NUCLEAR
ENGINEERING
MONOGRAPHS

NUCLEAR REACTOR SHIELDING

BY

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TEMPLE PRESS

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TEMPLE PRESS LIMITED
BOWLING GREEN LANE, LONDON E.C.1

FIRST PUBLISHED 1958

Made and printed in Great Britain by
William Clowes and Sons, Limited, London and Beccles

MADE IN GREAT BRITAIN

Preface

With the expansion of nuclear power into commercial and mobile reactor systems, the interest in reactor shielding has increased greatly. It is now necessary to design efficient reactor shields which are economic in the first case, and which for mobile systems satisfy (for example) minimum weight requirements. Consequently much work both experimentally and theoretically has been, and is being undertaken in shielding, and the "art" of shielding is being replaced by a scientific approach.

This Monograph is intended to provide an introduction to the principles and practice of reactor shielding, and it is hoped that it will be of interest to those who are entering the reactor field and who require a short survey of the subject.

The author is indebted to many of his colleagues for stimulating discussions, and in particular to Mr. W. G. Clarke for his constructive criticism of the text. He is also indebted to many authors for permission to publish their work, and in particular to Messrs. Price, Horton and Spinney for permission to use data from *Radiation Shielding Handbook*, published by Pergamon Press Ltd., and to the U.S.A.E.C. for permission to reproduce Table 8 in the text.

J. R. HARRISON

October 1958

Notation

A = Atomic weight
A, α_1, α_2 = Taylor build-up coefficients
b = Average cosine of scattering angle
B = Binding energy per nucleon
B = Build-up factor
c = Velocity of light
D = Diffusion coefficient
E = Energy of neutron or γ-ray
Ei = Exponential integral
h = Planck's constant
J = Current, neutron or γ
K_1 = Bessel function
I = Radiation intensity
L = Thermal neutron diffusion length
L_s = Fast neutron slowing down length
m = Mass
N = Number of atoms/cm^3
N = Average number of collisions made by neutrons in slowing down
n = Neutron density (neutrons/cm^3)
P_0 = Reactor operating power
P_s = Shut down power
r = Distance (spherical or cylindrical geometry)
r_0 = Classical electron orbital radius
R = Activity of sample
R_0 = Initial activity
S = Source strength, neutron or γ
seci = Secant integral
T = Temperature
T = Decay time
T_0 = Irradiation time
t = Shield thickness
$t_{1/2}$ = Half-life
V = Volume
v = Neutron velocity
x = Distance (plane geometry)
Z = Atomic number
$\alpha = \left(\frac{A-1}{A+1}\right)^2$
θ = Neutron scattering angle in centre of mass co-ordinates
ξ = Mean logarithmic energy decrement
ρ = Density (gm/cm^3)
σ = Microscopic cross-section
Σ = Macroscopic cross-section ($= N\sigma$)
ϕ = Flux
$\bar{\phi}$ = Average flux
ψ = Neutron scattering angle in laboratory co-ordinates
ν = Number of neutrons/fission
μ = Linear absorption coefficient
μ_e = Energy linear absorption coefficient
λ = Mean free path, or relaxation length
λ = Decay constant

Contents

Chapter One

NUCLEAR REACTOR SHIELDING

In the operation of a nuclear reactor, practically all the fundamental particles are emitted in copious quantity; and since the interaction of these particles with matter results in damage, their intensity must be reduced to a safe level. A radiation shield is provided for this purpose, which for power reactors will be several feet thick. In this monograph, an elementary treatment of the problems encountered and their solution is given. It is obviously not possible to give great detail; if this is required reference should be made to the recently published book by Price, Horton and Spinney.[1] In this first chapter, we shall be concerned with the enumeration of the problems involved, and some basic considerations.

PROBLEMS OF SHIELD DESIGN

The main points which must be considered in the design of a reactor shield are

1. The radiation level which may be permitted at the shield surface. This will depend on the type of reactor (i.e. research or power), since the instrumental tolerance level may be less than for personnel; and if governed by the latter, consideration must be given to the period for which personnel will be at any particular face of the shield.

2. The type of shield required. The shield may have functions other than purely a radiation barrier—it may be an integral part of the structure of the system. In addition, the shield may be optimized for minimum weight (e.g. for mobile systems), minimum thickness if high neutron beam intensities or ease of insertion of materials into the core are required, or minimum cost. This last-named consideration must include the overall cost of the reactor structure and building, as well as the shield materials and placement.

