Chapter 9

Radiation hazards and shielding

One of the most important characteristics of nuclear radiation such as alpha, beta, gamma radiation and neutrons is that these radiations and particles are a hazard to health in human beings and animals due to their ionizing effect in body tissue. Gamma radiation and neutrons, which have a high penetrating power, are an external hazard, and sources of these radiations must be shielded. Alpha and beta particles have a low penetrating power and are only dangerous if a substance emitting these particles is breathed or ingested into the body.

In view of this hazard it is important in the construction of a nuclear reactor to provide adequate shielding so that neutrons and gamma radiation originating in the reactor core are prevented from escaping into the reactor's surroundings where people are working. The same problem arises to a smaller extent with radioactive isotopes which are used on a large scale in science, engineering and medicine. In this case the size and strength of the radioactive source is very much less than the core of a nuclear reactor, but it is nevertheless necessary to reduce the radiation escaping from such a source to an acceptably low level.

In this chapter we will discuss first the effects of radiation on living tissue, the units by which these effects are measured, and the presently accepted limits of radiation dose. The interaction between gamma radiation (the most important radiation from the shielding point of view) and matter will be described in greater detail than was possible in Chapter 1. The shielding of simple radioactive sources will be studied to establish the relationship between source strength, shield thickness and dose rate. The problem of reactor shielding, which involves complex sources of both neutrons and gamma radiation, is beyond the scope of this book and will only be described qualitatively. Finally, the heating of shields due to the absorption of radiation will be discussed.

This energy is released in 0·001 293 gram of air. In one kg of air the energy released due to the same intensity of radiation is:

$$\frac{1{\cdot}133 \times 10^{-8}}{1{\cdot}293 \times 10^{-6}} = 8{\cdot}77 \times 10^{-3}\text{J/kg}$$

The roentgen may thus be defined as the dose of radiation which causes the release of $8{\cdot}77 \times 10^{-3}$ J/kg in air. This definition of the roentgen is based on effects in air whose properties are not the same as those of body tissue. The quantity of gamma radiation which releases $8{\cdot}77 \times 10^{-3}$J/kg in air releases about $9{\cdot}7 \times 10^{-3}$ J/kg in soft body tissue.

The use of this unit of radiation exposure is now more or less discontinued, and it has been described for historical interest.

The present unit of radiation dose which can be applied to any type of radiation is the gray (Gy). It is a measure of the energy of any radiation which is absorbed per unit mass in any material and is defined as:

1 gray = 1 joule of absorbed energy/kg

It is known that the energy released (or absorbed) is not the only factor to be considered in estimating radiation damage to the human body. Highly ionizing particles such as alphas or recoil protons (which are produced by the scattering of fast neutrons in hydrogenous materials) are more effective from the point of view of damaging human tissue than beta or gamma radiation, even when the amounts of energy released are in all cases the same. This factor is taken into account by specifying the Quality Factor (QF) of each type of radiation, based on a value of 1 for gamma radiation. The QF for several different types of radiation are given in Table 9.1.

Table 9.1. The Quality Factor of radiations and particles

Radiation	QF
X-, γ-radiation, betas	1
Thermal neutrons	3
Recoil protons, fast neutrons, alpha particles	20
Heavy recoil nuclei	20

The biologically equivalent dose is calculated by multiplying the absorbed dose in grays by the Quality Factor, the unit being called the sievert (Sv). The relationship is:

Equivalent dose in sieverts = Actual dose in grays × QF

As an example of this, it is evident that a fast neutron dose of 1 Gy is equivalent biologically to a dose of 20 Gy of gamma radiation.

Returning briefly to the list of radiation sources given above to which the public is exposed, the National Radiological Protection Board (NRPB), the UK agency responsible for matters connected with radiological protection, has estimated that the average annual dose equivalent to members of the UK population in the early nineteen-eighties from all sources was about 2·2 millisieverts (mSv). Of this, natural background radiation contributed 87 per cent, or just less than 2 millisieverts (mSv). Medical radiation contributed 11·5 per cent, and the smallest contribution, about 0·1 per cent, was due to the discharge of radioactive effluents and wastes. (NRPB – R159, 1984). There are, of course, non-typical groups of the population to which the above figures do not apply. For example, the citizens of Aberdeen, the granite city, receive a higher level of natural background radiation due to the traces of uranium in their buildings.

9.2 The biological effects of radiation exposure

The biological effects of exposure to radiation have been studied since the nineteen-twenties when it first became obvious that radiation could cause serious damage to health, and in extreme cases, death. The International Commission on Radiological Protection (ICRP) was established in 1928, and since then it has been the one internationally recognized body responsible for setting standards relating to radiation levels.

In the fifty years of its existence the ICRP has gathered data from many sources, including early workers using radioactive materials in a way which would now not be permitted, survivors of the two atomic bomb explosions over Japan in 1945, people who have received radiation diagnosis and therapy in hospital, workers in the nuclear industry and people living either near to a nuclear plant, such as power stations, or in areas of high natural background radiation. More recently, as a result of the serious reactor accident at Chernobyl in the USSR, another large group of exposed persons, those living near to that reactor at the time of the accident, will provide scientists with much data for years to come on the long-term effects of a reactor accident which resulted in a massive discharge of radioactive material into the environment.

During these fifty years the ICRP has continuously reviewed radiation dose levels that in the light of existing knowledge were regarded as having a negligible probability of causing serious illness. These levels have from time to time been reduced as more evidence has become available, and in 1977 the ICRP recommendation replaced the use of a maximum permissible dose by a more comprehensive system of dose limitation.

The 1977 recommendations may be summarized as:

1. No practice shall be adopted unless its introduction produces a positive net benefit.
2. All radiation exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.

