PROTECTION AGAINST NEUTRON RADIATION UP TO 30 MILLION ELECTRON VOLTS

Handbook 63

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PROTECTION AGAINST NEUTRON RADIATION UP TO 30 MILLION ELECTRON VOLTS

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Preface

In recent years the number of neutron sources in this country has increased greatly. It has also become evident that, because of their physical properties and biological effects, neutrons must be regarded as a special type of radiation hazard. On the other hand, the formulation of adequate protection regulations is made difficult because of the limited experience available and because of the variable output of many neutron sources. The contents of this Handbook are based on what is believed to be the best information presently available, and as some of this information is not easily obtainable it has been set forth in some detail. Because of the rapid development of neutron technology it was felt advisable to state recommendations rather than rules in many instances. However, the rules provided (section IV) are deemed essential for proper protection.

Subcommittee I of the National Committee on Radiation Protection and Measurements has recommended limits for the maximum permissible dose of ionizing radiations in NBS Handbook 59. Since the publication of this Handbook the limits have again come under consideration by both the National and the International Committees on Radiation Protection. In addition, the National Academy of Sciences and the British Medical Research Council have executed detailed reviews on the effects of ionizing radiations on human beings. The four groups have made quite similar recommendations.

The recommendations of this Handbook take into consideration the statement of January 8, 1957, by the National Committee on Radiation Protection and Measurements recommending a substantial lowering of the maximum permissible levels for radiation workers.[1] The requirements in section 21.5 are somewhat more stringent than those given in the January statement. In addition, the RBE's used for neutrons are those given previously, although there is some evidence that they are too high. At the present time, Subcommittee M-4 of the NCRP is considering the RBE problem. At some future time, the RBE's in this Handbook may need to be revised. This conservative attitude is considered desirable because much less is known about biological effects of neutron radiation as compared with X- and gamma rays.

[1] NBS Technical News Bulletin 41, 17, 1957; Radiology 68, 260, 1957.

The scope of this Handbook extends to neutron energies up to 30 Mev. Although both theoretical and experimental information is sparser beyond about 10 Mev, the higher limit was chosen because many neutron generators operate within the wider range. The comparatively few sources producing neutrons in excess of 30 Mev usually attain energies several times as great, and a substantially different protection problem is involved.

The subject of neutron protection at reactors has been limited to considerations arising in routine operations. The problems of safe design and construction are outside the scope of this Handbook.

The National Committee on Radiation Protection and Measurements (originally known as the Advisory Committee on X-ray and Radium Protection) was formed in 1929 upon the recommendation of the International Commission on Radiological Protection. The Committee is sponsored by the National Bureau of Standards and governed by representatives of participating organizations. Eighteen subcommittees have been established, each charged with the responsibility of preparing recommendations in its particular field. The reports of the subcommittees are approved by the Main Committee before publication.

The following parent organizations and individuals comprise the Main Committee:

American College of Radiology: R. H. Chamberlain and M. D. Schulz.
American Dental Association: R. J. Nelsen.
American Industrial Hygiene Association: E. C. Barnes and J. H. Sterner.
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Subcommittee chairmen: See below.

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Subcommittee 2. Permissible Internal Dose, K. Z. Morgan.
Subcommittee 3. X-rays up to Two Million Volts, T. P. Eberhard.
Subcommittee 4. Heavy Particles (Neutrons, Protons, and Heavier), H. H. Rossi.
Subcommittee 5. Electrons, Gamma Rays and X-rays above Two Million Volts, H. W. Koch.
Subcommittee 6. Handling of Radioactive Isotopes and Fission Products, H. M. Parker.
Subcommittee 7. Monitoring Methods and Instruments, H. L. Andrews.
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Subcommittee 13. Safe Handling of Cadavers Containing Radioactive Isotopes, E. H. Quimby.
Subcommittee 14. Permissible Exposure Dose Under Emergency Conditions, L. S. Taylor, Acting chairman.
Subcommittee M-1. Standards and Measurement of Radioactivity for Radiological Use, W. B. Mann.
Subcommittee M-2. Standards and Measurement of Radiological Exposure Dose, H. O. Wyckoff.
Subcommittee M-3. Standards and Measurement of Absorbed Radiation Dose, H. O. Wyckoff.
Subcommittee M-4. Relative Biological Effectiveness, W. Langham.

The present Handbook was prepared by the Subcommittee on Heavy Particles (Neutrons, Protons, and Heavier). The following are the subcommittee members:

H. H. Rossi, Chairman, Columbia University.
E. P. Blizard, Oak Ridge National Laboratory.
R. S. Caswell, National Bureau of Standards.
F. P. Cowan, Brookhaven National Laboratory.
D. B. Cowie, Carnegie Institution.
T. C. Evans, State University of Iowa.
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L. D. Marinelli, Argonne National Laboratory.
W. S. Snyder, Oak Ridge National Laboratory.
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A valued contribution to the early work of the Committee was made by T. N. White, Los Alamos Scientific Laboratory, now deceased.

A. V. Astin, *Director*.

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Protection Against Neutron Radiation up to 30 Million Electron Volts

I. Introduction

1. Definition of Terms

The following definitions are given for purposes of clarification of the contents of this Handbook. In some instances they may differ somewhat from common use.

Shall. Necessary or essential to meet currently accepted standards of protection.

Should. Is recommended. Indicates advisory requirements that are to be applied when practicable.

Absorbed dose. Absorbed dose of any ionizing radiation is the energy imparted to matter by ionizing particles per unit mass of irradiated material at the place of interest. The absorbed dose is measured in rads. In this Handbook the term will be used to express the energy absorbed per gram of tissue of composition given in 5.1.

Accelerator. A device for imparting large kinetic energy to charged particles such as electrons, protons, deuterons, and helium ions.

Accessible location. Any region around a source of ionizing radiation that can be reached without rupture of structures or without use of specially designed tools not generally available.

Anemia. Abnormally low number of red blood corpuscles or low haemoglobin.

Barn. A unit of area used in expressing a nuclear cross section. 1 barn $= 10^{-24}$ cm^2. Cross sections per atom are customarily measured in barns.

Capture. A process in which a neutron becomes part of the nucleus with which it collides without release of another heavy particle.

Controlled area. A defined area in which the occupational exposure of personnel to radiation or to radioactive material is under the supervision of a radiation safety (protection) officer.

Cross section. Effective target area for specified nuclear interaction. The cross section is a measure of the probability for the interaction. It is expressed in barns.

Dose. As used in this Handbook, either *absorbed* dose or RBE dose. Doses and dose rates will be said to exist at some point in space even if tissue may not actually be located at such a point. In such instances such doses

and dose rates are the maximum ones that would be obtained in a 30-cm-thick infinite slab of tissue. (see appendix 1).

Dose rate. The rate of dose delivered averaged over a stated period. This may be different from instantaneous dose rate (e.g., in the operation of a pulsed generator the instantaneous dose rate greatly exceeds the dose rate averaged over 1 hour. This in turn may be higher than the dose rate averaged over 1 week if generator operation is intermittent).

Elastic scattering. Collisions in which the kinetic energy of neutron plus nucleus is unchanged by the collision, and the nucleus is left in the same state as before the collision.

Electron volt (ev). A unit of energy equal to the energy gained by a particle having one electronic charge when it passes in a vacuum through a potential difference of 1 volt; 1 ev$=1.60\times10^{-12}$ erg.

Epilation. Temporary or permanent removal of hair.

Epithelium. The purely cellular, nonvascular layer covering all the free surfaces of the body, cutaneous, mucous, and serous, including the glands and other structures derived therefrom.

Erythema. Reddening of the skin, primarily due to dilation of small blood vessels and also due to other tissue damage.

Exposure dose. Exposure dose of X- or gamma radiation at a certain place is a measure of the radiation which is based upon its ability to produce ionization.

Fast neutrons. Neutrons of energies between 10 kev and 10 Mev.

Gamete. A mature germ cell, such as an unfertilized ovum or spermatozoon.

Inelastic scattering. Scattering collision of neutron with attendant loss of kinetic energy which causes excitation of the target nucleus, and subsequent release of gamma rays.

Intermediate neutrons. Neutrons of energies between 0.5 ev and 10 kev.

Kilo electron volt (kev). 1,000 ev.

Kilovolt (kv). A unit of electrical potential equal to 1,000 volts. The term is also used to characterize the radiation emitted by X-ray tubes operating at this potential.

LD_{50}. Dose of ionizing radiation required to kill 50 percent of the animals in a given group. A time limit of 30 days is usually applied (***$LD_{50/30}$***).

Leukemia. Disease of blood-forming organs, usually with greatly increased production of white blood cells; these cells are invasive, and body deterioration may be either chronic (slow) or acute (rapid).

Leukocyte. White blood corpuscle.

Linear energy transfer (LET). The linear rate of loss of energy (locally absorbed) by an ionizing particle traversing a material medium.

Lymphocyte. A variety of white blood cell formed in lymph glands and other lymphoid tissue. The nucleus is single and is surrounded by a thin layer of cytoplasm which is nongranular. Average life span is relatively short.

Million electron volt (Mev). 1 million electron volts.

Moderator. Material used in a nuclear reactor to moderate, i.e., slow down, neutrons from the high energies at which they are released. Neutrons lose energy by scattering collisions with nuclei of the moderator.

Neutron flux. The number of neutrons which, per unit time, traverse a sphere of unit cross-sectional area centered about the point of interest. It is usually expressed in n cm^{-2} sec^{-1}. The correct term for this quantity is neutron flux density, but in common use the incorrect term "flux" is almost invariably employed.

Nuclear reaction. Collision between nuclear particles leading to release of different particles.

Platelet. A small colorless corpuscle present in large numbers in the blood of all mammals, believed to play a role in the clotting of blood.

Qualified expert. A person suited by training and experience to perform dependable radiation surveys, to oversee radiation monitoring, to estimate the degree of radiation hazard, and to advise regarding radiation hazards.

Rad. Unit of absorbed dose. 1 rad is equal to 100 ergs/g.

Radiation safety (protection) officer. An individual in charge of radiation protection.

RBE dose. Product of absorbed dose (as measured in rads) and RBE. The RBE dose is measured in rems.

Radioactive neutron source. A neutron source consisting of a combination of radioactive nuclides and suitable target materials. Neutron production occurs as a result of an (α,n) or (γ,n) reaction.

Relative biological effectiveness (RBE). Biological potency of one radiation as compared with another. It is numerically equal to the inverse of the ratio of absorbed

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doses of the two radiations required to produce equal biological effect. The standard of reference used in this Handbook is 200 kv X-radiation, which thus has an RBE of 1.

Relativistic neutrons. Neutrons of energies above 10 Mev.

Rem. Unit used in the description of radiobiological effects on man. The dose in rems is numerically equal to the product of the absorbed dose in rads and the value of the RBE applicable for the radiation in question.

Thermal neutrons. Strictly, neutrons in the thermal equilibrium with their surroundings. In this Handbook, all neutrons with energies of less than 0.5 ev are included in this category.

Thrombocytopenia. Decreased number of platelets per cubic millimeter of circulating blood.

Week, calendar. 7 consecutive days.

Week, work. Any combination of time intervals adding up to 40 hours within 7 consecutive days.

II. Present Status of Physical and Biological Information

2. Classification of Neutrons and Primary Modes of Interaction

2.1. Neutrons are available from various types of sources in an energy range (per particle) from about 10^{-4} to 10^{9} ev, a variation in energy of a factor of 10^{13}. The types of interaction vary markedly with energy. It is therefore convenient to base the classification of neutrons on energy intervals where certain interactions predominate. The transition between these ranges is, however, not sharp, and consequently there is a certain degree of latitude in the choice of limits. The classification set forth below shall be adopted for the purposes of this Handbook.

2.2. *Thermal neutrons.* These are neutrons in thermal equilibrium with matter, usually at room temperature, which therefore have a Maxwellian distribution of velocities. The most probable velocity per unit velocity in this distribution is 2,200 meters per second, corresponding to an energy of 0.025 ev. In most instances it is sufficiently accurate to consider thermal neutrons as monoenergetic with this energy. The most important interaction with

matter is capture—usually with emission of gamma radiation.[1] Occasionally, nuclear reactions such as (n,p) or fission may occur. The N^{14} (n,p) C^{14} reaction is important in tissue.

2.3. *Intermediate neutrons.* The intermediate energy region is here defined as extending from 0.5 ev to 10 kev. The absorption spectrum of intermediate neutrons often exhibits resonant peaks and for this reason the term "resonance neutrons" is sometimes employed. Intermediate neutrons are usually obtained from a moderating material in which fast neutrons are slowed down by elastic collisions. This slowing-down process is the most important interaction between intermediate neutrons and matter, and it leads to a neutron flux inversely proportional to energy—the well-known dE/E spectrum. Capture and nuclear reactions may also occur.

2.4. *Fast neutrons.* The range of fast neutrons will be considered to extend from 10 kev to 10 Mev. The most important interaction with matter is elastic scattering. At the upper part of this energy range, inelastic scattering and nuclear reactions are comparable in frequency to elastic scattering. Although resonance phenomena may occur (particularly for light elements), cross sections vary slowly with energy in general.

2.5. *Relativistic neutrons.* All neutrons beyond 10 Mev energy will be considered relativistic neutrons. Their energy exceeds the binding energy of nucleons, and for this reason complex nuclear reactions such as spallation become important. In addition, the kinetic energy of neutrons is an appreciable fraction of the rest energy, and relativistic corrections must be applied. Elastic scattering also occurs, but it tends to be markedly asymmetric in the center-of-mass system.

2.6. The probability of any interaction between neutrons and matter is expressed quantitatively in terms of *cross sections.* The cross section, σ, may be considered as the *effective* target area of the nucleus if the neutron is assumed to have zero diameter. For a beam containing n neutron/cm^3 moving with velocity v (cm/sec) toward N nuclei, the rate of interactions per second will be $nvN\sigma$. The quantity nv is the neutron flux. The cross section is usually expressed in barns (10^{-24} cm^2).

2.7. There is a finite cross section for each possible nuclear interaction. In addition, cross sections for scattering

[1]Actually, elastic collisions are more prevalent, but at thermal levels neither neutrons nor the matter traversed gain or lose energy on the average.

processes may be further divided into various *differential* cross sections that express the probability for scattering in a particular direction. The number of neutrons interacting is determined by the sum of all processes that can take place, that is by the total cross section, σ_T.

3. Absorbed Dose

3.1. The energy imparted to tissues is the physical basis for quantitative correlation between exposure and biological effect. The energy per unit mass that is imparted to any material by ionizing radiation is denoted as the *absorbed dose*. It is expressed in rads.

3.2. In the absolute CGS system of units, specific energy imparted to matter is expressed in ergs per gram. One rad equals 100 ergs/g. One millirad (mrad) is one-thousandth of 1 rad. The rad was adopted as the international unit of "absorbed dose" in July 1953, at the meeting of the International Commission on Radiological Units in Copenhagen [1].[2]

3.3. Heretofore, the rep (roentgen-equivalent-physical) has been used extensively for the specification of permissible doses of ionizing radiations other than X-rays or gamma rays. Several definitions of the rep have appeared in the literature, but in the sense most widely accepted it is a unit of *absorbed dose* in soft tissue with a magnitude of 93 ergs/g. The difference in magnitude between the rep (93 ergs/g) and the rad (100 ergs/g) is negligible in the estimation of permissible doses. Therefore, the adoption of the rad to replace the rep does not necessitate a change in the numerical values of permissible doses stated in reps heretofore.

4. Other Dose Units Employed to Define Exposure

4.1. Although the presently recommended unit of neutron dose is the rad, a variety of other units are employed or have been employed in the past.

4.2. One of the most commonly used quantities is the neutron flux. Since the absorbed dose rate delivered by a given neutron flux depends on energy, the latter must be specified (at least within certain limits) to make possible an estimate of the absorbed dose. The relation between flux and the absorbed dose received by a small amount of tissue (first collision dose) is given in figure 1.

[2] Figures in brackets indicate the literature references at the end of this Handbook.

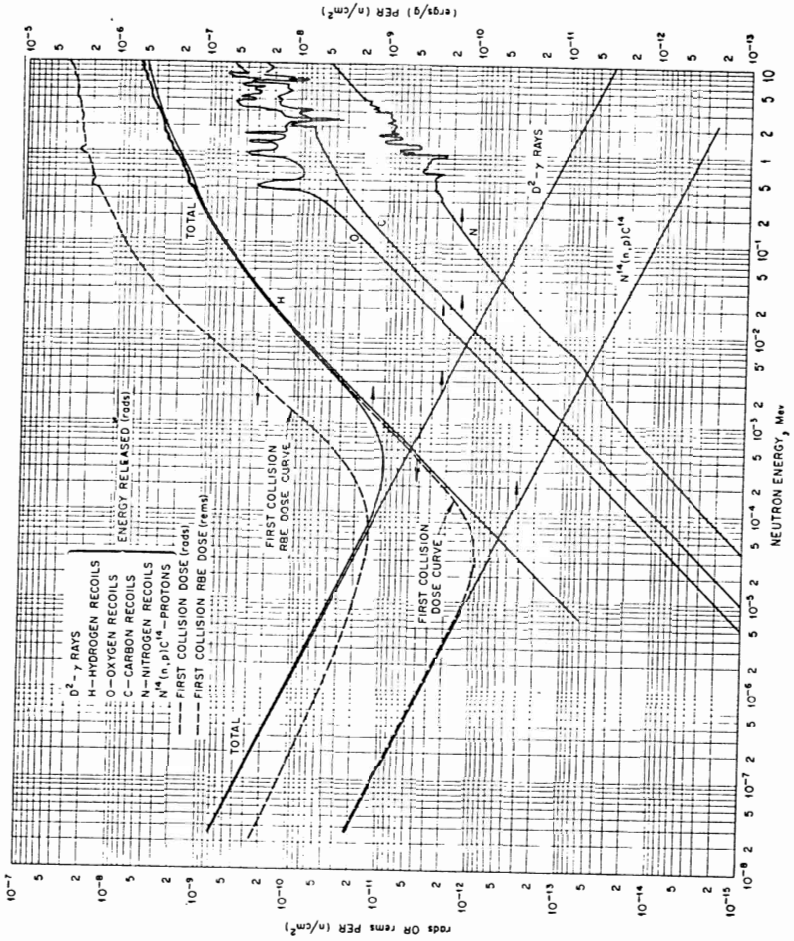
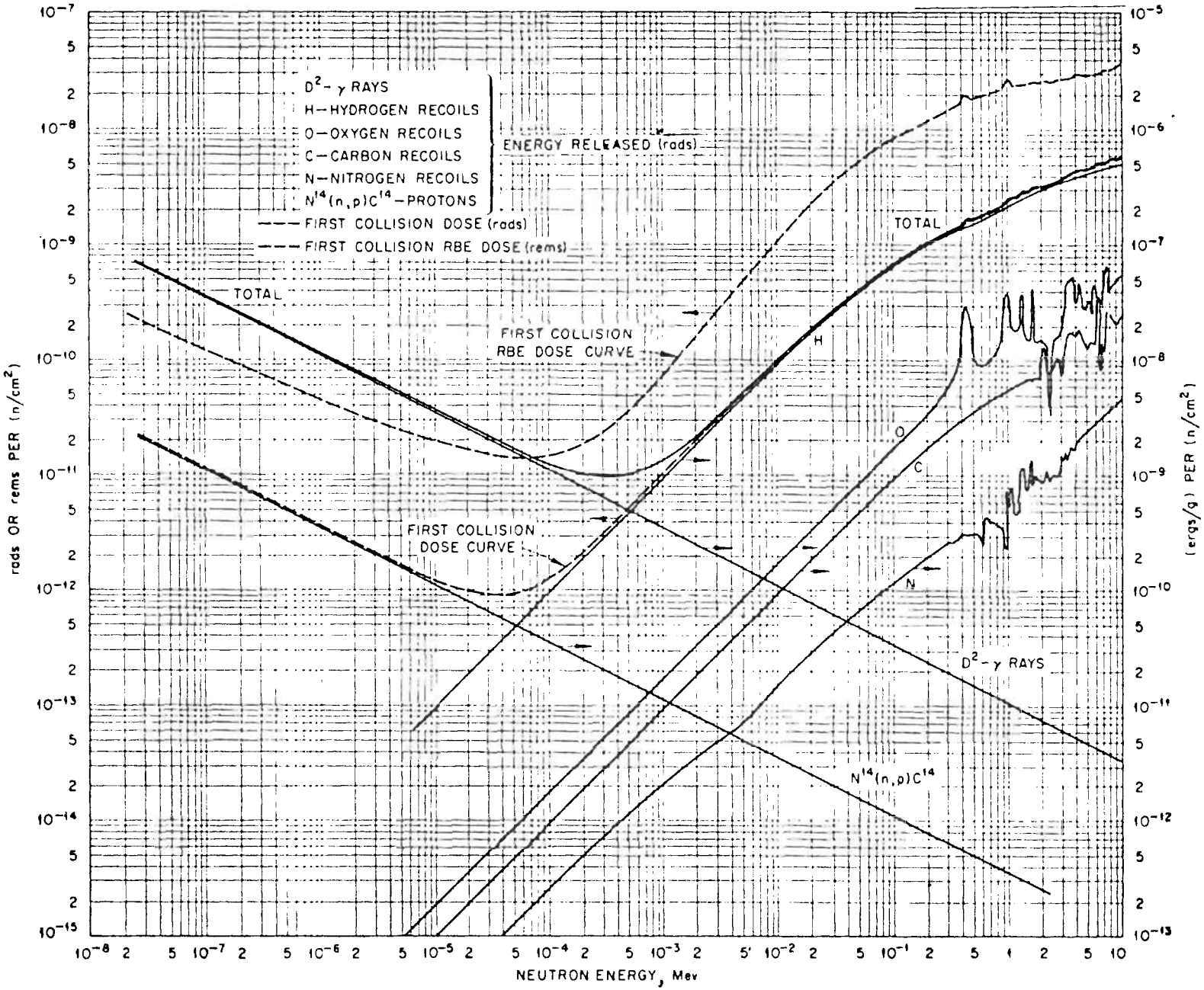


FIGURE 1. *First collision dose curve.*

Energy imparted to soft tissue by neutrons of energies between 10^{-8} and 10 Mev.

The curve marked "total" represents the entire energy deposited. Because of the penetration of the D^2 gamma rays most of their energy can escape from the point of neutron absorption. The energy dissipated in the immediate vicinity of the point of absorption is given by the "first collision dose", which includes all of the energy imparted to heavy ionizing particles. These include hydrogen, oxygen, carbon, and nitrogen recoils, as well as protons from the $N^{14}(n,p)C^{14}$ reaction. The individual contributions due to these particles are shown.

The "first collision RBE dose" curve was obtained by multiplying the "first collision dose" curve by the appropriate RBE. The ordinate is rems.


