INTERNATIONAL ATOMIC ENERGY AGENCY
UNITED NATIONS EDUCATIONAL, SCIENTIFIC AND CULTURAL ORGANIZATION
INTERNATIONAL CENTRE FOR THEORETICAL PHYSICS
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H4.SMR/773-16

College on Medical Physics:
Radiation Protection and Imaging Techniques

5 - 23 September 1994

Radiation Protection in Brachytherapy

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Trieste 5-23 September 1994

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Brachytherapy

Brachytherapy is a method of radiation therapy in which encapsulated sources are utilized to deliver radiation within a distance of a few centimeters by surface, intracavitary or interstitial applications.

The aim of this therapy is to enhance tumor sterilization while minimizing damage to normal tissue structures.

A number of advantages are provided by brachyterapy applications such as a more precise localization of dose and a dose distribution which conforms to irregular tumor shape.

Initially all brachytherapy was carried out with radium or radon sources. In recent years there has been increasing utilization of artificially produced radionuclides.

In table 1 and table 2 information on the radiation emitted by various radionuclide is reported.

The nuclides of shorter half-life which are most utilized in brachitherapy are Ir 192, Ta 182, I 125, and Au 198.

Among the long-lived radionuclides Cs 137 and Co 60 are the most utilized.

The development of new techniques, the use of radium substitutes, and the improvement of after-loading devices have increased interest in brachytherapy.

Radiations emitted by certain radionuclides

Radionuclide	Radiation		Range	
	Type	Energy	Air	Aluminum
		MeV	cm	mm
Naturally Occurring				
Radium-226	alpha	4.78, 4.60	3.4	—
	gamma	0.18	—	—
Radon-222	alpha	5.5	4.1	—
Polonium-218–Ra A	alpha	6.0	4.7	—
Lead-214–Ra B	beta	0.67, 1.03	—	0.9
	gamma	0.053–0.35	—	—
Bismuth-214–Ra C	alpha	5.5	4.1	—
	beta	3.26	—	5.9
	gamma	0.6–2.4	—	—
Polonium-214–Ra C'	alpha	7.69	7.0	—
Thallium-210–Ra C"	beta	2.3	—	4.0
	gamma	0.296–2.43	—	—
Lead-210–Ra D	beta	0.06	—	0.02
	gamma	0.047	—	—
Bismuth-210–Ra E	beta	1.16	—	1.9
Polonium-210–Ra F	alpha	5.3	3.9	—

Tab 1

Radiations emitted by certain radionuclides

Radionuclide	Radiation		Range	
	Type	Energy	Air	Aluminum
		MeV	cm	mm
Man-Made				
Californium-252	neutrons	fission	—	—
	alpha	6.12, 6.08	4.9	—
	gamma	prompt fission plus fission product	—	—
Cesium-137	beta	0.514, 1.176	—	2.0
	gamma	0.662	—	—
Chromium-51	electrons	0.315	—	0.3
	gamma	0.320	—	—
Cobalt-60	beta	0.314	—	0.4
	gamma	1.173, 1.332	—	—
Gold-198	beta	0.962	—	1.4
	gamma	0.412	—	—
Iodine-125	electrons	0.004, 0.030	—	—
	gamma	0.035	—	—
	Te x-rays		—	—
Iridium-192	beta	0.67	—	0.9
	gamma	0.296–0.612 (various)	—	—
Tantalum-182	beta	0.522	—	0.7
	gamma	0.068–1.23 (various)	—	—

Tab 2

Sealed sources

The accuracy of source calibration and of absorbed dose calculations in brachytherapy applications depends, in part, on a detailed description of the radioactive sources.

Therefore, it is important that the user obtain any information that permits to evaluated the potential implications for clinical dosimetry.

In general, this information is available from the manufacturer or from the literature.

- Physical and Chemical Form

The chemical composition of the radionuclide (e.g., Cs-137 absorbed onto ceramic microspheres, radium sulfate, etc.) and inert filler material should be known along with information on the physical characteristics of the material (e.g., density, effective mass energy-absorption coefficient, etc.).

This information is useful because:

although chemical instability and physical changes within a source are unlikely and are the responsability of the manufacturer, the possibility of such changes and the potential effects on patient treatments during the useful life of a source should not be ignored.

Beyond, dose correction for attenuation due to the self-absorption within a source may be desired although the effect is generally quite small. Third, the presence of radioactive impurities should also be known.

Some sources (e.g., Ir-192) require a storage period after initial production to allow the decay of short-lived impurities; users should ask the manufacturer if such procedures are followed. Finally, if the source should rupture, knowledge of the chemical form may aid in radiation safety considerations.

-Source calibration

Since tbe source encapsulation can influence source calibration, dose distribution, and source integrity, detailed knowledge of its configuration and composition is important for the overall accuracy of clinical dosimetry.

Such information should be available from the manufacturer. Encapsulation design may vary for the same radionuclide for different manufacturers as well as for different radionuclides. Most long-lived sources (Ra-226, Cs-137) are doubly encapsulated; other sources (Au-198) are singly coated, and others have a unique capsule design (I-125).

The effect of the encapsulation on dose distribution of various source has been investigated both experimentally and theoretically .

-Radionuclide Distribution and Source Uniformity

The distribution of radioactive material within the encapsulation may be continuous or in compartments or cells; the loading of radionuclide along a source may be uniform or non-uniform, by design or otherwise; the active length may or may not be centrally located along the source; the wall thickness of the casing may be non-uniforn in different areas.

These intricacies need to be considered for each type of source a.nd their implications relative to source calibrations and dose diitributions carefully assessed.