3. The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances by the Commission.

The present whole body dose equivalent limit recommended by the ICRP for members of the public (not including the dose due to natural background radiation and medical sources) is 5 millisieverts per year (mSv/y). In the case of workers undergoing occupational exposure, the corresponding dose limit recommended is 50 millisieverts per year (mSv/y).

The health effects of radiation exposure can be divided into two categories, namely acute and delayed effects. Acute effects are commonly called radiation sickness. The symptoms depend on the magnitude of the dose received. In order of increasing dose the symptoms are fever, internal bleeding, vomiting and diarrhoea, tremors, cramps and coma. The dose threshold for the onset of radiation sickness is about 1 Sv, and for a dose of 4 Sv about 50 per cent of the victims would probably die. Few people could survive an exposure of more than 6 Sv. Radiation sickness may not be evident for up to two weeks following exposure, depending on the dose, and death, if it occurs, may not happen until eight weeks after exposure. In cases of very severe exposure, however, death may follow within two days.

Delayed effects take the form of cancers and hereditary effects, and in discussing them the following terms should be noted:

1. Somatic effects: those effects in which the radiation induced sickness appears in the irradiated person.
2. Hereditary effects: those effects which are evident only in the children of the irradiated person as a result of radiation damage to the reproductive organs.
3. Stochastic effects: those effects for which the probability or occurrence is regarded as a function of the dose of radiation received.

9.3 The effects of inhaled and ingested radioactivity

Studies of the effects of accidents involving the release of radioactivity show that, with the exception of those very close to the scene of the accident at the time it happens, the main hazard to the population as a whole arises from the inhalation or ingestion of air or food which is contaminated by the radioactive material from the accident. For example, the accident at the Russian nuclear power station at Chernobyl in April 1986 produced severe acute somatic effects in a few hundred people, namely the power plant operators and firemen exposed to radiation at the time of the accident and the subsequent fire. In addition, many thousands of people may suffer in years to come from delayed stochastic effects from the radioactive debris of the accident (principally fission products) which was carried in the

atmosphere for hundreds of miles from Chernobyl and deposited on the ground by rain, where it entered the food chain of humans and animals. Fallout from nuclear weapons testing produces a similar hazard.

In order to make calculations about the risks of such exposure, it is necessary to consider the following points:

1. How much of each radioactive isotope is inhaled or ingested.
2. The organs of the human body in which each isotope may become concentrated.
3. The dose which each organ receives from a given quantity of inhaled or ingested radioactive material.
4. The dose–risk coefficient, known as the weighting factor, for each organ.

The first point depends on the amount of radioactivity released and its subsequent dispersion in the atmosphere and return to ground level. The second and third points have been the subject of medical research, and the following table shows the organs affected by each of three important radioactive fission products, and the dose (Sv) which results from the inhalation or ingestion of 1 GBq of the fission product concerned.

Table 9.2. The dose due to certain fission products

Isotope	Principal organ(s) affected	Inhalation factor (Sv/GBq)	Ingestion factor (Sv/GBq)
^{90}Sr	(i) Red bone marrow	330	190
	(ii) Bone	730	420
^{131}I	Thyroid	290	480
^{137}Cs	Whole Body	8	13

Source: ICRP 30, Supplement 1, 1979

The weighting factors mentioned above are the ratios of the stochastic risk resulting from radiation damage to an individual organ to the total risk when the body is uniformly irradiated. Values of the weighting factors recommended by the ICRP are given in Table 9.3.
These factors can be used to define the effective dose equivalent as

$$\text{Effective dose equivalent} = \sum_{\substack{\text{all}\\ \text{affected}\\ \text{tissues}}} \left(\begin{matrix}\text{Dose received by a}\\ \text{particular tissue}\end{matrix}\right) \times \left(\begin{matrix}\text{Weighting factor}\\ \text{for that tissue}\end{matrix}\right)$$

The effective dose equivalent is subject to the same recommendations and limitations as those mentioned earlier for the whole body dose equivalent.

Table 9.3. Weighting factors for individual organs

Tissue	Weighting factor
Red bone marrow	0·12
Bone surface	0·03
Lung	0·12
Thyroid	0·03
Breast	0·15
Gonads	0·25
Remainder	0·30

The most important delayed effect of radiation is cancer, which is serious and may be fatal. Cancer is, however, a relatively common disease from which millions of people die each year. Any person exposed to radiation has an increased probability of contracting cancer, this extra probability being dependent on the radiation dose received. The linear dose–risk relationship assumes (without complete justification) that the increased probability of serious effect such as cancer is proportional to the radiation dose received. This hypothesis enables risk factors to be expressed for exposure of various parts of the body to radiation.

The risk factor expresses the additional number of people in a population sample of stated size who are likely to suffer serious effects if the members of this population are each exposed to an additional dose of 1 Sv. For example, a risk factor of 1 in 200 per Sv or 5×10^{-3} per Sv means that in a group of 200 people each exposed to 1 Sv of radiation to a particular part of their bodies, it is probable that one person will suffer serious effects as a result of this exposure.

The ICRP has made the following recommendations for risk factors for serious hereditary effects and fatal cancers resulting from the irradiation of various tissues and organs.

Table 9.4. Risk factors for irradiation

Tissue or organ	Effect	Risk factor/Sv
Breast	Cancer	$2{\cdot}50 \times 10^{-3}$
Red bone marrow	Leukaemia	$2{\cdot}00 \times 10^{-3}$
Lung	Cancer	$2{\cdot}00 \times 10^{-3}$
Thyroid	Cancer	$5{\cdot}00 \times 10^{-4}$
Bone surfaces	Cancer	$5{\cdot}00 \times 10^{-4}$
Other tissues	Cancer	$5{\cdot}00 \times 10^{-3}$
Whole body, all cancer effects		$1{\cdot}25 \times 10^{-2}$
Gonads	Hereditary	$4{\cdot}00 \times 10^{-3}$
Whole body, all health effects		$1{\cdot}65 \times 10^{-2}$

The data in Tables 9.2, 9.3 and 9.4 can now be used in the following

example to calculate the additional probability that a person who drinks milk contaminated with ^{131}I having an activity of 1000 Bq/litre of milk at the rate of 0·5 litres per day for two weeks will develop a fatal cancer as a result.