D2 - γ RAYS
H—HYDROGEN RECOILS
O—OXYGEN RECOILS
C—CARBON RECOILS
N—NITROGEN RECOILS
N14(n,p)C14—PROTONS
ENERGY RELEASED (rads)
FIRST COLLISION DOSE (rads)
FIRST COLLISION RBE DOSE (rems)
TOTAL
FIRST COLLISION RBE DOSE CURVE
FIRST COLLISION DOSE CURVE
H
O
C
N
D2 - γ RAYS
N14(n,p)C14
rads OR rems PER (n/cm2)
(ergs/g) PER (n/cm2)
NEUTRON ENERGY, Mev

4.3. The Victoreen condenser r-meter, an instrument designed for X-ray measurement, has often been used to determine neutron exposure. The "n unit" is understood to be the amount of neutron radiation that discharges the ion chamber to the same degree as does 1 roentgen of X-rays. Sometimes a distinction is made between the 25-r chamber model (N unit) and the 100-r chamber (n unit). The number of rads corresponding to 1 n depends on neutron energy and may vary among individual chambers by ± 20 percent or more. For neutron energies above 1 Mev and up to several Mev, 1 n corresponds to approximately 2 rads. At lower neutron energies the number of rads per n probably becomes larger. The unit is now obsolete.

5. Interactions Between Neutrons and Tissue

5.1. The interactions that occur between neutrons and tissue depend upon the composition of the tissue involved. For purposes of this Handbook the atomic composition of the "standard man" has been used to evaluate the more probable interactions of neutrons and the atoms of tissue. The atomic composition of the "standard man" is given in table 1.

TABLE 1. *Atomic composition of the "standard man"*

Element	Weight, grams	Weight, percentage	Atoms per gram	Atoms, percentage
O	45,500	65	2.45×10^{22}	25.7
C	12,600	18	0.903	9.49
H	7,000	10	5.98	62.8
N	2,100	3	0.129	1.36
Ca	1,050	1.5	.0225	0.236
P	700	1.0	.0194	.204
S	175	0.25	.00470	.0494
K	140	.20	.00308	.0324
Na	105	.15	.00393	.0413
Cl	105	.15	.00255	.0268
Mg	35	.050	.00124	.0130
Fe	4	.0057	6.16×10^{17}	6.47×10^{-4}
Cu	0.1	.00014	1.35×10^{16}	1.42×10^{-5}
Mn	.02	.000029	3.13×10^{15}	3.29×10^{-6}
I	.03	.000043	2.03×10^{15}	2.13×10^{-6}
Total	69,500		9.52×10^{22}	

5.2. For both fast neutrons and relativistic neutrons up to 30 Mev, the most important interaction between neutrons and tissue is elastic scattering. The cross sections for absorption are small in comparison with the scattering cross sections, and inelastic scattering, although present at energies above several Mev, does not occur with hydrogen. Because the hydrogen cross section tends to decrease rather rapidly above 10 Mev, inelastic scattering and spallation

become more important as the neutron energy increases. However, there are few, if any, quantitative data on this point at present. The energy absorbed in tissue due to elastic collisions is transferred largely as a result of scattering by hydrogen atoms. This is so both because the hydrogen atoms occur more abundantly, but also because the average fraction of the energy lost in an elastic collision is given by $2M/(M+1)^2$, where M is the mass number of the atom struck, and thus the heavier atoms dissipate only a small fraction of the neutron's energy. Figure 1 gives the energy absorbed per gram of tissue when exposed to neutrons of energy E (Mev) with 1 neutron/cm² incident on a small volume element of tissue. Taking into account only the first collisions of the neutrons, the dose in rads due to collisions with an element of mass number M_i, cross section σ_i (barns), and atomic abundance N_i (atoms/g), is given by

$$D_i=E\frac{2M_i}{(M_i+1)^2}N_i\sigma_i 1.6\times10^{-8}\ \text{(rads)}. \qquad (1)$$

Figure 1 indicates that, in general, elastic scattering by hydrogen accounts for 80 to 95 percent of the energy transferred to tissue by fast neutrons.

5.3. For a beam of neutrons incident on a large mass of tissue, the formulas (of 5.2) must be corrected to take account of the attenuation of the incident beam. The effect of the buildup (enhancement by multiple collisions) on the dose is very difficult to assess precisely but can, fortunately, be bracketed with sufficient accuracy for most practical cases of protection. Because the total dose always includes the first collision dose given by summing the above formulas, it is clear that the first collision dose is a lower bound for the total dose. In the case of neutrons of energies between 0.1 and 10 Mev impinging on a large convex mass of tissue, the maximum dose is at or near the irradiated surface and is never more than twice the first collision dose at the surface. It is to be emphasized that this rule only applies to the maximum or surface dose and does not apply deep within an irradiated body where the first collision dose may be a much smaller fraction of the total dose.

5.4. For neutrons of intermediate or thermal energy, most of the dose is imparted in the process of neutron absorption. The principal interactions are the $H(n,\gamma)D$ and the $N^{14}(n,p)C^{14}$ reactions. The energy released by the interaction with hydrogen is much greater, because the γ quantum has an energy of about 2.2 Mev whereas the

proton has an energy of about 600 kev. In addition, the product of relative abundance and cross section is much larger in the case of the (n,γ) reaction. However, the γ-ray can travel through tissue for considerable distances before losing its energy, whereas the proton dissipates its energy in the immediate vicinity of its origin. Thus, for small masses of tissue the proton dose predominates, but for large masses of tissue the gamma dose is much larger. There is at present little precise information available about the variation of dose with the geometry of the irradiated body, or even a rough rule such as that given above for fast neutrons. For large masses of tissue (20 cm thick or more) the maximum dose is essentially independent of neutron energy up to 10 kev. Monte Carlo calculations have indicated that the percentage of neutrons that slow down to thermal energy in a thick slab does not vary greatly with the energy of the incident beam; and as, for energies up to 5 kev, these thermalized neutrons account for most of the dose, it follows that the dose is roughly constant.

5.5. The depth dose curves obtained by Monte Carlo calculations using a slab of tissue 30 cm thick are given in appendix 1. These may be regarded as useful approximations to the dosage patterns within the trunk of a human body.

6. Relative Biological Effectiveness

6.1. It has been found that equal absorbed doses delivered by different ionizing radiations may produce varying degrees of injury. The relative biological effectiveness (RBE) of one radiation with respect to another is defined as the inverse ratio of the absorbed doses required for equal effect. Thus, if induction of a given degree of damage requires an absorbed dose D_R by the reference radiation and D_N by the other radiation, the RBE of the latter is D_R/D_N.

6.2. The biological effectiveness of different kinds of ionizing radiation is usually indicated as relative to that of conventional therapeutic X-radiation (200 kv) as unity. In lethality studies it has been found that gamma radiation (from Co^{60} or from radium) apparently has an RBE from 0.6 to 0.8 of that of 250-kv X-radiation.

6.3. As measurements of RBE are based on comparisons of tissue dose, the chief physical variable which apparently accounts for the difference in RBE is the rate of loss of energy along the path of ionizing particles. It is assumed that biological effectiveness is dependent on spatial distribution of the energy transfer taking place in tissue. Consequently, it is now proposed to consider linear energy transfer (LET) as the physical factor responsible for the RBE.

6.4. The RBE for various biological effects varies with test objects, the type of effect studied, and often other factors (such as dose rate).

6.5. The term "rem" has been used as meaning "roentgen or rad equivalent man." The unit is used in an attempt to express dose in terms of biological rather than physical equivalence. It may be defined as the product of absorbed dose in rads times RBE. Thus an absorbed dose of 10 rads from a radiation having an RBE of 10 represents 100 rems. Because of the variability of RBE, equal doses of rads represent varying doses of rems depending on the effect under consideration. However, for purposes of this Handbook, the RBE dose in rems will be understood to be the absorbed dose in rads multiplied by the RBE (applicable for the radiation under discussion) pertaining to exposure of humans and as formulated for purposes of radiation protection.

6.6. Handbook 59 of this series recommends RBE values that are made dependent on the LET of the charged particles produced in tissue. An RBE of 1 is proposed for all LET values up to 3.5 kev/μ. The RBE is assumed to increase more or less linearly from 1 to 20 in the range from 3.5 to 175 kev/μ. No recommendations were made for values in excess of 175 kev/μ. Permissible doses in section 8 have been derived on the basis of these recommendations. The RBE of LET values beyond 175 kev/μ has been assumed to be 20.

6.7. Biological comparison based on $LD_{50/30}$ appears to indicate that the RBE of fast neutrons in acute exposures would be 2 to 4. Dose-effect ($LD_{50/30}$) curves for X-rays and neutrons are similar in shape for mice, rats, and other animals, but may differ for some such as the chick. Continuous low level of exposure (protraction) and fractionation appear to result in a higher RBE of fast neutrons, particularly in injury to the gonads or to the lens of the eye.

7. Biological Effects

7.1. Certain basic facts are generally accepted about the biological action of neutrons and other ionizing radiations: (a) The radiation penetrates throughout the cell, (b) the energy transferred is high enough to cause fundamental changes in atomic and molecular structure, (c) alterations are widely and randomly distributed throughout the cell.

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7.2. The cellular changes due to single or multiple doses may not be grossly observable for some time, and may accumulate as time goes on. Tissues vary in radiosensitivity and in ability to recover from radiation damage. They also vary in latent period (i.e., time from exposure to manifestation of change).

7.3. In man, effects that may appear early are (1) reduction in lymphocytes in blood and (2) damage to epithelial cells of intestine and skin. Other rather early changes are (1) reduction in leukocyte number in the blood, (2) erythema, epilation, and inhibition of gamete formation, and (3) damage to small blood vessels. Changes that occur more slowly and require more exposure are reduced production of erythrocytes (anemia) and of platelets (thrombocytopenia — leading to bleeding tendency) and general lack of new cells resulting in systemic deterioration. It should be pointed out that these effects are not necessarily indicative of radiation damage, because they may result from any of several abnormal conditions.

7.4. General considerations regarding biological effects of radiation and maximum permissible exposure conditions usually include the following: (a) The most serious condition is exposure of the whole body to penetrating radiation; (b) in general, percentage survival and survival time increase markedly as (1) portions of the body are protected, (2) the exposures are fractionated, and (3) the penetration of the radiation is decreased; (c) hereditary effects are not easily detected but they are cumulative and must be considered, especially if large population groups are exposed; (d) repeated exposures (each too small to produce demonstrable injury alone) may eventually reduce life span.

7.5. Skin cancer and leukemia are hazards of overexposure. Such malignant changes usually, but not always, are preceded by other indications of radiation damage. Again, it must be pointed out that the presence of the malignancy does not necessarily indicate radiation as the causative agent.

7.6. Certain fast-neutron hazards appear to be related to a more pronounced accumulation of damage from multiple exposures than occurs in the case of X-irradiation. Thus, tissues not likely to be replenished by cells from other organs (such as the lens epithelium and germinal epithelium) are especially vulnerable to fast neutrons in multiple exposures. Although there is usually some recovery following exposure to neutrons, it is less than in the case of X-rays.

7.7. Much of our information concerning biologic effects of radiation comes from clinical experience and from animal experimentation, but occasionally an accidental overexposure yields valuable data in this regard [2 to 6]. Much more information is needed from all three sources.

8. Permissible Exposure to Neutrons

8.1. Due to the action of cosmic radiation, there exists a constant neutron flux of roughly 50 n cm^{-2} hr^{-1} at sea level. This increases with altitude, reaching a value approximated by 500 n cm^{-2} hr^{-1} at 10,000 feet. The resultant dose is of the order of 10^{-2} mrad/week at sea level and 10^{-1} mrad/week at 10,000 feet.

8.2. Permissible exposure to neutrons is the dose that may be received without undue risk to the health of the individual and that of the population. Certain basic rules are given in Handbook 59. These state that for radiation workers the weekly dose delivered to the skin must not exceed 600 mrems, and that, in addition, any portion of the body beyond a depth of 5 cm as well as certain critical organs must receive no more than 300 mrems/week. In the energy range covered here, absorbed doses of neutrons are for practical purposes always maximal at or near the body surface; and because some of the critical organs, such as the lens of the eye and the male gonads, are at little depth in the body, these rules require that in whole-body exposure the maximum permissible weekly dose be 300 mrems as measured at or near the body surface. The rules permit somewhat larger doses to be received by certain body regions. Thus, hands and feet may receive 1,500 mrems/week. However, these larger local limits shall not be permitted unless special efforts have been made to assure that the head and the trunk are not exposed in excess of 300 mrems/week. In exceptional cases when it is necessary for a person to receive more than 0.3 rem in 1 week, he may receive 3 rems in 13 weeks.

8.3. The 1957 recommendations of the National Committee on Radiation Protection impose additional restrictions on the dose that may be incurred by radiation workers over long periods of time [7]. These may be expressed by the formula

$$D=5(N-18) \text{ rems,} \qquad (2)$$

where D is the RBE dose accumulated at age N years. The formula applies to all critical organs except the skin, for which the value $2D$ is permitted.

It will be noted that in the case of a radiation worker who, beginning at an early age, is routinely exposed to a substantially constant radiation level, the maximum yearly dose shall not exceed 5 rems and the average weekly dose shall not be more than 100 mrems. It will sometimes be necessary to plan radiation protection on the basis of these figures when protracted exposure of younger individuals is anticipated or must be considered likely. For this reason, certain data given below (e.g., table 2) are based on weekly exposures of 100 mrems as well as 300 mrems, although it will be understood that larger values are permissible as long as the restrictions in this and the preceding paragraph are adhered to.

8.4. Handbook 59 also contains a listing of applicable RBE values according to the specific ionization of the particles delivering the dose. Calculations taking into account the LET of secondary recoils arising from both primary and multiple scattered neutrons indicate that in a phantom 30 cm thick, the RBE depends both on neutron energy and, to some extent, on the depth in the phantom. However, in general, the highest RBE occurs near the regions where the dose is also maximal. Therefore, the highest RBE must be applied for purposes of protection. Figures 2 to 12 in appendix 1 show depth doses in both rads and rems for a number of neutron energies.[3] Table 2 gives RBE and maximum permissible average neutron flux as a function of energy for protracted exposure on the basis of a 40-hour week. Although an RBE of 10 might be slightly exceeded at neutron energies in the neighborhood of 1 Mev, it would seem sufficiently safe to derive maximum permissible doses for any neutron energy between thermal level and 10 Mev by linear interpolation between neighboring energies in table 2. In the absence of any definite information, a conservative limit has been adopted as the maximum permissible flux density between 10 and 30 Mev.

8.5. It must be realized that the values in table 2 apply only to monoenergetic neutrons incident normally on the major portions of the human body. Even when a neutron generator emits monenergetic neutrons, scattering by walls and other structures will cause degradation in energy. However, because this process will, *in the range of table 2,* almost always lead to decreased biological potency, it is safe to assume that all neutrons have the original maximum energy. Similarly, it may be assumed that all neutrons are incident normally (even if some may actually arrive obliquely).

[3] Subcommittee No. 4 is indebted to W. S. Snyder for having carried out the computations on which these figures are based.

TABLE 2. *Maximum permissible neutron flux*

Time-average flux for 40-hour week to deliver either 100 or 300 mrems.

Neutron energy	RBE	100 mrems	300 mrems
Mev		*n cm⁻² sec*	*n cm⁻² sec*
Thermal	3	670	2,000
0.0001	2	500	1,500
.005	2.5	570	1,700
.02	5	280	850
.1	8	80	250
.5	10	30	90
1.0	10.5	18	55
2.5	8	20	60
5.0	7	18	55
7.5	7	17	50
10	6.5	17	50
10 to 30		[a]10	[a]30

[a] Suggested limit.

8.6. In case sufficiently detailed information on neutron energy is not available, an RBE of 10 shall be assumed.

8.7. The maximum permissible weekly dose may be received within any time period within the week. Assuming a 40-hour work week, a steady exposure to 7.5 mrems/hr or the fluxes in the last column of table 2 represents the permissible limit.

8.8. It is clearly desirable to design neutron installations so that the above hourly rates are not exceeded in normally occupied areas, as it is then impossible for any personnel to be exposed beyond the permissible weekly limit. In areas where these values are exceeded, the occupancy by personnel or the duty period of the neutron generator must be restricted to avoid dosage in excess of the permissible annual limit. As failure to observe this restriction would create a hazard, the rules given below require certain safeguards. A greater potential hazard exists if the duty cycle of the equipment is such that a dose in excess of the *weekly* permissible limit may be received by a person who remains at the location for the full 40-hour work week. The rules below make a distinction between these conditions.

8.9. High dose rates, apart from inherent danger, often represent a psychological hazard because of anxiety or nervousness, and should be avoided whenever possible.

8.10. It may often occur that individuals who are in no way associated with the operation of the source are ex-

posed to detectable amounts of radiation. If such locations are outside the controlled area and accessible to the public, it may be difficult or impossible to regulate or accurately predict the occupancy. Because in addition a comparatively large number of individuals (including children) might be exposed, special safeguards must be provided.

8.11. Considering all available evidence, no significant radiation effects have been demonstrated in animals or humans at or below the maximum weekly permissible dose levels. By extrapolation from observations at much higher dose levels, it seems possible that long-continued irradiation, even at permissible levels, may have some deleterious effect. The magnitude of such effects is believed to be less than that due to many other physical and chemical health hazards to which we are daily exposed. Nevertheless, efforts should be made to reduce exposure as much below permissible dose levels as practicable.

The 1957 recommendations of the NCRP (see page 33) stipulate that "the maximum permissible dose to the gonads for the population of the United States as a whole for all sources of radiation, including medical and other manmade sources, and background, shall not exceed 14 million rems per million of population over the period from conception up to age 30, and one-third that amount in each decade thereafter. Averaging should be done for the population group in which cross-breeding may be expected." To achieve this goal it is further recommended that for persons nonoccupationally exposed to radiation in the environs of a radiation source, the maximum permissible accumulated dose shall not exceed one-tenth that for radiation workers. This is equivalent to an average per capita dose of 500 mrems per year. It should be noted that this lesser figure has been set primarily because of genetic effects rather than the likelihood of personal injury. Exposure within the limits set for radiation workers is believed to be an entirely acceptable individual risk, but if a large fraction of the population were exposed at these levels, long-term genetic effects might be too great. This is the reason why the limit has been given in terms of a million individuals; larger doses might be imparted to a few individuals who are not radiation workers.

In this Handbook a limit of 125 mrems per 3-month period is recommended for uncontrolled areas.

Obviously the best way to attain this objective is to avoid delivery of such a dose to any part of the areas in question regardless of any occupancy factors, and this

policy is strongly recommended. If more than 10 mrems can be received in such uncontrolled areas during a calendar week, regular checks are required to insure that individuals remain there for sufficiently limited periods of time so that exposure in excess of 125 mrems per 3-month period is unlikely. This limit applies equally if several generators contribute radiation to the same location.

8.12. It is to be noted that permissible exposure figures may be revised downward in the future, particularly if the number of persons exposed to radiation should increase markedly. Therefore, in the design of neutron protection it is advisable to make provision for more shielding to be added in the future. In the case of large permanent installations it may be much more economical to apply such additional protection at the time of original installation, i.e., to overdesign protection deliberately to insure compliance with possible future limits that may be lower. However, in the absence of definite knowledge no firm recommendations can be made on this point.

9. Gamma- and X-ray Hazards Arising in the Operation of Neutron Sources

9.1. In practice, the presence of neutrons in the laboratory is almost invariably accompanied by X- and gamma radiation. Because the relative intensity of this component may vary widely with experimental conditions, the hazard must be assessed directly by the operator and added to the neutron hazard with an RBE of unity. A brief description of the mechanism of photon production in various types of neutron installation is appended hereto for information of the reader.

9.2. *Low-voltage ion accelerators (400 kev).* Although used to produce neutrons free of gamma rays, such machines are usually strong sources of X-rays. These are engendered by electrons released at the walls and target of the apparatus by ion bombardment and accelerated toward the anode and supporting structures. This radiation may be reduced (but never entirely suppressed) by application of positive potential to the target assembly.

9.3. *High-voltage ion accelerators.* In addition to producing X-rays, these installations become sources of gamma radiation because of the increased likelihood of nuclear excitation of the target material and the walls of apparatus. The energies of the photon radiation are characteristic of the nucleus and may vary from a few hundred kev to about 20 Mev.

9.4. (α,n) *sources*. The (α,n) reactions are, with a few exceptions (e.g., Po^{210}, Pu^{239}), also sources of intense gamma radiation because of concomitant emission in the radioactive chain (Ra^{226}, Ra^{223}, Th^{228}, Ac^{227}, etc.).

9.5. *Photoneutron sources*. (γ,n) emitters, such as radioactive sources and betatrons, are particularly hazardous from this standpoint because the relatively small cross section of the reaction, as compared to the Compton and pair-production process, requires implicitly an overwhelming flux of photons for the production of a relatively small number of neutrons.

9.6. *Thermal neutron sources*. Such sources as nuclear reactors present the added hazard of gamma rays, which almost invariably follow the capture of neutrons by nuclei. The cross sections for this process vary widely (10^6 to 10^{-5} barn), and the energy of the gamma radiation emitted varies from a few tenths to about 10 Mev. Frequently the product nucleus is radioactive and emits gamma radiation also.

9.7. *Inelastic scattering*. Fast neutrons, of energy greater than that of the lowest excitation levels of a nucleus, can lose energy by excitation. This process (of cross section usually below 3 barns) results in the emission of gamma rays characteristic of the disturbed nucleus. This type of gamma radiation must be anticipated in all installations producing fast neutrons.

9.8. *Fission sources*. Since the fission process—even when engendered by thermal neutrons—is accompanied by gamma rays and by the production of radioactive fission products, it is reasonable to expect a definite gamma-ray hazard.

10. Neutron Detectors

10.1. For practical purposes the processes that reveal the presence (or previous presence) of neutrons may be classified as instantaneous or as delayed processes. In most cases the phenomenon that is detected is ionization, or something that is caused by ionization.

10.2. *Examples of instantaneous radiations and their detection.*

a. Fission: Fission recoils may be detected by their ionizing effect in counters or ion chambers; the attendant release of gamma radiation may be detected by its ionizing effect.

b. Recoil from neutron collision: The recoiling nucleus may cause detectable ionization.

c. Inelastic collision: On collision, part of the kinetic energy of the neutron may be converted to gamma ray energy.

d. Capture: Capture of a neutron by a nucleus is followed by emission of one or more gamma rays. Also, the new nucleus that is formed by capture may be so unstable as to disintegrate immediately, giving rise to one or more of the ionizing phenomena.

10.3. *Delayed processes*. The principal delayed processes are radioactive disintegrations of fission products and of nuclei produced by neutron capture.

10.4. If the presence of neutrons has not been foreseen, and provision for immediate measurement has therefore not been made, the delayed process may sometimes be of special usefulness.

10.5. Neutron detectors utilize the same devices (counters, ionization chambers, photographic emulsions, etc.) that are used for the detection of other ionizing radiations, although, as in the case of X-rays, any direct ionizing effect of the primary radiation is inconsequential.

