S/\rho)_t \Sigma E(\text{RBE})} \frac{\mu\text{Ci}}{\text{cc}}.$$

In the case of a γ or high-energy β ($E_{\text{max}} \geq 0.1 \text{ MeV}$) emitter, the total body will receive the exposure. Then, $R = 0.1 \text{ rem/wk}$, and for β and γ , $\text{QF(RBE)} = 1$. The value of $(S/\rho)_t$ can be taken as 1.13 and ρ_a as 0.0012 gm/cc . Using these values, one gets

$$(\text{MPC})_a = \frac{(1.2 \times 10^{-3})(0.1)}{41(1.13) \Sigma E} = \frac{2.6 \times 10^{-6}}{\Sigma E} \frac{\mu\text{Ci}}{\text{cc}}.$$

If low-energy radiation is involved, then skin becomes the organ of concern and

$$(\text{MPC})_a = \frac{1.56 \times 10^{-5}}{\Sigma E} \frac{\mu\text{Ci}}{\text{cc}}.$$

One should note that since this is an external source of exposure, the total dose equivalent varies directly with the exposure time. If the individual is exposed continuously to the radioactive cloud, replace 2.6×10^{-6} and 1.56×10^{-5} by the values 5.9×10^{-7} and 3.5×10^{-6} , respectively. From the protection standpoint, one would be concerned with concentrations many times these MPC values, even for short term exposures.

L. MPC Values for a Mixture of Radionuclides

Suppose that the air or water contains a mixture of radionuclides. Assume that both the identity and concentration of each of the members is known. Let C_i represent the concentration of the i^{th} member of permissible concentration $(\text{MPC})_i$. Then, if n radionuclides are present and the following inequality holds:

$$\frac{C_1}{(\text{MPC})_1} + \frac{C_2}{(\text{MPC})_2} \dots + \frac{C_n}{(\text{MPC})_n} \leq 1,$$

the mixture can be taken as within the limits.

For the cases in which either the identity or the concentration of one or all the members is not known, one may refer to the AEC Manual, Chapter 0524 - Standards for Radiation Protection. Annex 1 contains the rules which apply in these cases.

M. Dose Equivalent from Single Intake

At times one is faced with the problem of a single intake of a radionuclide rather than with long-term exposure. Two facets of the problem must be taken into account. First, for a large intake in a short time, a good portion of the substance may just pass through the body without absorption. For this condition, a portion of the gastrointestinal (GI) tract may receive the most exposure. This part of the problem is treated somewhat in the ICRP report⁽²⁾ and also in Ref. 9 and 10. Also treated is the problem of continuous exposure when the GI tract is the critical organ. This occurs most often when the inhaled or ingested substance is insoluble.

The other part of the problem concerns the portion of the radioactive material which is absorbed. A certain amount of the substance then ends up in some of the body organs. In the case of acute exposure, the radionuclide may be distributed much differently than in the case of long-term exposure, that is, one may find larger amounts of a bone-seeking radionuclide in other organs than in bone following an acute intake. This occurs because of the difference in distribution and retention. For a single intake, a larger fraction of the substance may be taken up by soft tissue than by bone. However, this matter may be quickly eliminated from soft tissue, whereas the portion in bone may be very slowly eliminated. Over a long period of time the amount in the bone will build up. This makes the bone the organ of concern for long-term exposure. For the acute intake case, then, one is interested in the short-term distribution in the body.

Let us assume that one has a measure of the amount $I_0(\mu\text{Ci})$ of a radionuclide in a given organ. If this is a single intake, our concern is only with this amount, that is, we ignore any past accumulation and assume no further accumulation. We will use the compartment model to describe the elimination of this product. The amount $I(\mu\text{Ci})$ which is present at any time t after the intake is completed will be given by

$$I(\mu\text{Ci}) = I_0 e^{-\lambda t}.$$

The dose equivalent rate in the given organ will be given by

$$\frac{d \text{ DE } \left(\frac{\text{rem}}{\text{day}} \right)}{dt} = \frac{I(\mu\text{Ci}) 2.22 \times 10^6 \left(\frac{\text{dis}}{\text{m-}\mu\text{Ci}} \right) \Sigma E(\text{RBE}) n \left(\frac{\text{MeV}}{\text{dis}} \right) 1.6 \times 10^{-6} \left(\frac{\text{ergs}}{\text{MeV}} \right)}{m(\text{gm})}$$

$$1.44 \times 10^3 \left(\frac{\text{m}}{\text{day}} \right) \frac{1 \text{ rad}}{100 (\text{erg/gm})} = \frac{51.15 I \Sigma E(\text{RBE}) n}{m}$$

