ICRU REPORT 26

Neutron Dosimetry for Biology and Medicine

INTERNATIONAL COMMISSION
ON RADIATION UNITS
AND MEASUREMENTS

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INTERNATIONAL COMMISSION ON RADIATION
UNITS AND MEASUREMENTS
7910 WOODMONT AVENUE
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U.S.A.

Preface

Scope of ICRU Activities

The International Commission on Radiation Units and Measurements (ICRU), since its inception in 1925, has had as its principal objective the development of internationally acceptable recommendations regarding:

(1) Quantities and units of radiation and radioactivity,

(2) Procedures suitable for the measurement and application of these quantities in clinical radiology and radiobiology,

(3) Physical data needed in the application of these procedures, the use of which tends to assure uniformity in reporting.

The Commission also considers and makes similar types of recommendations for the radiation protection field. In this connection, its work is carried out in close cooperation with the International Commission on Radiological Protection (ICRP).

Policy

The ICRU endeavors to collect and evaluate the latest data and information pertinent to the problems of radiation measurement and dosimetry and to recommend the most acceptable values and techniques for current use.

The Commission's recommendations are kept under continual review in order to keep abreast of the rapidly expanding uses of radiation.

The ICRU feels it is the responsibility of national organizations to introduce their own detailed technical procedures for the development and maintenance of standards. However, it urges that all countries adhere as closely as possible to the internationally recommended basic concepts of radiation quantities and units.

The Commission feels that its responsibility lies in developing a system of quantities and units having the widest possible range of applicability. Situations may arise from time to time when an expedient solution of a current problem may seem advisable. Generally speaking, however, the Commission feels that action based on expediency is inadvisable from a long-term viewpoint; it endeavors to base its decisions on the long-range advantages to be expected.

The ICRU invites and welcomes constructive comments and suggestions regarding its recommendations and reports. These may be transmitted to the Chairman.

Current Program

The Commission has divided its field of interest into eleven technical areas and has assigned one or more members of the Commission the responsibility for identification of potential topics for new ICRU activities in each area. A body of consultants has been constituted for each technical area to advise the Commission on the need for ICRU recommendations relating to the technical area and on the means for meeting an identified need. Each area is reviewed periodically by its sponsors and consultants. Recommendations of such groups for new reports are then reviewed by the Commission and a priority assigned. The Technical areas are:

Radiation Therapy
Radiation Diagnosis
Nuclear Medicine
Radiobiology
Radioactivity
Radiation Physics—X Rays, Gamma Rays and Electrons
Radiation Physics—Neutrons and Heavy Particles
Radiation Protection
Values of Factors—W, S, etc.
Theoretical Aspects
Quantities and Units

The actual preparation of ICRU reports is carried out by ICRU report committees. One or more Commission members serve as sponsors to each committee and provide close liaison with the Commission. The currently active report committees are:

Average Energy Required to Produce an Ion Pair
Dose Specification for Reporting
Dosimetry of Pulsed Radiation
Fundamental Quantities and Units
High Energy Radiation Dosimetry
Methods of Assessment of Dose in Tracer Investigations
Photographic Dosimetry in External Beam Therapy
Radiobiological Dosimetry
Scanning
Stopping Power
Visual Determination of Resolution of Screen-Film

Systems
Intercomparison of Neutron Instruments

ICRU Reports

In 1962 the ICRU, in recognition of the fact that its triennial reports were becoming too extensive and in some cases too specialized to justify single-volume publication, initiated the publication of a series of reports, each dealing with a limited range of topics. This series was initiated with the publication of six reports:

ICRU Report 10a, *Radiation Quantities and Units*
ICRU Report 10b, *Physical Aspects of Irradiation*
ICRU Report 10c, *Radioactivity*
ICRU Report 10d, *Clinical Dosimetry*
ICRU Report 10e, *Radiobiological Dosimetry*
ICRU Report 10f, *Methods of Evaluating Radiological Equipment and Materials*

These reports were published, as had been many of the previous reports of the Commission, by the United States Government Printing Office as Handbooks of the National Bureau of Standards.

In 1967 the Commission determined that in the future the recommendations formulated by the ICRU would be published by the Commission itself. This report is published by the ICRU pursuant to this policy. With the exception of ICRU Report 10a, the other reports of the "10" series have continuing validity and, since none of the reports now in preparation is designed specifically to supersede them, they will remain available until the material is essentially obsolete. All future reports of the Commission, however, will be published under the ICRU's own auspices. Information about the availability of ICRU Reports is given in page 129.