Autoradiography of a source is a simple and informative test; gross non-uniformity of the radionuclide within the source ir easily visualized. For radioactive seeds or grains, the unifornity of activity among seeds needs to be assessed. The spacing of seeds in ribbons as provided by the manufacturer may require monitoring.

-Source Identification

Correct identification of sources of the same radionuclide and capsule design but of different activities is essential. Ease in such identification will prevent errors and reduce the level ot personel exposure and anxiety.

At present, markings on sources are frequently difficult to read; color coding fades or disintegrates with time, and repeated handling of color satures tied to needles causes loss of effectiveness with age.

The user must work with the manufacturers to devise an acceptable identification system which is simple, easy to read and long lasting.

Calibration of brachytherapy sources

It is common practice to express source strength of brachytherapy sources in terms of activity or equivalent mass of radium.

The output of a brachytherapy source, in terms of exposure rate in free space, can be obtained by multiplying the source strength by the appropriate exposure-rate constant.

Several shortcomings of these methods has been pointed out by International Committee, the most important being that they do not specify strenght in terms of directly measured quantities.

In the following pages a part of AAPM Report n.21 is reported on the subject.

III. Source Specification Methods Currently in Use

Determination of the dose distribution from a brachytherapy source in tissue requires a knowledge of source strength and a number of conversion factors. In this section, currently used methods of source specification and the calculation of exposure rate in free space are described. Not treated is the calculation of dose rate in tissue, which may proceed from that of exposure rate in free space but is preferably based on dose measurements in phantom around a source of known strength.

A. Activity of the Radionuclide

The source strength can be specified in terms of the activity of the radionuclide in the source. For an activity A, the exposure rate at a point of interest in free space may be given by the expression

$$\dot{X}(r,\theta) = A \; (\Gamma_\delta)_X \; G(r,\theta) \quad \alpha(r,\theta) \tag{1}$$

where

$$G(r,\theta) = \begin{cases} \dfrac{1}{r^2} & \text{for point source approximation,} \\[2ex] \dfrac{\theta_2-\theta_1}{Lh} & \text{for line source approximation,} \end{cases}$$

and

$$\alpha(r,\theta) = \begin{cases} e^{-\mu t} & \text{for point source approximation,} \\[2ex] \dfrac{1}{\theta_2-\theta_1} \displaystyle\int_{\theta_1}^{\theta_2} e^{-\mu t \sec\theta} d\theta & \text{for line source approximation.} \end{cases}$$

The lengths and angles are shown in Fig. 1. L is the active length of the line source, t is the thickness of encapsulation material, μ is the effective absorption coefficient of the encapsulation material and $(\Gamma_\delta)_X$ is the exposure-rate constant for a bare point source of the radionuclide. $G(r,\theta)$ is a geometry factor and $\alpha(r,\theta)$ is a dimensionless quantity that accounts for absorption and scatter in the source material and encapsulation.

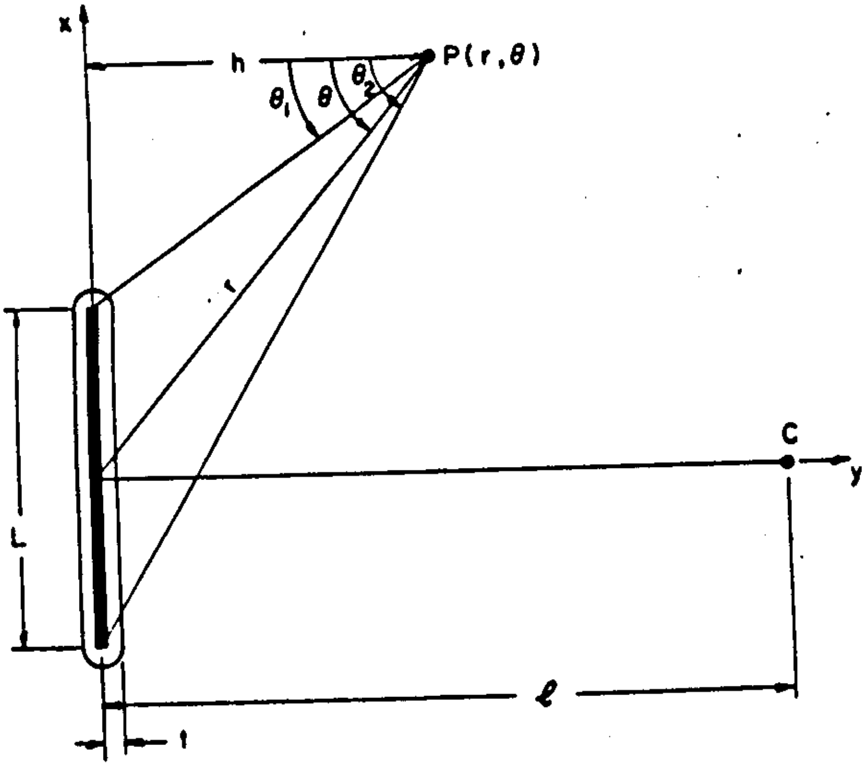


Fig. 1. Schematic drawing of the geometry of a line source calculation.

The concept of effective attenuation coefficient for these calculations has been a subject of many investigations, including recent studies by Williamson and Morin[14] and Cassell[15]. It should be noted that, for polyenergetic radionuclides, e.g. ^{226}Ra, the effective attenuation coefficient depends on absorber thickness, due to selective absorption of the lower energy gamma rays. In the case of ^{226}Ra, measured values of μ are available and may be used[16,17]. However, when such values are not available, calculated values using mass energy-absorption coefficients have to be used[14].