3. The required thickness of shield is then determined, and the heat distribution from the absorption of radiation in the shield calculated. If the heat produced is excessive, a re-design of the shield is necessary.

4. Special problems related to the shield design must be considered, such as radiation streaming along coolant ducts, etc.

5. Provision of auxiliary shielding, such as for fuel elements on removal from the reactor, and for circulating fuel or coolant, etc.

In some of these problems, because of the inadequacy of the theoretical methods and data, it is necessary to include a safety factor, and make approximations in the analysis. However, oversafe-design results in unnecessary shield costs, and care should be taken that cumulative approximations do not lead to overlarge errors.

Interaction of Radiation with Matter

The mode of interaction of radiation with matter depends on the type of radiation, its energy and the material through which it is passing. In the interaction, ionization is produced which may be fairly diffuse (as in γ-ray interaction), or localized as for fast-moving heavy charged particles. The resulting damage in human beings may be either temporary or permanent, and a maximum permissible level (m.p.l.) of exposure to radiation of various types has been recommended by the Commission on Radiological Protection.[2] Before considering these recommendations, we shall discuss some definitions and units of radiation intensity.

Flux and Cross-section

If in a material there are n neutrons/cm^3 each moving with a velocity v cm/sec, the number of neutrons absorbed by the material per cm^3 sec is proportional to the number of atoms/cm^3 in the material, and the total distance travelled in one second by the neutrons present in one cm^3, i.e. the number of absorptions/cm^3 sec $= Nnv\sigma_a$ where N is the number of atoms/cm^3 and σ_a, known as the absorption cross-section, is a function of v and characteristic of the material. σ_a has units cm^2, and is generally measured in barns, where one barn $= 10^{-24}$ cm^2.

N, the number of atoms/cm^3 $= (0{\cdot}6025 \times \rho \times 10^{24})/A$, where ρ is the density and A the atomic weight of the material, and $0{\cdot}6025 \times 10^{24}$ is Avogadro's number.

$\phi = nv$ is known as the neutron flux, and $\Sigma = N\sigma_a$ the macroscopic absorption cross-section of the material, and has units cm^{-1}. ϕ is measured in units of neutrons/cm^2 sec. Hence the number of absorptions/cm^3 sec can be written $= \phi\Sigma_a$.

For collimated radiation, the flux is just the number of neutrons crossing unit area/sec placed normally to the beam.

Current

The neutron current ($\mathcal{J}$) in any direction is the number of neutrons crossing a unit area per second placed normally to that direction. It has units of neutrons/cm^2 sec, and in the special case of collimated radiation, the current is equal to the flux. Current is a measure of the net number of neutrons/unit area crossing a plane per second. In an isotropic flux the current in any direction is $\phi/4$.

These definitions of flux, current and cross-section hold equally well for γ-radiation. In certain instances, it is convenient to define the energy current, which is measured in MeV/cm^2 sec, and for example in the case of γ-radiation equals the γ-ray current multiplied by the energy of the photons.

UNITS OF RADIATION INTENSITY

The Röntgen (r)

The intensity of γ-radiation is measured by determining the total ionization per unit volume. The unit is the röntgen which is defined by that amount of X- or γ-radiation which will produce, as a result of ionization, one electrostatic unit of charge of either sign in one cubic centimetre of air at standard temperature and pressure (20°C and 76 cm Hg). Since the biological effect of X- or γ-radiation is proportional to the ionization produced, the unit can be applied as a measure of dose. It can be shown that one röntgen corresponds to an energy release of $5{\cdot}24 \times 10^{13}$ eV or 83·8 ergs per gramme of air. In tissue, it corresponds more accurately to an energy release of 93 ergs/g. Note that the röntgen is a total dose, and not a dose rate.

In dealing with neutrons, and for practical reasons γ-rays of energy greater than 3 MeV, it is necessary to introduce a new unit, the rad.

The Rad

The rad is the unit of absorbed radiation, and is defined in terms of the amount of energy deposited by the radiation at the place of interest. The integral absorbed dose, i.e. the integrated energy absorption over a given region, is expressed in g-rad. One rad corresponds to an energy deposition of 100 ergs/g. The röntgen is retained as the unit of absorbed radiation for γ-rays in the energy range 0–3 MeV; one röntgen equals 0·93 rad.