The dose equivalent rate at any time t in terms of the initial amount I_0 is given by

$$\frac{d \text{ DE }}{dt} \left(\frac{\text{rem}}{\text{day}} \right) = \frac{51.15 I_0 \Sigma E(\text{RBE})n}{m} e^{-\lambda t}.$$

The total dose equivalent DE can be found by integrating the above expression with respect to t :

$$\text{DE} = \frac{74 I_0 T \Sigma E(\text{RBE})n}{m} (1 - e^{-0.693t/T}) \text{ rem.}$$

This equation gives the total dose equivalent DE delivered in the time t .

This completes the present discussion on internal dose. For a more thorough treatment of the problem one may refer to the ICRP report⁽²⁾ and the many sources of reference contained therein.

In an attempt to illustrate the use of the formulas presented and discussed in this section, a number of example calculations are included.

N. Sample Calculations of $(\text{MPC})_a$ and $(\text{MPC})_w$

(1) Calculate $(\text{MPC})_a$ and $(\text{MPC})_w$ for ${}^{170}\text{Tm}$ for bone as the critical organ. Referring to page 746, Ref. 14, one obtains the decay scheme for Tm^{170} :

The electron capture may be ignored in this case. One can write:

$$\Sigma E F(\text{RBE})n = \sum_{i,j}^n f_{\gamma_i} E_{\gamma_i} (1 - e^{-\sigma_i x}) F_i(\text{RBE})_i n_i + f_{\beta_j^-} E_{\beta_j^-} (0.33) \left(1 - \frac{Z^{1/2}}{50} \right) \left(1 + \frac{E_{\beta_j^-}^{1/2}}{4} \right) F_j(\text{RBE})_j n_j,$$

where $F_i = F_j = 1$; $(RBE)_\gamma = (RBE)_{\beta^-} = 1$; $n_\gamma = 1$; $n_{\beta^-} = 5$; $Z = 69$; $x = 5$ cm (from Table 8 - Ref. 2); and σ_i is obtained from Ref. 8. Then,

$$\Sigma EF(RBE)n = (0.24)(0.084)[1 - e^{-0.0265(5)}] (1)(1)(1) + (0.76)(0.968)(0.33)$$

$$\left(1 - \frac{\sqrt{69}}{50}\right) \left(1 + \frac{\sqrt{0.968}}{4}\right) (1)(1)(5) + (0.24)(0.884)(0.33)$$

$$\left(1 - \frac{\sqrt{69}}{50}\right) \left(1 + \frac{\sqrt{0.884}}{4}\right) x (1)(1)(5)$$

$$= 0.002 + 1.3 + 0.4 \cong 1.7 \text{ MeV/dis.}$$

Since the critical organ is bone and there is particulate radiation involved, the body burden is based on a comparison with Ra^{226} :

$$q = \frac{11}{f_2 \Sigma EF(RBE)n}$$

or

$$qf_2 = \frac{11}{1.7} = 6.47 \text{ } \mu\text{Ci}$$

in the critical organ. Thus,

$$(\text{MPC})_a = \frac{10^{-7} qf_2}{Tf_a(1 - e^{-0.693t/T})}$$

for occupational exposure. From Table 12 of Ref. 2, $T = 113$ days, and $f_a = 0.16$ for bone as the critical organ. For $t = 50$ yr,

$$[1 - e^{-0.693t/113}] \cong 1.$$

Therefore,

$$(\text{MPC})_a = \frac{10^{-7}(6.47)}{113(0.16)(1)} \cong 4 \times 10^{-8} \text{ } \mu\text{Ci/cc.}$$

Similarly, $f_w = 6.5 \times 10^{-5}$ from Table 12, so that

$$(\text{MPC})_w = \frac{9.2 \times 10^{-4} qf_2}{Tf_w(1 - e^{-0.693t/T})} = \frac{9.2 \times 10^{-4}(6.47)}{113(6.5 \times 10^{-5})(1)} \cong 0.8 \text{ } \mu\text{Ci/cc.}$$

(2) Calculate the $(MPC)_a$ for $^{41}_{18}\text{Ar}$. The decay scheme appears on page 618 of Ref. 14:

Since argon is an inert gas, one can use the formula for submersion, in which the dose is based on external rather than internal exposure. For this case, one can write

$$\Sigma E = \sum_{i,j}^n f_{\gamma_i} E_{\gamma_i} [1 - e^{-\sigma_i x}] + f_{\beta_j^-} E_{\beta_j^-} (0.33) \left(1 - \frac{Z^{1/2}}{50}\right) \left(1 + \frac{E_{\beta_j^-}^{1/2}}{4}\right).$$

The total body becomes the critical organ and the gamma travels through air. One must use σ_i for air, and x is then the radius of the gas cloud. For a cloud of large volume, $[1 - e^{-\sigma_i x}] \cong 1$. Then,

$$\begin{aligned} \Sigma E &= (0.99)(1.29)(1) + (0.009)(2.48)(0.33) \left[1 - \frac{\sqrt{18}}{50}\right] \left[1 + \frac{\sqrt{2.48}}{4}\right] \\ &\quad + (0.99)(1.2)(0.33) \left[1 - \frac{\sqrt{18}}{50}\right] \left[1 + \frac{\sqrt{1.2}}{4}\right] \\ &= 1.28 + 0.01 + 0.46 \cong 1.8 \text{ MeV/dis.} \end{aligned}$$

Therefore,

$$(MPC)_a = \frac{2.6 \times 10^{-6}}{\Sigma E} = \frac{2.6 \times 10^{-6}}{1.8} = 1.4 \times 10^{-6} \mu\text{Ci/cc.}$$

(3) Find the occupational $(MPC)_a$ and $(MPC)_w$ for $^{147}_{60}\text{Nd}$ plus daughters considering bone as the critical organ. The decay chain is

The factor F for each of the products appears in the effective energy term. Our first step will be to determine this factor for the daughter products. The factor for the parent, F_0 , is taken as 1. The others are obtained from the expression

$$F_i = \frac{\prod_{j=1}^i T_j / T_j^r}{1 - e^{-\lambda_0 t}} \sum_{n=0}^i \frac{T_n^i (1 - e^{-\lambda_n t})}{\prod_{\substack{p=0 \\ p \neq n}}^i (T_n - T_p)}$$

Hence,

$$F_{Nd} = F_0 = 1.$$

$$F_{Pm} = F_1 = \frac{T_1}{T_1^r (1 - e^{-\lambda_0 t})} \left[\frac{T_0 (1 - e^{-\lambda_0 t})}{T_0 - T_1} + \frac{T_1 (1 - e^{-\lambda_1 t})}{T_1 - T_0} \right].$$

$$F_{Sm} = F_2 = \frac{T_1 T_2}{T_1^r T_2^r (1 - e^{-\lambda_0 t})} \left[\frac{T_0^2 (1 - e^{-\lambda_0 t})}{(T_0 - T_1)(T_0 - T_2)} + \frac{T_1^2 (1 - e^{-\lambda_1 t})}{(T_1 - T_0)(T_1 - T_2)} \right. \\ \left. + \frac{T_2^2 (1 - e^{-\lambda_2 t})}{(T_2 - T_0)(T_2 - T_1)} \right].$$

From Table 12 - Ref. 2,

$$T_0 = 11.2 \text{ d}, T_1 = 570 \text{ d}, T_2 = 1500 \text{ d},$$

$$T_0^r = 11.3 \text{ d}, T_1^r = 920 \text{ d}, T_2^r = 4.8 \times 10^{13} \text{ d}.$$

For $t = 50 \text{ yr}$, the terms $(1 - e^{-\lambda_n t}) \approx 1$, and

$$F_1 = \frac{T_1}{T_1^r} \left[\frac{T_0}{T_0 - T_1} + \frac{T_1}{T_1 - T_0} \right] = \frac{T_1}{T_1^r} = \frac{570}{920} = 0.62$$

$$F_2 = \frac{T_1 T_2}{T_1^r T_2^r} \left[\frac{T_0^2}{(T_0 - T_1)(T_0 - T_2)} + \frac{T_1^2}{(T_1 - T_0)(T_1 - T_2)} + \frac{T_2^2}{(T_2 - T_0)(T_2 - T_1)} \right]$$

$$F_2 = \frac{(570)(1500)}{(920)(4.8 \times 10^{13})} \left[\frac{(11.2)^2}{(11.2 - 570)(11.2 - 1500)} + \frac{(570)^2}{(570 - 11.2)(570 - 1500)} \right. \\ \left. + \frac{(1500)^2}{(1500 - 11.2)(1500 - 570)} \right]$$

$$= 1.94 \times 10^{-11} [0.00015 - 0.63 + 1.63] = 1.94 \times 10^{-11}.$$

The effective energy of the chain will be the sum of the terms for each member of the chain. Again, from Ref. 14 - page 726,