For a point source with spherical encapsulation, α is a constant that depends upon photon energy and encapsulation thickness. For heavily filtered sources with non-spherical encapsulation, the effect of anisotropy may be large. Anisotropy correction factors have been calculated by using the angular exposure distribution, normalized to the calibration exposure rate. For ^{125}I and ^{192}Ir seeds, these factors can be significant.[18,19]

For a linear source, α is a function of r and θ , and is proportional to the Sievert integral indicated. Young and Batho[20] have shown that the exposure rate from any linear radium source can be expressed in the form:

$$\dot{X}(x,y) = \frac{A}{L^2} F(j,k,\mu t) \tag{2}$$

where j and k are cartesian coordinates of the point in terms of active length L (i.e. $j = x/L$ and $k=y/L$). The function F has been tabulated and these tables are used in some computerized dose calculation systems currently in clinical use. One of the key advantages of the Young and Batho scheme is that the same function F can be used for radium sources of any length. The same tables are also used for radium substitutes such as ^{137}Cs sources by some of the computer systems. For better accuracy, similar tables should be recalculated for each radionuclide.

B. Exposure Rate at a Reference Point in Free Space

Source strength can be specified in terms of exposure rate $\dot{X}_\ell$ at a specified distance ℓ (usually 1 m) along the perpendicular bisector of the line source. If the source calibration is performed at (or if the result is corrected to) a point that is at a distance of 1 m from the source center, the line-source geometry factor is negligibly (less than 0.01%) different from the point-source geometry factor $1/r^2$ and α reduces to $e^{-\mu t}$. Under these conditions, $\dot{X}_\ell$ is given by

$$\dot{X}_\ell = \frac{A}{\ell^2} \; (\Gamma_\delta)_X \; e^{-\mu t} \tag{3}$$

Exposure rate at any point in free space is given by

$$\dot{X}(r,\theta) = \frac{\dot{X}_\ell \, \ell^2}{e^{-\mu t}} \quad G(r,\theta) \; \alpha(r,\theta) \tag{4}$$

The exposure rate at any point (r,θ) is calculated from the measured exposure rate at the calibration distance ℓ. The source activity is not implicitly required in equation 4.

C. Equivalent Mass and Apparent Activity

Brachytherapy sources for which exposure is well defined, can be specified in terms of equivalent mass of radium, which is defined as that mass of radium, encapsulated in 0.5 mm Pt, that produces the same exposure rate at the calibration distance as the source to be specified.

For a source with an equivalent mass of radium m_{eq} the exposure rate at the calibration distance ℓ is given by the expression,

$$\dot{X}_\ell = \frac{m_{eq}}{\ell^2} \; (\Gamma_\delta)_{X,Ra} \tag{5}$$

provided the specification point is far enough away that the source may be treated as a point source. $(\Gamma_\delta)_{X,Ra}$ is the exposure-rate constant for radium encapsulated in 0.5 mm Pt, in terms of exposure rate at a unit distance per unit mass of radium. This method of specification is thus equivalent to the specification in terms of exposure rate, as described above. The gamma-ray energy of radium is so high that the air attenuation at one meter can be neglected.

A similar method is that of specifying source strength as the activity of a bare point source (of the same radionuclide) that produces the same exposure rate at the calibration distance as the source to be specified. In analogy with the equivalent mass of radium method, the "apparent" source activity A_{app} is defined by its calibration-distance exposure rate, i.e.,

$$\dot{X}_{\ell} = \frac{A_{app}}{\ell^2} (\Gamma_\delta)_X \tag{6}$$

This method has been extensively applied for ^{125}I seeds[21].

D. Air-Kerma Rate at a Reference Point in Free Space

Source strength can be specified in terms of air-kerma rate at a specified distance in free space (usually 1 m) along the perpendicular bisector of the line source. BCRU[8] and CFMRI[6] have recommended this method of specification. ICRU has employed this method of specification of source strength in its report on intracavitary therapy[24] and has recommended the name "reference air-kerma rate" for source strength. This method of specification of source strength is currently being used in most of Europe.

Quantities and Units in Teletherapy and Brachytherapy

Quantity	SI Unit Name	SI Unit Symbol	usual multiple	Symbol of customary or special units
length	meter	m	mm, cm	cm
time	second	s	min,h,d	h
mass	kilogram	kg	g	g
energy	joule	J		MeV
absorbed dose	gray	Gy	cGy	rad, mrad
exposure	coulomb per kilogram	C/kg		R, mR
activity	becquerel	Bq	MBq	mCi
temperature	degree Celsius	°C		°F
pressure	pascal	Pa	kPa	mbar atm mmHg

CONVERSION FACTORS

1 rad = 1cGy

1 Gy = 100 rad

1 R = 0.258 mC/kg

1 C/kg ≈ 3.88 kR

1 mCi = 37 MBq

1 MBq ≈ 27.0 μCi

1 MeV ≈ 1.602×10^{-13} J

1 J ≈ 6.24×10^{12} MeV

$1 \frac{\mathrm{R\ cm^2}}{\mathrm{mCi\ h}} \approx 1.937 \times 10^{-13} \frac{\mathrm{(C/kg)\ m^2}}{\mathrm{MBq\ s}}$

$1 \frac{\mathrm{(C/kg)\ m^2}}{\mathrm{MBq\ s}} \approx 5.16 \times 10^{12} \frac{\mathrm{R\ cm^2}}{\mathrm{mCi\ h}}$

$1 \frac{\mathrm{R\ cm^2}}{\mathrm{mCi\ h}} \approx 7.51 \times 10^{-10} \frac{\mathrm{R\ m^2}}{\mathrm{MBq\ s}}$

$1 \frac{\mathrm{R\ m^2}}{\mathrm{MBq\ s}} \approx 1.332 \times 10^{9} \frac{\mathrm{R\ cm^2}}{\mathrm{mCi\ h}}$

$t_C = (t_F - 32)/1.8$

$t_F = 1.8t_C + 32$

where t_C is the temperature in degrees Celsius and t_F is the temperature in degrees Fahrenheit.

