CHAPTER 9

RADIATION SAFETY

Two Lectures

D.A. Woods

PART A

IONISING RADIATIONS

by

D. A. Woods

1. IONISATION AND IONISING RADIATIONS

When nuclear radiation falls on an atom, there is a statistical probability that an electron will be removed from the atom, leaving a positive ion. The electron remains free for a very short time and usually attaches itself to another atom, forming a negative ion. This is called ionisation and an ion pair has been created. Radiations which produce ionisation are known as ionising radiations.

When ionising radiation falls on biological tissue, ionisation which can lead to biological injury takes place.

2. DOSE UNITS

In radiation dosimetry there are three different dose units, namely, exposure, absorbed dose, and dose equivalent.

Exposure is a measure of the amount of ionisation in air. The unit of exposure is the *roentgen* (R), which was originally defined as the amount of X or γ radiation required to produce one electrostatic unit of charge of either sign in air per cubic centimetre of air at standard temperature and pressure.

In more up-to-date units:

$$1 \text{ R} = 2.58 \times 10^{-4} \text{ C kg}^{-1}$$

The new SI unit of exposure is the *coulomb per kilogram*, a much larger unit than the roentgen.

Absorbed dose is a measure of the energy absorbed from ionising radiations per unit mass of the absorbing material. The unit of absorbed dose is the *rad*, which was originally defined as the amount of ionising radiation required to produce 100 ergs per gram in the absorbing medium:

$$1 \text{ rad} = 0.01 \text{ J kg}^{-1}$$

The new SI unit of absorbed dose is the *gray* (Gy), which is the amount of ionising radiation required to produce one joule per kilogram in the absorbing medium:

$$1 \text{ Gy} = 1 \text{ J kg}^{-1} = 100 \text{ rad}$$

In biological systems the same degree of damage is not necessarily produced by the *same* absorbed dose of *different types* of ionising radiation. To take account of this we use the dose equivalent unit:

dose equivalent = absorbed dose x quality factor (QF)

This was originally called the RBE (relative biological effectiveness) dose. The unit of dose equivalent is the *rem* (roentgen equivalent man), and the new SI unit is the *sievert* (Sv):

$$1 \text{ rem} = 1 \text{ rad} \times \text{QF}$$

$$1 \text{ Sv} = 1 \text{ Gy} \times \text{QF} = 100 \text{ rem} \quad .$$

Table 1 indicates the quality factors for various types of ionising radiation and table 2 summarises radiation dose units.

TABLE 1

QUALITY FACTORS

Type of Radiation	Quality Factor
beta	1
alpha	20
X-rays	1
γ-rays	1
thermal neutrons	3
fast neutrons	10

TABLE 2

RADIATION DOSE UNITS

Type	Old Unit	Symbol	New Unit	Symbol	Conversion
Exposure	roentgen	R	coulomb per kilogram	$C\ kg^{-1}$	$1\ R = 2.58 \times 10^{-4}\ C\ kg^{-1}$
Absorbed dose	rad	r	gray	Gy	1 Gy = 100 r
Dose	rem	rem	sievert	Sv	1 Sv = 100 rem

3. NATURAL BACKGROUND RADIATION

Everyone is exposed to natural sources of ionising radiation. This natural background radiation varies from place to place, depending on the radioactive content of the rocks and soils in the locality, the altitude, the latitude, the building materials, *etc.* Small amounts of natural radioactive material, mainly potassium-40, are incorporated within our bodies. The food we eat, the air we breathe, the water we drink all contain trace quantities of naturally occurring radioactive elements.

Table 3 indicates typical annual whole body doses caused by natural background radiation.

TABLE 3

TYPICAL ANNUAL WHOLE BODY DOSES
FROM NATURAL BACKGROUND RADIATION

Source	Annual Dose	
	(mSv)	(mrem)
Terrestrial radiation	0.5	50
Cosmic radiation	0.3	30
Internal radiation	0.2	20
Total	1.0	100

One millisievert per year (or 100 mrem per year) is an averaged world figure and is rounded so that it is easy to remember. Obviously the natural background varies from place to place. In two places, Espirito Santo State in Brazil and Kerala in Southern India, the natural background dose rate is as high as 20 mSv per year (2000 mrem per year); it is caused by beach sands (monazite) containing natural thorium.

4. ICRP DOSE LIMITS

The International Commission on Radiological Protection (ICRP) publishes safety recommendations periodically. Various countries may then adopt these recommendations into their own national legislation, usually in the form of a radioactive substances act. The most recent publication giving dose limits for radiation workers is ICRP 26 (adopted in 1977); before that, the recommendations of ICRP 9 (adopted in 1959) were used. Table 4 summarises the ICRP 9 dose limits and table 5 summarises the ICRP 26 dose limits.

TABLE 4

ICRP 9 DOSE LIMITS (pre 1977)

Organ or Tissue	Maximum Permissible Dose for Radiation Worker	Dose Limit for Individual Members of the Public
Gonads and bone marrow	5 rem/year 3 rem/quarter	0.5 rem/year
Skin and bone	30 rem/year 15 rem/quarter	3 rem/year
Thyroid	30 rem/year 15 rem/quarter	3 rem/year except for children < 16 years for whom 1.5 rem/year
Hands, forearms, ankles and feet	75 rem/year 40 rem/quarter	7.5 rem/year
Other single organs	15 rem/year 8 rem/quarter	1.5 rem/year

TABLE 5

ICRP 26 DOSE LIMITS (post 1977)

Organ or Tissue	Dose Equivalent Limit for Radiation Workers (mSv/year)	S = Stochastic NS = Non-stochastic
Whole body	50	S
Gonads	200	S
Breast	330	S
Red bone marrow	417	S
Lung	417	S
Thyroid	500	NS
Eyes	300	NS
Other single organs	500	NS
Skin	500	NS

In ICRP 26, stochastic and non-stochastic effects are defined as follows:

Stochastic effects : Probability of effect occurring, rather than its severity, is regarded as a function of dose, without threshold.

Non-stochastic effects : Severity of effect varies with the dose, and for which a threshold may occur.

5. DOSE RATE

The annual whole body dose equivalent limit for radiation workers is 5 rem or 50 mSv (unchanged from ICRP 9 to ICRP 26). Pro-rating this over 50 weeks gives 100 mrem/week or 1 mSv/week. Pro-rating this over a 40 h week gives 2.5 mrem h^{-1} or 25 μSv h^{-1}; *i.e.* a person working for 40 h per week for 50 weeks per year with a dose rate of 2.5 mrem h^{-1} or 25 μSv h^{-1} will receive the annual limit of 5 rem or 50 mSv.

Note: To convert mrem h^{-1} to μSv h^{-1} simply multiply by 10.

6. POTENTIAL HAZARDS OF IONISING RADIATIONS

Human senses cannot detect ionising radiations, therefore, we must rely on instruments capable of detecting them to give us warning of potential exposures.

Radiation injury to people can be classed in two main ways:

(a) *Somatic effects*, where the effects occur in the exposed individual, and

(b) *Genetic effects*, where the effects occur in the exposed individual's descendants.