10.6. Discrimination between ionization due to neutrons and ionization due to gamma rays, when both are present, is not difficult in instruments that are designed solely for detection (as distinct from flux or dosage measurement). The burst of ionization that can be produced by a recoiling or disintegrating nucleus is much greater than any burst that is likely to be produced in a detector by gamma radiation. The latter can be discriminated against by the electrical circuit of an instrument. Very high sensitivity can be obtained in instruments based on these processes, but the sensitivity usually depends strongly on the energy of the neutrons, which is usually unknown. Devices that depend on the delayed process can usually be counted upon to discriminate against everything but neutrons, and can be made quite sensitive also, but they suffer from the same dependence on neutron spectrum. In spite of this deficiency, these high-sensitivity detectors are of value in rapidly delimiting large areas where the neutron flux is too low to justify any efforts at quantitative measurements.

10.7. One note of warning: Any instrument that depends upon the counting of ionization bursts that exceed some predetermined size must be suspected when it is used for surveying in the vicinity of a pulsed generator. It will read correctly only if capable of time resolution of the pulses received during the active part of the cycle of the machine. If these pulses are not resolved, the counting

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rate of the instrument may be independent of intensity and signify only the pulsing rate of the generator.

11. Measurement of Neutron Flux

11.1. In the case of monoenergetic neutrons of known energy, the flux may be determined provided the cross section for the interaction employed in detection is known and the number of interactions occurring in unit time can be measured in an absolute manner.

11.2. The flux measurement in neutron beams involving an unknown energy distribution can be made only if the detection efficiency is independent of neutron energy. This requirement is met in certain threshold detectors by the existence of substantially constant cross section above the threshold, and in the "long counter" by the adjustment of absorbing and scattering characteristics of the moderator surrounding the sensing element. The complex characteristics of this structure make it virtually impossible to determine the response absolutely, and a relative calibration must be performed. On the other hand, the comparatively wide range (about 0.1 to 3 Mev) in an important energy interval makes the long counter an important tool.

11.3. It is to be noted that the permissible flux figures given in table 2 apply to incident neutron fluxes rather than the combination of incident and reflected neutrons that exist near the body surface. The latter figure is higher, and if personnel monitoring is performed on the basis of flux at the body surface, data in appendix 1 should be applied.

11.4. The characteristics of various flux detectors as well as their calibration are discussed in appendix 2.

12. Dose Measurement

12.1. Because the permissible limit of exposure to neutrons is related to absorbed dose, it is obviously preferable to measure this quantity directly, particularly as this can usually be done with little or no information on neutron energy. Like other aspects of neutron technology, dosimetry has not been developed to the point where universally applicable methods of measurement are available. Nevertheless, in many instances dose determinations can be made more accurately by direct rather than by indirect means (such as flux determinations).

12.2. The most direct dose measurement may be performed utilizing the Bragg-Gray theorem. This relates

the ionization produced in a cavity with the energy imparted by ionizing radiation to the walls surrounding the cavity. In the special case when wall and gas have the same atomic composition, the cavity may be of any size (provided the radiation field is constant in the region immediately surrounding the collecting volume). If the walls have the same composition as tissue (e.g., are made of tissue-equivalent plastic), the absorbed dose is proportional to the ionization per unit mass of gas and the factor of proportionality is W, the average energy expended in the production of an ion pair. This quantity is known within about 2 to 3 percent for all gases of interest and varies little with energy or nature of the ionizing particle.

12.3. This method has been used for the determination of fast neutron doses with the use of ionization chambers constructed of hydrogenous materials. Use of tissue-equivalent materials (i.e., substances having the same atomic composition as tissue) permits dose measurement of neutrons of any energy. The principal practical limitation of the technique is that such chambers do not discriminate against other radiations, and consequently the total tissue dose is measured in a mixed radiation field where gamma radiation is also present.

12.4. Chambers having nonhydrogenous walls may be employed in an effort to measure the gamma radiation selectively and to obtain the neutron dose by subtraction. However, if the atomic number of the wall material is low enough to exhibit an approximately air- or tissue-equivalent response at low photon energies, the chamber also has a certain neutron response beyond neutron energies of about 0.5 Mev. Theoretical computations indicate that between 1 and 10 Mev the response of a chamber having nonhydrogenous walls and filled with CO_2 should vary between 5 and 25 percent of that of a tissue-equivalent chamber. Because of the wide variation and the fact that it is an erratic function of neutron energy, corrections are difficult to apply. Therefore, in a mixed radiation field the response of such a chamber may be merely interpreted as an upper limit of gamma contamination. A subtractive measurement of neutron dose is consequently inaccurate, particularly when the relative amount of gamma radiation is large. However, at low neutron energies a graphite or Teflon chamber filled with CO_2 provides an excellent means of selective measurement of contaminating gamma radiation.

12.5. Methods of dosimetry have been developed in which an attempt is made to distinguish between electrons

and heavy particles by employing proportional counters and discriminating circuits. Instruments of this type, such as the count-rate dosimeter and the pulse-energy integrating dosimeter, are discussed in appendix 4.

12.6. As explained in section 5, multiple scattering of fast neutrons leads to an increase in dose if the mass of tissue exposed is increased. Many neutron dosimeters contain only enough hydrogenous material to establish proton equilibrium, and thus record essentially only the first collision dose. In order to determine the tissue dose of interest here, it is necessary to use such devices in properly designed tissue-equivalent phantoms. For fast and relativistic neutrons, the use of such a phantom may be omitted if the curves used in figure 15 (appendix 1) are applied to the reading obtained with the bare dosimeter. However, with intermediate and thermal neutrons, geometrical exposure conditions are so critical that phantoms are essential.

12.7. Appendix 4 gives further information on instruments that may be employed in dose measurement.

III. Radiation Protection in Installation and Operation of Neutron Sources

13. Types of Sources

13.1. Radioactive sources are of two types, (α,n) and (γ,n). In an (α,n) source, the alpha emitter is mixed with the target material.

13.11. Polonium–210 and plutonium–239 are the most suitable radioactive alpha sources[4] from the standpoint of low gamma-ray activity and compactness. Both elements, however, are among the most dangerous ones when ingested or inhaled. Special precautions must be taken to prevent their escape by providing durably sealed containers.

13.12. Ra^{226}, in equilibrium with its daughters, is another source of α-rays of convenient half life, but of equally high toxicity. Its containment is exceptionally important because of the dangers due to its daughter product Rn. In addition, due to its copious photon emission, it represents a gamma-ray hazard.

13.13. The target materials most commonly used are Li, Be,[5] and B. These materials in powdered form, intimately mixed with the alpha emitters, are sealed in metal containers.

[4] In the polonium sources Po^{210} may be present as daughter product of long-lived RaD (Pb^{210}).

[5] Be constitutes a recently recognized chemical toxicity hazard.

13.14. (γ,n) sources are usually called photoneutron sources. They consist of sealed containers enclosing emitters of gamma rays of energy high enough to detach neutrons from target nuclei. The latter, usually deuterium or beryllium, are placed in spheres or cylinders which surround the gamma emitter. Photoneutron sources have been used mostly to generate moderate fluxes of neutrons of fairly homogeneous energies below 1 Mev.

13.2. *Constant-voltage accelerators,* as used in the production of neutrons, are usually either Van de Graaff or Cockroft-Walton machines. They are essentially the same as those used for the production of X-rays, except that the polarity of the high-voltage electrode is reversed, and this electrode is provided with a positive ion source instead of a negative electron source. In the Van de Graaff generator the charge is conveyed to the high-voltage electrode mechanically by means of a belt; in the Cockroft-Walton it is conveyed by a cascaded sequence of voltage-doubling circuits, each comprising a condenser and a rectifier. Most Van de Graaff generators operate below voltages of 5 Mev; Cockroft-Walton generators below about 1 Mev.

13.3. *High-frequency accelerators* include the cyclotron, synchrocyclotron, betatron, synchrotron, microtron, and linear resonance accelerator.

13.31. The cyclotron, or magnetic resonance accelerator, is a device for accelerating light ions. The ions are kept in a spiral orbit by a constant magnetic field and are given successive acceleration when they traverse the gap between the dee-shaped electrodes. The operation is usually continuous in the sense that a beam is produced in each cycle. Light ions, particularly H^+, H^{2+}, and He^{4++}, are accelerated up to 15 Mev per nucleon.

13.32. The synchrocyclotron is a cyclotron modified to allow for the relativistic increase in the mass of the accelerated particles in higher energy ranges. In the frequency-modulated cyclotron the particles are accelerated in pulses of about 1 microsecond duration and the oscillator frequency is modulated. Particle energies range from 15 to several hundred Mev per nucleon. Usual particles accelerated are: H^1, H^2, and He^4.

13.33. The betatron is a circular electron accelerator that has been used to accelerate electrons to energies up to 100 Mev or more. Electrons are injected in a pulse about 1 microsecond in duration. After injection, the electrons are

continuously accelerated and held in a circular path by a changing magnetic field.

13.34. The synchrotron operates on the same fundamental principles as the frequency-modulated cyclotron. The frequency of the oscillator is matched with the frequency of cycling charged particles being accelerated in a closed path. When the synchrotron is used to accelerate protons, both the oscillator frequency and the magnetic field are varied to allow for the large relativistic mass change. Particles to be accelerated are injected into the machine from a smaller accelerator, such as a Van de Graaff accelerator for protons. For electrons when the final velocity is near to the velocity of light, it becomes impracticable to change the oscillator frequency. The frequency is kept fixed and the magnetic field is varied.

13.35. The microtron is a variation of the electron synchrotron in which the magnetic field is held constant and the orbit radius is allowed to increase with increasing electron energy.

13.36. In a linear resonance accelerator the particles travel in a straight path and are accelerated by the electric field of an electromagnetic wave which travels down the accelerating tube. An advantage of the linear resonance accelerator over circular-orbit accelerators is the ease of bringing the beam into field-free space.

13.37. Although the primary hazard from high-energy electron accelerators is usually X-rays, the neutron dose may be comparable to or greater than the X-ray dose outside shielding material of high atomic number, such as lead, which is used to attenuate the X-rays. The chief sources of neutrons are the machine itself, and the point where the beam strikes the wall.

In designing a shield for an electron accelerator, both the X-ray and the neutron hazards must be considered. The neutron hazard may be neglected in comparison to the X-ray hazard for accelerators that operate only below 10 Mev, because the thresholds for the neutron-producing reactions are at about this energy. For quantitative information on neutron production and shield design at higher energies, see appendix 7. Detailed discussion of the shielding problems arising in the operation of electron accelerators is given in Handbook 55 of this series.

13.4. *Reactors.* Neutron production in reactors occurs as a result of the fission process, which is maintained at a high rate by means of a carefully controlled chain reaction. The usual operating mode is one in which the reactor is

critical, in which case the number of fissions occurring is substantially constant in time. This is achieved by a definite arrangement of fuel elements (uranium or plutonium) and adjustment of neutron absorbers (control rods).

14. Neutron Production

14.1. Neutrons are produced in accelerators by the interaction of high-speed nuclear particles (usually positively charged) and target nuclei. Depending on the nature of the reaction involved, there may be evolution or absorption of energy. The net gain or loss of energy is usually denoted by the symbol Q. Q is defined by

$$Q=\sum_{i} m_i c^2 - \sum_{r} m_r c^2, \qquad (3)$$

where m_i are the masses of the interacting particles, m_r the masses of the resultant particles, and c the speed of light. A list of important reactions used in neutron production as well as corresponding Q values is given in appendix 3.

14.2. *The fission process.* Given sufficient excitation energy, certain heavy nuclei are prone to divide into two or more large pieces. This energy can be obtained by absorption of a gamma ray, but is attained more commonly following production of a fissionable isotope by absorption of a neutron. In this case, the binding energy of the neutron constitutes the excitation energy to induce fission. The most notable example of this process is the absorption of a slow neutron by U^{235} to produce U^{236} in a highly excited state which immediately fissions. At the time of fission, neutrons, gamma rays, and occasionally a high-speed proton are given off, in addition to the two (and sometimes more) large fission fragments. The neutrons are, on the average, 2½ in number and of mean energy of about 2.5 Mev. The total gamma-ray energy per fission has an average value of about 7.5 Mev.

15. Other Radiation Hazards Associated with Neutron Production

15.1. In section 9, a brief description of the various types of reactions leading to associated gamma-ray hazards has been given. Protection against photon radiation per se is discussed in two publications of the NCRP. Specifically, Handbook 60, X-ray Protection, and Handbook 55, Protec-

tion Against Betatron-Synchrotron Radiation Up to 100 Million Electron Volts, should be consulted as guides to protective measures.

15.2. Radioactive photoneutron sources represent, from the protection standpoint, gamma-ray hazards exclusively. Thus, a Na^{24}-D_2O source delivers a neutron dose rate of the order of only 10^{-5} times the gamma-ray dose rate at the same distance. In most cases the gamma dose to be expected from a 1-curie source at 1 meter varies within narrow ranges, i.e., 0.1 to 2.0 rads/hour.

15.3. As mentioned, the transmutations produced by neutrons in the vicinity of sources lead to two modes of gamma-ray emission. One is capture radiation that is emitted simultaneously with neutron absorption. A list of the gamma-ray energies emitted in capture is given in appendix 4.

15.4. In addition, nuclei produced as a result of capture are often radioactive, and particle accelerators of high energy and particle flux—such as cyclotrons—represent producers of radionuclide sources of considerable activity and significant half lives. Any part of the accelerator or its surroundings is potentially a source of beta and/or gamma radiation, which must be evaluated by competent beta- and gamma-ray monitoring. In addition, the air and loose dust in the room can be activated sufficiently to require delay in entering the room. The responsible officer should in either case be guided by recommendations in Handbook 42 (Safe Handling of Radioactive Isotopes) and in Handbook 52 (Maximum Permissible Amounts of Radioisotopes in the Human Body) concerning the over-all operation of these machines.

16. Radiation Protection Considerations in the Design of Neutron Sources

16.1. Because of the diversity of source types and reactions it is impossible to furnish detailed instructions on the safe design of all types of neutron sources. The following recommendations are designed to cover the majority of conditions likely to be encountered in practice and to serve as a general guide for the remaining situations.

Neutron sources may be classed into essentially four groups of increasing output. As the protection problems involved are somewhat different, these types will be discussed separately.

16.2. *Radioactive sources.*

16.21. As mentioned above, these can be classed into (γ,n) and (α,n) sources. For (γ,n) sources, gamma radiation is usually the primary hazard and protection should be designed according to recommendations contained in other handbooks in this series. This is also true for Ra-Be sources where alpha particles are actually employed as bombarding particles. However, when the gamma radiation from a Ra-Be source is reduced by massive shielding, particularly lead which absorbs very few neutrons, the neutron hazard may become appreciable and should not be overlooked.

16.22. For the other (α,n) sources(Po-Be, Po-B, Pu-B, etc.), the neutron hazard is usually the primary one. Sources exceeding dose rates of 30 mrems/hr at the surface shall be stored in labeled containers (see Rules below). Usually paraffin in considerable thickness is employed for sources exceeding 100 μc (see appendix 5). The paraffin should be encased to facilitate handling.

16.23. Sources employed in routine use must be sealed hermetically because of serious problems of chemical toxicity. Recommendations on the design of sealed sources are given in Handbook 54 of this series.

16.3. *Constant-voltage accelerators.*

16.31. This group comprises Van de Graaff generators, Cockcroft-Walton generators, and similar installations that involve accelerating voltages up to 5 Mev and beam power up to about 100 watts. Machines of this type represent the most difficult problem in radiation protection because of the variety of possible reactions involved. Many of these may produce but a slight hazard, but others (such as $H^3(d,n)He^4$) make the installation a dangerous source of neutrons. There is often a frequent change from one type of reaction to another, and the schedule of operations may be intermittent or erratic.

16.32. It is beyond doubt safest and simplest to design the protection around such machines so that the dose rate of 2½ mrems/hr is not exceeded outside the shielding when the machine is run at maximum neutron output (usually $H^3(d,n)He^4$). The shielding required is of the order of a few feet of concrete, which adds comparatively little to the cost of a special building designed to house the machine. Such a structure is highly desirable because experience indicates that efforts to fit such generators into existing structures result in cluttered arrangements that tend to be inconvenient for purposes of research, and unsafe radiologically as well as otherwise (electrical and mechanical hazards).

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16.33. On the other hand, it is realized that at many of these installations the provision of desirable shielding is structurally impossible or otherwise very difficult, particularly when it becomes necessary to fit them into existing buildings. Consequently the dose rate of 2½ mrems/hr might be exceeded at accessible locations, perhaps even at the operator's console. At present it does not seem warranted to categorically declare this practice unacceptable, but it certainly is not recommended. Furthermore, at installations where this condition exists it becomes mandatory that constant checks be performed, that operations be limited to insure that not more than the maximum permissible weekly dose is received by any personnel, and that operations be suspended if this would occur otherwise.

16.34. Location of the target below the ground level is usually preferable. Some experimenters prefer low-scatter flooring and building walls made of thin sheets or grids. Directing the beam away from occupied areas, and particularly the operator's console, often results in some shielding economy. Access to the region around the target should also be from other than the beam direction.

16.35. Because the ion beam produced by these machines can be "piped" over considerable distances with comparatively little effort, it is quite practicable to bring it into a shielded enclosure containing the target. However, such an enclosure should permit sufficient space for any experimentation that can be foreseen. In addition, effort should be made to shield any other structures that could intercept the beam.

16.36. In the design of these generators it should be kept in mind that secondary electrons are likely to produce X-rays and that modifications in basic design might lessen the protection requirements for this radiation.

16.4. *High-frequency accelerators.*

16.41. In virtually all of these machines massive shielding is mandatory, and few new installations can be planned that do not require a special building designed for the purpose. Because of structural considerations, there is a tendency to keep shielding of the space immediately above the machines at a minimum. As a result, a large fraction of the neutrons observed outside the shield may originally have escaped through the top and then been scattered downward by other structures or the air above. Most circular accelerators require shielding of the entire machine. The shielded enclosure should permit sufficient room for experimentation, particularly in the beam direction.

It is highly desirable to provide space for additional shielding that may become necessary in case methods are found (as has been the case in the past) to boost the radiation output.

16.42. No installation should be designed in which the dose rate of 2½ mrems/hr is exceeded outside the shielding under conditions that can be foreseen. This is particularly necessary because additional exposure may be incurred during periods when the beam is off, as personnel, when working inside the enclosure, are likely to be exposed to gamma and beta radiation arising from induced activity.

16.43. Installations that are subterranean or built into a hillside are likely to result in appreciable economy because advantage may be taken of the shielding effects of soil.

16.44. Access to the shielded enclosure may be provided through either a maze or movable (usually power driven) shielding blocks. The latter design requires less space and is likely to be more economical.

16.45. Nuclear reactors: Because of the complexity of reactor design and the great variety of reactor types in existence, it appears impractical to provide any definite recommendations. Protection design should be based on pertinent experiences gained with existing types.

17. Stationary Shields

17.1. The shielding of neutron sources is at present not as well understood as that of gamma sources. Consequently the information on shield thicknesses is less exact, and shields should in general be more overdesigned. The formulas given below will include adequate safety factors for general use. It is a fact of practical importance that adequate shielding against neutrons will in general suppress gamma radiation to permissible levels at both reactors and accelerators. Water and other hydrogenous shields constitute an important exception to this rule. In the use of radioactive sources, and in particular photoneutron sources, gamma shielding is a separate and often more important problem.

17.2. In the design of buildings planned to contain neutron generators, shielding should be the first item considered because of both size and weight.

17.3. Ordinary or heavy aggregate concrete or earth are the recommended materials in most installations. Any economy by the use of water-filled tanks is likely to be offset by maintenance difficulties. In addition, evaporation rep-

resents a serious hazard, although it may be retarded by the addition of oil.

17.4. Paraffin or oil is a fire hazard, and neither should be used in large stationary shields.

17.5. Methods of shielding calculations are outlined in appendix 5.

18. Movable Shields

18.1. It is often necessary to operate with temporary shielding, in which the shield is not cast into place but rather is built up of separate blocks. As all such installations are subject to flaws or cracks which are left in the assembly, a detailed survey should be made prior to routine operation of the source. In general it is found that a carefully laid unmortared concrete block shield is nine-tenths as effective as a monolithic poured structure. Lead bricks laid with care show similarly reduced attenuation. All vertical cracks should be staggered to reduce leakage. Gravity is usually sufficient to keep horizontal cracks closed.

19. Unusual Hazards

19.1. Even though shields may be satisfactory when installed, they may deteriorate, either suddenly or gradually, so that it is necessary to monitor the radiation outside routinely. Examples of such deteriorations are the loss of water from a shield tank or from a hydrogenous shield material, the development of cracks in concrete due to settling, or the loss of hydrogen due to radiation damage in paraffin or oil. Reactors in water pools should be equipped with proper monitor systems to warn of lowered water level.

20. Procedures to be Implemented in Case of Overexposure

20.1. According to principles discussed at length in Handbook 59 of this series, the tolerance status of an individual is altered if, once in his lifetime and within a period of 1 month or less, he is exposed to absorbed doses exceeding 25 rems to the whole body or a major part thereof. For the purposes of this Handbook, one-half of this dose (12.5 rems) represents the limit above which exposures shall become a problem that must be referred, for joint consideration and appraisal, to recognized experts in medical radiology, radiobiology, and radiological physics.

20.2. Although the clinical management of such an individual is obviously the province of the physician, its form and course, as well as the individual's tolerance status, will be influenced by the magnitude of the dose received. Hence, every effort should be made to evaluate it as accurately and as soon as possible.

20.3. The extent of this effort will in turn depend on the availability of suitable personnel-monitoring devices on the body of the person involved. On the basis of the very scant information on the subject, and the unexpected nature of accidental exposures, no hard and fast rules can be given as to the experimental approach.

20.4. Ideally, a suitable personnel-monitoring device is an apparatus of ample dosimetric range capable of registering separately doses of gamma rays and of neutrons. Nuclear track emulsions, specially packed, tissue-equivalent and graphite-ionization chamber pairs, and other dosimeters that fulfill these requirements to a great extent, are described in appendix 4. If available, the readings given by these instruments can be interpreted directly in the evaluation of body doses, once the influence of shielding by the operator's body and the spatial characteristics of the radiation field are independently established. The unique merits of a true dosimeter are readily appreciated whenever the acute exposure is the consequence of sudden damaging overload of the neutron generator, leading to delay or to physical inability to undertake dosimetric studies under operating conditions duplicating those prevailing at the time of the accident..

20.5. Whenever true personnel dosimeters are, for any reason, unavailable, other integrating personnel detector readings can be used to advantage.

Dosimetric evaluation is then best done with proper dosimeters in the presence of radiations that are as nearly identical as possible to those emitted during the accident, and by relying on personnel detectors as integrating monitors. Because the spectral characteristics of a neutron generator may be critically dependent upon ion energy, particular attention must be paid to duplicating this factor. In this as well as in the previous case, the influence of the operator's body on the instrumental readings can be established with the use of experimental mock-ups simulating in size and composition the person's body.