Relative Biological Efficiency (r.b.e)

The damage due to ionizing radiation depends not only on the amount of energy absorbed, but also on the microscopic density of ionization produced. For the same amount of energy absorbed, the dense local ionization produced by a fast proton is more damaging than the diffuse ionization produced by γ-radiation. This effect is taken into account by a unit known as the relative biological efficiency (r.b.e.) which is defined as

$$\frac{\text{the dose in rads to produce a given effect with } \gamma\text{-radiation}}{\text{the dose in rads to produce the same effect with the radiation considered}}.$$

Since the r.b.e. depends on the density of ionization, it would be expected to be larger for the heavily ionizing particles, and for fast neutrons, because the damage caused by fast neutrons depends on recoil particles (in particular protons).

Table 1 summarizes the r.b.e. values for the common radiations.

TABLE 1. RELATIVE BIOLOGICAL EFFICIENCY

Radiation	*r.b.e.*
γ-rays	1
Protons ($\rightarrow$ 10 MeV)	10
Fast neutrons ($\rightarrow$ 10 MeV)	10
α particles (internally)	10
Heavy recoil nuclei	20

Röntgen Equivalent Man (rem)

This is a unit in common use, and equals the product of rads multiplied by r.b.e. It is of use in summing the effects due to different types of radiation.

MAXIMUM PERMISSIBLE LEVEL (m.p.l.)

For the total body dose, in particular for neutrons and γ-radiation, the Commission on Radiological Protection [2] recommends that

1. The total dose received by any one person should not exceed 50 rem before the age of 30 years. If more than 2 per cent of the population might receive this dose, the figure should be reviewed.
2. The total dose received by any one person during his lifetime should not exceed 200 rem.
3. The maximum weekly dose is 0·3 rem for a consecutive period of 13 weeks.

The report gives fuller details for exposure of specific regions of the

TABLE 2. γ-RAY FLUX FOR 0·3r IN 40 hr (7·5 mr/hr)

γ energy (MeV)	0·05	0·1	0·5	1	2	3	4	5
Photons/ cm^2 sec	6×10^4	$4{\cdot}6\times10^4$	7×10^3	$3{\cdot}8\times10^3$	$2{\cdot}2\times10^3$	$1{\cdot}7\times10^3$	$1{\cdot}4\times10^3$	$1{\cdot}2\times10^3$

body. For a 40-hr working week, 3. above corresponds to a dose rate of $7 \cdot 5 \times 10^{-3}$ rem/hr or 7·5 mr/hr.

The shield designer works in terms of flux, and the fluxes for exposure to one m.p.l. for γ-rays and neutrons are given in Tables 2 and 3 respectively.

TABLE 3. NEUTRON FLUX FOR 0·3 rem IN 40 hr

Neutron energy	0·025 eV	1·0 eV	10 eV	0·1 MeV	0·5 MeV	1 MeV	2 MeV	3–10 MeV
n/cm² sec	2000	2000	1000	200	80	60	40	30

CURIE

The strength of a radioactive source is often expressed in *curies*. One curie can be defined as the disintegration rate of a radioactive source of $3 \cdot 7 \times 10^{10}$ disintegrations/sec. It should be noted that the unit refers to a disintegration rate of active nuclei, and (not necessarily) to a rate of emission of γ (or β) radiation. For example ^{60}Co decays with the emission of two γ-rays per disintegration, and thus one curie of ^{60}Co emits $7 \cdot 4 \times 10^{10}$ γ-rays/sec.

An alternative unit occasionally encountered is the rutherford, which is a disintegration rate of 10^6 disintegrations/sec.

Chapter Two

GAMMA-RAY ATTENUATION

The problem of X- and γ-ray shielding has been the subject of a considerable amount of work over several years. The nature of the interaction with matter is well understood, and is amenable to theoretical calculation. In this chapter, the sources of the radiation met with in reactor shielding are considered, and a discussion of the mechanism of interaction follows. The attenuation of narrow and broad beams of radiation is then described.