The dose due to the quantity of contaminated milk is:

$$0{\cdot}5 \times 1000 \times 14 \times 480 \times 10^{-9} = 3{\cdot}36 \times 10^{-3}\ \text{Sv}$$

Since ^{131}I is concentrated in the thyroid, for which the weighting factor (Table 9.3) is 0·03, the effective dose equivalent is approximately 10^{-4} Sv.

Using the risk factor (Table 9.4) of $1{\cdot}25 \times 10^{-2}$ for all cancer causing effects, the probability of a person suffering a fatal cancer is

$$10^{-4} \times 1{\cdot}25 \times 10^{-2} = 1{\cdot}25 \times 10^{-6},$$

or approximately one person in a million who drink the contaminated milk is likely to suffer a fatal cancer.

One uncertainty associated with the above calculation is the impossibility of checking its accuracy, since the risk from the dose specified is so small. Any increase in mortality rates from small radiation doses is lost in the normal incidence of death from cancer, which is about 1 in 4 of the population, and only a minute fraction of these deaths is due to natural or man-made radiation.

The allowable quantity of any radioactive isotope which may be taken into the body per year, either by ingestion or inhalation, and results in a whole body effective dose equivalent of 0·05 Sv is called the annual limit of intake (ALI). This quantity is expressed in terms of its radioactivity.

The derived airborne concentration (DAC) is the concentration of a radioactive isotope in air which would result in a person inhaling one ALI in a year. In calculating this quantity it is assumed that the person is occupationally exposed to the radioactive substance in the atmosphere of his working environment which he breathes for 250 days per year at the rate of 10 m^3 per 8 hour working day. Values of annual limits of intake and derived airborne concentrations published by the ICRP are given in Table 9.5.

In concluding this section, it is emphasized that for workers who are occupationally exposed in a radiation environment the total dose rate from external exposure, inhalation and ingestion must not exceed 0·05 Sv/year.

Table 9.5. Annual limits of intake and derived airborne concentrations

Isotope	ALI by ingestion (Bq)	DAC (Bq/m^3)
^{90}Sr	1×10^6	300
^{131}I	1×10^6	700
^{137}Cs	4×10^6	2000

Usually the dose rate is kept well below this level, 0·015 Sv/year being a typical value permitted at nuclear installations. Actual dose rates recorded may be even less. For example, at the fuel processing works of British Nuclear Fuels Ltd at Sellafield and the adjacent Calder Hall nuclear power station the average annual dose of the workers in the years 1977 to 1983 was in the range from 6 to 10 mSv (NRPB – R173, 1984).

Studies of mortality among radiation workers in the UK have shown no detectable excess cancer deaths, in spite of the fact that these workers regularly receive small doses of radiation. Some authors believe that the ICRP assumption of a linear relationship between dose and risk is pessimistic at low doses. It has even been suggested that low doses of radiation might be beneficial to health. A minority of researchers take the opposite view that the ICRP assumption underestimates the risk at low doses.

The effective dose equivalent for members of the public arising from nuclear installations is kept below 1 mSv per year per person for the most exposed group of individuals, and the dose received by the public as a whole from this source is much less than this level, being about 0·002 mSv/year per person in the early nineteen-eighties. This figure may be compared with natural background radiation which, as stated earlier, leads to an average dose of 2 mSv/year per person.

9.4 The interaction of gamma radiation and matter

The three processes known as Compton scattering, pair production and the photoelectric effect have already been mentioned in Chapter 1. The rates at which these processes occur depend on the cross-sections or absorption coefficients for each process. These cross-sections are similar to neutron cross-sections and will be considered in detail shortly.

At low gamma ray energies (less than 0·5 MeV) the most important process is the photoelectric effect. In this interaction the photon interacts with an orbital electron and transfers all its energy to it. The electron is ejected from its orbit and the photon disappears and may be considered to be absorbed. The photoelectric cross-section decreases rapidly as the photon energy E_γ increases, being roughly proportional to $E_\gamma^{-7/2}$. The cross-section also depends on the atomic number of the target material, being proportional to Z^n, where the index n has (depending on the photon energy and the element in question) a value between 3 and 5. In general the photoelectric effect is most important with heavy elements and low photon energies.

The pair production process occurs with gamma radiation of energy greater than 1·02 MeV, and the photon disappears with the production of a positron and an electron. The positron is shortlived, and when it annihilates itself with an electron two gamma photons of 0·51 MeV energy are formed. These can generally be neglected from the shielding point of view because of their low energy and isotropic emission. Thus pair production can be regarded as an absorption process like the photoelectric effect. The threshold energy for pair production is 1·02 MeV, which is equivalent to the mass of a positron and an electron. The cross-section is roughly proportional to Z^2. In view of this, it is evident that pair production is more important in heavy than light elements.

In the Compton scattering process it may be assumed that a photon collides with an orbital electron and that as a result of the collision another photon of lower energy emerges. The difference between the energies of the incident and emergent photons is transferred to the electron. This is not an absorption reaction as a photon of reduced energy remains after the interaction. The Compton scattering cross-section per atom is inversely proportional to the energy E_γ in the range 0·5 to 2 MeV, and is proportional to the atomic number Z of the material.