ICRU's Relationships With Other Organizations

In addition to its close relationship with the International Commission on Radiological Protection, the ICRU has developed relationships with other organizations interested in the problems of radiation quantities, units and measurements. Since 1955, the ICRU has had an official relationship with the World Health Organization (WHO) whereby the ICRU is looked to for primary guidance in matters of radiation units and measurements and, in turn, the WHO assists in the world-wide dissemination of the Commission's recommendations. In 1960 the ICRU entered into consultative status with the International Atomic Energy Agency. The Commission has a formal relationship with the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), whereby ICRU observers are invited to attend UNSCEAR meetings. The Commission and the International Organization for Standardization (ISO) informally exchange notifications of meetings and the ICRU is formally designated for liaison with two of the ISO Technical Committees. The ICRU also corresponds and exchanges final reports with the following organizations:

Bureau International des Poids et Mesures
Commission of the European Communities
Council for International Organizations of Medical Sciences
Food and Agriculture Organization
International Council of Scientific Unions
International Electrotechnical Commission
International Labor Office
International Union of Pure and Applied Physics
United Nations Educational, Scientific and Cultural Organization

The Commission has found its relationships with all of these organizations fruitful and of substantial benefit to the ICRU program. Relations with these other international bodies do not affect the basic affiliation of the ICRU with the International Society of Radiology.

Operating Funds

In the early days of its existence, the ICRU operated essentially on a voluntary basis, with the travel and operating costs being borne by the parent organizations of the participants. (Only token assistance was originally available from the International Society of Radiology.) Recognizing the impracticability of continuing this mode of operation on an indefinite basis, operating funds were sought from various sources.

Financial support has been received from the following organizations:

B.A.T. Cigaretten-Fabriken GMBH
Commission of the European Communities
Council for International Organizations of Medical Sciences
Eastman Kodak
Ford Foundation
General Electric Company
International Atomic Energy Agency
International Radiation Protection Association
International Society of Radiology
Japan Industries Association of Radiation Apparatus
John och Augusta Perssons stiftelse
National Cancer Institute of the U.S. Department of Health, Education and Welfare
N.V. Philips Gloeilampenfabrieken
Picker Corporation

Radiological Society of North America
Rockefeller Foundation
Siemens Corporation
Society of Nuclear Medicine
Statens laegevidenskabelige Forskningsrad
U.S. Bureau of Radiological Health of the Food and Drug Administration
World Health Organization

In recognition of the fact that its work is made possible by the generous support provided by these organizations, the Commission expresses its deep appreciation.

Composition of the ICRU

It is of interest to note that the membership of the Commission and its subgroups totals 90 persons drawn from 13 countries. This gives some indication of the extent to which the ICRU has achieved international breadth of membership within its basic selection requirement of high technical competence of individual participants.

The current membership of the Commission is as follows:

H. O. Wyckoff, *Chairman*
A. Allisy, *Vice Chairman*
K. Lidén, *Secretary*
R. S. Caswell
H. J. Dunster
P. Edholm
J. R. Greening
D. Harder
P. Harper
A. Kellerer
H. H. Rossi
W. K. Sinclair
A. Wambersie
L. S. Taylor, *Honorary Chairman and Member Emeritus*

Composition of Report Committee Responsible for the Drafting of this Report

This report was prepared by the Report Committee on Neutron Dosimetry for Biology and Medicine. Serving on the Report Committee during the preparation of this report were:

J. J. Broerse, *Chairman*
D. K. Bewley
L. J. Goodman
D. Greene
B. J. Mijnheer

L. L. Anderson, G. Burger, R. S. Caswell, J. J. Coyne, J. A. G. Davids, A. M. Kellerer, J. Law, A. C. Lucas, M. L. Randolph, and W. L. Zijp served as consultants to the Committee.

H. H. Rossi and W. K. Sinclair served as Commission Sponsors for the Report Committee.

The Commission wishes to express its appreciation to the individuals involved in the preparation of this report for the time and effort they devoted to this task.