Radiation Sources

In determining the escape of γ-radiation from a shield, it is necessary to know the source strength and the energy of the radiation, since the absorption in the shield depends on the energy (and, of course, on the shield material). In a reactor, γ-radiation will be in the range 0·1–10 MeV, and will result from

(i) The fission process.
(ii) Fission products.
(iii) Radiative neutron capture, i.e. (n, γ) reactions, throughout the reactor system.
(iv) Decay of radioactive isotopes.
(v) Inelastic scattering of fast neutrons.
(vi) *Bremsstrahlung* and positron annihilation.

Of these, (v) and (vi) are relatively unimportant as sources of γ-radiation, and will not be considered in detail.

The Fission Process

γ-rays emitted at the instant of fission are known as prompt γ-rays, and several measurements of their energy distribution and total energy have been made.[1, 2] The various data are in good agreement for $E_\gamma > 0{\cdot}2$ MeV, and are summarized in Fig. 1 and Table 4. The energy distribution is independent of the fissile nucleus, and of the energy of the incident neutron for $E_n < 1{\cdot}0$ MeV. The total energy released per fission as prompt γ-rays has been measured at 7·5 MeV[1] and 12 MeV,[2] the difference being due to the low energy γ-rays. This difference is generally not important for shield problems, since the low energy photons are absorbed in the core, and do not enter the shield.

TABLE 4. PROMPT FISSION γ-RAYS

γ-energy (MeV)	Number of photons/fission
1	3·2
1·5	0·8
2·3	0·85
3	0·15
5	0·2

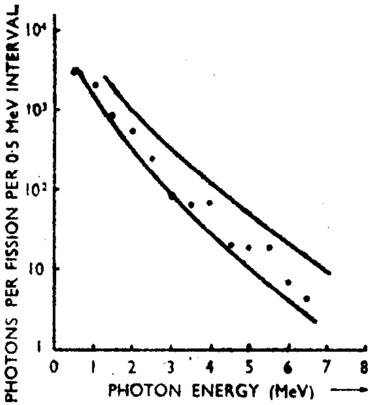


FIG. 1. *Gamma-rays per ^{235}U fission (from Price[9])*

Fission Product γ-rays

The distribution of fission products for the three fissile isotopes is well established, and their decay schemes known. By considering each individual product it should be possible to determine the β and γ yield after various times of irradiation and decay. This has been done for the γ-radiation by Moteff[3] in a report which gives graphs of the decay of different energy groups for various irradiation times. These results are reproduced in Fig. 2. This mode of presentation is very convenient for certain shielding calculations, as for example determining the shielding

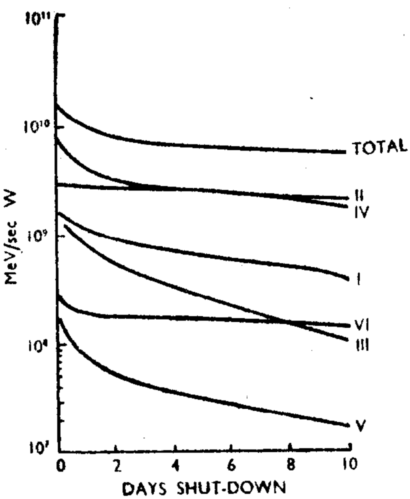


FIG. 2

Energy group distribution of the gamma decay rates for 1000-hr operation as a function of time after shut-down (from Moteff[3])

Group	Energy (MeV)
I	0·10–0·40
II	0·40–0·85
III	1·10–1·35
IV	1·65
V	1·8–2·2
VI	2·4–2·60
VII	2·90

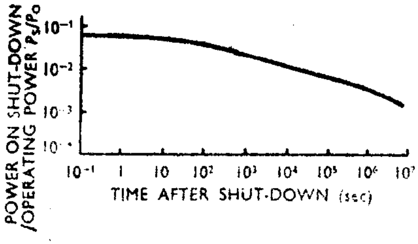


FIG. 3

Shut-down power in ^{235}U for infinite operation (from Untermeyer and Weills[4])

necessary for irradiated fuel elements. The results do not cover within 2 hr after shut-down.

An alternative evaluation of similar data has been given by Untermeyer and Weills,[4] who give the observed decay of heat generation after irradiation of ^{235}U and natural uranium for various irradiation times. This is shown in Figs. 3 and 4.