Other gammas are emitted but have not been included because their contribution is small. One then gets

$$\begin{aligned}
 [\Sigma EF(RBE)n]_{\text{Nd}} &= \sum_{i,j} f_{\gamma_i} E_{\gamma_i} (1 - e^{-\sigma_i x}) F_i(RBE)_i n_i + f_{\beta_j} E_{\beta_j} (0.33) \left(1 - \frac{Z^{1/2}}{50}\right) \\
 &\quad \left(1 + \frac{E_{\beta_j}^{1/2}}{4}\right) F_j(RBE)_j n_j. \\
 &= 0.85(0.091)(1 - e^{-(0.026)5})(1)(1)(1) + (0.13)(0.532) \\
 &\quad (1 - e^{-(0.032)5})(1)(1)(1) + (0.79)(0.81)(0.33) \left(1 - \frac{\sqrt{60}}{50}\right) \\
 &\quad \left(1 + \frac{\sqrt{0.81}}{4}\right)(1)(1)(5) + (0.02)(0.5)(0.33) \left(1 - \frac{\sqrt{60}}{50}\right) \left(1 + \frac{\sqrt{0.5}}{4}\right) \\
 &\quad (1)(1)(5) + (0.15)(0.38)(0.33) \left(1 - \frac{\sqrt{60}}{50}\right) \left(1 + \frac{\sqrt{0.38}}{4}\right)(1)(1)(5) \\
 &\quad + (0.04)(0.22)(0.33) \left(1 - \frac{\sqrt{60}}{50}\right) \left(1 + \frac{\sqrt{0.22}}{4}\right)(1)(1)(5).
 \end{aligned}$$

$$[\Sigma EF(RBE)n]_{Nd} = 0.01 + 0.01 + 1.09 + 0.02 + 0.09 + 0.01 = 1.23 \text{ MeV/dis.}$$

The decay scheme of Pm^{147} is found on page 727 of Ref. 14:

From this, one gets:

$$\begin{aligned} [\Sigma EF(RBE)n]_{Pm} &= f_{\beta^-} E_{\beta^-} (0.33) \left(1 - \frac{Z^{1/2}}{50}\right) \left(1 + \frac{E^{1/2}}{4}\right) F_{Pm}(RBE)_{\beta^-} n_{\beta^-} \\ &= (1)(0.223)(0.33) \left(1 - \frac{\sqrt{61}}{50}\right) \left(1 + \frac{\sqrt{0.223}}{4}\right) \\ &\quad (0.62)(1)(5) = 0.22 \text{ MeV/dis.} \end{aligned}$$

The decay of Sm^{147} is by alpha emission to stable Nd^{143} (p. 729 - Ref. 14):

The effective energy term will include the contribution from the recoil atom:

$$\begin{aligned} [\Sigma EF(RBE)n]_{Sm} &= f_{\alpha} E_{\alpha} F_{Sm}(RBE)_{\alpha} n_{\alpha} + f_{\alpha} E_{\alpha} \frac{m_{\alpha}}{m_r} F_{Sm}(RBE)_{r} n_r \\ &= (1)(2.2)(1.9 \times 10^{-11})(10)(5) + (1)(2.2) \frac{4}{143} \\ &\quad (1.9 \times 10^{-11})(20)(5) \\ &= 2.09 \times 10^{-9} + 0.12 \times 10^{-9} = 2.21 \times 10^{-9} \text{ MeV/dis.} \end{aligned}$$

The effective energy term for the chain then becomes

$$\begin{aligned} \sum_{i=0}^k E_i F_i(RBE)_{i n_i} &= [\Sigma EF(RBE)n]_{Nd} + [\Sigma EF(RBE)n]_{Pm} + [\Sigma EF(RBE)n]_{Sm} \\ &= 1.23 + 0.22 + 2.21 \times 10^{-9} = 1.45 \text{ MeV/dis.} \end{aligned}$$