Somatic effects can be subdivided into:

(a) *Acute effects*, which occur when a large exposure is received over a very short time. Here we must protect the worker from large, accidental exposures.

(b) *Late effects*, which can occur when low exposures are received continuously over a long period of time. Here we must keep the worker's radiation exposure within the acceptable dose limits set by ICRP.

When persons are exposed to ionising radiations which are outside the body, this is known as an *external radiation exposure*. When persons take radioactive material into their bodies (by inhalation, ingestion or absorption through the skin) this is known as an *internal radiation exposure*.

Radioactive sources may be sealed (*e.g.* γ radiography source) or unsealed (*e.g.* radioactive tracer). Unsealed radioactive sources may give rise to surface or airborne contamination. The working environment

must be monitored for external radiation, surface contamination and/or airborne contamination when these hazards are likely. The worker must be monitored for personal radiation exposures and personal contamination.

7. TYPES OF IONISING RADIATIONS

Alpha-particles are helium nuclei. They have a very short range in air and are easily shielded. They penetrate less than one tenth of a millimetre in human tissue, which is less than the thickness of the dead layer of skin, and therefore are not considered an external hazard. However, if alpha-emitting material is taken into the body, alphas can be considered a very significant internal hazard.

Beta-particles are electrons and have a greater penetrating ability than alphas. Their range varies with energy. To penetrate the outer layer of skin, a beta-particle must have an energy greater than 70 keV. The more energetic betas can travel a few millimetres in human tissue and, therefore, they represent an external hazard to skin or eye. They cannot penetrate the skin to the more sensitive internal organs and are easily shielded. However, if beta-emitting material is taken into the body, betas can present a significant internal hazard.

Gamma-rays and X-rays are highly penetrating electromagnetic radiations. They are very significant external radiation hazards. If gamma-emitting radioactive material is taken into the body it presents an internal radiation hazard, irradiating the whole body.

Neutrons are uncharged highly penetrating particles and represent a significant external hazard.

8. CONTROL OF INTERNAL RADIATION EXPOSURES

To prevent radioactive material entering the body the following general rules are applied:

(a) Provide proper and adequate containment for unsealed sources.

(b) Carry out regular monitoring for contamination.

(c) Use suitable protective clothing.

(d) Decontaminate immediately after spillages.

(e) Maintain good housekeeping.

(f) Do not smoke, eat, drink, use cosmetics, or pipette by mouth in potentially contaminated areas.

(g) Check yourself for personal contamination on leaving a potentially contaminated area.

9. CONTROL OF EXTERNAL RADIATION EXPOSURES

There are three fundamental rules to remember:

TIME : The less time spent in a radiation environment the smaller is the radiation exposure.

DISTANCE : The greater the distance from a source of radiation the smaller is the radiation exposure.

SHIELDING : If a suitable absorbing material is placed between you and the source of ionising radiation, your exposure is less.

The essential aspect of the TIME rule is to plan your work to avoid unnecessary exposure. If necessary, a dose rate measurement or estimate can be made and a time limitation set so that the worker receives no more than the acceptable dose for the particular operation in question:

$$\text{Time limit} = \frac{\text{Acceptable Dose}}{\text{Dose rate}}$$

Example: If the dose rate in an area is 10 mSv h^{-1} and the permitted dose per worker for the operation is 1000 μSv, then each worker can spend only

$\frac{1000}{10\ 000}$ hours or $\frac{1000}{10\ 000}$ x 60 minutes, *i.e.* 6 minutes in the area.

For DISTANCE, the inverse square law applies, *i.e.* for an isotropic point source of ionising radiation the dose rate at a given distance from the source is inversely proportional to the square of the distance. This may be expressed as

$$I_1 r_1^2 = I_2 r_2^2$$

where I_1 = dose rate at distance r_1 from the source,

and I_2 = dose rate at distance r_2 from the source.

$$\text{If } r_1 = 1 \text{ then } I_2 = \frac{I_1}{r_2^2}$$

Distance is a very effective protective measure.

Provision of proper SHIELDING enables individuals to work much closer to a source of ionising radiation, and for longer periods, than if no shielding were provided. For β radiation, only small thicknesses of low density material (*e.g.* Perspex or aluminium) are required, but for γ radiations larger thicknesses of dense material (*e.g.* lead or iron) are required.

10. PERSONAL DOSEMETERS

It is normally required by law that radiation workers be provided with personal dosemeters and that each individual's accumulated dose be entered in a personal dose record.

The most common personal dosemeter used is the UKAERE film badge (figure 1), which is normally worn on the chest. The UKAERE film badge consists of a hinged plastic cassette containing several filters and a Kodak radiation monitoring film. The film has a fast emulsion for measurement of low doses on the front side of the film base, and a slow emulsion for measurements of high doses on the reverse side. The two emulsions have significantly different melting points and the fast one can be removed from the film base, after processing, by immersion in water at 50°C and wiping with a tissue.

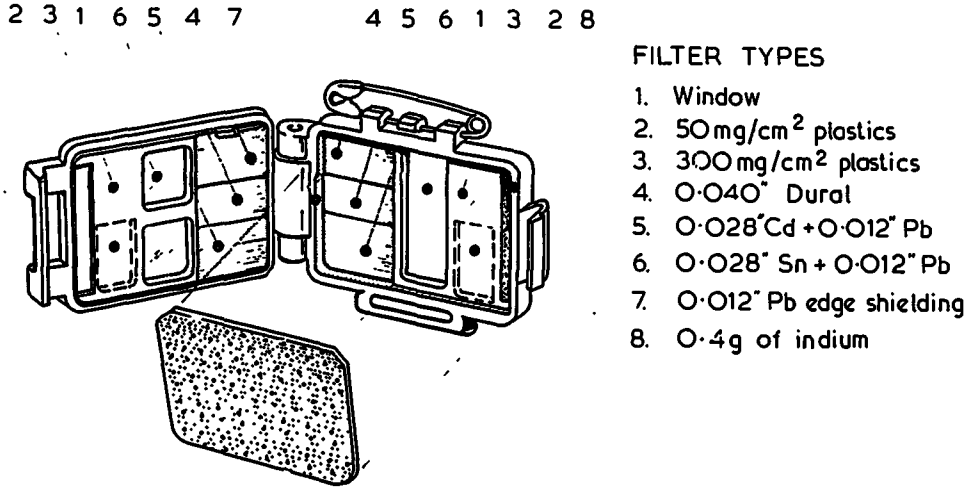

FIGURE 1

STANDARD UKAERE FILM BADGE

The film badge can be used to measure radiation exposures due to slow neutrons, β radiation and X- and γ-rays. The dose range is 10 mR to 1000 R.

The filter system consists of seven filters; an open window, a thin plastic filter, a thick plastic filter, a dural (aluminium and copper alloy) filter, a tin/lead filter, a cadmium/lead filter, and an indium strip.

The photographic film when exposed to ionising radiation appears black after processing. By comparing the amount of blackening (optical density) under the different filters and using calibration films we can calculate the amount and type of radiation to which the film badge has been exposed.