760 mmHg ≈ 1 atm = 1013.25 mbar = 101.325 kPa

100 kPa = 1000 mbar ≈ 0.987 atm ≈ 750 mmHg

Radiation safety

The fundamental objective of the use of sealed sources is to obtain optimum therapeutic effec with minimum exposure to the patient, the radiological personnel concerned, and the general public.

The exposure of individuals can be greatly reduced by the correct application of radiation protection techniques.

Reduction of radiation exposure to an individual from external sources of radiation may be achieved by one or any combination of the following measure:

-increasing the distance of the individual from the source,

-reducing the duration of exposure,

-using protective barriers between the individual and the source.

In brachytherapy all these measure are useful.

Categories requiring radiation safety measures in brachytherapy are the following:

-Facility

Receipt and inventory of the sources
Storage and work areas (shieldind, carrier design)
Transportation (shielding, carrier design)

-Maintenance

Inventory
Source identification
Cleaning (especially safety aspects)
Leak test
Disposal

-Clinical application

Preparation, sterilization and transfer of the sources and source applicator
Application to patient
Removal of sources from patient (patient and room survey)
Return of sources to storage area
Personnel monitoring
Patient discharge

-Emergencies and Special Precautions

Source breakage and contamination
Loss of sources
Cardiac and respiratory arrest
Emergency surgery
Death of patient (autopsy, cremation, embalming)
Notification of the location of radioactive sources to local fire department

-Education and training

Physician and nursing staff
Ancillary personnel (including housekeeping)

In table 3 some data useful to evaluate the thickness of barriers are reported.
Table 4 gives the required thickness for various conditions.
Table 5 gives the distance and time for unshielded sources to yield an exposure of 0.1 R.
Table 6 and table 7 give dose estimate which can be useful in case of handling the sources.
Approximated exposure rates in a region around the bed of a patient containing several amount of radiactive are shown in figure 2.
In table 8 exposure rate of Ra 226 at various distance is presented.

The exposure rates for other radionuclides than Ra 226 can be calculated by multiplication of the values from the table by the following values which correspond to the relationship of specific gamma constant ($\Gamma_X / \Gamma_{Ra226}$).

Specific γ-ray constants Γ

Co 60 1.58 Cs 137 0.39 Ta 182 0.83 Au 198 0.283 Ir 192 0.62

Table 9 shows the relation between distance and millicurie-hours for an exposure of 0.1 R from an unshield source.
Table 10 shows the transmission factor B for various sources and for different distances. The factor B may be employed to determine the required thickness of shielding material from figure 3.

Advantage of Remote Afterloading

Remote afterloading improves radiation control and provide technical advantage, such as isodose distribution optimization, that improve patient care. Replacing manual afterloading with remote afterloading reduces the radiation exposure to radiation oncologist, physicist, attending physicians, source curator, nurses and other personnel.

In figure 4 a dedicated remote afterloading roon is shown.

Previous data and information are taken from AAPM Report n.13, AAPM Report n.21, AAPM Report n.41 NCRP Report n.40, NCRP Report n.49.

TABLE 3—*Selected gamma-ray sources*

Radionuclide	Atomic Number	Half-Life	Gamma-Energy	Gamma-Ray Exposure Rate at 1 meter	Half Value Layer[a]		Tenth Value Layer[a]	
					Lead	Concrete	Lead	Concrete
			MeV	R/Ci h	cm	inches	cm	inches
Cesium-137	55	30.0 years	0.662	0.33	0.65	1.9	2.1	6.2
Chromium-51	24	27.8 days	0.320	0.015	0.2		0.7	
Cobalt-60	27	5.26 years	1.17, 1.33	1.30	1.2	2.5	4.0	8.1
Gold-198	79	2.7 days	0.412	0.23	0.33	1.6	1.1	5.3
Iodine-125	53	60.2 days	0.035	0.123 [39]	0.04			
Iridium-192	77	74.2 days	0.296–0.612	0.48	0.3	1.7	2.0	5.8
Radium-226 (with daughters: 0.5 mm Pt filter)	88	1602 years	0.047–2.4	0.825	1.4	2.7	4.6	9.2
Tantalum-182	73	115.1 days	0.068–1.23	0.68	1.2	2.6	4.0	8.6

[a] Approximate values obtained with large attenuation.

TABLE 4—*Protection requirements for various gamma-ray sources—thicknesses of lead or concrete required to reduce radiation exposure to 100 mR at distances specified* ( )

a. *Cesium-137*[b]. *Thicknesses of lead required (in cm)*

Millicurie Hours	Distance from source: 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	1.1	—	—
300	2.1	—	—
1000	3.2	1.1	—
3000	4.2	2.1	0.9
10000	5.3	3.2	1.9

b. *Cesium-137. Thicknesses of concrete required (in inches)*

Millicurie Hours	Distance from source: 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	3.4	—	—
300	6.3	—	—
1000	9.6	3.2	—
3000	12.5	6.1	2.6
10000	15.7	9.3	5.7

c. Cobalt-60[c]. Thicknesses of lead required (in cm)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	4.5	0.4	—
300	6.5	2.6	—
1000	8.6	4.4	2.4
3000	10.0	6.4	4.0
10000	12.6	8.4	6.0

d. Cobalt-60. Thicknesses of concrete required (in inches)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	9.3	0.9	—
300	13.1	4.8	—
1000	17.4	9.2	4.3
3000	21.2	12.9	8.7
10000	25.5	17.0	12.2