They quote the formulae

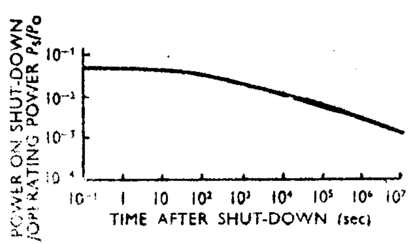
$$\frac{P_s}{P_o} = 0{\cdot}1\{(T+10)^{-0\cdot2} - 0{\cdot}087(T+2.10^7)^{-0\cdot2}\} - \\ -0{\cdot}1\{(T+T_0+10)^{-0\cdot2} - 0{\cdot}087(T+T_0+2.10^7)^{-0\cdot2}\}$$

$$\frac{P_s}{P_o} = \Big[0{\cdot}1(T+10)^{-0\cdot2} - 0{\cdot}087(T+2.10^7)^{-0\cdot2} - \\ -0{\cdot}0025\exp\Big\{-\Big(\frac{T}{2040}\Big)\Big\} - 0{\cdot}0013\exp\Big\{-\Big(\frac{T}{290000}\Big)\Big\}\Big] - \\ -\Big[0{\cdot}1(T+T_0+10)^{-0\cdot2} - 0{\cdot}087(T+T_0+2.10^7)^{-0\cdot2} - \\ -0{\cdot}0025\exp\Big\{-\Big(\frac{T+T_0}{2040}\Big)\Big\} - 0{\cdot}0013\exp\Big\{-\Big(\frac{T+T_0}{290000}\Big)\Big\}\Big]$$

for natural uranium and ^{235}U respectively in the range $T = 1$ to 10^8 sec. P_s and P_o are the powers on shut-down and during operation respectively, T_0 and T (sec) the irradiation and decay times. In the case of

FIG. 4

Shut-down power in natural uranium after infinite operation (from Untermeyer and Weills[4])



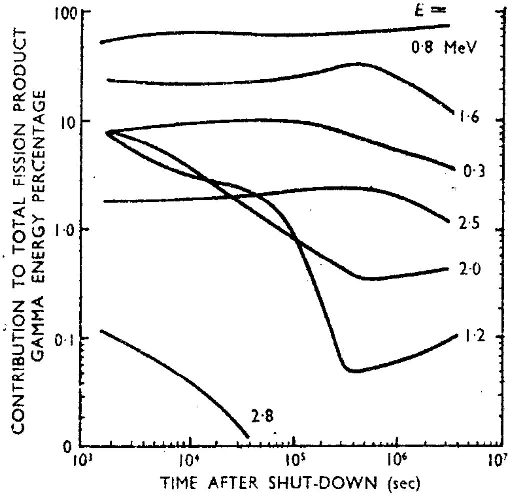


FIG. 5. *Distribution of gamma-radiation from fission products at short times after shut-down (reproduced from Rockwell*[6]*)*

natural uranium, the formula includes the effect of the decay of ^{239}U and ^{239}Np. ($^{238}U + n \rightarrow {}^{239}U \xrightarrow{23.5\ \text{min.}} {}^{239}Np \xrightarrow{2.3\ \text{d}} {}^{239}Pu$). A simpler though limited formula is given by Way and Wigner.[5]

It has been shown that the total energy released in the decay of the fission products is divided between β and γ-radiation in the ratio $\frac{\gamma \text{ energy}}{\text{total energy}} = 0{\cdot}65$ at about 20 minutes after irradiation. This ratio falls to approximately 0·5 after approximately 50 days after a short irradiation. For longer irradiation times the relative spectra have not been investigated, but approximately equal division of the total energy may be assumed. The accuracy of these figures is probably ± 50 per cent immediately on shut-down, but improves to approximately ± 30 per cent after longer decay times.

By using these figures in conjunction with data quoted by Rockwell[6] for the energy distribution of γ-radiation at short times after irradiation, an approximate estimate of the intensity and energy distribution of the γ-radiation can be obtained for short decay periods. Rockwell's data is shown in Fig. 5. From these results, it can be seen that immediately after shut-down approximately 6 MeV of γ-radiation is emitted per fission. The spectrum is not accurately known, but the dominant energy is about 0·7 MeV. During operation, again the spectrum is not well ascertained, but the following has been given[6]:

TABLE 5. γ-RADIATION FROM FISSION PRODUCTS DURING REACTOR OPERATION

MeV	0·15	0·2	1·2	1·6	2	2·4	2·8
Photons/fission	6·0	3·3	0·54	0·67	0·34	0·12	0·011

[Table 2.2 of Rockwell[6]]

This gives a total energy of 7·3 MeV per fission; since the dominant energy is fairly low, the radiation does not affect the overall shield dimensions, but may be important in determining the heat production in the inner parts of the system.