The total cross-section per atom for gamma ray interaction is the sum of the three individual cross-sections:

$$\sigma_t = \sigma_{PE} + \sigma_{PP} + \sigma_{CS} \tag{9.1}$$

The energy absorption cross-section, which gives a measure of the rate at which the energy of gamma radiation is attenuated, is the sum of the photoelectric and pair production cross-sections plus a fraction of the Compton scattering cross-section which represents the average fraction f of the photon energy lost in this process:

$$\sigma_{tE} = \sigma_{PE} + \sigma_{PP} + f\sigma_{CS} \tag{9.2}$$

The fraction f may be determined by analysing the Compton scattering process.

The cross-sections per atom are analogous to microscopic neutron cross-sections. It is more usual to use absorption coefficients which are analogous to macroscopic neutron cross-sections. Thus, if the number of atoms per cubic centimetre of an element is N, these absorption coefficients are:

the linear absorption coefficient, $\mu = N\sigma_t$; and
the linear energy absorption coefficient, $\mu_E = N\sigma_{tE}$.

The units of linear absorption coefficients are cm^2/cm^3 or cm^{-1}.

The cross-sections per unit mass are known as mass absorption coefficients, and are obtained by dividing the linear absorption coefficients by the material density (g/cm^3). Thus the mass absorption

coefficient is $\chi = \mu/\rho$, and the mass energy absorption coefficient is $\chi_E = \mu_E/\rho$ cm²/g.

The rate of interaction of gamma radiation in matter is given by equations that are similar to those for neutron interactions. Consider a collimated beam of gamma radiation of intensity ϕ_γ photons/cm² s incident on a target material of thickness dx and area A normal to the beam, containing N atoms/cm³. The following equations apply:

the rate of interaction in the target = $N\sigma_t\phi_\gamma A\ \mathrm{d}x$,
the rate of interaction per unit volume = $N\sigma_t\phi_\gamma = \mu\phi_\gamma$,
the rate of energy absorption per unit volume = $\mu_E\phi_\gamma E_\gamma$,

where E_γ is the energy of the gamma radiation:

the rate of interaction per unit mass = $\chi\phi_\gamma$; and
the rate of energy absorption per unit mass = $\chi_E\phi_\gamma E_\gamma$.

The attenuation of the beam of gamma radiation is given by the equation:

$$\phi_\gamma(x) = \phi_{\gamma 0}\, e^{-\mu x} \tag{9.3}$$

where $\phi_\gamma(x)$ is the flux of radiation of initial intensity ϕ_{γ_0} which penetrates a distance x into the target material without interacting. This equation assumes that all interaction processes result in the gamma radiation being removed from the original beam.

Figure 9.1 shows the variation of the individual and total absorption coefficients for aluminium and lead, which are typical light and heavy elements. It is of interest to note that for radiation of about 1 or 2 MeV

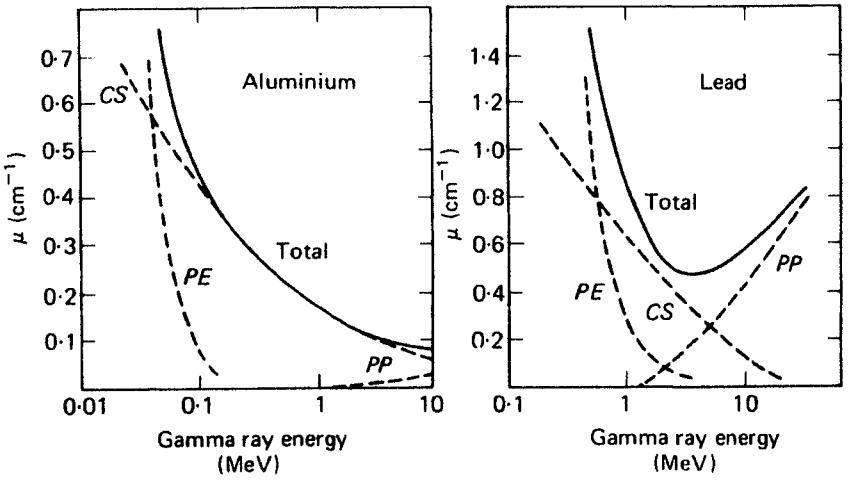

Figure 9.1. Gamma radiation absorption coefficients for aluminium and lead

energy, the Compton scattering process predominates for both elements, and the total cross-section is approximately equal to the Compton scattering cross-section. The Compton scattering cross-section is proportional to the atomic number Z, which for most elements is approximately half the atomic mass (hydrogen and the very heavy elements are exceptions). According to Avogadro's Hypothesis the number of atoms per unit mass of any element is inversely proportional to its atomic mass. Consequently the Compton scattering cross-sections per unit mass, and thus the mass absorption coefficients, of most elements are nearly equal for gamma radiation of about 1 MeV energy. The value is about 0·06 cm²/g. The same is true for mass energy absorption coefficients, whose value at 1 MeV is about 0·027 cm²/g. From this it is evident that equal masses of all materials have approximately the same shielding effect for gamma radiation of 1 or 2 MeV energy.