A film badge is also available for measuring fast neutrons. This consists of an aluminium cassette, a Kodak personnel neutron type 'A' film and a lead filter. The film has a very thick emulsion with a high hydrogen content and when exposed to fast neutrons, the neutrons collide with the hydrogen nuclei (protons), causing them to recoil. These 'knock-on' protons move through the emulsion in straight tracks producing developable photographic grains. When the film is processed, the tracks can be viewed and counted using a microscope. The number of tracks can be related to dose using a calibration curve of dose *v.* number of tracks.

The film has a sensitive range of 50 mrem to about 100 rem (above this tracks are too numerous to be counted), and is useful for neutrons in the energy range 600 keV to 10 MeV. If more than 5 rem gamma is also present, the film will be too black to be counted.

Another type of personal dosemeter which is becoming more popular, and in some countries has replaced the film badge, is the thermoluminescent dosemeter (TLD). Thermoluminescent dosemeters when heated give off light in proportion to the amount of ionising radiation they have received.

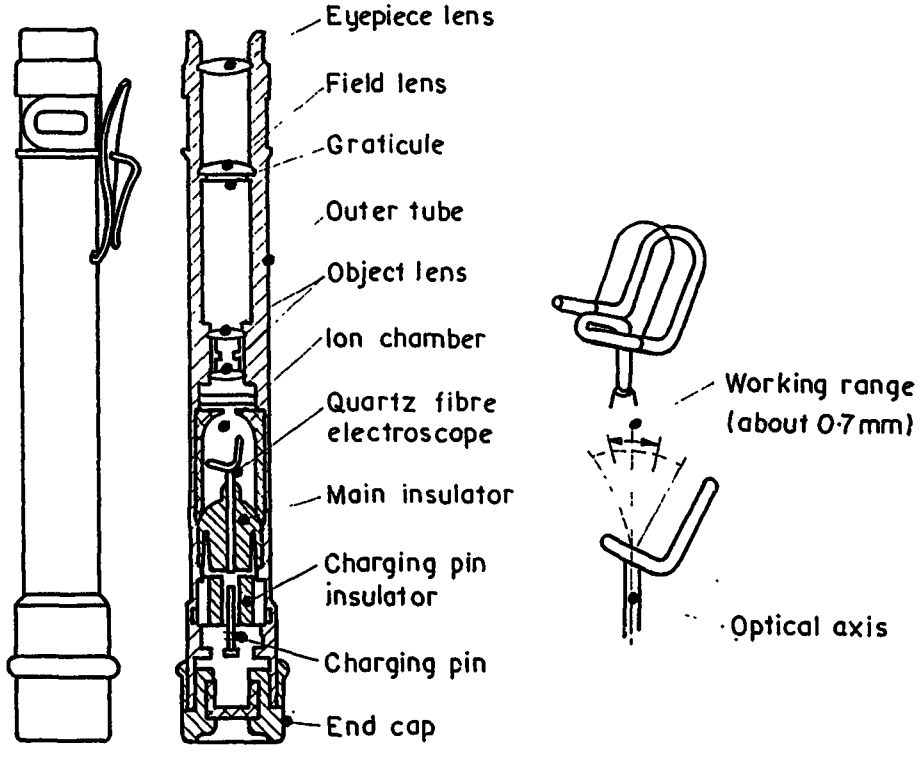

FIGURE 2

QUARTZ FIBRE ELECTROSCOPE

Thermoluminescent material is available in powder, chip and strip or disc form. Lithium fluoride and calcium fluoride are two common thermoluminescent materials. A TLD reader is used to heat the dosemeters and measure their light output. These dosemeters measure beta, X and gamma radiation, are physically small, and can be worn on the finger to measure finger doses.

A fourth type of personal dosemeter (figure 2) is the quartz fibre electroscope (QFE), which is similar in size to a fountain pen and is normally worn in a chest pocket. It has a small ionisation chamber which, when fully charged, reads zero. When ionising radiations enter the ionisation chamber, ions are produced, it discharges and a quartz fibre moves along a calibrated scale, which can be seen by holding the QFE up to the light and looking through a small microscope incorporated in it. It is designed to measure X or gamma radiation up to 500 mR.

11. THE AIMS OF RADIATION PROTECTION

All unnecessary personal radiation exposures should be avoided. Occupational exposures to ionising radiations should be kept as low as is reasonably achievable, social and economic factors being taken into account. The recommended dose equivalent limits should not be exceeded. Whenever sources of ionising radiations are used, the benefit accrued from that use must be greater than the risk associated with their use.

PART B

SOME HEALTH PHYSICS CONSIDERATIONS

by

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1. INTRODUCTION

When gamma cameras, thickness gauges, *etc.* are used, the sources are sealed and radioactive contamination is unlikely. The concepts of time, distance and shielding (discussed in the previous lecture) should be used to protect the worker from unnecessary radiation exposure.

When radioactive tracers or other unsealed radioactive material is handled, precautions must be taken, such as wearing protective clothing to avoid personal contamination. Where highly active stock solutions are concerned the concepts of time, distance and shielding also apply.

2. USEFUL FACTS, FORMULAE AND RULES OF THUMB

2.1 Alpha-particle Range

$$R\alpha = 0.56E \quad (E < 4 \text{ MeV})$$

$$R\alpha = 1.24E - 2.62 \quad (4 < E < 8 \text{ MeV})$$

where $R\alpha$ is the range in cm of air at 1 atm and 15°C, and E is the energy in MeV.

2.2 Beta-particle Range

For $0.01 \leqslant E \leqslant 2.5$ MeV

$$R = 412E^{1.265 - 0.0954 \ln E}$$

$$\ln E = 6.63 - 3.2376 \left[10.2146 - \ln R\right]^{\frac{1}{2}}$$

where R is the range in mg cm^{-2} and E is the maximum energy in MeV.

For $E \geqslant 2.5$ MeV

$R = 530E - 106$ where R and E are the same as above.

Sargent's rule ($E > 0.8$ MeV)

$R = 0.526E - 0.094$ where R is in g cm^{-2}.

Feather's rule ($E > 0.6$ MeV)

$R = 0.542E - 0.133$ where R is in g cm^{-2}.

2.3 Bremsstrahlung

Fraction (F) of beta energy converted to bremsstrahlung

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

where Z is the atomic number of the absorbing material and E_{max} is the maximum beta energy in MeV.

2.4 Radioactive Decay

$$\lambda T = 0.693$$

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

$$A = Ao\, e^{-\frac{0.693t}{T}}$$

$$A = \frac{Ao}{2^n}$$

where Ao is the initial activity (t = 0), A is the activity at time t, T is the half-life for the particular radionuclide, λ is the decay constant for the particular radionuclide, n is the number of half-lives, and e is the base of natural logs (2.718).

The decay constant λ represents the fraction of the total number of atoms in a radioactive source which decay per unit time. The activity of a radioactive source is reduced to less than 1 per cent of its original activity after 7 half-lives ($2^{-7} = 0.8\%$).

2.5 Specific Activity

The specific activity (SpA) of a radionuclide (disintegrations per unit time/unit mass) is calculated from the basic equation:

$$SpA = \lambda N = \frac{\ell n\ 2\ N}{\tau_{\frac{1}{2}}}$$

where N is the number of atoms per unit mass, and $\tau_{\frac{1}{2}}$ is the half-life.