e. Gold-198[d]. Thicknesses of lead required (in cm)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	0.5	—	—
300	1.0	—	—
1000	1.5	0.4	—
3000	2.1	0.9	0.3
10000	2.7	1.5	0.8

f. Gold-198. Thicknesses of concrete required (in inches)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	2.3	—	—
300	4.6	—	—
1000	7.4	2.0	—
3000	9.9	4.4	1.4
10000	12.7	7.2	4.2

g. Iridium-192[e]. Thicknesses of lead required (in cm)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	1.5	—	—
300	2.4	0.4	—
1000	3.4	1.4	0.2
3000	4.4	2.4	1.2
10000	5.5	3.4	2.2

h. Iridium-192. Thicknesses of concrete required (in inches)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	4.3	—	—
300	7.1	1.1	—
1000	10.0	4.1	0.6
3000	12.8	6.9	3.4
10000	16.0	9.8	6.4

i. Radium[f]. Thicknesses of lead required (in cm)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	4.6	—	—
300	6.5	1.4	—
1000	9.0	4.2	1.4
3000	11.1	6.4	3.6
10000	13.5	8.8	6.0

j. Radium. Thicknesses of concrete required (in inches)

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	8.7	—	—
300	13.0	3.6	—
1000	18.0	8.4	2.9
3000	22.0	12.8	7.2
10000	27.0	17.6	12.0

TABLE 4—*Continued*

k. *Tantalum-182*[g]. *Thicknesses of lead required (in cm)*

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	3.5	—	—
300	5.4	1.2	—
1000	7.4	3.3	1.0
3000	9.4	5.2	2.8
10000	11.4	7.3	4.9

l. *Tantalum-182. Thicknesses of concrete required (in inches)*

Millicurie Hours	Distance from source 1 ft 30 cm	3.2 ft 1 m	6.5 ft 2 m
100	7.4	—	—
300	11.5	2.7	—
1000	16.0	7.2	2.0
3000	20.1	11.3	6.1
10000	24.6	15.7	10.6

[a] This table is modified somewhat from the form used in NCRP Report No. 24 [6].
[b] *TVL* = 2.1 cm lead, or 6.2 inches concrete.
[c] *TVL* = 4.0 cm lead, or 8.1 inches concrete.
[d] *TVL* = 1.1 cm lead, or 5.3 inches concrete.
[e] *TVL* = 2.0 cm lead, or 5.8 inches concrete.
[f] *TVL* = 4.6 cm lead, or 9.2 inches concrete.
[g] *TVL* = 4.0 cm lead, or 8.6 inches concrete.

TABLE 5—*Relation between distance and millicurie-hours for an exposure of 0.1 R from an unshielded source*[a]

Source	Millicurie-Hours 10	30	100	300	1000	3000	10000
	Distance to Source—Feet						
Cesium-137	0.60	1.00	1.9	3.2	6.0	10.2	18.6
Chromium-51	0.13	0.22	0.4	0.70	1.3	2.2	4.0
Cobalt-60	1.18	2.05	3.8	6.5	12.0	20.5	37.5
Gold-198	0.50	0.87	1.6	2.7	5.0	8.9	15.8
Iodine-125	0.36	0.63	1.2	2.0	3.6	6.4	11.5
Iridium-192	0.73	1.27	2.3	4.0	7.3	12.7	23.3
Radium-226 (with daughters, filter 0.5 mm Pt)	0.94	1.63	3.0	5.1	9.4	16.3	30.0
Tantalum-182	0.86	1.50	2.7	4.7	8.6	15	27

[a] This table is modified somewhat from the form used in NCRP Report No. 24 [6].

TABLE 6—*Approximate gamma-ray dose rates to the hand for 1 curie in a sealed source*[a]

Nuclide	β max (principal) MeV	γ (principal) MeV	Γ R/mCi-h at 1 cm	Surface Dose Rate[b] R/min	Dose Rate at 1 cm tissue depth R/min	Dose Rate at 3 cm tissue depth R/min
^{137}Cs	0.51, 1.2	0.662	3.26	513	28	3.7
^{60}Co	0.31	1.17, 1.33	13.00	2075	114	16.0
^{192}Ir	0.67	0.468	4.80	813	43	5.5
^{226}Ra	0.4–3.2	0.047–2.4	8.25	1310	72	9.7

[a] Industrial source housings are usually of stainless steel and for the purpose of the calculations, the activity is considered to be a point source. In considering these dose estimates, there is assumed a capsule of outside diameter ¼ inch, with a wall of stainless steel (type 304) which is $\frac{1}{32}$ inch thick.