The $(MPC)_a$ and $(MPC)_w$ for a decay chain are obtained from:

$$(MPC)_a = \frac{2.8 \times 10^{-10} \text{ mR}}{f_a T_0 (1 - e^{-0.693t/T_0}) \sum_{i=0}^k E_i F_i (RBE)_{ini}} \mu\text{Ci/cc},$$

and

$$(MPC)_w = \frac{2.58 \times 10^{-6} \text{ mR}}{f_w T_0 (1 - e^{-0.693t/T_0}) \sum_{i=0}^k E_i F_i (RBE)_{ini}} \mu\text{Ci/cc}.$$

The parameters in this case become:

$$\begin{aligned} \text{for bone as the critical organ} & \quad \left\{ \begin{array}{l} R = 0.56 \text{ rem/wk} \\ m = 7 \times 10^3 \text{ gm} \end{array} \right. \\ \text{with Nd as the parent of the chain} & \quad \left\{ \begin{array}{l} T_0 = 11.2 \text{ d} \\ f_a = 0.09 \\ f_w = 3.5 \times 10^{-5} \end{array} \right. \end{aligned}$$

One then solves the expressions with $t = 50$ yr. This gives $(1 - e^{-0.693t/T_0}) \cong 1$, and one arrives at

$$(MPC)_a = \frac{2.8 \times 10^{-10} (7 \times 10^3) (0.56)}{(0.09) (11.2) (1) (1.45)} \cong 8 \times 10^{-7} \mu\text{Ci/cc},$$

and

$$(MPC)_w = \frac{2.58 \times 10^{-6} (7 \times 10^3) (0.56)}{(3.5 \times 10^{-5}) (11.2) (1) (1.45)} \cong 20 \mu\text{Ci/cc}.$$

BIBLIOGRAPHY

1. Hine, G. J. and Brownell, G. L., RADIATION DOSIMETRY, New York: Academic Press Inc. (1956).
2. REPORT OF COMMITTEE II ON PERMISSIBLE DOSE FROM INTERNAL RADIATION (1959), London: Pergamon Press (1959).
3. BACKGROUND MATERIAL FOR THE DEVELOPMENT OF RADIATION PROTECTION STANDARDS, REPORT No. 1, Staff Report of the Federal Radiation Council, United States Government Printing Office, Washington, D. C., May 1960.

4. Report of the RBE Committee to the International Commission on Radiological Protection, HEALTH PHYSICS, 9:357-386 (1963).
5. Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959), HEALTH PHYSICS, 3:1 (1960).
6. Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, NATIONAL BUREAU OF STANDARDS HANDBOOK 69, United States Government Printing Office, Washington, D. C., June 1959.
7. Addendum 1 to National Bureau of Standards Handbook 69, HEALTH PHYSICS, 9:565-568 (1963).
8. Rockwell, T., REACTOR SHIELDING DESIGN MANUAL, Princeton: D. Van Nostrand Co., Inc. (1956).
9. Morgan, K. Z., Determination of Exposures, RADIATION HYGIENE HANDBOOK, edited by H. Blatz, New York: McGraw-Hill Book Company, Inc. (1959).
10. Morgan, K. Z., Dosimetry of Internal Radioactive Isotopes, SELECTED MATERIALS ON RADIATION PROTECTION CRITERIA AND STANDARDS: THEIR BASIS AND USE, Joint Committee on Atomic Energy - Congress of the United States, United States Government Printing Office, Washington, D. C., 317-337 (May 1960).
11. Safe Handling of Radioactive Luminous Compounds, NATIONAL BUREAU OF STANDARDS HANDBOOK 27, United States Government Printing Office, Washington, D. C.
12. Evans, R. D., and Dudley, R. A., The Radium-226 Standard (0.1 microgram) for Bone-seekers, SELECTED MATERIALS ON RADIATION PROTECTION CRITERIA AND STANDARDS: THEIR BASIS AND USE, Joint Committee on Atomic Energy - Congress of the United States, United States Government Printing Office, Washington, D. C., 437-441 (May 1960).
13. REPORT OF THE UNITED NATIONS SCIENTIFIC COMMITTEE ON THE EFFECTS OF ATOMIC RADIATION, Seventeenth Session, Supplement No. 16 (A/5216), United Nations, New York (1962).
14. Strominger, D., Hollander, J. M., and Seaborg, G. T., Table of Isotopes, REVIEWS OF MODERN PHYSICS, 30, No. 2, Part II: 585-904 (1958).