9.5 The shielding of sources of gamma radiation

Consider a point source of gamma radiation, for example a small capsule of a radioactive isotope, emitting S photons per second isotropically. The absorption of radiation in air may be neglected because of the very low density of air, and if the source is unshielded, the flux of gamma radiation at a distance r from the source is:

$$\frac{\text{The number of photons emitted per second by the source}}{\text{The surface area of a sphere of radius } r}$$

Therefore

$$\phi_\gamma = \frac{S}{4\pi r^2} \tag{9.4}$$

If the source is shielded by a spherical shield of thickness x (see Figure 9.2), then according to equation (9.3) the fraction of gamma photons which pass through the shield without interacting is $e^{-\mu x}$. Combining

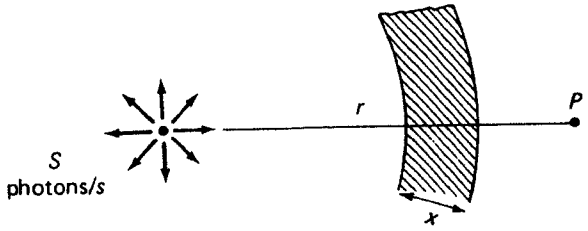

Figure 9.2. Shielding of a point source of radiation

equations (9.3) and (9.4), the flux of gamma radiation at a point P distant r from a source emitting S photons per second, shielded by a

thickness x of material whose linear absorption coefficient is μ is given by the equation:

$$\phi_\gamma = \frac{S}{4\pi r^2} e^{-\mu x} \qquad (9.5)$$

This equation neglects the effect of multiple scattering in the shield as a result of which a photon after one or more scattering collisions may contribute to the flux at P. This effect will be discussed later, however equation (9.5) is valid if the thickness of the shield is small enough for multiple scattering to be insignificant.

The dose rate due to a point source of gamma radiation may now be determined. Consider an unshielded source of strength C curies emitting one photon per disintegration of energy E_γ. The flux at a distance 1 metre from the source is, using equation (9.4):

$$\phi_\gamma = \frac{3{\cdot}7 \times 10^{10} \times C}{4\pi \times 10^4}$$

$$= 2{\cdot}94 \times 10^5 \times C \quad \text{photons/cm}^2\,\text{s}$$

Taking a value of $0{\cdot}027$ cm^2/g for χ_E, which is approximately correct for most materials with gamma radiation of energy about 1 MeV, the rate of energy absorption per gramme is:

$$0{\cdot}027 \times 2{\cdot}94 \times 10^5\, CE_\gamma$$

$$= 7{\cdot}94 \times 10^3\, CE_\gamma \text{ MeV/g s}$$

Expressed in the S.I. unit of absorbed dose, this expression becomes:

$$\text{Absorbed dose rate} = 7{\cdot}94 \times 10^3 \times 3600 \times 1{\cdot}6 \times 10^{-13}\, CE_\gamma$$
$$= 4{\cdot}57 \times 10^{-3}\, CE_\gamma \text{ Gy/h}$$
$$= 4{\cdot}57\, CE_\gamma \text{ mGy/h}$$

This equation gives the dose rate 1 metre from an unshielded point, source of radiation of strength C curies and energy E_γ MeV per photon. A more general expression for the dose rate due to a shielded point source is obtained from equation (9.5). At a distance r metres from the source, if the shield thickness is x cm and its linear absorption coefficient μ cm^{-1}, the dose rate is:

$$D = \frac{4{\cdot}57 C E_\gamma e^{-\mu x}}{r^2} \quad \text{mGy/h}$$

Finally, it is easy to verify that in terms of the flux and the energy of gamma radiation, and the mass energy absorption coefficient, the dose rate is given by the equation:

$$D = 5{\cdot}76 \times 10^{-4}\, \phi_\gamma E_\gamma \chi_E \quad \text{mGy/h}$$

Equation (9.5) for a point source of radiation can be used to solve problems involving more complicated source geometries. Let us consider, for example, an irradiated fuel element which after undergoing fission in a reactor for a long time is intensely radioactive due to the build up of fission products in the fuel. Such a fuel element might be a rod 2·5 cm diameter and 1 metre long, and it can be regarded as a finite line source of radiation. Figure 9.3 shows the fuel element enclosed in a container of wall thickness x.

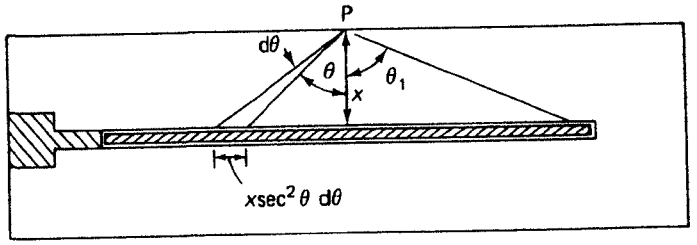

Figure 9.3. A shielded fuel element – a line source of radiation

If the activity of the source is S photons/cm s, and, referring to Figure 9.3, the element of length of the source is $x \sec^2\theta\, d\theta$, then the flux of gamma radiation at the joint P on the surface of the container is given by:

$$\phi_\gamma = 2\int_0^{\theta_1} \frac{S \times \sec^2\theta\, d\theta}{4\pi x^2 \sec^2\theta} e^{-\mu x \sec\theta}$$

$$= \frac{S}{2\pi x}\int_0^{\theta_1} e^{-\mu x \sec\theta}\, d\theta \qquad (9.6)$$

$\int_0^{\theta_1} e^{-\mu x \sec\theta}\, d\theta$ is known as the Secant Integral, sec i $(\mu x, \theta_1)$ and values of this integral for various values of θ_1 and μx are tabulated in the radiation shielding literature.

Finally, we will consider an infinite plane source of radiation emitting S photons/cm^2 s isotropically, with an infinite slab shield of thickness x, see Figure 9.4.