By definition

$$N = \frac{6.0225 \times 10^{23}}{A}$$

where A is the mass number of the radionuclide and 1 Ci = 3.7×10^{10} disintegrations per second (dps).

Substituting

$$SpA = \frac{0.693N}{\tau_{\frac{1}{2}}} = \frac{0.693}{\tau_{\frac{1}{2}}} \times \frac{6.0225 \times 10^{23}}{A} \times \frac{1}{3.7 \times 10^{10}}\ Ci\ g^{-1}$$

$$= \frac{1.128 \times 10^{13}}{\tau_{\frac{1}{2}}\ A}\ Ci\ g^{-1},\ \text{where } \tau_{\frac{1}{2}} \text{ is in seconds}$$

Also

$$SpA = \frac{1.880 \times 10^{11}}{\tau_{\frac{1}{2}}\ A}\ Ci\ g^{-1},\ \text{where } \tau_{\frac{1}{2}} \text{ is in minutes}$$

$$SpA = \frac{3.134 \times 10^{9}}{\tau_{\frac{1}{2}}\ A}\ Ci\ g^{-1},\ \text{where } \tau_{\frac{1}{2}} \text{ is in hours}$$

$$SpA = \frac{1.306 \times 10^{8}}{\tau_{\frac{1}{2}}\ A}\ Ci\ g^{-1},\ \text{where } \tau_{\frac{1}{2}} \text{ is in days}$$

$$SpA = \frac{3.578 \times 10^5}{\tau_{\frac{1}{2}} A} \text{ Ci g}^{-1}, \text{ where } \tau_{\frac{1}{2}} \text{ is in years}$$

2.6 Gamma-ray Absorption

(i) Narrow beam

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

(ii) Broad beam

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

where Io is the intensity before absorption, I is the intensity after absorption, x is thickness of absorber (cm), μ is linear absorption coefficient (cm^{-1}), and B is build-up factor.

2.7 Half-value Layer

This is the thickness of a particular shield material which will reduce the intensity of the radiation by a factor of two:

$$I = \frac{Io}{2^n}$$

where Io is the intensity before absorption, I is the intensity after absorption, and n is the number of half-value layers.

2.8 Tenth-value Layer

This is the thickness of a particular shield material which will reduce the intensity of the radiation by a factor of ten:

$$I = \frac{Io}{10^n}$$

where I and Io are as above and n is the number of tenth-value layers.

2.9 Dose Rate from a Point Beta Source

$$D = \frac{10^6\, C\, N}{3d^2} \quad \text{(valid for betas > 0.5 MeV)}$$

where D is the dose rate in rad h^{-1}, C is the activity in Ci, N is the number of betas per disintegration, and d is the distance from source (cm).

2.10 Dose Rate from a Point Gamma Source

For a point source of C curies emitting one gamma per disintegration, the dose rate at d (m) is

$$D = \frac{0.55\, C\, E}{d^2} R\, h^{-1}$$

where E is the gamma energy in MeV (valid for 0.3 MeV < E < 3 MeV).

For more than one gamma per disintegration

$$D = \frac{0.55\ C\ \Sigma\ (fE)}{d^2}\ R\ h^{-1}$$

where the particular nuclide emits gammas of energy E_1 in f_1 % of its disintegrations, *etc.*

2.11 Specific Gamma-ray Constant

Γ = exposure rate at a given distance from an unshielded gamma source : activity.

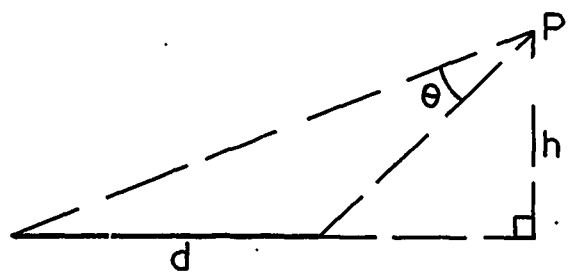
Units: $R\ h^{-1}\ Ci^{-1}$ at 1 m

$R\ h^{-1}\ mCi^{-1}$ at 1 cm

Dose rate from a point gamma source = $\frac{\Gamma\ A}{d^2}\ R\ h^{-1}$

where A is activity and d is distance.

line gamma source

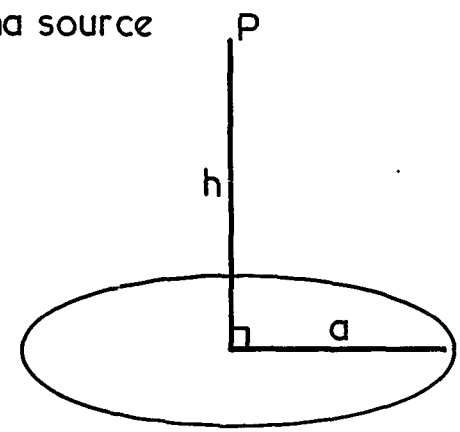


For a line gamma source

$$\gamma Dp = \frac{\Gamma\ A\ \theta}{dh}\ R\ h^{-1}$$

where A is in mCi, d and h in cm and θ in radians.