20.6. Whenever personnel monitor readings are unavailable, dosimetric evaluations become much more difficult to obtain and approach the complexity of a research prob-

lem. The presence of area monitors, located in the radiation field, may be of substantial aid in establishing experimental conditions relevant to a dosimetric study, i.e., conditions of exposure bearing quantitatively known relation to those of the accident. Readings from area gamma-ray monitors should be sought and recorded because they may serve in some instances as integrator detectors.

20.7. Tools, apparatus, and other objects containing chemical elements such as Sn, Sb, Mn, Cu, Al, Cd, Hg, Ni, Au, Fe, may serve as neutron monitors if their induced radioactivity can be measured at known and preferably short times after the exposure. The movement of exposed individuals, however, must be established as accurately as possible, and pertinent questioning of personnel and witnesses should be undertaken promptly and testimony recorded with the least reliance on memory. Similarly, the radioactivity induced in any of the objects worn may serve the purposes of a neutron integrating detector, once the itinerary of the objects in the radiation field and its location on the person's body is established. Gold jewelry is particularly suited for this purpose; hence radioactivity in rings, wristwatch cases, bracelets, earrings, medals, etc., should be investigated. To a lesser extent, money (Cu, Ni, Ag) and other base metals likely to be on the person (in the form of identification badges, fountain pens, pencils, belt buckles, garters, costume jewelry, etc.) may serve the same purpose.

20.8. Because most of the radioactivities induced in these elements are short-lived and the result of small cross sections, speed and sensitivity of measurement are likely to prove critical in estimates of this sort. Very important also to ultimate interpretation is the critical analysis of the activities present, hence radiometric and spectrometric analysis of the samples are highly desirable.

20.9. The specific activity of Na^{24}—and to a lesser extent of P^{32}—in the blood serum and urine of the exposed individuals has been utilized for this purpose. The specific activity of the former should be evaluated by external gamma-ray measurements. This will require considerable accuracy and sensitivity, but will eliminate any uncertainty as to the rapidity of exchange with the other Na deposits. In procedures of this type care should be taken to remove contamination that may be present on the skin, hair, nails, and clothing of the individual.

20.10. It should be realized that most of the induced radioactivities thus far mentioned are caused in overwhelming measure by thermal neutrons and, therefore, they can be generated in part also by fast neutrons moderated by the body. A better indication of the fast neutron flux can be obtained from the threshold reaction P^{31} (n,p) Si^{31}. This element is readily available in matches and in urine.

IV. Rules for Protection against Neutron Radiation

Scope of rules. The rules set forth below are considered essential for the avoidance of hazards attending exposure to neutron radiation. They apply to other radiations only insofar as they might occur simultaneously with neutrons and add to the exposures incurred. Other handbooks in this series deal more explicitly with protection against other ionizing radiations. The present rules are not concerned with any electrical, mechanical, toxicological, and other nonradiation hazards that might arise in the operation of neutron sources, except as they affect radiation safety.

A further restriction applies to the case of reactors. The rules set forth below extend only to protection during normal operation at power levels anticipated. The prevention of abnormal conditions that entail particularly severe radiation hazards is a complex technological problem that will not be discussed here.

Finally, for obvious reasons, no specific recommendations can be made regarding protection in the vicinity of classified assemblies emitting neutrons.

21. Maximum Permissible Dose

21.1. For a radiation worker of age N, the accumulated RBE dose shall not exceed $5(N-18)$ rems in the blood-forming organs, the lens of the eye, and the gonads. In the skin it shall not exceed $10\ (N-18)$ rems.

21.2. The weekly RBE dose incurred by a radiation worker shall not exceed 300 mrems. In exceptional cases where it is necessary for a person to receive larger doses, the unit of time may be extended to 13 weeks, provided that the dose accumulated during this period does not exceed 3 rems.

21.3. If detailed information on the nature of the ionizing radiations is not available, the RBE shall be assumed to be 10.

21.4. If the portion of the tissue dose contributed by the various radiations is known, each dose in mrads shall

be multiplied by the appropriate RBE to obtain the dose in mrems. The RBE of electrons (whether primary or produced by electromagnetic radiation) shall be taken as 1.0. The RBE of neutrons shall be taken as 10, except that, if the distribution of neutron energies is known, the RBE values in table 2 may be applied.

21.5. Any person while occupying regions outside the controlled area shall not incur a dose of more than 125 mrems in a 3-month period. If such regions contain a residence or regular place of work, the dose rate in any building located therein shall be less than 125 mrems in a 3-month period. It is recommended that even in the absence of such buildings the dose rate should be less than 125 mrems in a 3-month period. If it is more, it shall be the duty of the radiation protection officer to assure himself that there is no likelihood that any person while remaining in these areas will receive doses in excess of 125 mrems in a 3-month period.

21.6. In *exceptional* cases where operations of the source would be virtually impossible otherwise, the permissible dose received outside the installation may be averaged over 1 year, provided the dose received in the period of 1 calendar week does not exceed 300 mrems.

22. Radiation Protection Officer

22.1. Personnel responsible for work with neutron sources shall also be responsible for radiation safety. If a neutron source is capable of delivering more than 300 mrems per work week due to all ionizing radiations emitted in accessible regions inside or outside of any externally applied shielding, a radiation protection officer shall be designated by the management concerned. His responsibilities shall include:

a. Furnishing of technical assistance in the planning and executing of work insofar as radiation safety considerations are involved.

b. Appraisal of operation of the source with regard to the radiation safety rules set forth below.

c. Notification to personnel working near the source of any special hazards that may exist.

d. Awareness of exposure of such personnel from additional sources of ionizing radiation.

e. Reporting of radiation hazards or unsafe practices to the proper authorities for suitable action whenever necessary.

The radiation protection officer should be familiar with the contents of this Handbook, and shall have sufficient training and experience to understand and apply pertinent provisions. A user of the source or a person employed in other capacities may qualify as radiation protection officer. A radiation protection officer may delegate duties but not responsibility. He shall be guided by advice from qualified experts if necessary.

22.2. The radiation protection officer shall be informed of any changes in the mode of operation of the source if these affect the radiation hazard.

22.3. The radiation protection officer should keep records of personnel exposure and area dose levels.

23. Radioactive Sources

23.1. Neutron sources containing materials that constitute a potential hazard of inhalation and/or ingestion due to their radiological toxicity shall be sealed securely or handled under conditions that otherwise eliminate the hazards involved.

23.2. A neutron source having a surface dose exceeding 10 mrems per calendar week, due to all ionizing radiations emitted, shall be marked with a label or stored in a labeled container. The label shall contain information on the nature and intensity of the source.

23.3. Any neutron source having a surface dose of more than 200 mrems/hr shall be normally stored in a labeled container conforming with 23.4.

23.4. When the source is in a storage container, less than 200 mrems/hr shall be delivered at container surface and less than 10 mrems/hr at 1 meter from the container.[6] These requirements need not be fulfilled if the regions around the source are marked as described in 23.5 and 23.6.

23.5. When such a source is removed from its storage container, any accessible location in which more than 7.5 mrems/hr are delivered shall be segregated by a clearly marked barrier or means equally effective in impeding unintentional access.

23.6. When the source is removed from a storage container, any accessible location in which more than 2.5 mrems/hr are delivered shall be clearly marked with signs that indicate the hazard.

[6] These figures have been chosen so that containers may be used for purposes of source shipment in accordance with ICC regulations.

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24. Accelerators

24.1. If dose rates in excess of 2.5 mrems/hr may be incurred in accessible regions outside the shielding, or in the absence of shielding at any accessible location, such regions shall be clearly marked with signs that indicate the hazard.

24.2. Any accessible region inside or outside shielding where a dose rate in excess of 7.5 mrems/hr exists shall be segregated by marked barriers that impede unintentional access. Such areas shall be provided with clear visual indication whether the accelerator is on.

24.3. If the installation is provided with a shield segregating accessible locations, and if a dose rate in excess of 7.5 mrems/hr can be received inside the shield, additional provisions shall be made for audible indication that may be either intermittent or continuous, lasting for at least a 10-second period prior to turning on the beam.

24.4. If exit from the shield cannot be effected without motion of doors or other similar impediments, provisions shall be made that:

a. Such doors are interlocked with the accelerator controls in such a way that neutron production is impossible with the door open.

b. The doors can be opened from the inside of the enclosure.

c. At least one clearly marked crash button is provided inside the enclosure to suspend accelerator operation in such a way as to make neutron production impossible. It shall be possible to readily reach one such button from any point inside the enclosure within 5 seconds of onset of the audible warning signal.

24.5. At any installation where dose rates in excess of 100 mrems per work week may be received outside the shielding, it shall be the duty of the radiation protection officer to insure that a constant check is made that persons do not receive doses in excess of the permissible levels (see sections 21.1, 21.2). "Persons" includes operators, experimenters, visitors, and individuals employed in maintenance or other duties not directly associated with machine operation. Such checks shall include a daily assessment of the operations of the accelerator.

25. Reactors

25.1. On starting up a reactor for the first time the efficiency of the shielding shall be checked minutely. Every

accessible region should be surveyed for radiation, and appropriate adjustments should be made in the shielding.

25.2. Because it is possible for radiation leaks to develop, surveys should be made at least annually and whenever changes have been made in the shield or its perforations.

25.3. Because on startup of a reactor radiation levels increase strongly, an audible warning system shall be installed to insure that all personnel in the vicinity of the reactor are made aware that startup is planned. Personnel must be able to communicate with the reactor operator within a period of time that is less than the warning period.

25.4. In the vicinity of the reactor, continuous visual indication shall be provided to inform personnel whether the reactor is in operation.

25.5. Rigid procedures shall govern the changes of shielding. These shall include consultation with the radiation protection officer.

25.6. Upon first removal of a shielding block, or first testing of a beam trap placed behind a movable shutter, monitoring shall be performed to assess the existing radiation hazard.

25.7. If dose rates in excess of 2.5 mrems/hr can be incurred in accessible locations, such regions shall be clearly marked with signs that indicate the hazard.

25.8. Any accessible region where a dose in excess of 7.5 mrems/hr exists shall be segregated by marked barriers that impede unintentional access.

25.9. Because in the event of an accident it may be necessary to quickly evacuate the building, and probably also the area outside, a plan shall be devised for this evacuation. Responsible persons shall be designated and available at all times for its execution. All personnel normally in the environs shall be made aware of the plan.

26. Surveys at Accelerators and Reactors

26.1. During tune-up and initial operations, surveys of both the gamma- and neutron-radiation dose rate at accessible locations outside the shielding shall be performed as soon as radiation intensities in excess of 2.5 mrems/hr are likely to be produced.

26.2. Prior to routine operation, every accelerator or reactor shall be surveyed. Such surveys shall be repeated whenever operating conditions are changed in such a way that the neutron or gamma hazard may change. In the absence of any such changes, surveys shall be made at

least once a year. In installations where liquid shields are employed, a survey shall be performed at least once every 6 months.

26.3. At installations where liquid shields are employed, special precautions shall be taken to insure that the liquid is maintained at the desired level.

26.4. When personnel are exposed to dose rates in excess of 30 mrems/hr, continuous personal monitoring shall be performed unless reliable estimates of the dose received can be obtained by other means.

27. Health

27.1. Prior to beginning of employment, each person working more than occasionally in areas where significant neutron doses may be received shall have a medical examination. It should be recognized that the examination is directed toward determining the normal or presumed "pre-irradiation" condition of the worker, and toward taking especial precautions if any abnormalities exist that might later be confused with radiation damage. The examination shall include a complete blood count, with determination of erythrocyte, leukocyte, and platelet levels, clotting time, and differential W.B.C. An eye examination shall include a determination of vision with and without glasses, dilation of pupils and examination of the lenses with the slit lamp (corneal microscope). Routine examinations as above shall be done once a year, unless more frequent examinations are indicated by readings of monitoring instruments.

28. Overexposure

28.1. Any exposure in excess of the permissible limit shall receive the immediate attention of the radiation protection officer and the management. Corrective measures shall be instituted to prevent reoccurrence and steps shall be taken to reduce individuals' average exposure.

28.2. An individual who has been exposed to more than 12.5 rems of ionizing radiations in a period of less than 1 month shall be deemed to have suffered an overexposure sufficiently grave to require execution of the steps set forth in 28.3.

28.3. Considerable effort shall be made to determine the dose received, the portion of the body exposed, and the character of the radiation. The legal representative of the institution shall be notified of the amount of exposure, the

extent of the present injury, and the probable outcome. A competent panel, including a physician familiar with radiation risk and injury, a physicist, the radiation protection officer, and a responsible administrator, shall review the medical findings and decide on the advisability of renewed examinations. They shall investigate the reasons for the overexposure and undertake all reasonable efforts to prevent repetition of overexposures.

Appendix 1. Depth Dose

Experimental data on the neutron depth doses resulting from exposure to monoenergetic neutrons are not available. However, results have been reported from Po-Be and Po-B sources. In these two instances, the neutron dose I at a depth X, in terms of the dose I_0 at the surface of a phantom situated at a distance D from the source, can be expressed as

$$I = \frac{I_0}{(D+X)^2} b e^{-x/L}, \tag{4}$$

where b and L are parameters characteristic of the source and the size of the tissue equivalent phantom, as follows:

Source	Phantom	b	L	References[a]
			cm	
Po–B	Small	1.20	10	(1)
Po–B	Semi-infinite	1.56	7.5	(2)
Po–Be	Small	1.33	11.33	(1)

[a] References: (1) W. A. Mills and G. S. Hurst, Nucleonics 12, 17 (April 1954); (2) T. A. Barr and G. S. Hurst, Nucleonics 12, 33 (August 1954).

Theoretical depth dose curves have been computed by Snyder for a variety of neutron energies. The curves shown in figures 2 to 14 apply to an infinite slab of tissue-equivalent material 30 cm thick, irradiated by a broad beam of monoenergetic neutrons with 1 neutron/cm^2 entering normal to one face of the slab.