The area of a 'ring element' of the source subtended by an angle $d\theta$ at P is $2\pi r^2 \sin\theta \sec\theta\, d\theta$, and the flux of gamma radiation at point P due to the 'ring source' is given by:

$$d\phi_\gamma = \frac{2\pi S r^2 \sin\theta \sec\theta\, d\theta}{4\pi r^2} e^{-\mu x \sec\theta}$$

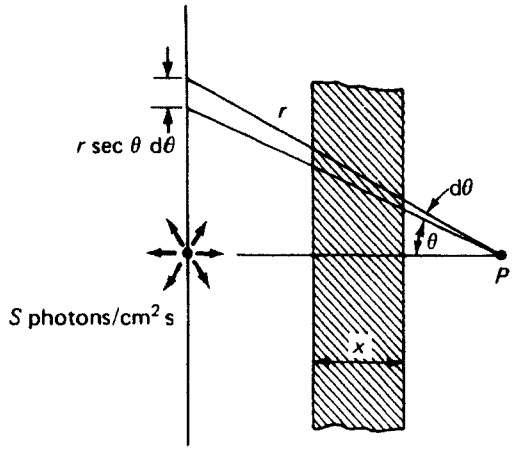

Figure 9.4. An infinite plane source of radiation

The flux of gamma radiation at P due to the entire plane source is given by:

$$\phi_\gamma = \frac{S}{2}\int_0^{\pi/2} \tan\theta\, e^{-\mu x \sec\theta}\, d\theta$$

If $y = \mu x \sec\theta$, $dy = \mu x \tan\theta \sec\theta\, d\theta$, and the preceding equation for the flux becomes:

$$\phi_\gamma = \frac{S}{2}\int_{\mu x}^{\infty} \frac{e^{-y}}{y}\, dy \qquad (9.7)$$

$\int_{\mu x}^{\infty} (e^{-y}/y)\, dy$ is known as the Exponential Integral, $E_1(\mu x)$, and values of this integral for various values of μx are tabulated in shielding literature.

The preceding equations (9.5), (9.6) and (9.7) refer to the uncollided flux of gamma radiation at the point of measurement P. If the beam is broad, or the shield thick enough to cause multiple scattering, the actual flux is greater than the uncollided flux due to the fact that Compton scattering may result in some photons contributing to the flux at P after one or more such collisions. This effect is known as build-up, and allowance for it is made by the introduction of build-up factors.

The flux build-up factor, B_F, is defined by the equation

$$B_F = \frac{\text{Total gamma flux at the point of interest}}{\text{Uncollided gamma flux at the point of interest}}$$

and the equation for the attenuation of a parallel beam of gamma radiation, (9.5), is modified to become:

$$\phi_\gamma = \phi_{\gamma_0}\, e^{-\mu x}\, B_F$$

The dose build-up factor, B_D, is defined by the equation

$$B_D = \frac{\text{Total dose at the point of interest}}{\text{Dose due to uncollided gamma radiation only}}$$

The determination of build-up factors depends on the following points:

(i) the type of build-up of interest (flux or dose),
(ii) the material through which the radiation is passing,
(iii) the thickness of the absorber,
(iv) the energy of the gamma radiation,
(v) the source geometry (point or distributed).

It is possible by the numerical analysis of multiple scattering collisions to calculate build-up factors, and the results of these calculations are shown in Figure 9.5 for water and lead. The abscissa of these

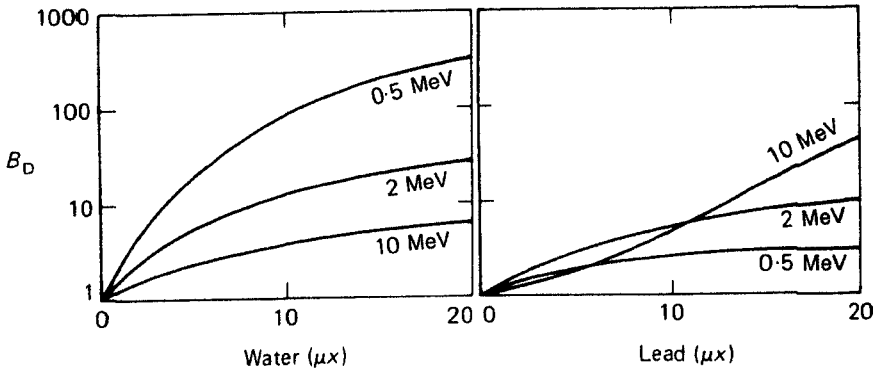

Figure 9.5. Build-up factors for a point source of radiation in water and lead

graphs are in units of (μx), the number of mean free paths through the absorber. It is noticeable that the lighter material, water, has in general larger build-up factors because Compton scattering (which is the process responsible for build-up effects) is more significant in light than heavy elements.

From the results of the numerical analysis it has been found possible to express build-up factors as the sum of two exponential terms by the equation:

$$B_D = A\, e^{-\alpha_1 \mu x} + (1 - A)\, e^{-\alpha_2 \mu x} \qquad (9.8)$$

where A, α_1 and α_2 are constants for any material and gamma ray energy, and are selected to fit the numerical results referred to above. Equation (9.8) may be incorporated into the uncollided flux equations;

for example, for the purposes of calculating dose rate, equation (9.5) becomes:

$$\phi_\gamma = B_D \frac{S}{4\pi r^2} e^{-\mu x}$$

$$= \frac{S}{4\pi r^2} \{A\, e^{-(1+\alpha_1)\mu x} + (1 - A)\, e^{-(1+\alpha_2)\mu x}\} \qquad (9.9)$$

The equation for the flux with build-up is thus similar to that for the uncollided flux, with amended exponents and weighting factors.