disc gamma source



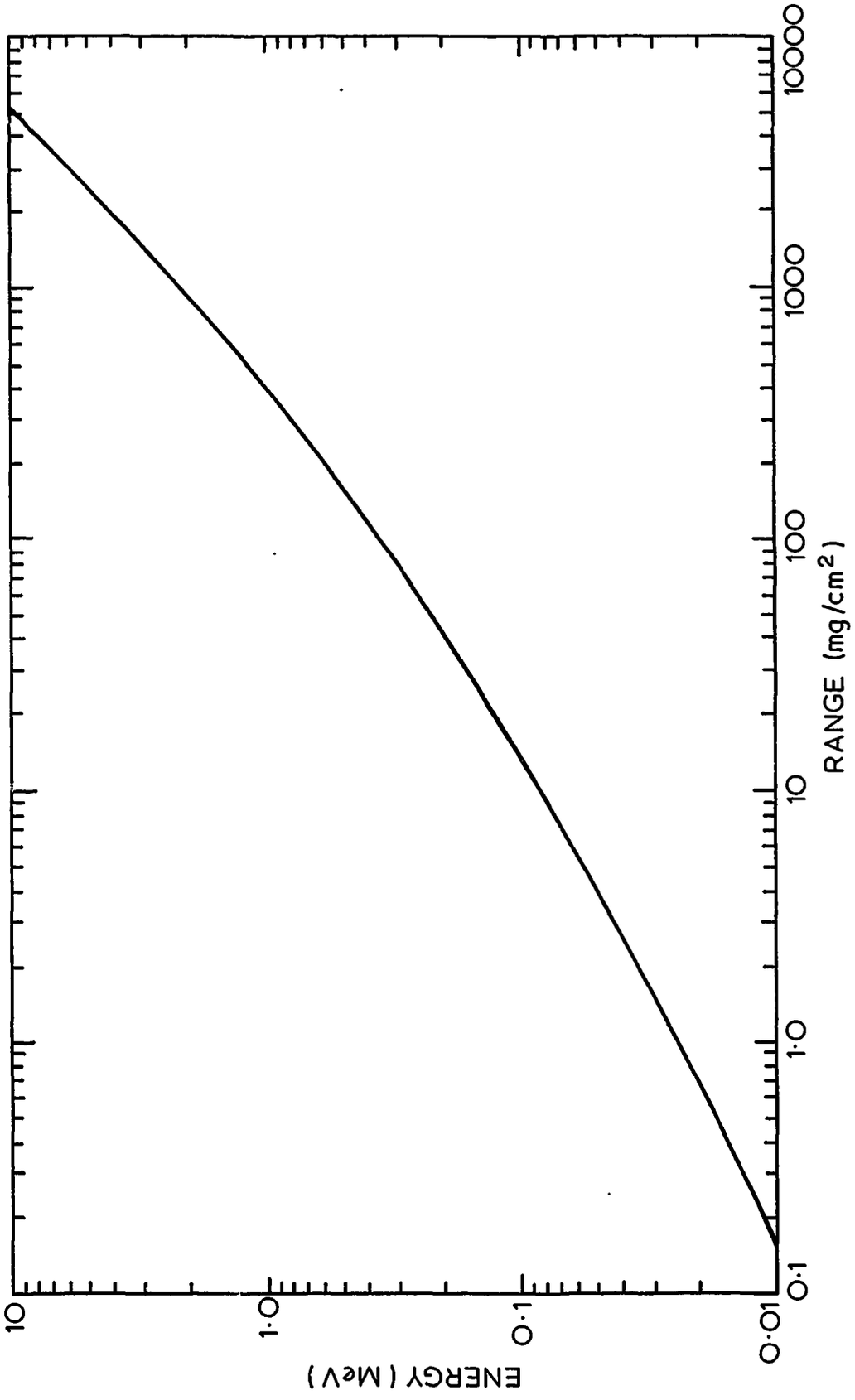
For a disc gamma source

$$\gamma Dp = \frac{\Gamma\ A}{a^2}\ \ell n\ \left(\frac{h^2 + a^2}{h^2}\right)\ R\ h^{-1}$$

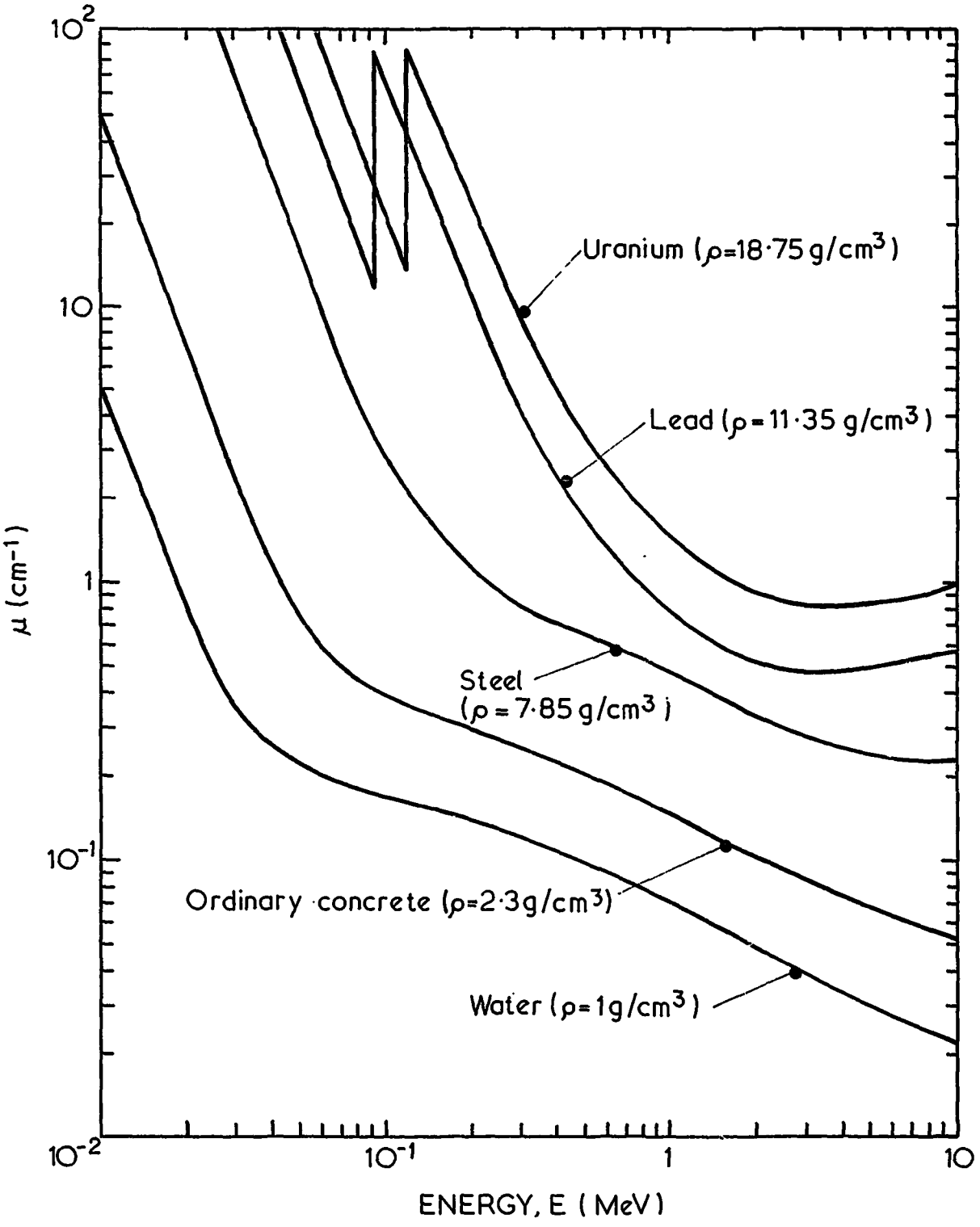
where A is in mCi, a and h in cm and Γ in $R\ h^{-1}$ from 1 mCi at 1 cm.