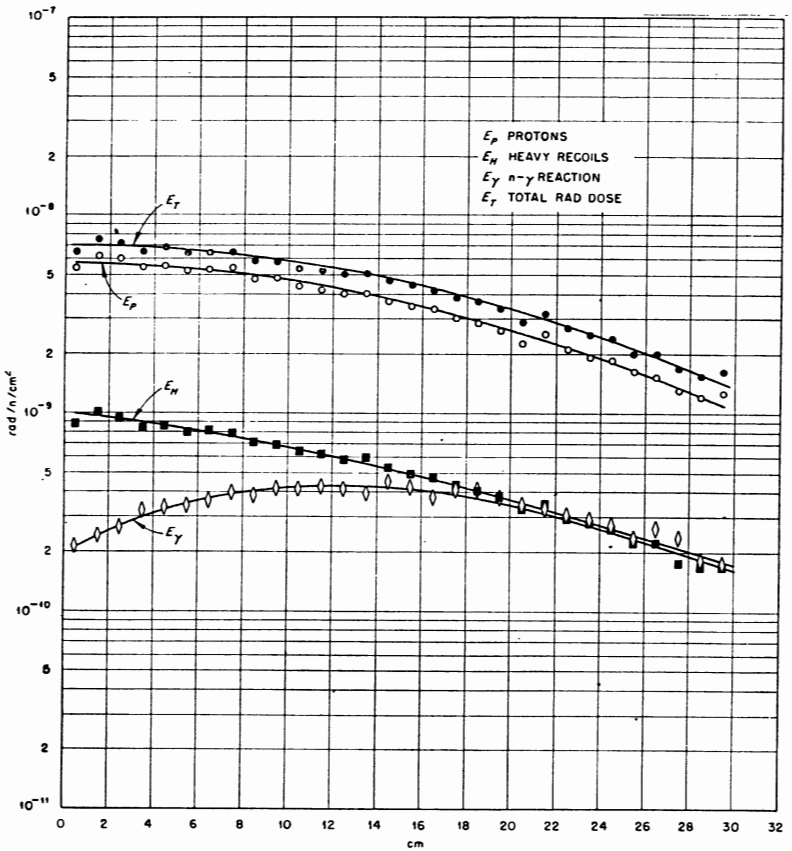

FIGURE 2A. *Absorbed dose (rads) from 10-Mev neutron beam.*

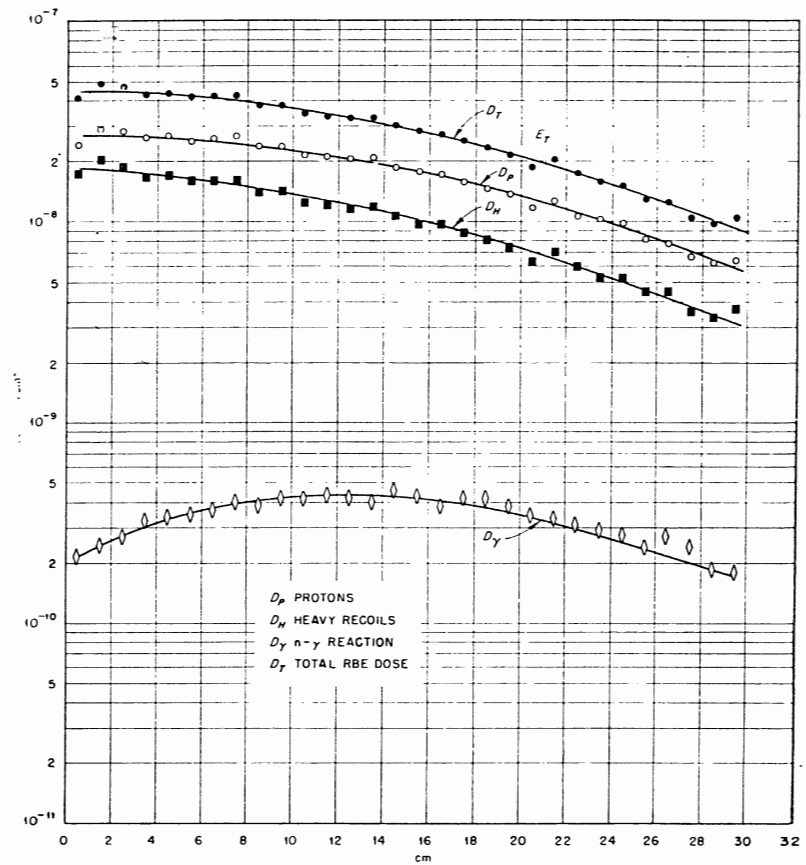

FIGURE 2B. *RBE dose (rems) from 10-Mev neutron beam.*

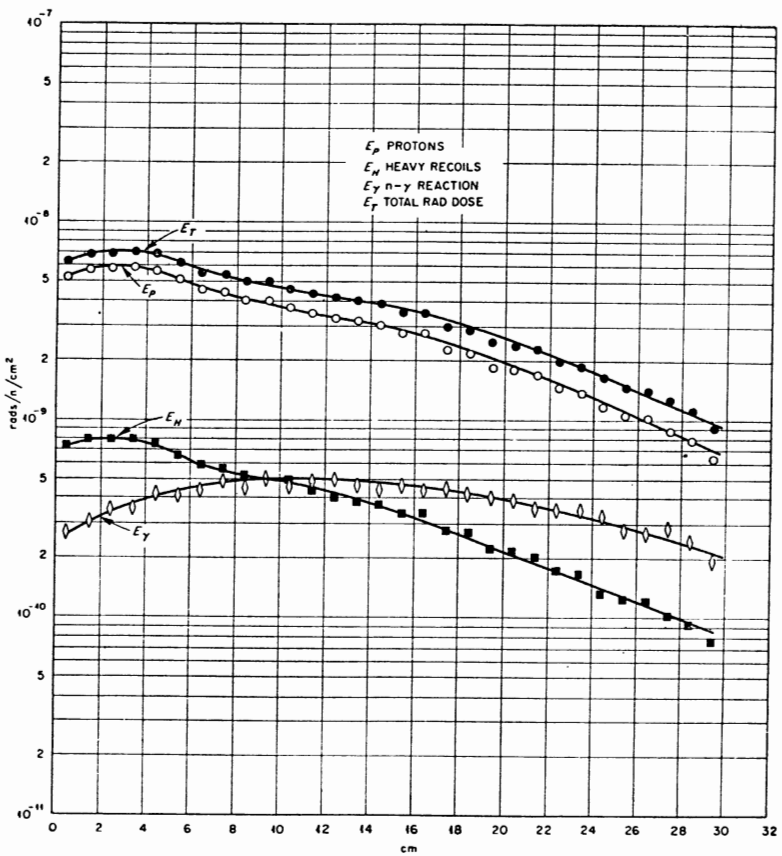

FIGURE 3A. *Absorbed dose (rads) from 7.5-Mev neutron beam.*

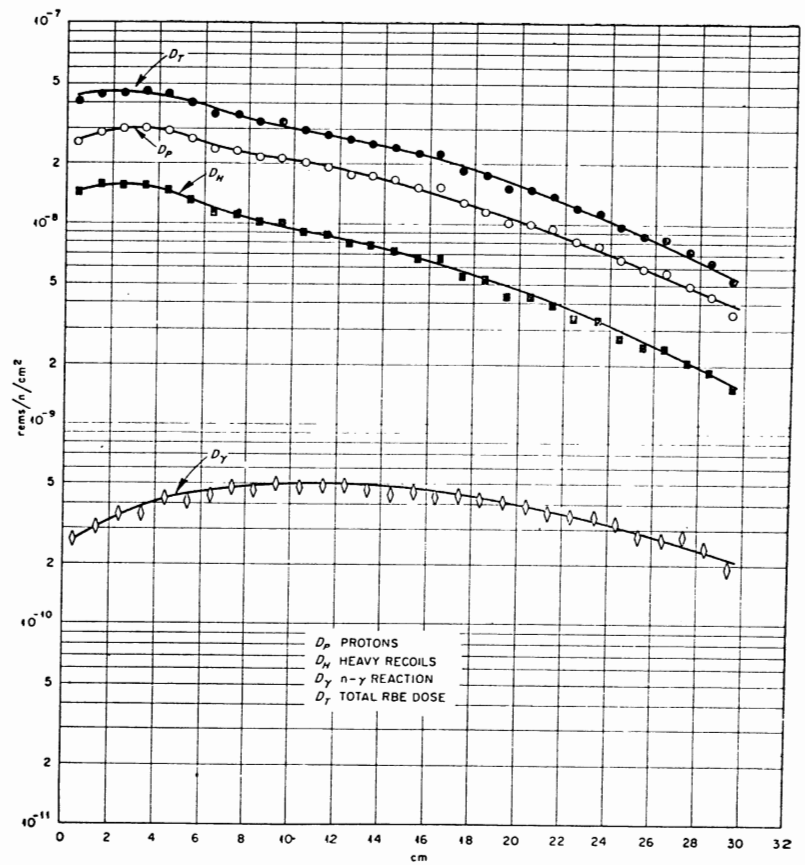

FIGURE 3B. *RBE dose (rems) from 7.5-Mev neutron beam.*

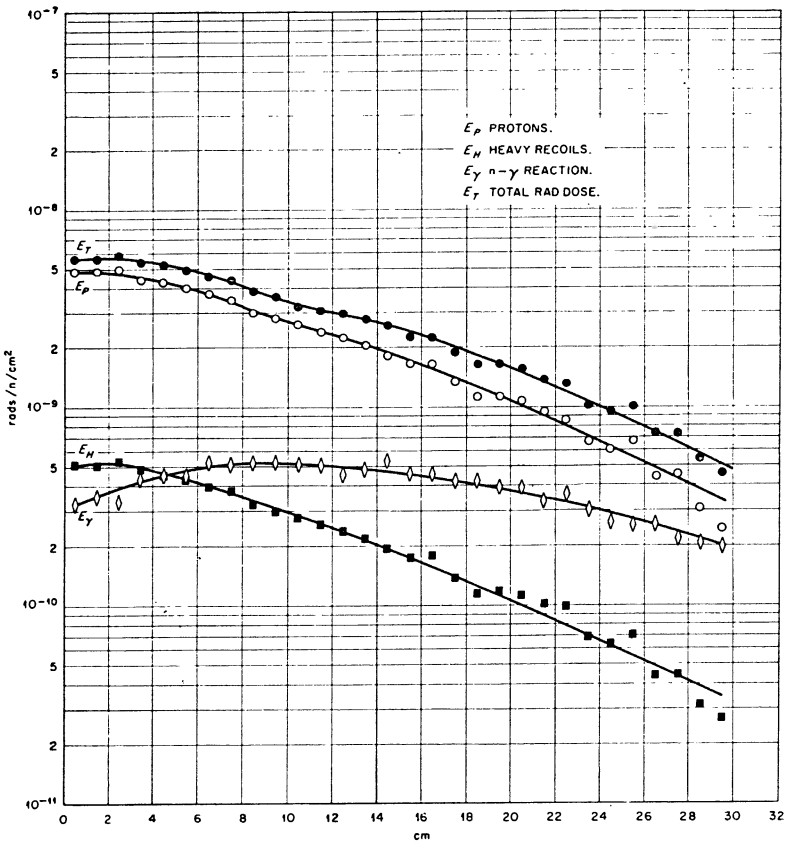

FIGURE 4A. *Absorbed dose (rads) from 5-Mev neutron beam.*

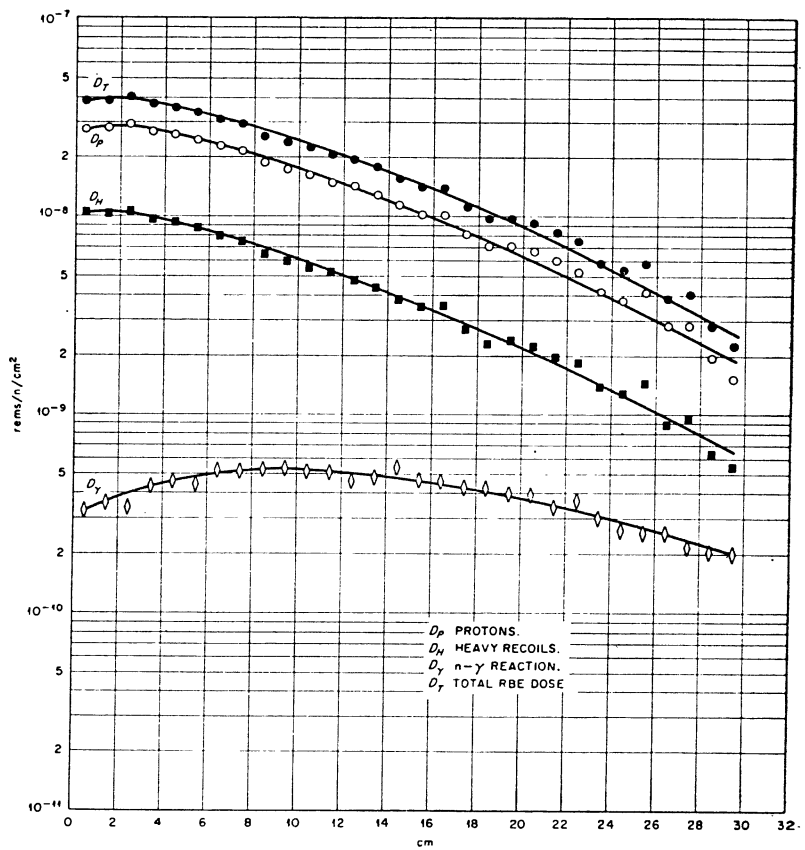

FIGURE 4B. *RBE dose (rems) from 5-Mev neutron beam.*

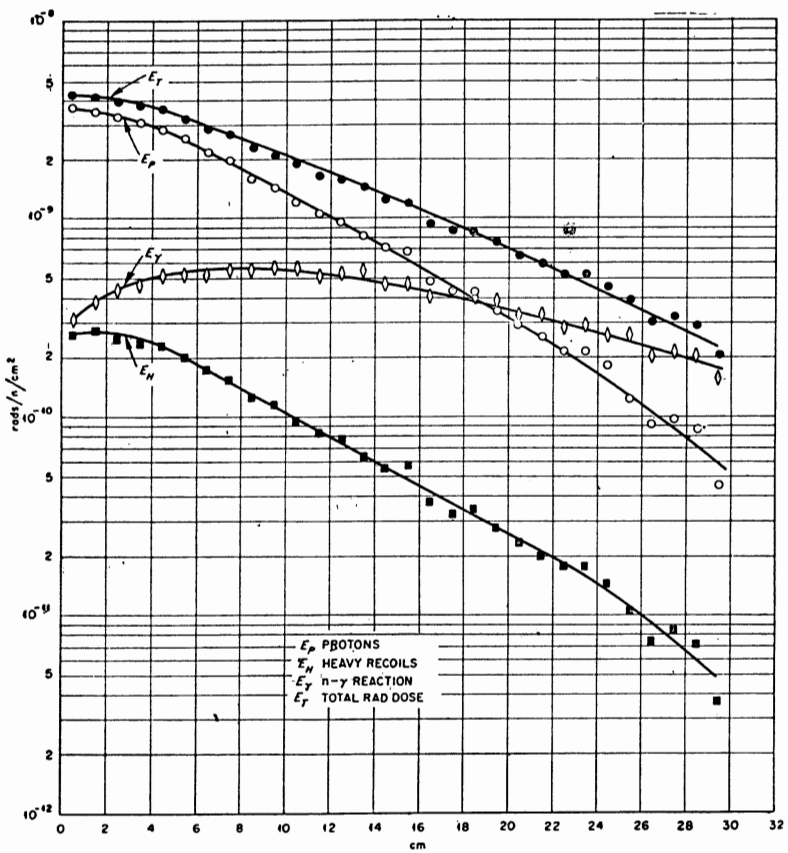

FIGURE 5A. *Absorbed dose (rads) from 2.5-Mev neutron beam.*

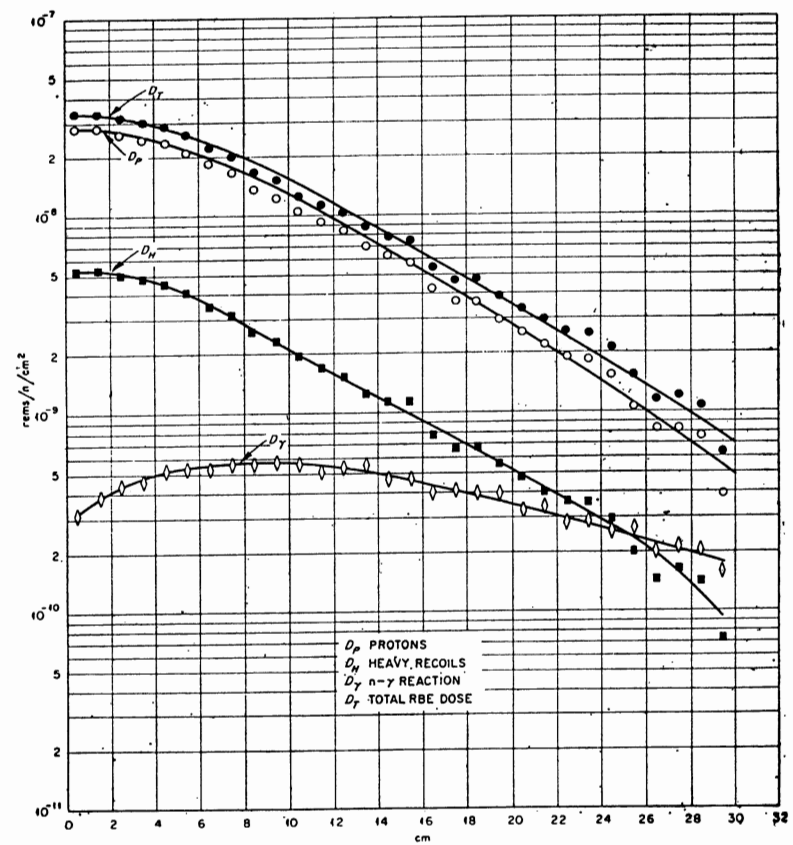

FIGURE 5B. *RBE dose (rems) from 2.5-Mev neutron beam.*

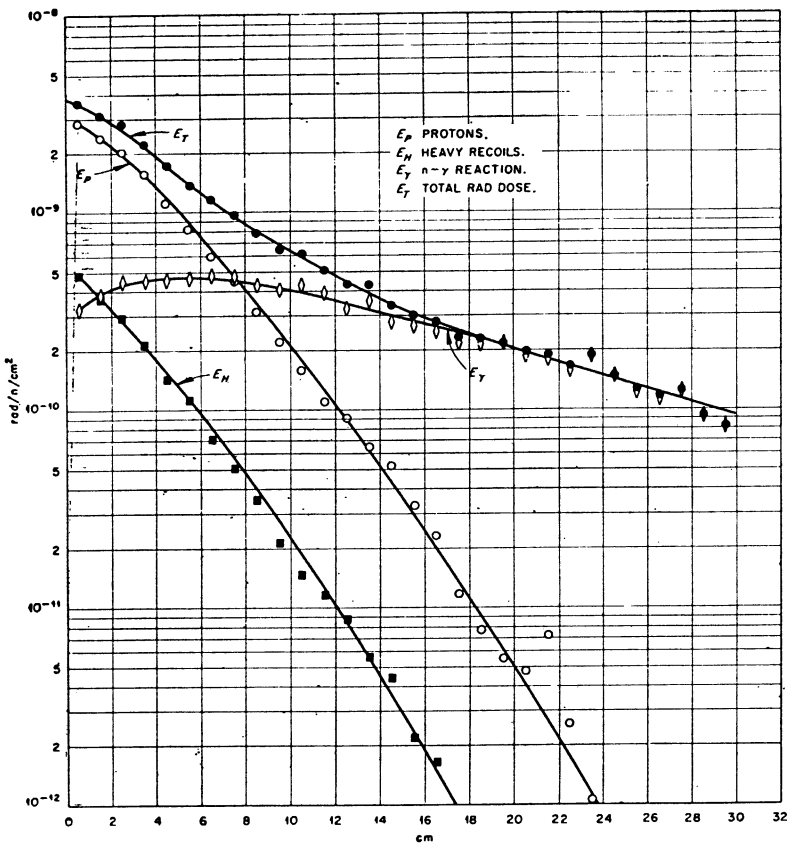

FIGURE 6A. *Absorbed dose (rads) from 1-Mev neutron beam.*

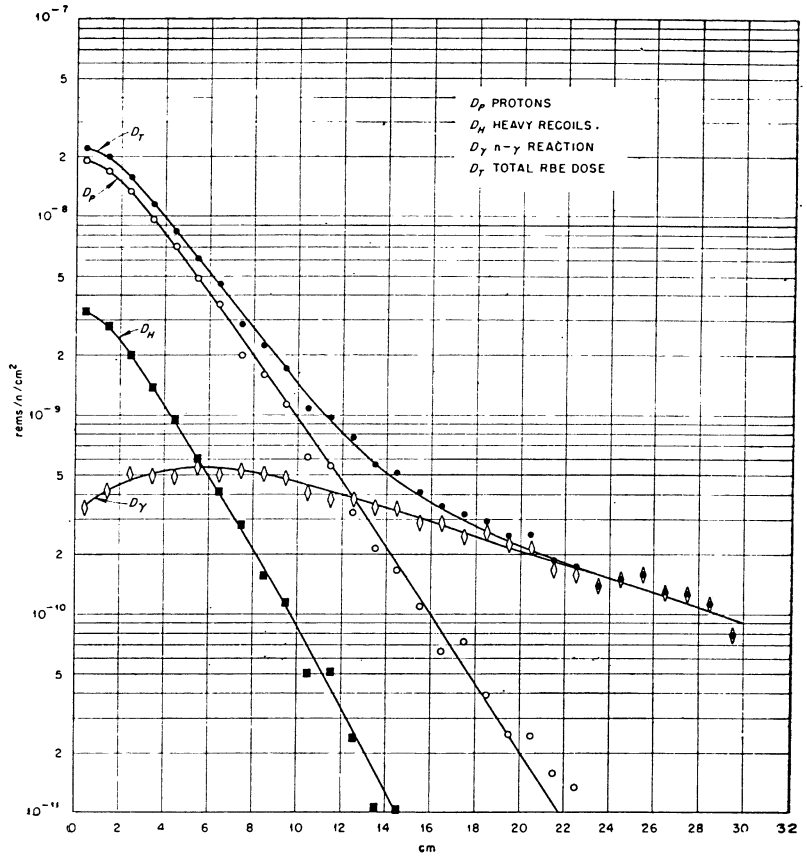

FIGURE 6B. *RBE dose (rems) from 1-Mev neutron beam.*

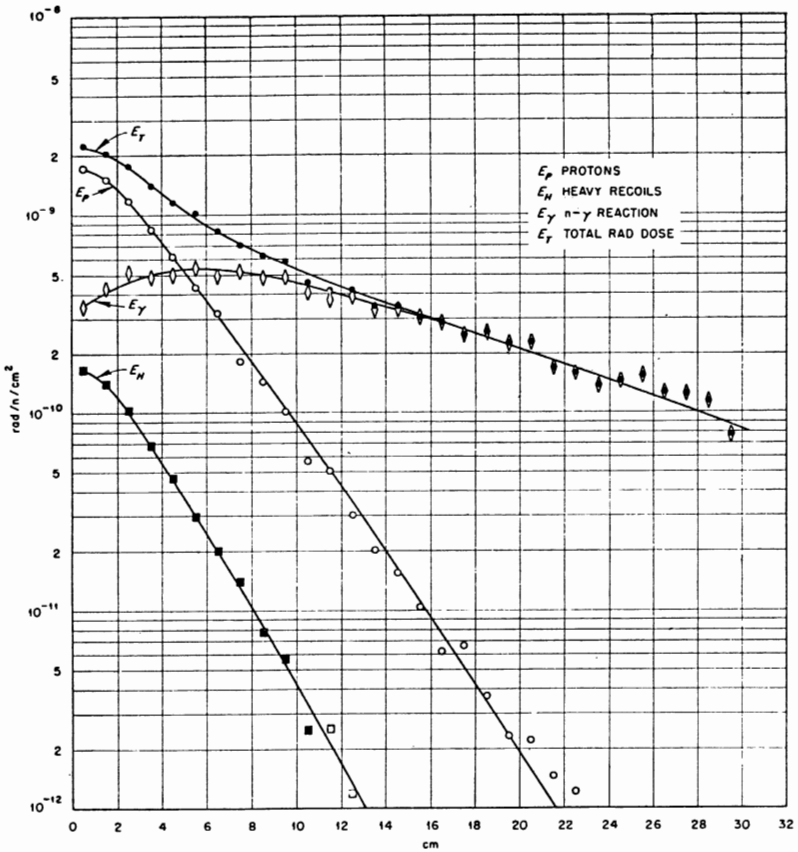

FIGURE 7A. *Absorbed dose (rads) from 0.5-Mev neutron beam.*

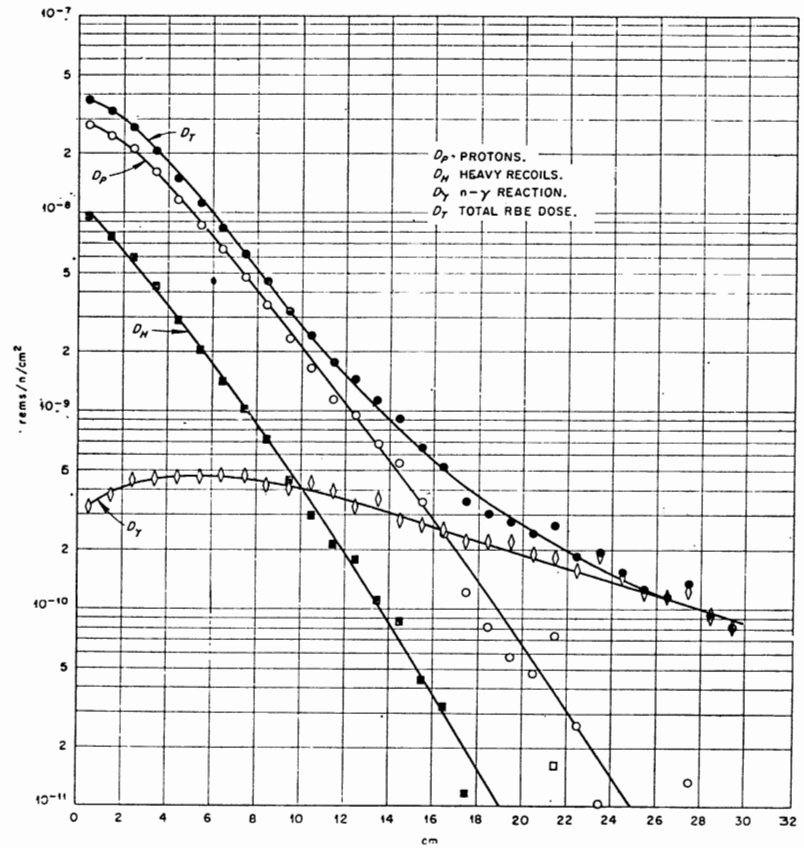

FIGURE 7B. *RBE dose (rems) from 0.5-Mev neutron beam.*