This formulation is very convenient if integration is carried out over a distributed source, for example the infinite plane source considered earlier. For such a source the uncollided flux at point P due to an elementary 'ring source' was previously shown to be given by (refer to Figure 9.4).

$$d\phi_\gamma = \frac{S}{2} \tan\theta\, e^{-\mu x \sec\theta}\, d\theta$$

With build-up taken into account for the purpose of calculating dose rate:

$$d\phi_\gamma = \frac{S}{2} \tan\theta\, e^{-\mu x \sec\theta}\, B_D\, d\theta$$

and expressing B_D by equation (9.8):

$$d\phi_\gamma = \frac{S}{2} \tan\theta \{A\, e^{-(1+\alpha_1)\mu x \sec\theta} + (1 - A)\, e^{-(1+\alpha_2)\mu x \sec\theta}\}\, d\theta$$

Making the substitution $y = \mu x \sec\theta$ and integrating, we get:

$$\phi_\gamma = \frac{S}{2} \int_{\mu x}^{\infty} \sum_{i=1}^{2} \frac{A_i}{y} e^{-(1+\alpha_i)y}\, dy$$

$$= \frac{S}{2} \sum_{i=1}^{2} A_i E_1(\mu_i x) \qquad (9.10)$$

where $E_1(\mu_i x)$ is the Exponential Integral previously introduced, and $A_1 = A$, $A_2 = (1 - A)$, $\mu_1 = (1 + \alpha_1)\mu$, and $\mu_2 = (1 + \alpha_2)\mu$.

Comparing equations (9.9) and (9.10) with (9.5) and (9.7), it is evident that the general rule for the application of build-up factors of the form given by equation (9.8) is:

If the uncollided flux is expressed by the equation:

$$\phi_\gamma = F(r)G(\mu x)$$

where $F(r)$ is a function of the distance from the source to the point of measurement, and $G(\mu x)$ is a function of the shield thickness, then the total flux at the same point is given by:

$$\phi_\gamma = F(r) \sum_{i=1}^{2} A_i G\{(1 + \alpha_i)\mu x\} \qquad (9.11)$$

As an example of the application of build-up factors, we will consider an irradiated fuel element from a nuclear reactor, stored under water in a cooling pond. The radioactive fission products emit gamma radiation with a wide spectrum of energies, however we will simplify matters by considering radiation of a single energy.

Example: An irradiated fuel element 70 cm long, which may be regarded as a uniform line source of radiation, is stored under 3 m of water. The activity of the fuel element is 2000 curies, and the energy of the emitted radiation is 2 MeV per photon. Calculate the dose rate at the surface of the water immediately above the fuel element.

For gamma radiation of 2 MeV energy, μ for water = 0·0493 cm^{-1}, and the values of the constants in equation (9.8) are A = 6·4, α_1 = −0·076 and α_2 = 0·092.

For the given geometry (see Figure 9.3):

$$\tan\theta_1 = \frac{35}{300} = 0{\cdot}117 \qquad \theta_1 = 6{\cdot}7°$$

and

$$\mu x = 300 \times 0{\cdot}0493 = 14{\cdot}8$$

$$\phi_\gamma = \frac{2000 \times 3{\cdot}7 \times 10^{10}}{70 \times 2\pi \times 300} \times$$

$$[6{\cdot}4 \sec i\{(1 - 0{\cdot}076)14{\cdot}8, 6{\cdot}7°\} - 5{\cdot}4 \sec i\{(1 + 0{\cdot}092)14{\cdot}8, 6{\cdot}7°\}]$$

The values of the Secant Integrals are tabulated in handbooks of reactor shielding:

$$\sec i(13{\cdot}68, 6{\cdot}7°) = 1{\cdot}3 \times 10^{-7}$$

and

$$\sec i(16{\cdot}18, 6{\cdot}7°) = 1{\cdot}1 \times 10^{-8}$$

Using these values:

$$\phi_\gamma = 434 \text{ photons/cm}^2 \text{ s at the surface of the water}$$

The mass energy absorption coefficient for 2 MeV radiation in air is 0·023 cm^2/g, and the dose rate is (using the equation derived earlier):

$$D = 5{\cdot}76 \times 10^{-4} \times 434 \times 2 \times 0{\cdot}023$$

$$= 0{\cdot}0115 \text{ mGy/h}$$

9.6 Reactor shielding

The problem of reactor shielding is a great deal more complicated than that of shielding a simple source of gamma radiation. In a reactor the principal source of radiation, neutrons and gamma radiation, is in the core itself, however radioactive coolant flowing in ducts and heat exchangers, and irradiated fuel elements in storage pits may provide additional sources of gamma radiation.

Neutrons and gamma radiation produced in the core have a wide spectrum of energies, and neutrons passing from the core into the shield produce more gamma radiation as a result of (n, γ) reactions in the shield. Beta radiation is produced by fission product decay, but only high energy beta radiation is important because as it is slowed down, gamma radiation known as Bremsstrahlung (braking radiation), is produced.

The important sources of radiation in a reactor can be summarized as follows:

1. *Neutrons.* The characteristics of prompt fission neutrons and delayed neutrons have been described in Chapters 2 and 8.
2. *Prompt gamma radiation.* This is the gamma radiation emitted at the instant of fission; the average number of photons emitted per fission is 8 with an energy spectrum up to about 7 MeV. The average energy per photon is about 1 MeV.
3. *Fission product gamma radiation.* This is the gamma radiation emitted during the decay of radioactive fission products. The radiation from short-lived fission products is important during reactor operation and rapidly decays when the reactor is shut down. The radiation from long-lived fission products may be of more importance after shutdown, particularly in the cooling of fuel in the core and the shielding of irradiated fuel elements.
4. *Capture gamma radiation.* This radiation is emitted as a result of (n, γ) reactions in the core and (to a lesser extent) in the shield. It provides a source of radiation in the shield itself.
5. *Activation product gamma radiation.* If the product of an (n, γ) reaction is radioactive, it will provide a further source of gamma radiation as it decays. Reactor coolants which become radioactive as a result of (n, γ) reactions in the core are a source of radiation in external ducts and heat exchangers both during operation and after shutdown. Sodium is an important example of such a coolant.