3. SOME USEFUL GRAPHS AND TABLES

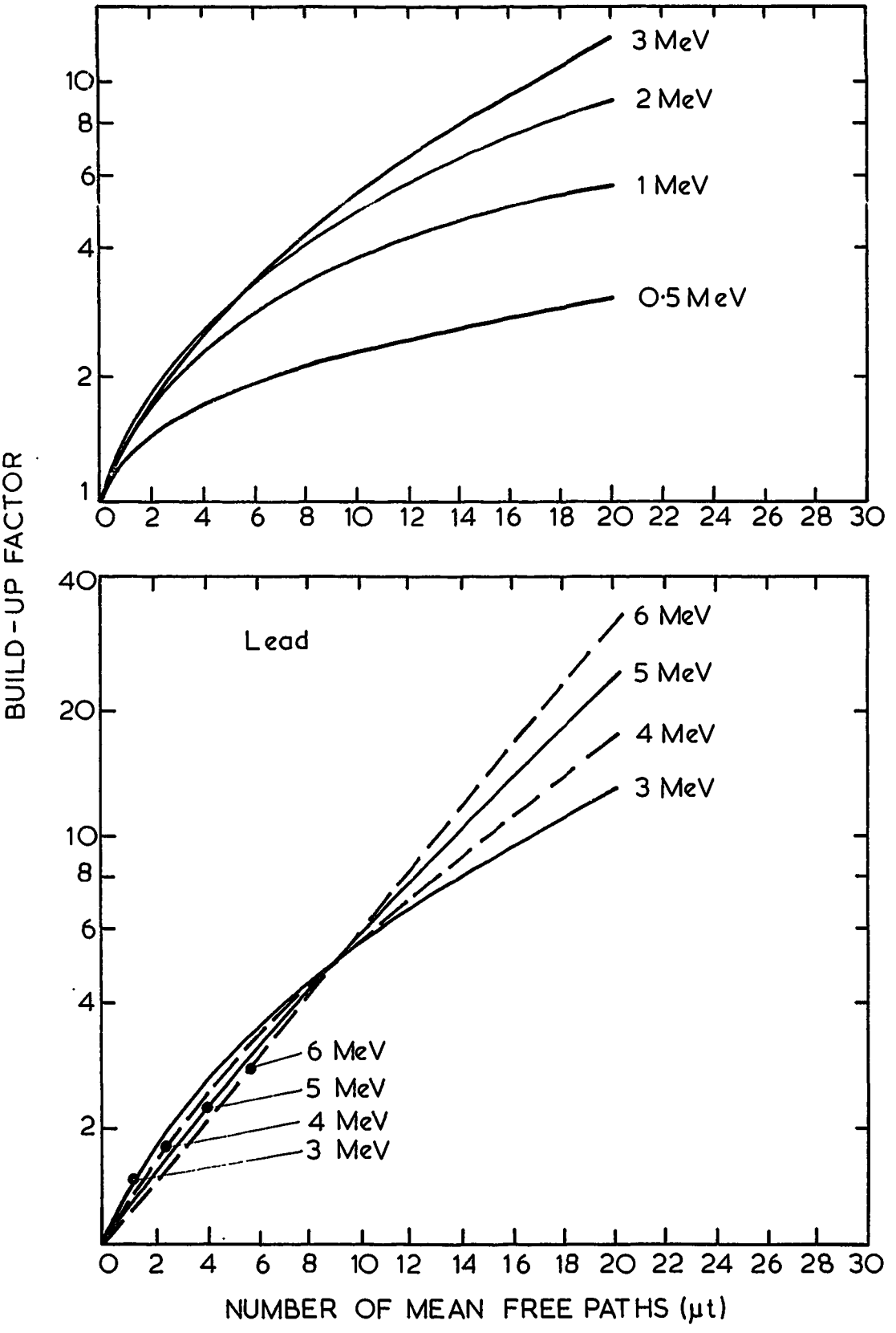
3.1 Beta-particle Range Energy Curve



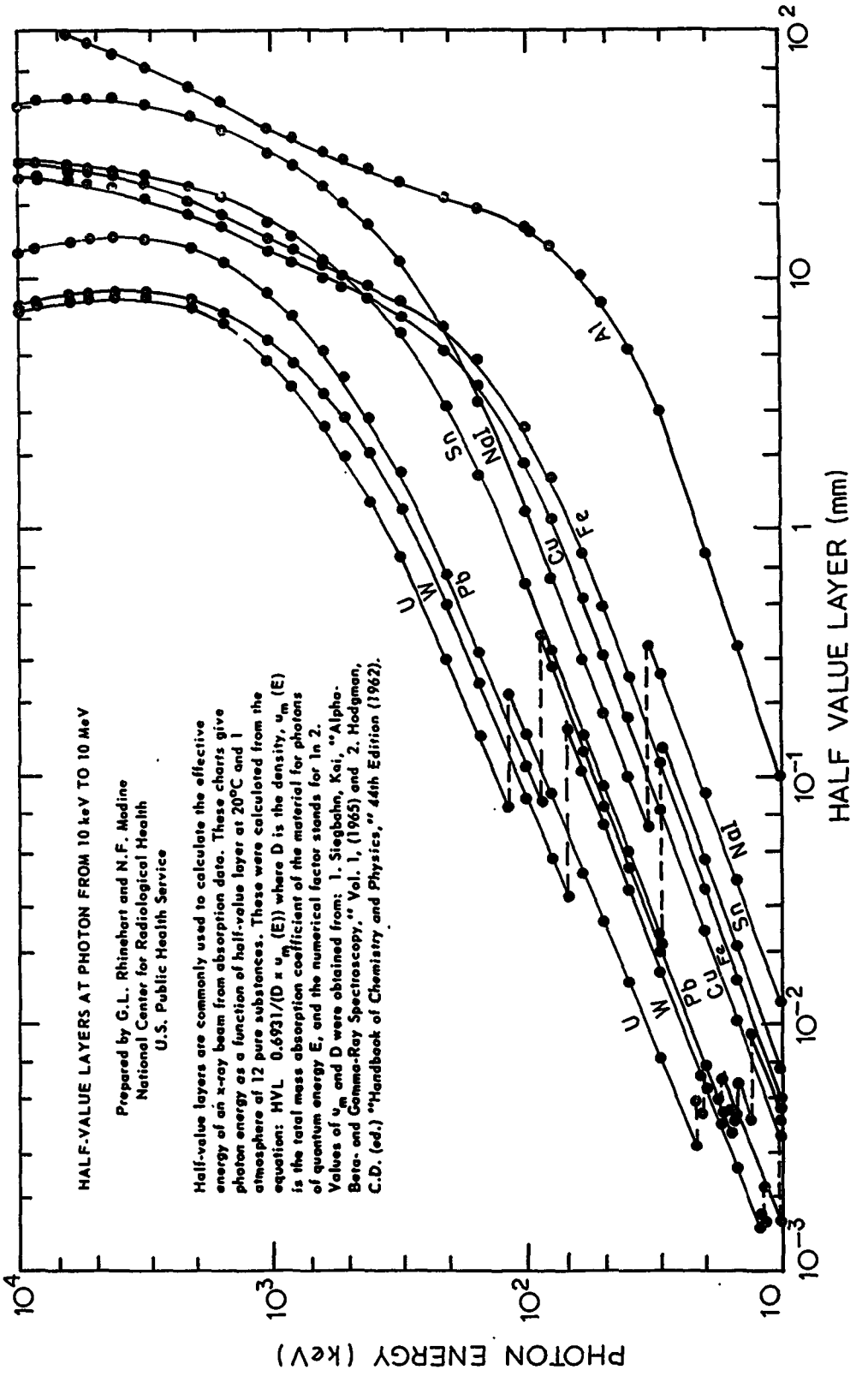
3.2 Gamma Linear Attenuation Coefficient *versus* Energy