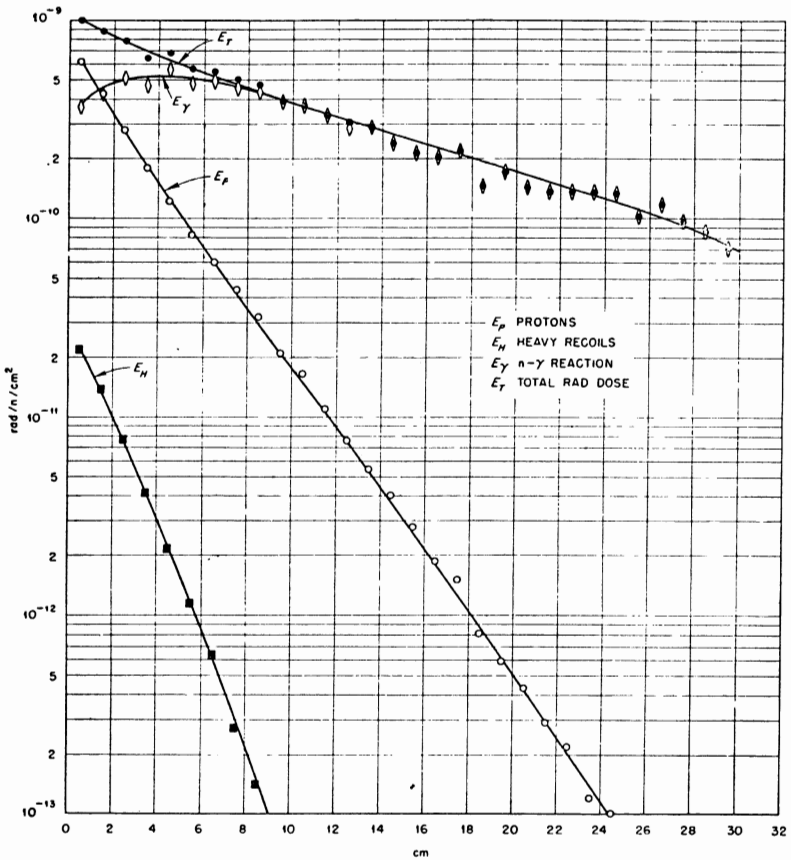

FIGURE 8A. *Absorbed dose (rads) from 0.1-Mev neutron beam.*

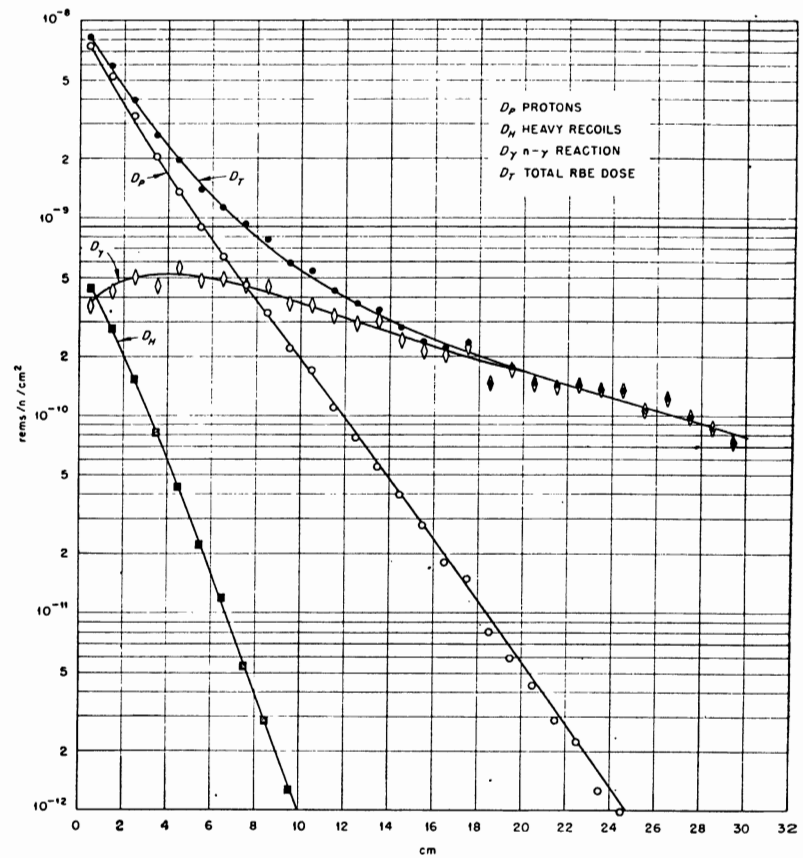

FIGURE 8B. *RBE dose (rems) from 0.1-Mev neutron beam.*

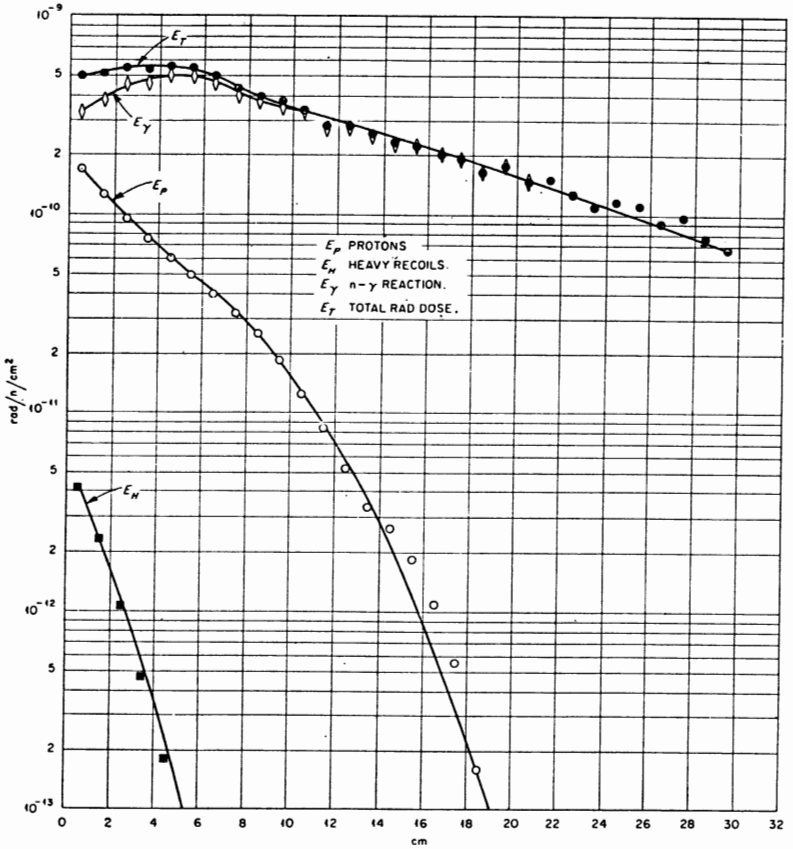

FIGURE 9A. *Absorbed dose (rads) from 0.02-Mev neutron beam.*

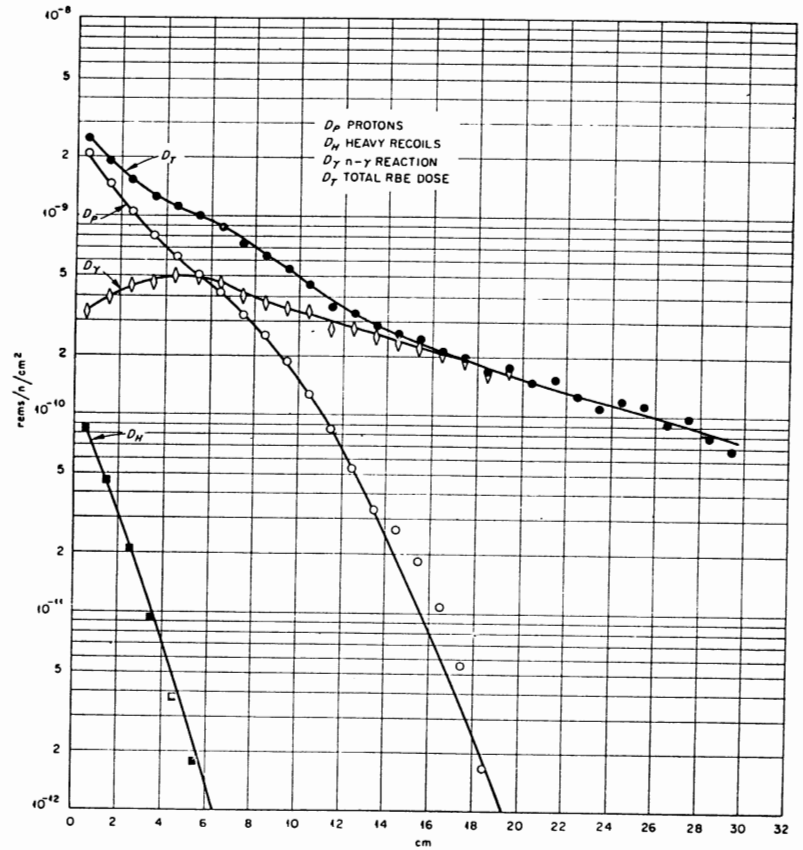

FIGURE 9B. *RBE dose (rems) from 0.02-Mev neutron beam.*

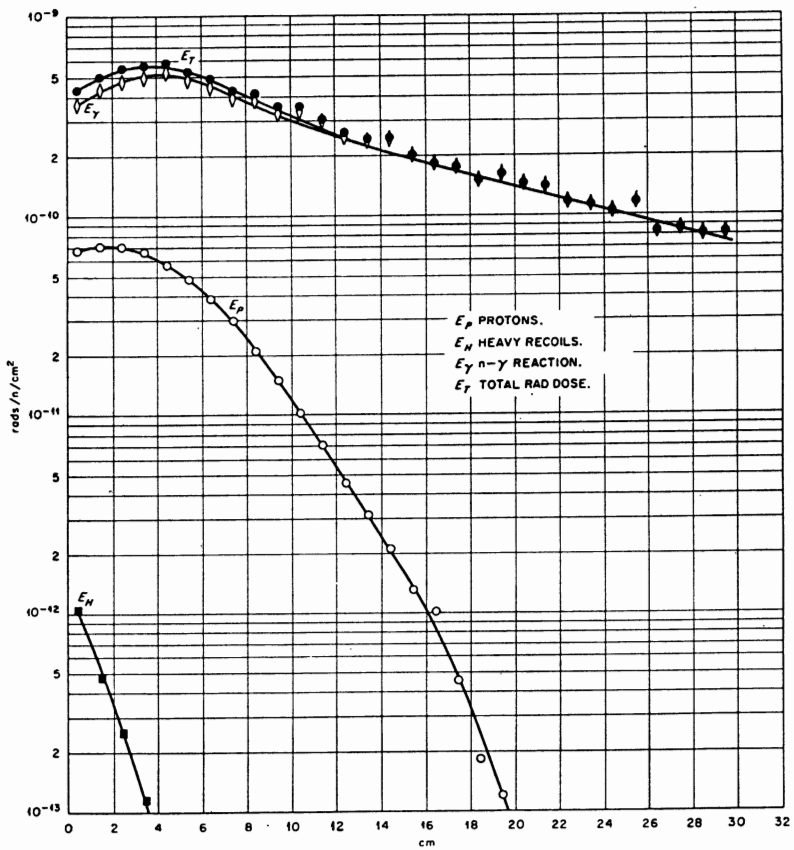

FIGURE 10A. *Absorbed dose (rads) from 0.005-Mev neutron beam.*

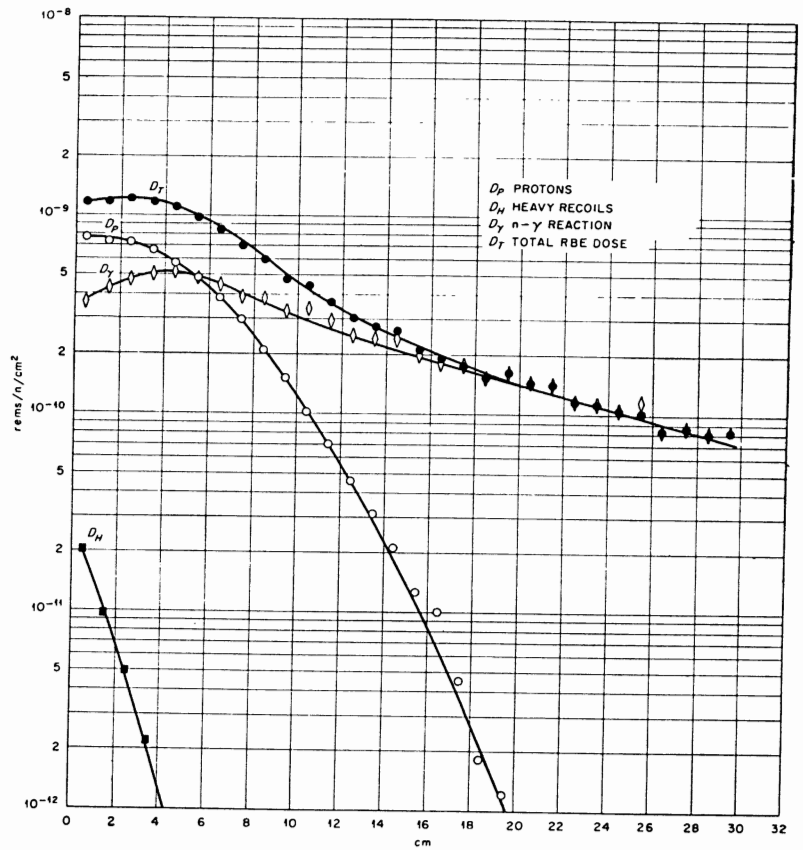

FIGURE 10B. *RBE dose (rems) from 0.005-Mev neutron beam.*

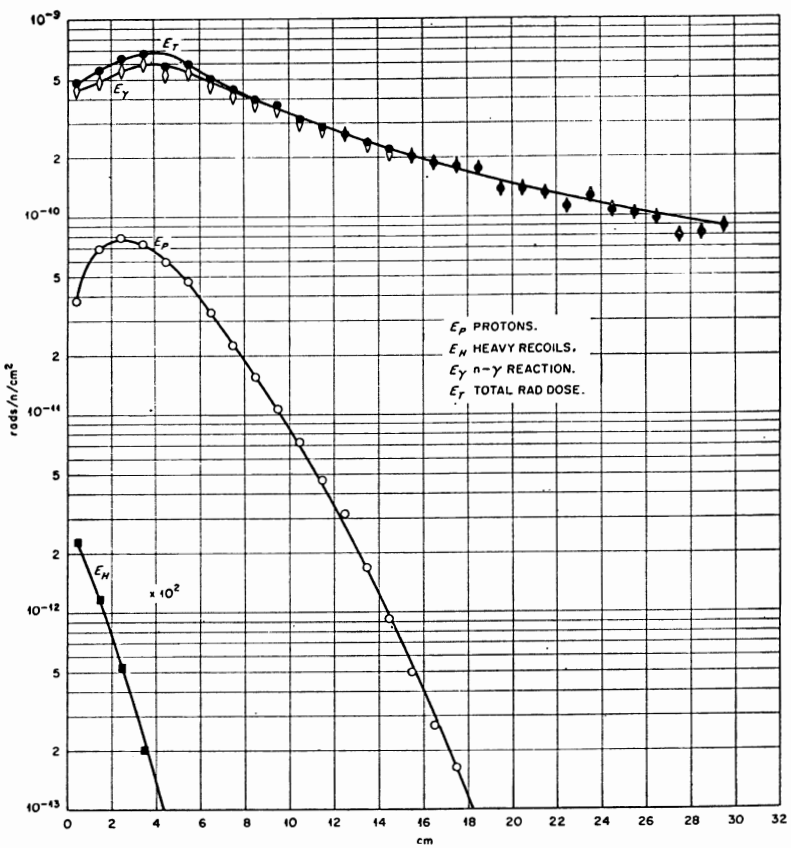

FIGURE 11A. *Absorbed dose (rads) from 0.0001-Mev neutron beam.*

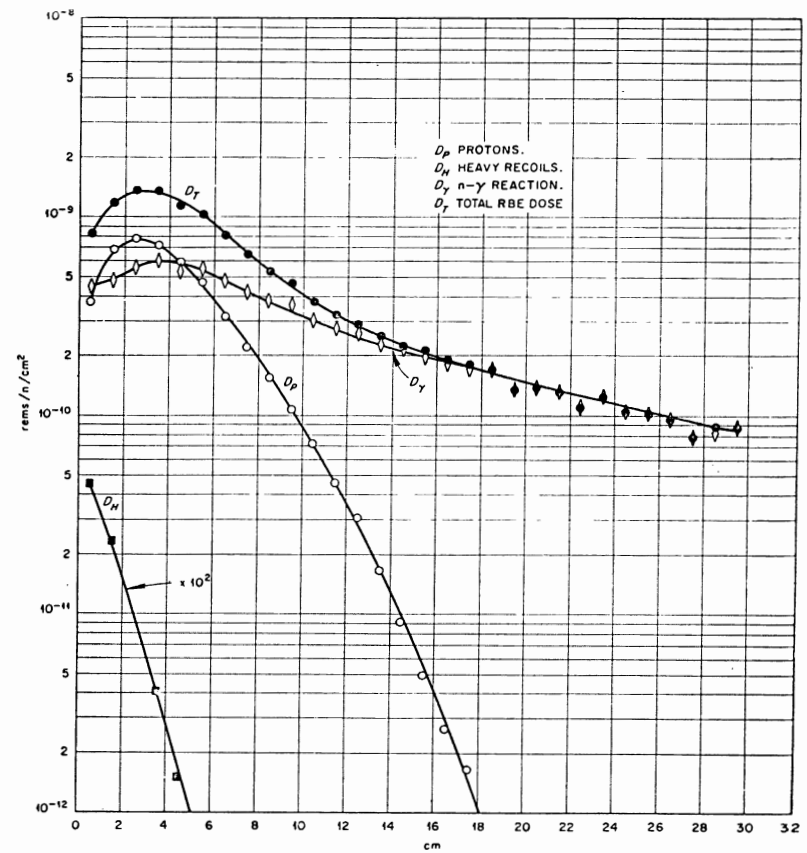

FIGURE 11B. *RBE dose (rems) from 0.0001-Mev neutron beam.*

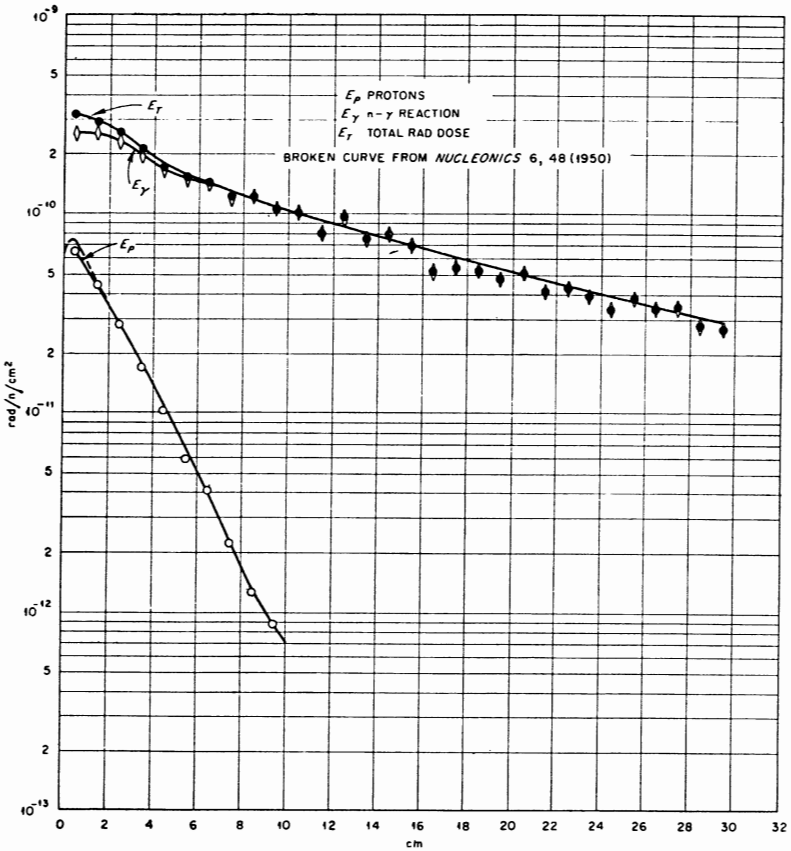

FIGURE 12A. *Absorbed dose (rads) from thermal neutron beam.*

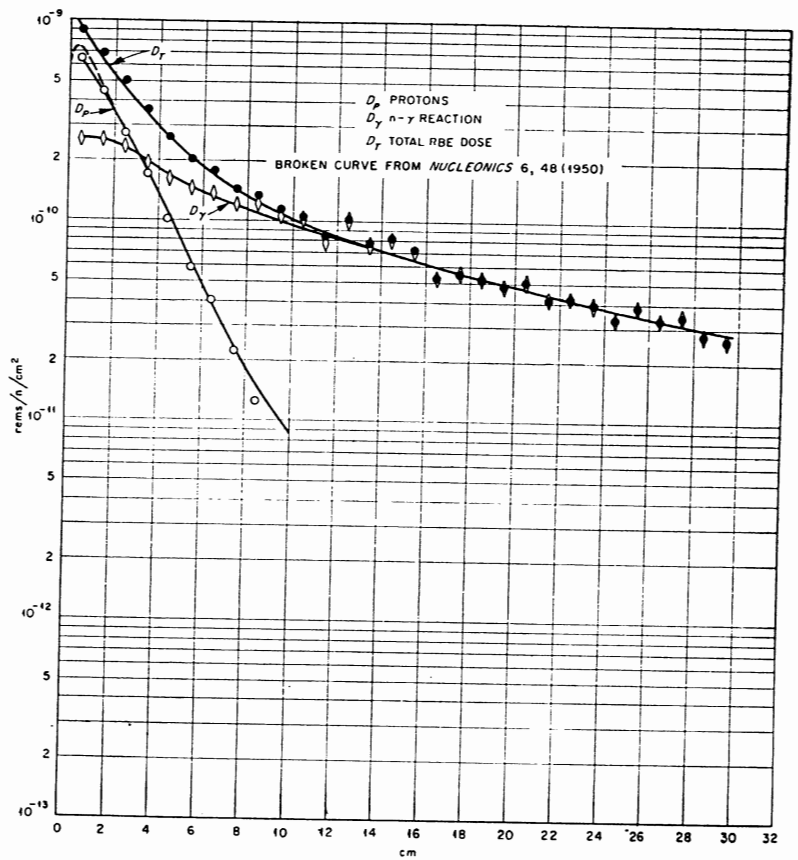

FIGURE 12B. *RBE dose (rems) from thermal neutron beam.*

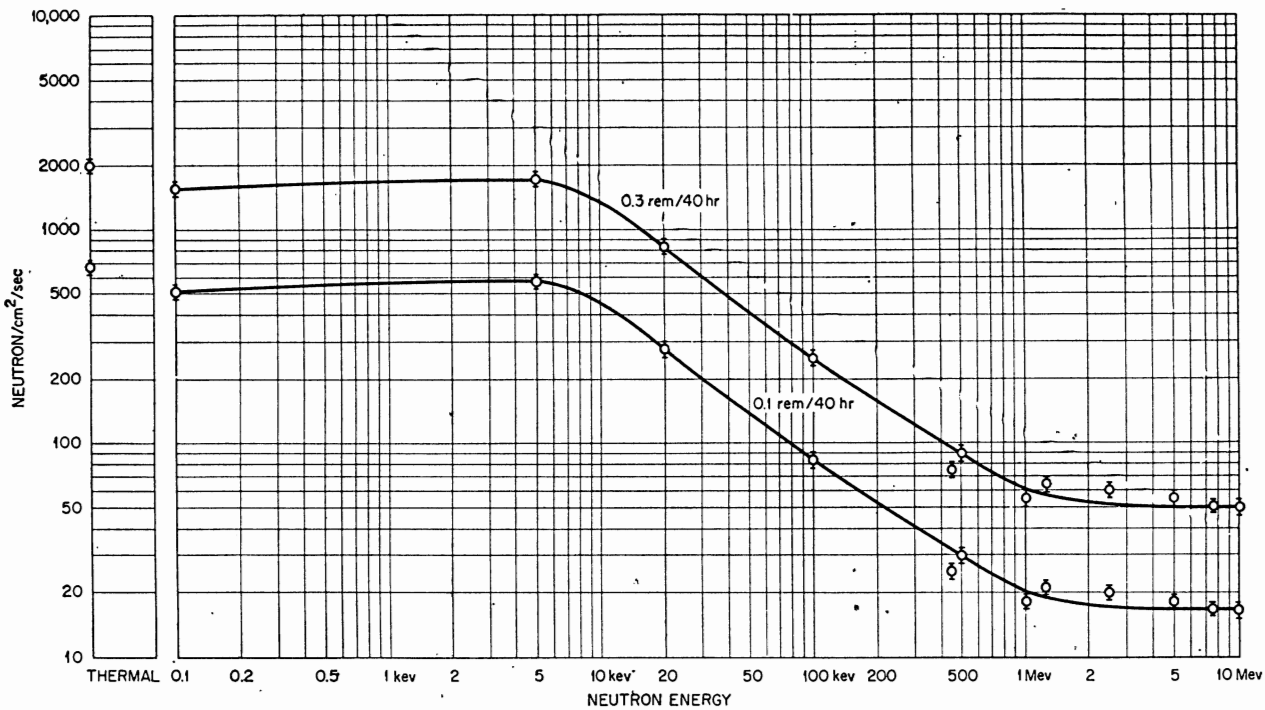

FIGURE 13. *Flux of neutrons to deliver the stated dose rate as a function of neutron energy.*