The activity of any particular fission product may be found in a report by Howlett *et al.*[7]

Neutron Capture (n, γ *Reactions*)

When a neutron is captured by a nucleus, the binding energy is released as γ-radiation. The total energy emitted, and the spectrum depends on the capturing nucleus. The former varies fairly smoothly over the periodic table, except for low-atomic-weight nuclei, and the so-called magic-number nuclei (i.e. those containing either 16, 50, 82 or 126 neutrons or protons). It is about 8 MeV for medium-weight nuclei, and drops to about 6 MeV for the heavy nuclei. Fig. 6 shows the binding energy as a function of atomic weight.

The spectra of the emitted radiation have been extensively investigated, and a collection of the available data given by Mittelman and Leidtke.[8] This data for the more important nuclei in shielding is given in Table 5. From the Table, it can be seen that the binding energy may be released as a single high-energy photon (e.g. ^{56}Fe), or as a continuous spectrum (e.g. ^{112}Cd). It will be noticed in the Table

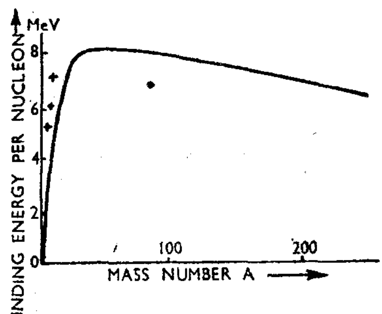


FIG. 6

The average binding energy per nucleon plotted against the mass number (From Mansfield[17]) that no accurate data is given for $E_\gamma \lesssim 3$ MeV. This is due to the experimental difficulties of measurement below this energy. It is reasonable to assume that the difference between the binding energy and a summation over the spectrum quoted for any particular nucleus will be given off as γ-rays below 3 MeV.

TABLE 6. CAPTURE γ-SPECTRA

Element	*Photons in energy interval* (MeV) *quoted per* 100 *captures*					*Highest energy γ-ray*
	0–1	1–3	3–5	5–7	> 7	(MeV)
Aluminium	?	> 13	77	21	35	7·724
Antimony*	?	~ 80	36	12	0	6·80
Barium*	?	~ 80	75	14	1	9·23
Beryllium	—	—	2 at 3·41 in 25	75 at 6·81 MeV	—	6·81
Bismuth	0	0	100 at 4·17 MeV	0	0	4·17
Cadmium*	> 120	20	73	17	1	9·046
Carbon	0	0	26 at 3·68 MeV	65 at 4·95 MeV	—	4·95
Chromium*	> 37	16	12	18	69	9·716
Cobalt	?	?	36	49	8	7·486
Copper	?	?	> 23	22	42	7·914
Iron*	?	< 10	24	22	50	10·16
Lead*	0	0	0	7	93	7·38
Manganese*	?	?	> 27	30	27	7·261
Molybdenum	?	?	84	26	3	9·15
Nickel	?	?	> 14	30	72	8·947
Niobium	?	?	54	14	0	7·19
Potassium	?	36	36	32	12	9·28
Silicon	?	> 100	229	41	16	10·55
Sodium	~ 100	> 50	61	29	0	6·41
Sulphur	?	~ 140	62	49	13	9·22
Titanium*	> 50	100	33	99	10	9·39
Tungsten*	?	?	53	14·5	0·5	7·42
Zinc*	?	?	48	29	17	9·51

[From Mittelman and Leidtke.[8]]

* For these elements, the Table gives the number of unresolved photons; elsewhere only the intensity of the resolved lines is given.

A more complete compilation can be found in Price.[9]

Decay of Radioactive Isotopes

Certain nuclei become radioactive upon neutron absorption, and emit β and/or γ-radiation. This source of radiation is generally not important however for reactor shielding. Table 7 gives data on many