Less important sources of radiation are:

1. *Gamma radiation*—from the inelastic scattering of fast neutrons.
2. *Photo-neutrons*—resulting from (γ, n) reactions in reactors containing significant quantities of heavy water of beryllium. (^{2}H and ^{9}Be are the isotopes responsible for this reaction.)

3. *Annihilation radiation.* This is the name given to the 0·51 MeV gamma radiation which results from the annihilation of positrons with electrons. The positrons may originate from high energy gamma radiation undergoing pair production processes.
4. *Bremsstrahlung.* This gamma radiation is produced by the deceleration of high energy beta particles. It may be of importance if ^{7}Li is used as a reactor coolant because neutron capture in ^{7}Li produces ^{8}Li which decays by the emission of high energy beta particles.

The main radiation shield of a reactor is known as the biological shield. Its function is to reduce the intensity of neutrons and gamma radiation escaping from the reactor to a level which is acceptable from the health physics point of view. The design of the biological shield involves the following steps:

1. Determine the neutron flux distribution in the core of the reactor, and hence determine the distribution of sources of gamma radiation in the core, and the energy spectrum of these sources.
2. Determine the current and energy of neutrons leaking out of the core into the shield, and hence the rate of slowing down and absorption of neutrons in the shield. This establishes the sources of gamma radiation in the shield.
3. The neutron current through the outer surface of the shield is determined.
4. Once the sources of gamma radiation in the core and shield are known, the gamma flux and energy in the shield, and the current through its outer surface are determined.
5. The dose rate at the outer surface of the shield is calculated from the neutron and gamma currents through it. The shield thickness is designed so that this dose rate is within acceptable limits.
6. The rate of energy absorption of gamma radiation and neutrons in the shielding must be determined. This energy absorption results in the heating of the shield, and in the case of power reactors this heating may lead to unacceptable temperatures and thermal stresses.

To reduce the heating effect in the biological shield, a comparatively thin thermal shield is placed between it and the core of the reactor. The function of the thermal shield, which is cooled by a suitable flow of coolant, is to reduce the neutron and gamma currents from the core into the biological shield, and thereby reduce the rate of energy absorption and the temperature rise in this shield. The thermal shield should have a good thermal conductivity, and be able to withstand thermal stresses. Steel is commonly used as the material of the thermal shield.

The biological shield should contain some hydrogen compound to slow down fast neutrons, and be dense enough to attenuate gamma

radiation effectively. Concrete satisfies both these requirements fairly well and is suitable for land-based reactors. Barytes concrete, containing the heavy element barium, and steel-shot concrete have been used for biological shields. They are more dense than ordinary concrete, with improved shielding properties, however their higher cost offsets this advantage. The biological shield for a marine reactor, which is usually a fairly compact pressurized water reactor, must satisfy a minimum space and weight requirement. This leads to a shield design which consists typically of alternate layers of water (for fast neutron slowing) and steel (for gamma ray attenuation).

9.7 Shield heating

As we have seen, the slowing down of neutrons and the absorption of gamma radiation are energy absorbing processes in which the energy lost by the radiation is transferred to the material through which it is passing, and results in an increase in the temperature of the material. The first step in determining the temperature rise in a reactor shield involves a knowledge of the distribution of the energy sources, i.e. the spatial variation and the energy spectrum of neutrons and gamma radiation in the shield. Once the distribution of energy sources is known, the temperature variation in the shield may be found by application of the general heat conduction equation, equation (6.5), with suitable boundary conditions.

As a simple illustrative example we will consider the effect of a parallel beam of gamma radiation passing through a slab-shaped shield. The spatial variation of the energy source, assuming exponential attenuation of the radiation, is given by:

$$H(x) = 1{\cdot}6 \times 10^{-13}\,\mu_E E_\gamma \phi_{\gamma 0}\, e^{-\mu x}\ \mathrm{W/cm^3}$$

the x coordinate being normal to the face of the shield.

The heat conduction equation for this problem is (neglecting temperature variation in the y and z directions):

$$\frac{\mathrm{d}^2T}{\mathrm{d}x^2} = -\frac{C}{k_s}e^{-\mu x} \tag{9.12}$$

where $C = 1{\cdot}6 \times 10^{-7}\,\mu_E E_\gamma \phi_{\gamma_0}$, and k_s is the thermal conductivity of the shield material, W/m °C.

Equation (9.12) is integrated twice, and the boundary conditions applied that $T = T_w$, the temperature of the inner and outer surfaces of the shield when $x = 0$ and L, the thickness of the shield. The solution for T is:

$$T - T_w = \frac{C}{\mu^2}\left[(1 - e^{-\mu x}) - \frac{x}{L}(1 - e^{-\mu L})\right] \tag{9.13}$$

The magnitude of this effect can be illustrated by considering a concrete shield 2 metres thick with an incident energy flux of gamma radiation of 10 mW/cm². (These figures are typical of a power reactor of the Calder Hall type which has a 15 cm thick steel thermal shield inside the concrete biological shield.) The linear absorption coefficient and thermal conductivity of concrete are taken as 0·085 $\mathrm{cm^{-1}}$ and 2·0 W/m °C respectively, and it is assumed that all gamma interactions are absorption processes. If equation (9.13) is evaluated for these figures, the result for the maximum concrete temperature is:

$$T_{max} - T_w = 4{\cdot}3°\mathrm{C}$$

This temperature difference is quite acceptable from the point of view of thermal stresses. However, to consider again the example of the Calder Hall reactor, if there were no thermal shield then the energy flux entering the biological shield would be about five or ten times greater, and the maximum temperature difference would be about ten times greater than the figure just calculated, namely about 40°C. This temperature difference would cause a significant thermal stress in the concrete shield, and it is to prevent this that the thermal shield is used.