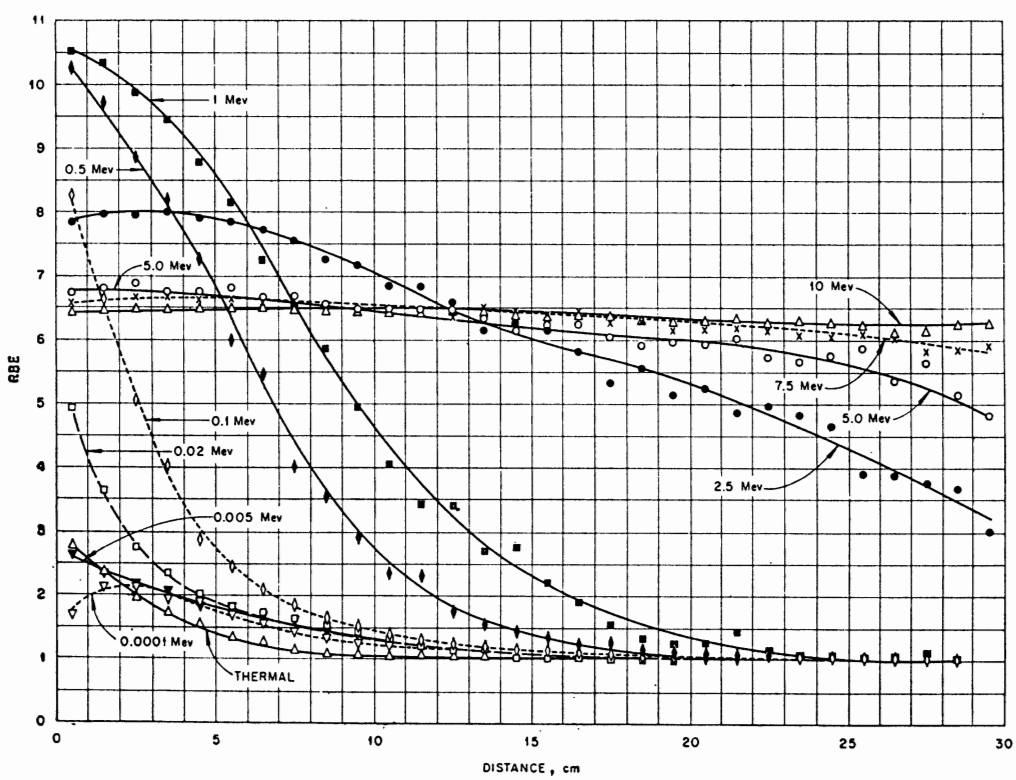

FIGURE 14. *RBE* ($=D_T/E_T$) *as a function of depth and neutron energy.*

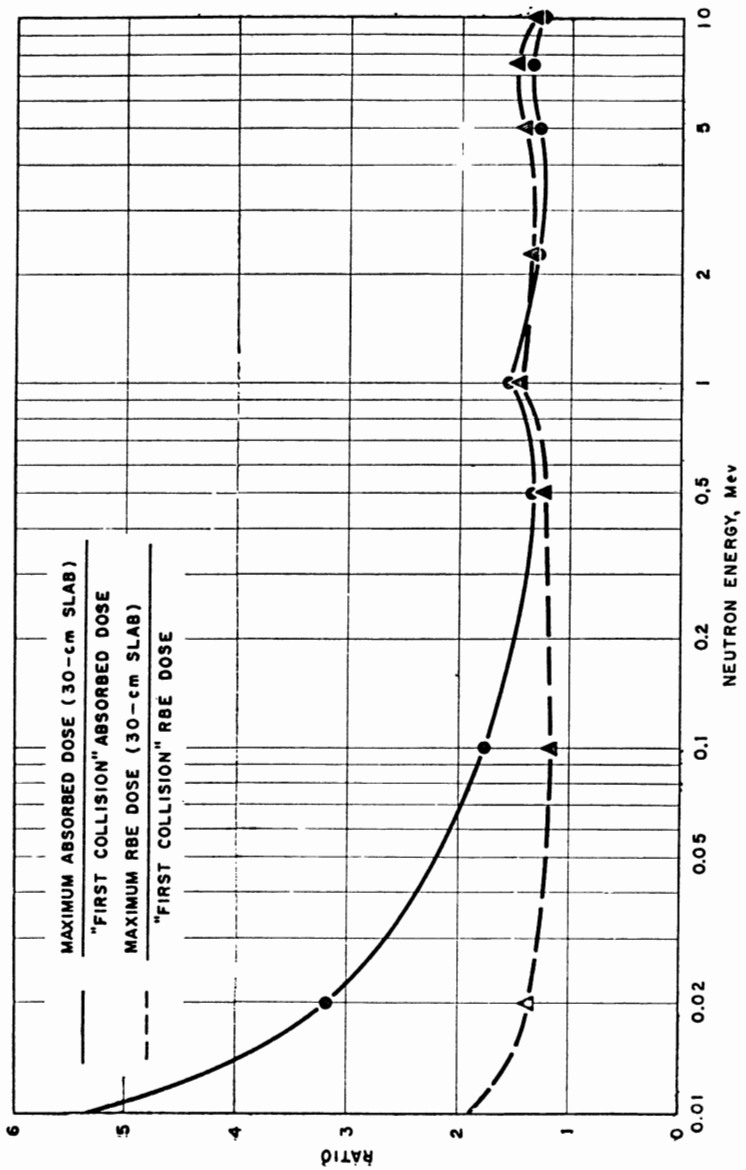

FIGURE 15. *Effect of buildup on neutron dose.*

These curves were obtained by computing neutron histories in an unbiased manner and averaging the absorbed dose delivered from a sample of such histories. The minimum sample size for a given neutron energy was 4,000 such histories. In computing these histories, tissue was considered to consist of hydrogen and one heavier element. The percentage by weight of this heavier element was taken as the sum of the percentage weights of all body constituents other than hydrogen, and the cross section for this element was a composite of the cross section of these elements. All scattering was considered to be isotropic in the center of mass system of coordinates and the only reactions considered were elastic scattering, the (n,p) reaction with nitrogen, and the (n,γ) reaction with hydrogen. The energy of the recoil atoms and of the protons produced by the (n,p) reaction was considered as absorbed at the site and the absorbed energy was tabulated and averaged over each centimeter of depth to obtain the dose curves.

Handbook 59 of this series has prescribed the RBE as a function of specific ionization. The specific ionization as a function of particle energy has been given by Livingston and Bethe [8] and for the heavier elements of principal interest here has been approximated by Neufeld and Snyder [9]. The latter reference gives RBE as a function of the particle energy as well as the biological dose delivered by a particle of given energy. These curves were approximately by polynomials, and the biological data were computed for each particle produced by the neutrons in the course of their histories. The γ-rays produced were followed also using the Monte Carlo or sampling method, and their absorbed dose (=RBE dose) computed for each centimeter of depth.

The points given on the graphs are the computed points, and the smooth curve indicates the general trend. In some cases where the penetration was small, few or no events occurred in some intervals and the curves were not continued where the data did not seem significant. Figure 14 gives the RBE (=RBE dose/absorbed dose) as a function of energy and depth. The flux to deliver a dose of 0.3 rem in 40 hours or of 0.1 rem in 40 hours has been computed and is shown in figure 13. The ratios of maximum dose for the 30-cm slab to first collision dose for both absorbed dose and for RBE dose are shown in figure 15. The low values at 1 Mev and 0.44 Mev are not entirely due to statistical fluctuations but are largely due to the resonance peaks in the oxygen cross-section curve. As a check on

this, the points at 1.2 Mev and at 0.5 Mev were computed, and this confirmed that the actual curve will reflect much of the detailed structure of the cross-section curves. However, the accuracy of the present study would not warrant attempting to predict the amount involved in detail.

For thermal neutrons an earlier study by Snyder [10] indicated a sharp peak in the absorbed dose due to the (n,p) reaction. The present calculations averaged out this peak. Figure 12 shows the peak as well as the average computed in the present study. In computing the thermal value for figure 13 the peak value was used.

The reflection of thermal neutrons incident on a large slab of tissue is approximately 80 percent. Hence the reading obtained with pocket dosimeters worn for purposes of thermal-neutron dosimetry should be corrected accordingly (i.e., the reading should be divided by 1.8 before the hazard is evaluated according to the data presented here).

Appendix 2. Flux Detectors and Their Calibration

As the methods of flux measurement are different for the various energy ranges, even though the principles involved are about the same, they will be considered according to neutron energy. Flux measurement of thermal neutrons has received most effort because it is basic to most flux measurements in the other energy classifications.

A. Thermal Neutrons

One approach to the absolute measurement of thermal neutron flux is closely related to source calibration, because it is possible to produce a flux with a standard source that is known almost as accurately as the emission rate of the source. The thermal flux is produced in a "standard pile," which is merely a large block of graphite in which the source, usually a Ra-Be source, is placed. The fast neutrons emitted from the source are moderated in the graphite and produce a certain spatial distribution of slowing-down density, q. When thin foils of indium, wrapped in cadmium, are activated at various places in the standard pile, the activation is proportional to the neutron flux at the resonance energy of indium, 1.44 ev, hence to the q at 1.44 ev. The absolute q values are obtained in terms of the source strength Q because the integral of q over the volume of the standard pile, assuming negligible capture and no escape of resonance neutrons from the pile, must equal Q. Once the slowing down density q is known in the standard pile it is a simple matter to compute the thermal flux from q, for it is necessary to know only the lifetime of the neutrons in graphite, easily obtained from the scattering constants of carbon.

The standard pile flux, while reasonably well known, is quite low, and activations intense enough for accurate foil counting can be obtained only with foils that are so thick that an appreciable change in the flux at a given point results from the neutron absorption of the foil. The flux perturbation does not affect the determination of an unknown flux relative to that in the standard pile if the unknown flux is also in graphite and if the foil geometry is identical, for in that case both fluxes will be equally perturbed. However, a correction for the perturbation must be made when fluxes in open beams are to be measured relative to the standard pile, for instance, and for this geometry the correction leaves an error of several percent.

A second method of absolute measurement of neutron flux is by means of the reaction rate of an element whose cross section is known, for

$$\text{reactions/sec} = nvN\sigma, \qquad (5)$$

where N is the number of atoms of reaction cross-section σ in the flux nv. The absorption cross section of boron, for example, which is essentially the (n,α) reaction, as the (n,p) and (n,γ) are negligible, has been accurately determined from the total cross section. The "best value" of the boron absorption cross section, from the AEC compilation, is 750 barns at 2,200 cm/sec, accurate to about 1 percent. The number of (n,α) reactions taking place in a certain amount of boron, located in a particular neutron flux, then gives the flux in terms of the reaction rate. Again, as for the standard pile, the rate of (n,α) reactions in boron depends on the neutron density, independent of the velocity, because boron is strictly $1/v$. There are several other elements for which the absorption cross section can be accurately determined from the total cross section by subtraction of scattering, for instance gold or indium. These elements differ from boron as flux standards because their absorption is a result of radioactive capture rather than the (n,α) process, and in addition they are not strictly $1/v$.

The reaction rate in a known amount of boron located in a neutron flux can be measured by a BF_3-filled proportional

counter or pulse-counting ion chamber, if one can be sure of a one-to-one correspondence between disintegrations and counts. The counting rate corresponding to complete detection of the disintegrations in ionization chambers has been obtained with an accuracy of about 2 percent. To this error must be added the uncertainty in the boron cross section, so the final accuracy is probably about the same as that of the standard pile flux. The absolute disintegration rate in boron has not been used in any determination of neutron flux to better than 5 percent, although the accuracy could definitely be increased.

The simplest method of *relative* flux measurement is by means of activation of a $1/v$ material in the form of a thin foil. Manganese, for instance, is a good material for relative determination of high flux because it has a convenient half life (2.6 hours), and a cross section (12 b) that is small enough so that excessively thin foils are not necessary. A thin foil of manganese activated in any thermal flux, and in a standard flux, will give the former flux (again, actual density times 2,200 cm/sec) correctly, regardless of the physical arrangement of the neutron flux, whether it be in graphite or some other moderator, in a parallel beam, or any arbitrary angular distribution. Because the foil is thin, there is no flux depression caused by its presence nor any self-protection of the foil itself. The activation of a thin foil does not depend on the direction of neutron travel through the foil. This is shown in the activation formula above, where flux is just the neutron density multiplied by the velocity, with no directional effects involved.

B. Intermediate Neutrons

There has been no extensive work on flux measurements for intermediate neutrons, primarily because there has been little need for accurate flux values in this energy range. For the purpose of neutron research, in which primarily total cross sections have been measured, only relative flux measurements, used in transmission measurements, are required. The intermediate neutrons in bulk matter, as in shields or in tissue, are soon slowed to thermal, especially in light elements, and spend most of their lifetime as thermal neutrons. Because of the rapid moderation, and the strong "self-protection" that reduces interaction in bulk material, intermediate neutrons are much less important than fast or thermal neutrons.

C. Fast and Relativistic Neutrons

Because fast neutrons represent such a wide energy range, over which detector response varies widely, accurate flux measurement is extremely difficult. For these neutrons the simplest method of flux measurement is by comparison with the flux emitted by a standard neutron source, usually a Ra-Be source.[7] For this comparison the standard source is not used in the standard pile but is placed in the center of a large room, and the flux of fast neutrons at a certain distance from the source is calculated from geometrical considerations. This flux is thus established with essentially the accuracy of the source calibration, about 5 percent usually. The unknown flux is then compared with this standard flux by some type of detector that will not be affected greatly by any energy difference between the unknown and the standard flux. Unfortunately, most fast neutron detectors are not energy-independent. The long counter is specifically designed for flux comparison, however, and can be used in the energy region 0.1 to 3 Mev, in which its sensitivity varies by less than 5 percent.

Fast neutrons, in contrast to those of lower energy, can be detected by means of the recoil protons they produce, and in principle these protons can be used for flux measurement. It is difficult, however, to measure the recoil protons quantitatively, for example, in an ionization chamber, and thus obtain the absolute flux incident on the recoil counter. In practice an accuracy of about 10 percent can be reached by this method. In a similar manner, an absolute flux can be measured by counting recoil protons in a photographic emulsion, but again measurements better than 10 percent are very difficult.

Activation of foils of known cross section, which is so useful for thermal flux measurement, is much more difficult to apply to fast neutrons, primarily because of the wide energy range of fast neutrons. In bulk material, moderation produces an energy spread, and a foil that has a cross section varying rapidly with energy does not give a readily interpretable result. Use of threshold reactions as (n,p), (n,α), (n,2n), and (n,f) can give approximate information on the flux value in different energy regions, depending on the reaction thresholds. Table 4 gives occasionally employed reactions and their thresholds.

[7] In computing the neutron flux from a standard source or in measuring its strength relative to another source by means of the long counter, it must be remembered that the emission of neutrons from sources is often anisotropic.

TABLE 4. *Threshold detectors*

Detector	Reaction	Product	[a]T½	[b]Et	[c]σ	References[d]
				Mev	*mb*	
Np^{237}	(n,f)	Many	Many	0.4	1500	(2)
Th^{232}	(n,f)	Many	Many	1.3	170	(2)
Pu^{239}	(n,f)	Many	Many	0.01	2000	(2)
U^{238}	(n,f)	Many	Many	1.3	550	(2)
P^{31}	(n,p)	Si^{31}	2.7 hr	1.0	120	(1)
Ag^{109}	(n,p)	Pd^{109}	13 hr	0.4	2.0	(1)
S^{32}	(n,p)	P^{32}	14.3 days	1.0	300	(1)
Ti^{48}	(n,p)	Sc^{48}	44 hr	1.3	23	(1)
Mg^{24}	(n,p)	Na^{24}	15 hr	2.1	39	(1)
Fe^{56}	(n,p)	Mn^{56}	2.6 hr	2.9	18.5	(1)
Ag^{107}	(n,2n)	Ag^{106}	24.5 min	9.6	42	(1)
C^{12}	(n,2n)	C^{11}	20.5 min	20.2	25	(1)

[a]T½=half life of product nucleus.
[b]Et=threshold neutron energy.
[c]σ=approximate cross sections at energies well above threshold.
[d]References: (1) B. L. Cohen, Phys. Rev. 81, 184 (1951); Nucleonics 8, No. 2, 29 (1951); (2) BNL-325; Rev. Sci. Instr. 27, 153, (1956).

Appendix 3. Reaction Employed in Neutron Production

Table 5 summarizes pertinent data on reactions utilized in neutron production. Both cross sections and neutron yield are a function of bombarding voltage, which is given in Mev in parentheses; at other bombarding energies corresponding values may be quite different.

The information in the last two columns is qualitative only and is furnished as a rough guide.

For further information see references [11 to 16].

Appendix 4. Practical Use of Radiation Instruments

For health protection in the vicinity of a neutron source, measurements are needed of the radiation hazard from fast neutrons, intermediate neutrons, thermal neutrons, and gamma rays. A brief discussion of fundamentals will be given here. For further information the reader is referred to several review articles [17, 18, 19], and to the references below.

A. Instruments

1. *Fast neutrons and relativistic neutrons (10 kev to 30 Mev).* As most tissue damage by fast neutrons is probably

due to ionization from recoil protons, the fast-neutron dose in rads is usually assessed with an ionization chamber or counter filled with a hydrogen-rich gas. Fast neutrons may be distinguished from gamma rays by (a) use of two ionization chambers, one sensitive to both neutrons and gamma rays, and the other sensitive to gamma rays alone or having a known residual neutron sensitivity (the difference in reading permits evaluation of the neutron dose), and (b) use of a proportional counter which is biased to discriminate against small pulses due to gamma rays. Examples of the ionization-chamber method are tissue-equivalent chambers [18] and CH_2-C and CH-C pairs of chambers [19]. Proportional counter instruments reading fast-neutron dose include the polyethylene-ethylene proportional counter [20] and the count-rate dosimeter [21].

The polyethylene-ethylene proportional counter may be used with a special circuit in which the pulses produced by heavy particles are weighted according to pulse height and a count results that is proportional to dose. This is sometimes called the pulse energy integrating dosimeter. This instrument is not directional, but requires somewhat complicated electronic circuitry. The count-rate dosimeter, however, is directional. Both dosimeters, being pulse-operated instruments, usually cannot be employed at sources that produce radiation in very short bursts for reasons that are given above (10.7). A scintillation-counter version of the count-rate dosimeter, in which an attempt has been made to eliminate directional response, has been reported [22].

Other fast-neutron survey instruments have been widely used, for example, the "Hornyak button" [23]. Caution: Such instruments do not in general read dose, and therefore knowledge of neutron spectrum is required to use them.

More detailed information about fast-neutron hazards may be obtained from instruments that measure the fast-neutron spectrum, and from instruments that measure the LET distribution of the ionization due to neutron recoils. Approximate measurements of neutron energy distributions may be made by activation of fast-neutron threshold detectors [24, 25, 26]. More accurate determination of neutron spectra may be made by measuring the proton recoil distribution in a cloud chamber or photographic emulsion [27], or in a proportional-counter spectrometer [28]. Information about the LET distribution and the dose is obtained from the tissue-equivalent wall, spherical pro-

TABLE 5. *Data on neutron production*

Target	Residual nucleus	Q	Reaction yields: Thick target yield or nominal cross section	Reaction yields: Corresponding incident particle energy
(p,n) Reactions				
H^3	He^3	—0.76	0.50	3.0 (peak)
Li^7	Be^7	—1.65	0.50	2.2 (peak)
Be^9	B^9	—1.85		
(d,n) Reactions				
H^2	He^3	3.26	2.5×10^6 n/μcoul 0.05 b	0.25 .25
H^3	He^4	17.6	~108 n/μcoul 5 b	.20 .10
Li^6	Be^7	3.38		
Li^6	He^3+He^4	1.79	1.7×10^7 n/μcoul	.6
Li^7	He^4+He^4	15.1	0.06	.2
Be^9	B^{10}	4.36	4×10^6 n/μcoul	.4
B^{10}	C^{11}	6.47		
B^{11}	C^{12}	13.7		
C^{12}	N^{13}	0.28		
C^{13}	N^{14}	5.3		
N^{14}	O^{15}	5.12		
F^{19}	Ne^{20}	10.65		
(α,n) Reactions				
Li^7	B^{10}	—2.78	5×10^4 n/sec/c	Po–Li
Be^9	C^{12}	5.7	2.3×10^6 n/sec/c 1.7×10^7 n/sec/c	Po–Be Ra–Be
B^{10}	N^{13}	1.07	6×10^5 n/sec/c	Po–B
B^{11}	N^{14}	0.15	7×10^6 n/sec/c	Ra–B
F^{19}	Na^{22}	—2.3	2×10^5 n/sec/c 2.5×10^6 n/sec/c	Po–F $RaBeF_4$
Na^{23}	Al^{26}	—3.88	4×10^4 n/sec/c	Po–Na
Mg^{25}	Si^{28}	2.67	3×10^4 n/sec/c	Po–Mg
Mg^{26}	Si^{29}	0.04		
Al^{27}	P^{30}	—2.92	2×10^4 n/sec/c	Po–Al

TABLE 5. *Data on neutron production*—Continued

Degree of gamma-ray contamination	Gamma-ray energies	Residual activity or comparative dosage
(p,n) Reactions		
	Mev	
Considerable ~10^{-8} γ/proton	20	None.
Considerable ~2×10^{-8}/proton	17	Considerable (Be^7).
Considerable 10^{-9} γ/proton		None.
(d,n) Reactions		
None	None	None.
Slight		None.
0.3 γ/n	0.4	Moderate (Be^7)
	.4	None.
.03 γ/n	4.9	None.
	3.6 ; 2.2 ; 1.7 ; 1.0; 0.7 ; 0.4.	Small (Be^{10}).
	8.9; 6.7; 6.4 ; 4.5.	Small (C^{11}).
		None.
4.45		Slight (N^{13}).
	6.1; 5.0; 3.3 ; 2.3 ; 1.6 ; 0.7.	None.
None		Slight (O^{15}).
	10.6 ; 9.1; 8.4 ; 6.4; 5.2; 3.5 ; 2.8; 1.6; 0.5.	
(α,n) Reactions		
	0.8 ; 0.48	$n/\gamma\simeq0.25$.
	0.8 ; 4.45	$n/\gamma\simeq4.0$.
Many		$n/\gamma<0.01$.
	0.8 ; 3.7; 2.4	$n/\gamma\simeq5.0$.
Many		$n/\gamma<0.01$.
	0.8 ; 1.3 ; 1.5	$n/\gamma\simeq1.0$.
Many		$n/\gamma<0.01$.
	0.8; 0.4; 1.1; 1.8; 2.6.	$n/\gamma\simeq0.3$.
	0.8 ; 1.8 ; 4.0	$n/\gamma\simeq0.15$.
	0.8; 1.3; 2.3 ; 3.6.	$n/\gamma\simeq0.1$.

portional counter [29, 30], or by calculation from the measured neutron flux and energy distribution [31].

2. *Intermediate neutrons (0.5 ev to 10 kev).* Intermediate-energy neutron flux is usually not important for health protection because the biological effect per neutron is small compared to that for fast neutrons, and the intermediate neutrons are quickly moderated in the human body to become slow or thermal. The usual instrument for flux measurements is a boron counter in a moderator, such as the "long counter" [32, 33] or certain survey instruments [34]. The long counter is also useful for fast-neutron and thermal-neutron flux measurement.

3. *Thermal neutrons.* Thermal-neutron flux may be determined by absolute measurement of the induced radioactivity in any one of several elements [25, 35]. Boron-walled ionization chambers or counters may be calibrated by means of foil activation.

4. *Gamma rays.* Gamma-ray dose may be measured in carbon-walled chambers which often have only a slight response to neutrons (which must be corrected for). Because the interaction of neutrons with tissue can produce gamma rays, a precise determination of the dose requires appropriate phantoms. Commonly used gamma-ray survey meters and dosimeters cannot be used when large numbers of neutrons are present, as they possess considerable but unknown neutron sensitivity.

5. *Personnel monitoring.* Area monitoring of neutrons around permanent installations is especially important for fast neutrons, because often neither gamma-ray film badges nor pocket dosimeters give sufficient information to evaluate the dose. Fast-neutron nuclear-track film badges [36] are useful in case of accident and, although rather insensitive, are of some value in the permissible dose range. Slow neutrons may be adequately monitored by a cadmium-covered emulsion in a gamma-ray film badge or by boron-lined pocket chambers.

B. Calibration

With most neutron-monitoring instruments it is essential that the instrument be calibrated frequently in a known neutron flux. Extremely large errors are possible with many commercial instruments unless properly calibrated. A convenient method for calibration is the use of a portable radioactive neutron source. A primary standard source of a laboratory may be calibrated for total neutron emission rate by submission to the National Bureau of Standards.

Radioactive neutron sources may also be calibrated for dose rate at a fixed distance using a fast neutron dosimeter calibrated with an internal alpha source [20] or calibrated with monoenergetic neutrons. Known thermal fluxes for calibration may be obtained using the calibrated standard source in a graphite "standard pile" [37], or by use of a moderating geometry to produce a thermal-neutron flux, which may then be compared to a known thermal-neutron flux such as the National Bureau of Standards standard thermal-neutron density [38].