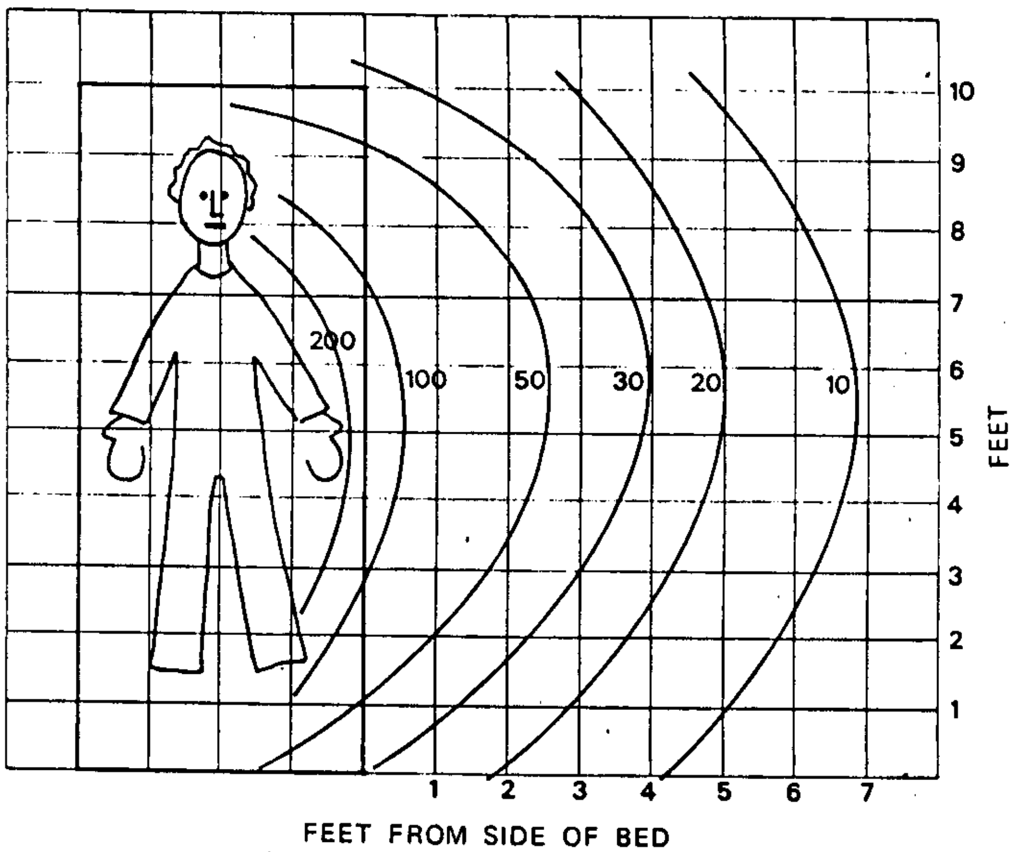
[b] The total surface dose rate for the ^{226}Ra source is 1900 R/min based on a 45 percent increase due to electron production in the stainless steel wall [38]. For the other nuclides given in the table, the increase in surface dose rate due to electron production in the stainless steel wall is estimated to be between 25–45 percent.

TABLE 7—*Surface dose rate calculations for radium needles and tubes*[a]

Weight mg	Active length cm	Outside diameter mm	Wall thickness mm Pt	Surface Gamma Dose Rate R/h	Total Surface[b] Dose Rates R/h
Needles					
1	1	1.7	0.5	393	656
2	2	1.7	0.5	393	656
3	3	1.7	0.5	393	656
Tubes					
5	1.15	2.7	1.0	700	1170
10	1.30	2.8	1.0	1240	2070
15	1.45	2.9	1.0	1665	2780
25	1.10	3.5	1.0	3660	6110

[a] Numerically integrated to correct for oblique filtration.

[b] Includes 67 percent increase in surface dose rate due to electron production in platinum wall [38].



• Approximate exposure rates (mR/h) in region around the bed of a patient containing 100 mg of radium or 300 mCi of gold (^{198}Au) or 300 mCi of iodine (^{131}I). Note that although only one side is shown, the exposure pattern completely surrounds the body. Furthermore, if the patient's bed is near a wall of beaverboard or other light material, the radiation penetrates the wall with little reduction, and the exposure pattern continues beyond the wall.

Fig 2

Activity	Exposure rates of Ra 226 at various distances (R/min)								
	Distance								cm
Ci	1	1.5	2	2.5	3	4	5	6	8
1	138	61.5	34.5	22.1	15.3	8.6	5.5	3.83	2.15
1.2	166	73.8	41.5	26.5	18.4	10.4	6.6	4.60	2.57
1.4	194	86.0	48.4	31.0	21.4	12.1	7.7	5.3	3.00
1.6	221	98.5	55.3	35.4	24.5	13.8	8.8	6.1	3.45
1.8	249	110	62.3	39.8	27.5	16.6	9.9	6.9	3.87
2	276	124	69	44.2	30.6	17.2	11.0	7.7	4.3
2.5	345	154	86.4	55.2	38.3	21.6	13.8	9.6	5.4
3	415	185	104	66.5	46.0	25.9	16.5	11.5	6.5
3.5	484	216	121	77.5	53.6	30.2	19.2	13.4	7.5
4	552	247	138	88.6	61.2	34.4	22.0	15.3	8.6
4.5	622	277	155	99.6	68.8	38.8	24.8	17.2	9.6
5	690	308	172	111	76.6	43.2	27.5	19.2	10.8
6	830	370	207	133	92.0	51.8	33.0	23.0	12.9
7	970	432	242	155	107	60.4	38.5	26.8	15.1
8	1104	493	276	177	122	68.8	44.0	30.6	17.2
9	2242	554	310	199	138	77.8	49.5	34.5	19.4
TBq	1	1.5	2	2.5	3	4	5	6	8
1	37.2	16.5	9.3	6.0	4.1	2.32	1.48	1.03	0.58
1.2	44.8	19.8	11.2	7.2	5.0	2.78	1.77	1.23	0.70
1.4	52.1	23.1	13.0	8.4	5.8	3.25	2.07	1.44	0.81
1.6	59.7	26.5	14.9	9.6	6.6	3.72	2.37	1.64	0.93
1.8	67.2	29.7	16.8	10.8	7.5	4.17	2.67	1.85	1.04
2	74.4	33.0	18.6	12.0	8.3	4.65	2.95	2.05	1.16
2.5	93.0	41.3	23.3	15.0	10.4	5.8	3.70	2.56	1.45
3	111	49.5	27.9	18.0	12.4	6.7	4.45	3.08	1.74
3.5	130	57.8	32.5	21.0	14.5	8.1	5.2	3.60	2.04
4	148	66.1	37.2	24.0	16.5	9.3	5.9	4.10	2.32
4.5	167	74.4	41.9	27.0	18.6	10.5	6.7	4.60	2.62
5	185	82.8	46.5	30.0	20.6	11.6	7.4	5.1	2.90
6	222	99.0	55.8	36.0	24.8	14.0	8.9	6.2	3.48
7	259	116	65.1	42.0	28.9	16.2	10.4	7.2	4.05
8	296	132	74.5	48.0	33.0	18.6	11.8	8.2	4.65
9	334	148	83.8	54.0	27.1	20.9	13.3	9.2	5.2

For activities other than those listed in the table, the exposure rates can easily be calculated in direct proportion to the activity and in inverse proportion to the square of the distance by means of decimal factors.