3.3 Build-up Factor *versus* Number of Mean Free Paths



3.4 Half-value Layer *versus* Photon Energy



3.5 Specific Gamma-ray Constants, Gamma Energies and Half-lives for Some Selected Radionuclides

Nuclide	Symbol	Half-life	Principal Gamma Energies, MeV (% abundances)	Unshielded Dose Rate from 1 Ci at 1 m(R h^{-1})	Unshielded Dose Rate from 1 GBq at 1 m(μSv h^{-1})
Americium-241	$^{241}_{95}Am$	458 y	0.060 (36)	0.013	3.6
Barium-133	$^{133}_{56}Ba$	10.7 y	0.08 (36) 0.302 (14) 0.356 (69)	0.24	65
Barium-139	$^{139}_{56}Ba$	83.2 min	0.166 (22.6)	0.02	5.4
Bromine-82	$^{82}_{35}Br$	35.4 h	0.55 (72.5) 0.62 (40) 0.69 (28) 0.78 (83.2) 0.83 (24) 1.04 (28) 1.31 (27) 1.48 (17)	1.46	394
Caesium-134m	$^{134m}_{55}Cs$	2.9 h	0.031 (31) 0.127 (14)	0.02	5.4
Caesium-137	$^{137}_{55}Cs$	30 y	0.66 (84.6)	0.31	83.7
Chromium-51	$^{51}_{24}Cr$	27.8 d	0.32 (9.8)	0.016	4.3
Cobalt-57	$^{57}_{27}Co$	270 d	0.014 (9) 0.122 (87) 0.136 (11)	0.09	24
Cobalt-60	$^{60}_{27}Co$	5.26 y	1.17 (100) 1.33 (100)	1.33	359
Copper-64	$^{64}_{29}Cu$	12.9 h	0.51 (37) 0.008 (14)	0.12	32.4
Gadolinium-153	$^{153}_{64}Gd$	242 d	0.041 (92) 0.047 (18) 0.097 (30) 0.103 (23)	0.1	27
Gallium-67	$^{67}_{31}Ga$	78.1 h	0.09 (40) 0.18 (20) 0.30 (15)	0.1	27
Gold-198	$^{198}_{79}Au$	2.7 d	0.41 (95.5)	0.23	62
Iodine-125	$^{125}_{53}I$	60.1 d	0.027 (57) 0.031 (10)	0.01	2.7
Iodine-128	$^{128}_{53}I$	25 min	0.443 (17.5)	0.04	10.8
Iodine-131	$^{131}_{53}I$	8 d	0.364 (82.4) 0.637 (7)	0.22	59.4

(Continued)

Nuclide	Symbol	Half-life	Principal Gamma Energies, MeV (% abundances)	Unshielded Dose Rate from 1 Ci at 1 m(R h^{-1})	Unshielded Dose Rate from 1 GBq at 1 m(μSv h^{-1})
Iridium-192	$^{192}_{77}Ir$	74 d	0.30 (60) 0.31 (31) 0.32 (86) 0.47 (51)	0.48	130
Iron-59	$^{59}_{26}Fe$	45 d	1.1 (56) 1.29 (44)	0.64	173
Manganese-56	$^{56}_{25}Mn$	2.58 h	0.85 (99) 1.81 (30) 2.1 (15)	0.9	243
Mercury-197	$^{197}_{80}Hg$	64.1 h	0.067 (20) 0.069 (36) 0.077 (32)	0.04	10.8
Mercury-197m	$^{197m}_{80}Hg$	23.8 h	0.134 (30)	0.02	5.4
Mercury-203	$^{203}_{80}Hg$	46.6 d	0.279 (81.5)	0.12	32.4
Molybdenum-99	$^{99}_{42}Mo$	66.6 h	0.14 (5) 0.018 (9.8) 0.74 (13) 0.78 (4.7)	0.15	40.5
Potassium-42	$^{42}_{19}K$	12.4 h	1.52 (18)	0.14	37.8
Radium-226	$^{226}_{88}Ra$	1600 y	See Uranium/ Radium (4n+2) Series	0.825	223
Rubidium-86	$^{86}_{37}Rb$	18.7 d	1.07 (8.8)	0.05	13.5
Scandium-46	$^{46}_{21}Sc$	83.8 d	0.889 (100) 1.121 (100)	1.09	294
Sodium-24	$^{24}_{11}Na$	15.0 h	1.37 (100) 2.75 (100)	2.18	589
Technetium-99m	$^{99m}_{43}Tc$	6.02 h	0.14 (85)	0.07	18.9
Ytterbium-169	$^{169}_{70}Yb$	32.0 d	0.0635 (85) 0.110 (18) 0.131 (11) 0.177 (22) 0.198 (40) 0.308 (10)	0.11	29.7
Zinc-65	$^{65}_{30}Zn$	243.8 d	1.115 (50.7) 0.511 (from β+)	0.30	81

3.6 Neutron Dose and Dose Rates for Particular Neutron Energies

Neutron Energy MeV	Quality Factor	Time Ave. Flux (n cm^{-2} s^{-1}) ≡ 2.5 mrem h^{-1}	Integrated Flux (n cm^{-2}) ≡ 1 rem
Thermal	3	670	9.6×10^8
0.0001	2	500	7.2×10^8
0.005	2.5	570	8.15×10^8
0.02	5	280	4.08×10^8
0.1	8	80	1.2×10^8
0.5	10	30	4.32×10^7
1.0	10.5	18	2.64×10^7
2.5	8	20	2.88×10^7
5.0	7	18	2.64×10^7
7.5	7	17	2.40×10^7
10	6.5	17	2.40×10^7

NOTE that the neutron dose depends on neutron energy and neutron flux.

3.7 Characteristics of Some Radioactive Neutron Sources

Source	Reaction	Half-life	Average Neutron Energy, MeV	Yield per Ci, neutrons s^{-1}	Gamma Dose Rate per 10^6 n s^{-1} (mR h^{-1} at 1 m)
^{241}Am-Be	α, n	458 y	4.5	2.2×10^6	1
^{252}Cf	Spontaneous fission	2.65 y	2.35	4.3×10^9	< 1
^{210}Po-Be	α, n	138 d	4.2	2.5×10^6	0.04
^{238}Pu-Be	α, n	86 y	4.5	2.3×10^6	< 1
^{239}Pu-Be	α, n	24 360 y	4.0	2.2×10^6	5
^{226}Ra-Be	α, n	1620 y	4.0	1.3×10^7	60

4. HEALTH PHYSICS MONITORING PROGRAM CONSIDERATIONS

When considering a monitoring program each facility must be considered individually on its merits, taking the following into account:

(i) the type of ionising radiation likely to be encountered;

(ii) the type and degree of shielding provided to minimise external radiation exposure to personnel;

(iii) the type of containment provided for work with unsealed radioactive materials and the radioactivity of these materials;

(iv) the type of work to be carried out (e.g. research, routine or production);

(v) the safety features of the working area (e.g. ventilation, working surfaces); and

(vi) the training and experience of the staff working in the area.

A monitoring program may include:

(a) monitoring the workplace for external radiation (e.g. dose-rate meter);

(b) monitoring personnel for external radiation (e.g. film badge);

(c) monitoring working surfaces, floors, walls, machines, etc. for surface contamination (e.g. contamination monitor);

(d) monitoring skin and clothing of personnel for contamination (e.g. hand and clothing monitor);

(e) monitoring air in working area for contamination (e.g. air sampler);

(f) monitoring waste (liquid, solid, gaseous); and

(g) special monitoring for a particular operation (e.g. during maintenance, repair, etc.) in high dose rate areas.