Appendix 5. Neutron-Capture Gamma Rays

Table 6 summarizes gamma-ray energies encountered in neutron capture.

Appendix 6. Shielding Calculations and Data

The same fundamental safety factors that were described [39] for gamma rays apply to neutrons: (a) Distance from source, (b) time of exposure, and (c) attenuation by shielding or other material. Because the several types of neutron sources differ considerably, they will be treated separately.

Reactors[8]

The neutrons from reactors come from (a) the fission process (prompt neutrons) and (b) radioactive nuclides produced in fission (delayed neutrons).

Prompt Neutrons from Fission

Prompt neutrons from fission are produced at all energies up to about 17 Mev, distributed as shown in figure 16. More accurate figures are given in table 7. The attenuation process consists primarily of collisions in which either large energy degradation is accomplished or the neutron is widely deflected so that its total escape path is significantly increased. Following degradation, many successive collisions take place in a relatively short further excursion before the neutrons are absorbed at low energy. Because in general cross sections decrease with increasing energy,

[8] Note: The shielding of reactors is a complicated subject beyond the scope of this Handbook, but the treatment given here will give approximate answers to simple problems and will be adequate for estimating the effects of small changes in shields.

TABLE 6. *Capture gamma rays*[a]

Target	Thermal (n, γ) cross section	Highest energy gamma-ray	Average number of photons per capture
	Barns	*Mev*	
Aluminum	0.215	7.724	~2
Antimony	6.4	6.80	
Arsenic	4.1	7.30	
Barium	1.17	9.23	2.7
Beryllium	0.009	6.814	
Bismuth	.015	4.17	
Boron-10	3,990	0.478	
Cadmium	3,500	9.046	4.1
Calcium	0.406	7.83	
Carbon-12	.0045	4.95	1.3
Chlorine	32	8.56	3.1
Chromium	2.9	9.716	2
Cobalt	34.8	7.486	
Copper	3.59	7.914	2.6
Fluorine	0.009	6.63	
Gadolinium	36,300	7.78	3.9
Gold	94	6.494	3.5
Hydrogen-1	0.330	2.230	
Indium	190	5.86	3.3
Iron	2.43	10.16	1.7
Lead	0.17	7.38	
Lithium-6	910	0	
Magnesium	0.059	9.216	
Manganese	12.6	7.261	2.6
Mercury	380	6.446	3.3
Molybdenum	2.4	9.15	
Nickel	4.8	8.997	
Niobium	1.1	7.19	2.6
Nitrogen-14	0.1	10.8	
Phosphorous	0.193	7.94	
Platinum	8.1	7.920	
Potassium	1.89	9.28	
Praseodymium	11.2	5.83	
Rhodium	150	6.792	
Samarium	10,600	7.89	5.6
Scandium	22	8.85	
Selenium	11.8	10.483	
Silicon	0.160	10.55	
Silver	60	7.27	2.9
Sodium	0.47	6.41	<2
Strontium	1.16	9.22	
Sulfur	0.49	8.64	
Tantalum	21.3	6.07	
Thallium	3.3	6.54	
Tin	0.65	9.35	
Titanium	5.8	9.39	
Tungsten	19.2	7.42	
Vanadium-51	4.7	7.305	2.5
Zinc	1.06	9.51	
Zirconium	0.18	8.66	

[a]These data are taken from P. S. Mittelman and R. A. Liedtke, Nucleonic 13, No. 5, 50-51 (1955).

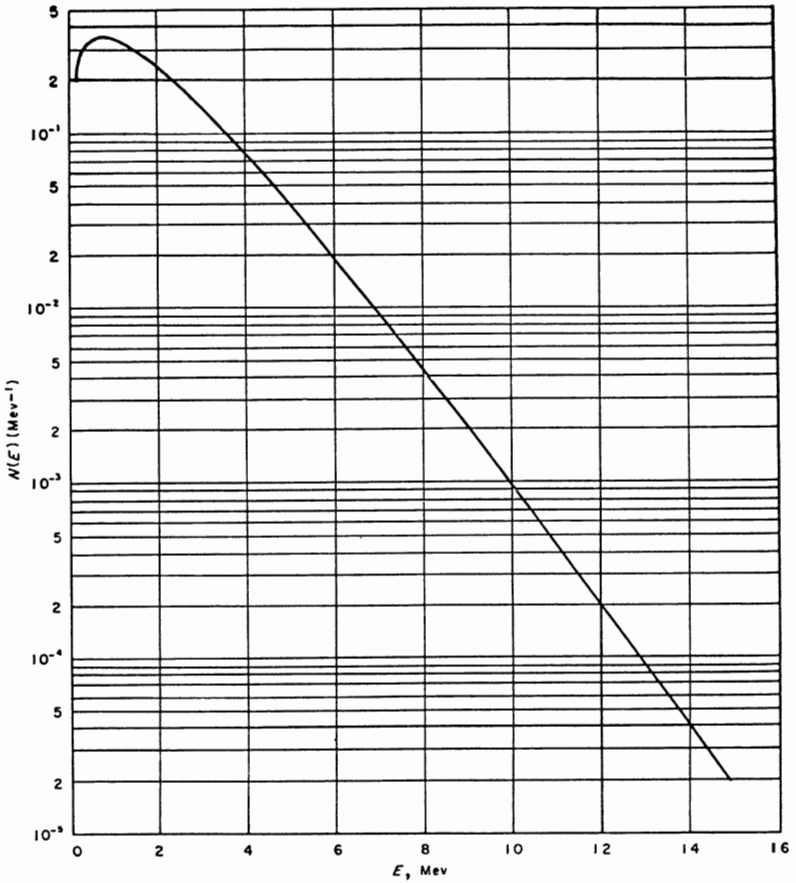

FIGURE 16. *Distribution of prompt fission neutrons.*
Number of neutrons per Mev energy interval versus neutron energy.

the neutrons produced at higher energy have the best chance of penetration. Balancing this is the original distribution in which the lower energies predominate. For most shields the neutrons produced at about 8 Mev are most likely to penetrate, although they may have had a collision near the outer shield surface and hence emerge at lower energy.

A simplified calculation of attenuation is made on the basis of the neutrons at about 8 Mev. Some allowance is necessary for the neutrons that have collided but still escape, a factor usually called the buildup factor. This is taken account of by a compensating factor obtained by

TABLE 7. *The* U^{235} *fission spectrum of prompt neutrons*[a]

$N(E)dE$ is the fraction of neutrons of energy E to $E+dE$ from thermal-neutron fission of U^{235}.

E	$N(E)$	$\int_0^E N(E)dE$	$\int_E^\infty N(E)dE$
Mev	*Mev*$^{-1}$		
0.1	0.2023	0.014	0.9860
.2	.2676	.377	.9623
.3	.3068	.663	.9337
.4	.3300	.987	.9013
.5	.3450	.1321	.8679
.6	.3528	.1669	.8331
.7	.3557	.2027	.7973
.8	.3542	.2385	.7615
.9	.3513	.2729	.7271
1.0	.3446	.3082	.6918
1.1	.3368	.3425	.6575
1.2	.3271	.3759	.6241
1.3	.3179	.4074	.5926
1.4	.3073	.4383	.5617
1.6	.2841	.4983	.5017
1.8	.2604	.5631	.4369
2.0	.2371	.6028	.3972
2.5	.1839	.7073	.2927
3.0	.1384	.7876	.2124
3.5	.1026	.8476	.1524
4.0	.07472	.8914	.1086
4.5	.05381	.9192	.0808
5.0	.03847	.9463	.0537
5.5	.02729	.9624	.0376
6.0	.01916	.9739	.0261
6.5	.01336	.9821	.0179
7.0	9.316×10^{-3}	.9876	.0124
7.5	6.436	.9916	.0084
8.0	4.423	.9943	.0057
8.5	3.034	.9960	.0040
9.0	2.076	.9973	.0027
9.5	1.418	.9982	.0018
10.0	9.53×10^{-4}	.9988	1.211×10^{-3}
11.0	4.384	.9994	5.501×10^{-4}
12.0	1.994	.9998	2.473
13.0	8.953×10^{-5}	.9999	1.103
14.0	3.997	1.0	4.875×10^{-5}
15.0	1.771	1.0	2.148

[a] Calculated from the formula of B. E. Watt, Phys. Rev. 87, 1037, (1952). See also J. E. Evans, Fast neutron spectra from the water boiler, Los Alamos Scientific Laboratory, LA-1395 (AECD 3073) (March, 1951).

counting all neutrons as if they were in the high-energy group. This simplifies the calculation, is reasonably accurate, and is conservative for shields that contain reasonable quantities of moderating material. Examples of such materials are the water-bearing concretes such as ordinary concrete, barytes concrete, magnesium oxychloride concrete, and water, masonite, moist earths, etc. Other materials

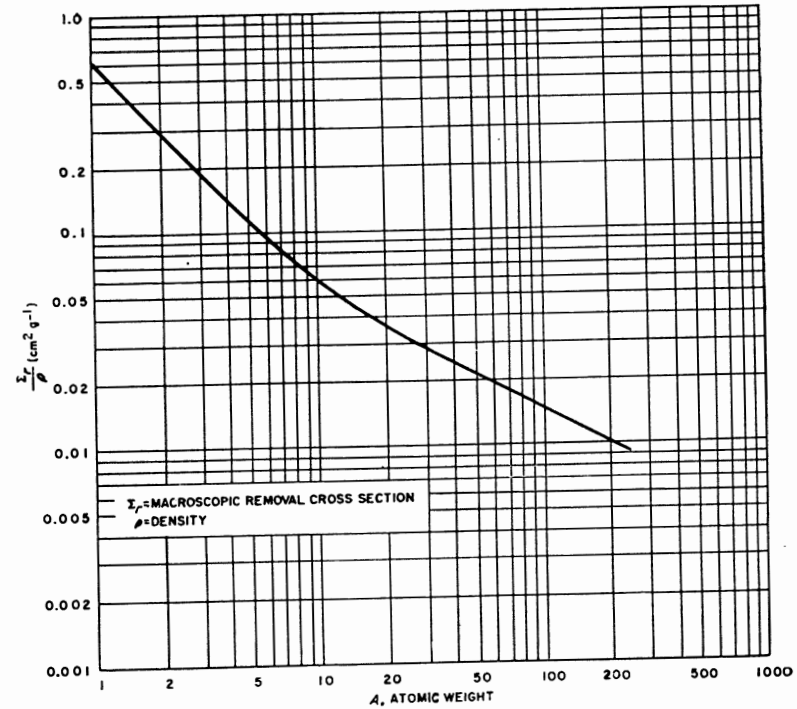

FIGURE 17. *Fission neutron attenuation coefficients.*
(Macroscopic removal cross sections divided by density) versus (Atomic weight).

which are not moderators (large atomic weight, $A>16$), such as iron, lead, etc., can be counted as well, provided they are followed with moderating material on the outside.[9]

Removal Cross Sections: Removal cross sections are the cross sections to be used in calculating the attenuation of fast neutrons. The microscopic removal cross section (σ_r) is roughly three-quarters of the total cross section at 8 Mev, an exception being hydrogen, for which the fraction is somewhat higher.

The macroscopic removal cross section is given by

$$\Sigma_r = \frac{0.602\sigma_r\rho}{A} \qquad (\text{cm}^{-1}), \qquad (6)$$

[9] Caution: These nonmoderating materials are relatively transparent to low-energy (10 to 500 kev) neutrons, which will produce capture gamma radiation when reaching the moderating material; these hard gamma rays must be shielded out subsequently by other material.

where

σ_r=microscopic removal cross section (barns),
ρ=density (g/cm³),
A=atomic weight.

It will be noted that Σ_r/ρ is a quantity dependent only on the microscopic nuclear properties. This is a smoothly varying function of the atomic weight, which is given in figure 17. From this curve the removal cross section can be obtained for any element.

The macroscopic removal cross section for a material of several elements is obtained by simple summation over its constituents:

$$\Sigma_r, \text{ compound} = \left(\frac{\Sigma_r}{\rho}\right)_1 \rho_1 + \left(\frac{\Sigma_r}{\rho}\right)_2 \rho_2 + \ldots, \qquad (7)$$

where

$\left(\frac{\Sigma_r}{\rho}\right)_1$=value from figure 17 for element 1 of compound, (cm²/g), etc.
ρ_1=density of element 1 (g/cm³), etc.

Example: Find Σ_r for $CaCO_3$; density = 2.711, molecular weight = 100.09.

Element	A	ρ	Σ_r/ρ	$(\Sigma_r\rho)_n \cdot \rho_n$
		g/cm³		
Ca	40.08	$\frac{40.08}{100.09} \times 2.711 = 1.087$	0.0242	0.0263
C	12.01	$\frac{12.01}{100.09} \times 2.711 = 0.325$	.0510	.0166
O	16	$\frac{3 \times 16}{100.09} \times 2.711 = 1.299$	.0410	.0533
Total				0.0962
Σ_r, $CaCO_3$ = 0.0962 cm⁻¹				

Examples of removal cross-section calculations for common shield materials are given below. Table 8 shows the pertinent steps for barytes and ordinary concretes, starting with elemental assays shown in the second columns. In table 9 the results are summarized for several common materials.

The core of the reactor is defined to be the volume in which the fissions take place, and the shield is the material

TABLE 8. *Calculations of neutron removal cross sections* (Σ_r)

Element	A, Atomic weight	ρ, Density in composition	Σ_r/ρ	Σ_r
Bartyes block concrete				
		g/cm³		
Fe	55.85	0.307	0.0200	0.0061
H	1.0	.015	.602	.0090
O	16.0	1.090	.041	.0447
Mg	24.32	0.013	.032	.0004
Ca	40.08	.159	.024	.0038
Ba	137.36	1.470	.124	.0182
Na	23.0	0.005	.033	.0002
Si	28.06	.061	.0295	.0018
Al	26.97	.020	.0301	.0006
Mn	54.93	.003	.0202	.0001
S	32.07	.348	.0275	.0096
Total		3.491		0.0945.
Ordinary concrete				
O	16.0	1.103	0.041	0.0452
Si	28.06	0.2815	.0295	.0083
Al	26.97	.0330	.0301	.0010
Fe	55.85	.0183	.0200	.0004
Ca	40.08	.7712	.024	.0185
C	12.01	.0761	.050	.0038
Na	23.00	.0116	.033	.0004
K	39.10	.0079	.0245	.0002
H	1.0	.0250	.602	.0150
Mg	24.32	.0426	.032	.0014
Total		2.3702		0.0942

TABLE 9. *Macroscopic neutron removal cross sections* (Σ_r) *and attenuation lengths of some shielding and reactor materials*

Material	Σ_r, Removal cross section	Attenuation length
	cm⁻¹	cm
Water	0.103	9.7
Iron	.1576	6.34
Ordinary concrete	.0942	10.6
Barytes block concrete	.0945	10.6
Graphite (ρ=1.54)	.785	12.7

external to it, including reflector, pressure shell, if any, and of course the shield proper.

The dose external to the shield is now calculated to be

$$D \text{ (rems/hr)} = 5.4 \times 10^6 \frac{pF}{T\Sigma_{rc}\bar{\Sigma}_r} e^{-T\bar{\Sigma}_r}, \qquad (8)$$

where

p=power density in core (watts/cm^3).

Σ_{rc}=macroscopic removal cross section of core material, calculated as above (cm^{-1}).

T=shield thickness (cm).

Σ_r=average macroscopic removal cross section for shield (cm^{-1}).

$=(1/T)\ [(\Sigma_r x)_i+(\Sigma_r x)_j+\Sigma_r)_k+\ldots]$.

$(\Sigma_r x)_i$=product of macroscopic removal cross section Σ_r and thickness x for the ith shield layer.

5.4×10^6=(7.5×10^{10} fission neutrons per joule)$\times$(½, geometrical factor)+(7,000 neutrons/cm^2/sec per rem/hr).

F=geometric factor taking account of the core shape, as follows:

=1 for a large flat core surface adjacent to shield.

$=a/(T+a)$ for a spherical core of radius a.

$=[a/(T+a)]^{1/2}$ for a cylinder of radius a, measurement at midplane.

For small cores of whatever shape the formula for a sphere of equal volume may be used, provided the maximum diameter (e.g., corner to corner) d_m fulfills the following criterion:

$$d^2_m<\frac{8T}{\Sigma_r}. \qquad (9)$$

Delayed Neutrons

Delayed neutrons are produced from certain radioactive nuclides among the fission products. The half lives, energies, and yields of the dominant emitters (groups) are given in table 10. They are not important by comparison with the prompt fission neutrons, but for cases in which the fission products are removed from the reactor to regions outside its shield shortly after being produced they may be very important. Examples of such removal are experimental fuel loops and pneumatic sample handlers in which fission takes place in the sample.

For delayed neutrons the only really satisfactory shields are hydrogenous materials, and the effectiveness depends only on the hydrogen thickness of the shield. Because the neutrons are of too low energy for inelastic scattering, ele-

TABLE 10. *Delayed neutrons from fission products*

Group	Half life	λ_i, Decay constant	Energy	q_i, Dose per unit flux	$y_i\times10^4$, Absolute yields per 10^4 fissions		Σ_rH, Removal cross section of hydrogen
					U^{235}	Pu^{239}	
	sec	*sec*$^{-1}$	*kev*	*rems/hr per n/cm²/sec*			*Barns*
1	55.6	0.0125	250	6.1×10^{-5}	6.25	4.2	7.6
2	22.0	.0315	570	9.7	41.5	31.5	5.2
3	4.51	.154	412	8.1	53.2	37.8	6.0
4	1.52	.456	670	10.5	60.2	.....	4.7
5	0.43	1.61	400	8.0	21.2	35.7	6.1
Total	.....	.....	.....	.....	182.35	109.2	.....

ments other than hydrogen are relatively ineffective and are ignored. For convenience the removal cross section of hydrogen for each energy is also given in table 10.

The dose due to the delayed neutrons is just the sum of the contributions from several emitters:

$$D_{total}=D_1+D_2+D_3+D_4+\ldots, \qquad (10)$$

where 1, 2, 3, etc., refer to the different groups of delayed neutrons. The contribution from one emitter is calculated as follows:

$$D_1\ (\text{rems/hr})=\frac{S_1 q_1 e^{-\overline{\Sigma}_{rH}T}}{4\pi R^2}, \qquad (11)$$

where

R=distance from center of delayed neutron source to place where D_1 is to be determined.

T=shield thickness (cm).

$\overline{\Sigma}_{rH}$=average macroscopic removal cross section (cm^{-1})

$=1/T[(\Sigma_{rH}x)_i+(\Sigma_{rH}x)_j+\ldots]$.

$(\Sigma_{rH}x)_i$=product of hydrogen removal cross section and thickness for the ith shield layer.

q_1=dose, rems/hr, per unit neutron flux (neutrons/cm^2/sec) for the delayed neutrons of group 1, table 9.

S_1=source strength, neutrons/sec, of neutrons from group 1.

For the common case of a uranium-bearing liquid (or wire) loop circulating into and out of a reactor, the source,

S_1, is just the rate of neutron release outside the reactor shield (but of course inside the special loop shield). Then, after many cycles, for each group there is an expression of the following form:

$$S_1 = \frac{f y_1 \ (1-e^{-\lambda_1 t_f})\,(e^{-\lambda_1 t_s})\,(1-e^{-\lambda_1 t_r})}{\lambda_1 t_f\,(1-e^{-\lambda_1 T})}, \qquad (12)$$

where

f=fission rate, fission/sec (total).

y_1=yield, neutrons produced in group 1 per fission.

λ_1=decay constant, sec^{-1}, for group 1.

t_f=time each loop particle spends in the neutron field per cycle.

t_s=time each loop particle spends on the way out through the shield.

t_r=time each loop particle spends in the exterior part of the loop, from which it radiates.

T=cycle time.

Radioactive Neutron Sources

The dose rate from a radioactive neutron source is best calculated somewhat differently from that from other sources. Owing to the small intensities available, relatively thin shields are needed and hence the scattered neutrons offer a very significant contribution to the total dose. To take account of this, a buildup factor is used. For Po-Be and Po-B sources with water or paraffin shields at least 20 cm thick, the value of this buildup factor is approximately 5. For these common situations, the following formula gives the dose rate:

$$D \ (\text{rems/hr}) = \frac{SBqe^{-\Sigma_r T}}{4\pi R^2}, \qquad (13)$$

where

S=source strength (neutrons/sec).

R=distance from source to point for which the dose rate is to be calculated.

B=buildup factor, $\backsim 5$.

q=dose per unit flux for neutrons of the source energy (rems/hr per neutron/cm^2/sec).

Σ_r=macroscopic removal cross section for the given source and shield, obtained from table 8 where possible, or calculated using nine-tenths of the hydrogen cross section for shields and sources not included thereon.

T=shield thickness (cm).

Accelerators

Accelerators which yield neutrons can be shielded according to formulas which have been developed for reactors. For neutron energies up to 30 Mev the removal cross sections can be taken to be about three-quarters of the total cross section.

Appendix 7. Neutron Protection Near High-Energy Electron Accelerators

Because there are many types of high-energy electron accelerators and because installations vary widely, it is not practicable to give general rules for neutron protection which would be valid for all cases. However, two chief factors may be pointed out that govern the neutron shielding problem. These factors are (1) the energy of the photons and (2) the type of shielding material used. The primary sources of neutrons near a high-energy electron accelerator are the machine itself and the point where the beam strikes the wall [40].

Neutron production. To liberate a neutron from the nucleus requires that the absorbed photon energy exceed the binding energy of the particle in the nucleus (of the order of 6 to 18 Mev, mostly about 10 Mev). Therefore electron accelerators of energy below 10 Mev will usually have very small or no neutron backgrounds. Common photoneutron threshold energies and information on neutron yields for various elements are given in Handbook 55 of this series.

For electron accelerators of energies much larger than photoneutron threshold energies, the yield of neutrons by photons incident on lead may be approximated by [41, 42, 43] $Y_{Pb} \simeq 0.4 W_0$ where Y= the number of neutrons produced and W_0 is the total incident photon energy (i.e., number of photons times photon energies) in Bev. The yield Y varies approximately as the square root of the atomic number. The above relation may therefore be rewritten $Y_z \simeq 0.044\sqrt{Z}\,W_0$. The photons have energies of the order of 20 Mev. The neutrons are produced with a wide distribution in energy, the average for lead being about 2 to 3 Mev, and for carbon about 5 Mev [40]. As photon

energy increases, the neutron protection problem becomes more serious, presumably because of a high-energy tail to the neutron distribution. As the mean free path for fast neutrons becomes larger (approximately as the energy) as the neutron energy increases, the choice of shielding may, for very high energy electron accelerators, actually be dictated by the fast-neutron hazard.

Shielding materials. Concrete or other hydrogen-containing materials are desirable for neutron shielding. However, many electron accelerators are shielded with lead, frequently as an integral part of the machine itself. The lead acts both as a strong source of photoneutrons and as a poor neutron shield.

After the neutron production near the high-energy electron accelerator has been established (remembering that neutrons may be produced in the shield itself), the neutron shielding requirements may be determined according to the methods outlined in appendix 6.

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