Tab 8

TABLE 9 —*Relation between distance and millicurie-hours for an exposure of 0.1 R from an unshielded source*

Millicurie-Hours	Gamma-Ray Source				
	Radium	Cobalt-60	Cesium-137	Iridium-192	Gold-198
	Distance to Source in Meters				
10	0.28	0.37	0.18	0.22	0.15
30	0.49	0.64	0.31	0.39	0.27
100	0.91	1.16	0.57	0.70	0.48
300	1.55	1.98	0.98	1.22	0.83
1,000	2.87	3.65	1.77	2.26	1.52
3,000	4.9	6.35	3.08	3.87	2.70
10,000	9.1	11.6	5.7	7.07	4.82

TABLE 10 —*Protective barrier transmission factor (B) per curie for 2.5 mR per hour (corresponding to 100 mR per 40 hours weekly exposure)*

Distance m	^{60}Co	^{226}Ra	^{192}Ir	^{137}Cs	^{198}Au
0.5	0.00048	0.00076	0.00114	0.00188	0.0027
1	0.00192	0.00303	0.00455	0.0075	0.0107
1.5	0.0043	0.0068	0.0102	0.0169	0.024
2	0.0077	0.012	0.0182	0.0301	0.043
2.5	0.012	0.019	0.028	0.047	0.067
3	0.0173	0.027	0.0404	0.068	0.096

Notes: (a) Applying the appropriate value of B, the necessary thickness of shielding material may be obtained from Fig. 11, 12, or 13 (Appendix D).

(b) If protection is required for non-occupational exposure, add one tenth value layer thickness for the shielding material chosen.

(c) For quantities of material other than a curie, the permissible transmission (B) should be adjusted in inverse proportion.

Example:

The required barrier thickness for a radiation worker at a 2-meter distance from a 50 mCi (0.05 Ci) cobalt-60 source is determined from:

$$B = 0.0077 \text{ (from table)} \times \frac{1 \text{ Ci}}{0.05 \text{ Ci}} = 0.154$$

For concrete, the value of B for cobalt-60 indicates a thickness (Figure 12, Appendix D) of 21 cm of concrete. To protect persons other than radiation workers, one tenth value layer of concrete or 20.6 cm would be added, totaling about 42 cm.