5. ACCIDENTS WITH RADIOACTIVE MATERIALS

Prime objectives following an accident are:

(i) to minimise the exposure of persons to ionising radiations and radioactive materials; and

(ii) to return conditions to normal as soon as possible.

Where necessary the following should be implemented:

(a) evacuate area;

(b) set up barriers, restrict access to area;

(c) measure radiation and contamination levels to determine the hazard and delineate the accident area;

(d) use suitable protective clothing (e.g. overshoes, boots, respirators);

(e) carry out decontamination;

(f) return radioactive sources to shielding;

(g) monitor persons involved in accident; and

(h) monitor clean up personnel.

Personnel dosemeters should be used during all clean-up operations where external radiation is present and care taken to minimise hazards to persons engaged in these operations.

6. PERSONNEL DECONTAMINATION

When a person becomes contaminated, every effort must be made to remove the contamination as soon as possible. Persistent skin contamination should be referred to qualified medical staff for treatment.

Simple soap and water washes are often effective if applied as soon as possible after occurrence of contamination.

For more stubborn contamination, a 1 per cent Cetavlon solution may be applied followed by washing or a 2 per cent potassium permanganate solution applied and left for about one minute; hands are then washed thoroughly and decolourised with 5 per cent sodium metabisulphite solution.

If at any time during decontamination the skin shows signs of cracking or becoming red-raw, medical attention should be sought.

7. LICENSING

In most countries users of radioactive materials and devices which emit ionising radiations are required to have a licence which controls their use, storage, transport and disposal.

8. TRANSPORT OF RADIOACTIVE MATERIAL

The 1973 IAEA Transport Regulations form the basis of most arrangements for the safe transport of radioactive materials throughout the world. (Note: This discussion covers the subject briefly; for firm information refer to the IAEA Transport Regulations.)

There are four basic requirements for packages containing radioactive materials:

(a) Adequate containment of radioactive material.

(b) Adequate shielding against radiation emitted by the material.

(c) The dissipation of heat generated by high-activity radioactive material.

(d) Prevention of nuclear criticality when the material is fissile.

Containment

The toxicity of radionuclides varies by a factor of about 10^8, so

there is clearly a need for a number of packaging standards. Packages have therefore been divided into five main types: Type A, Type B, low specific activity, low level solid, and exempt.

Type A packaging is designed to withstand the normal transport conditions. In an accident, however, it is accepted that the containment may be breached and that some of the contents may escape. The maximum activity of each radionuclide which can be transported in a Type A package is therefore limited so that, in the event of an accident, the risk to transport workers and members of the public will not be unacceptable.

Type A packaging must be capable of passing a series of prescribed tests which are intended to simulate the damage caused by driving rain and minor mishaps that would be encountered during rough handling of packages under normal transport conditions. The tests include a water spray test, a free drop test, a compression test and a penetration test. Type A packaging for liquid or gaseous materials, which are more dispersible than solids, must be capable of passing additional tests including a 9 metre drop test.

Type B packaging is intended to retain adequate containment and shielding, even in the event of a severe accident such as a drop while loading, a vehicle or ship collision, derailment followed by impact with a bridge or other abutment, or an air to ground crash. There is therefore no regulatory upper limit for the activity which can be transported in a suitably designed Type B package.

Type B packaging must be capable of passing the Type A tests and, in addition, mechanical tests in which a specimen package is dropped onto a flat target from a height of 9 metres and then dropped onto the end of a circular metal bar from a height of one metre, followed by a thermal test in which the specimen is exposed to a temperature of 800°C for 30 minutes. A separate specimen must also be capable of passing a water immersion test in which the specimen is immersed under a head of water of at least 15 metres for a period of not less than eight hours.

The design, and in some cases the shipment, of Type B packages requires the approval of the national competent authority because of the greater potential hazard of such packages compared with Type A packages. Type B packages are subdivided into two groups, Type B(U) and Type B(M), depending on whether the package design warrants the approval of all competent authorities en route, *i.e.* Type B Multilateral, or whether the approval of the competent authority of the country of origin can reasonably

be held to be binding on others, *i.e.* Type B Unilateral.

Type B(U) packages must meet a series of design criteria as specified in the IAEA Transport Regulations and must also require no operational controls during transport. Approval of the design of Type B(U) packages by the competent authority of the country of origin only is required. Type B(M) packages on the other hand do not meet all the above design criteria, or else require operational controls during transport. Approval of the design of Type B(M) packages, and for certain large shipments approval of the shipment, by the competent authorities of the country of origin and of all countries through or into which the package will be transported, is required.

Low specific activity materials are materials which are regarded as inherently safe because their specific activity is so low that it is considered inconceivable that, under any circumstances arising in transport, a sufficient mass of material could be taken into the body to give rise to a significant radiation dose. Uranium and thorium ores and their concentrates are an example of low specific activity materials. These materials can be transported either in bulk as a full load, or in commercial packages which meet less stringent requirements than those for Type A packages.

Low level solid radioactive materials represent an extension of the low specific activity material concept to include certain types of consignments of low and medium level radioactive wastes. Such materials are not inherently safe and so must be transported in strong industrial packaging under full load conditions.

Exempt items consist of small quantities of radioactive materials, such as samples and radioactive components of instruments, and articles which have a low potential hazard. These items are free from most regulatory requirements.

Shielding

All packages are classified into three categories based on the external radiation at the surface of the package and at a distance of 1 metre from the surface. The radiation level at a distance of 1 metre from the surface of the package is referred to as the transport index. The three categories are as follows:

Category I - White : Radiation level at surface < 0.5 mR h^{-1} and package not Fissile Class II or Class III.

Category II - Yellow : Radiation level at surface between 0.5 and 50 mR h^{-1}, transport index < 1.0, and package not Fissile Class III.

Category III - Yellow : Radiation level at surface between 50 and 200 mR h^{-1} and transport index < 10.

The above surface radiation levels have been adopted on the basis of safe operating experience. The level of 0.5 mR h^{-1} for Category I - White packages for example was determined on the basis that an exposure of 10 mR is the maximum that could be accepted for undeveloped photographic film. It has been assumed that 24 hours would be the longest period for which boxes of such film would be close to packages of radioactive material during transport. Category I - White packages can therefore be handled and transported with no requirements for segregation from persons or film.

The above radiation categories are identified with three defined labels as illustrated in the IAEA Transport Regulations. On the Category II - Yellow and Category III - Yellow labels it is important that the transport index be inserted on the label. The transport index is used to control the number of packages which can be grouped together in order to ensure that the external radiation level from a group of packages does not exceed safe levels and also as a criticality control device.

Higher external radiation levels, up to 1000 mR h^{-1} in some circumstances, are allowable on the external surface of a package when it is transported under full load conditions, *i.e.* for a load from a single consignor having the sole use of a vehicle and in respect of which all initial, intermediate and final loading and unloading is carried out in accordance with the directions of the consignor or consignee.

Similar provisions also exist for identifying freight containers with Category I - White, Category II - Yellow and Category III - Yellow labels.

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