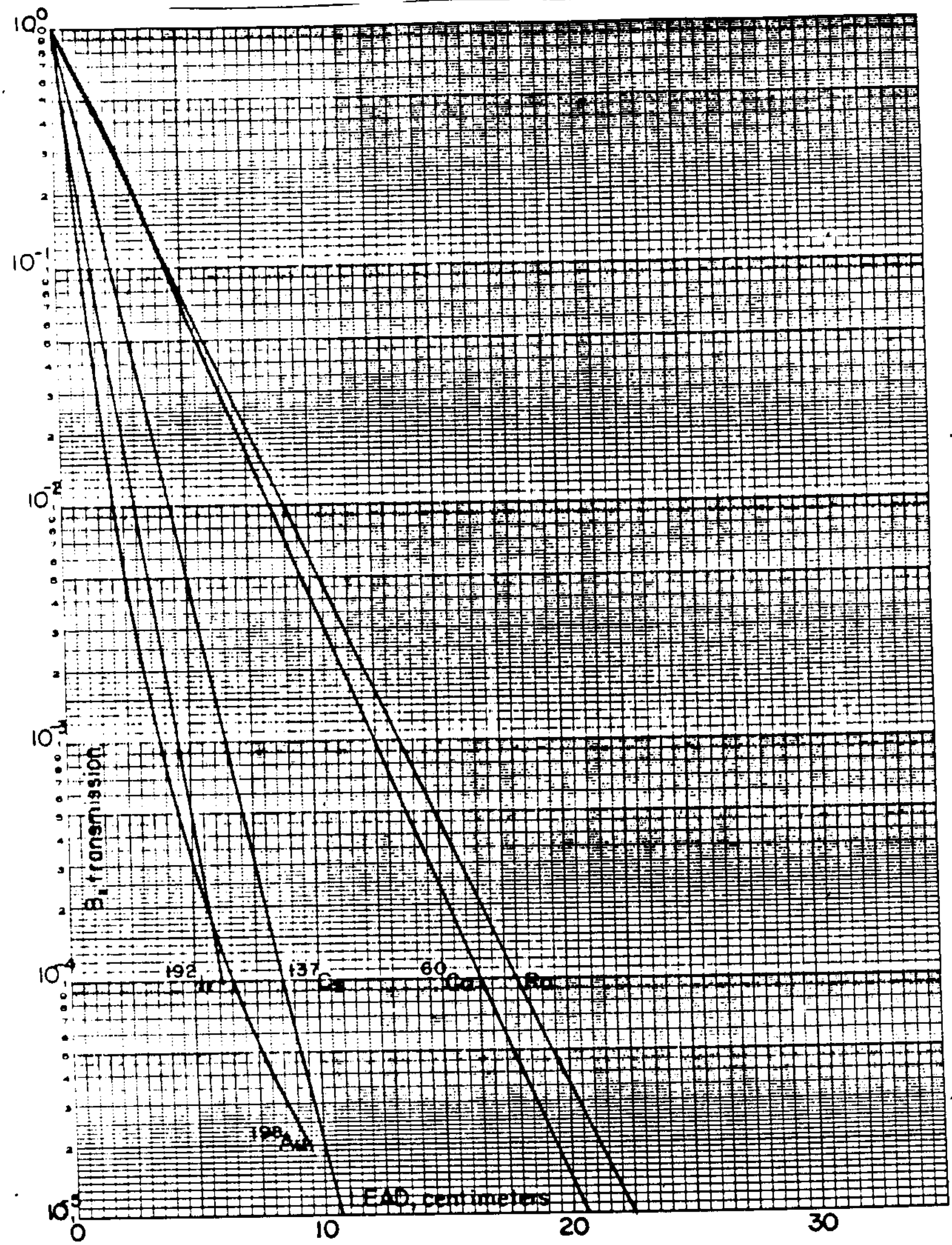


Fig. 3 Transmission through lead of gamma rays from selected radionuclides. Radium (Wyckoff and Kennedy [23]); cobalt-60, cesium-137, gold-198 (Kirn *et al.* [13]); iridium-192 (Ritz [24]).

[Data courtesy of the authors, Radiology, Journal of Research NBS, Non-Destructive Testing (now known as Materials Evaluation) and with permission of The American Society for Nondestructive Testing, Inc.]

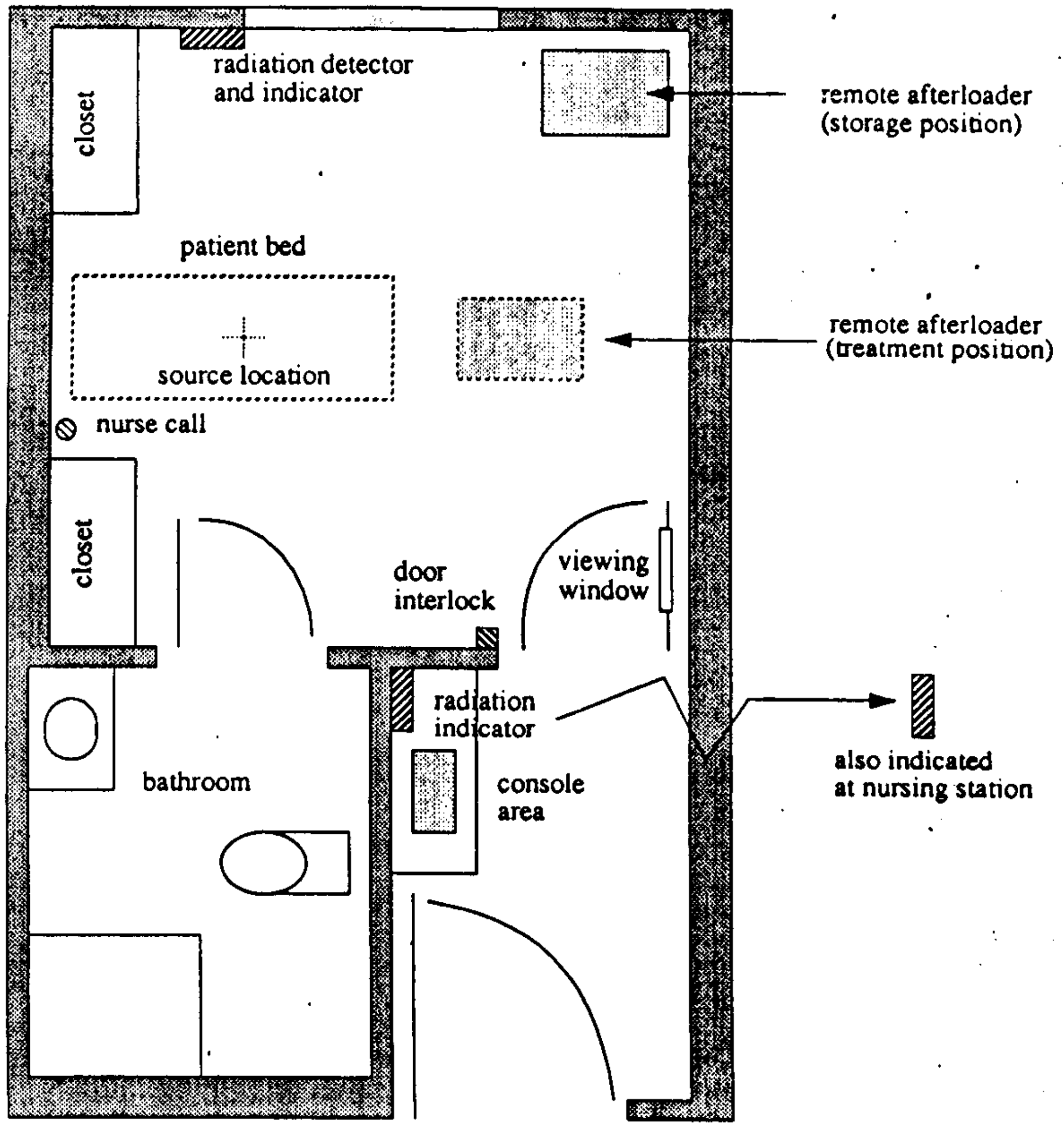


Figure 4. A dedicated LDR remote afterloading room with a small maze entryway and viewing window. (Modified from B. M. Wilson et al., Med Phys 13, 608, 1986, courtesy of J. D. Bourland, Rochester, MN).