Harold O. Wyckoff
Chairman, ICRU

Washington, D.C.
15 May 1976

List of Symbols

a = cross-sectional area of a sphere
A = activity
A_0 = change in optical density
A_w = response function of a dosimeter for a specific neutron spectrum relative to its response function for a known calibration field
A_{Cd} = activity of a material irradiated under a cadmium cover
A_{int} = activity induced by neutrons of intermediate energy
A_p = nuclear mass of particle p
B = ratio of mass of an ion to mass of a neutron
b = distance
c_m = molarity
c = instrument calibration relating energy to channel number of a pulse height analyzer
C_N = conversion factor used to determine the absorbed dose of neutrons from the response of a dosimeter used for a neutron spectrum different from that with which it was calibrated
d = diameter
$\bar{d}$ = mean chord length
$d(L)$ = differential distribution of absorbed dose in LET L
$d(y)$ = differential distribution of absorbed dose in lineal energy y
$d_1(z)$ = differential distribution of absorbed dose in events of specific energy z
D = absorbed dose
D_c = absorbed dose in calibration field c
D_g = absorbed dose in a gas g
D_G = absorbed dose of photons
ΔD_G = uncertainty of absorbed dose of photons
$D(L)$ = integral distribution of the quotient of absorbed dose with lineal energy up to L by the total absorbed dose
$D(L_1/L_2)$ = absorbed dose between two values of LET, L_1 and L_2
D_m = absorbed dose in a medium m
D_N = absorbed dose of neutrons
ΔD_N = uncertainty of absorbed dose of neutrons
D_w = absorbed dose delivered by radiation field w
$D(y)$ = integral distribution of the quotient of absorbed dose delivered by events up to lineal energy y by the total absorbed dose
e = electronic charge
E = kinetic energy
E_{Cd} = cadmium cut-off energy
E_d = deuteron energy
E_{fl} = total energy, exclusive of rest energies
E_m = Maxwellian energy
E_{max} = maximum energy
E_{min} = minimum energy
E_N = neutron energy
$\bar{E}_N$ = mean neutron energy
ΔE_N = increment of neutron energy
E_{tr} = energy transferred to kinetic energy of charged particles
$\bar{E}_\phi$ = mean energy of fluence spectrum
$\bar{E}_\psi$ = mean energy of energy fluence spectrum
f = fraction of total absorbed dose due to gamma rays
$f(y)$ = differential distribution of the number of events in lineal energy y
$f_1(z)$ = differential distribution of the number of events in specific energy z
$F(y)$ = integral distribution of events resulting in lineal energy up to y
g = correction factor for deviation from $1/v$ law
G_a = geometrical attenuation factor
G = quotient of number of ions produced by mean energy imparted
G_G = quotient of number of ions produced by mean energy imparted by gamma rays
G_M = quotient of number of ions produced by mean energy imparted by mixed neutrons and gamma rays
G_N = quotient of number of ions produced by mean energy imparted by neutrons
h = pulse height
h_T = relative photon sensitivity of a dosimeter having approximately the same sensitivity to neutrons and photons
h_U = relative photon sensitivity of a dosimeter having lower sensitivity to neutrons than to photons
h_1 = lower limit of pulse height for determination of absorbed dose
h_2 = upper limit of pulse height for determi-

nation of absorbed dose

I_{tot} = total resonance integral cross section for activation

$I_{1/v}$ = contribution to the resonance integral cross section arising from $1/v$ part of σ_E

I' = resonance integral cross section for activation, excluding $1/v$ part of σ_E

J_g = quotient of ionization charge by mass of gas

k_T = relative neutron sensitivity of a dosimeter having approximately the same sensitivity to neutrons and photons

k_U = relative neutron sensitivity of a dosimeter having lower sensitivity to neutrons than to photons

Δk_U = overall uncertainty of relative neutron sensitivity of a dosimeter having lower sensitivity to neutrons than to photons

K = kerma

K_{CO_2} = kerma in carbon dioxide

K_G = kerma of gamma rays

K_m = kerma in material m

K_N = kerma of neutrons

K_t = kerma in tissue t

l = length or distance

L = linear energy transfer

L_Δ = restricted linear energy transfer or restricted linear collision stopping power, up to cut-off energy Δ

L_1 = lower limit of LET

L_2 = upper limit of LET

L_∞ = total linear energy transfer or total linear collision stopping power

$\bar{L}_D$ = dose average linear energy transfer

$\bar{L}_T$ = track average linear energy transfer

m = mass

m_c = mass of gas during calibration in field c

M = molar mass

N = number of particles

N_A = Avogadro constant

$N(h)$ = number of pulses accumulated in channel h of pulse height analyzer

$N(L)$ = number of pulses in the channel corresponding to the pulse height observed when a particle of LET L traverses a major diameter of a spherical proportional counter

N_{th} = quotient of number of thermal neutrons by volume

$N(x)$ = number of neutrons which have not undergone an interaction after traveling a distance x

$N(0)$ = number of neutrons at a distance $x = 0$

P = absolute gas pressure

P_c = absolute gas pressure in calibration field c

Q = energy released by particle transformation or nuclear reaction

Q = charge

Q_c = ionization charge produced in calibration field c

Q_N = ionization charge produced by neutrons

Q_T = total ionization charge produced by neutrons and gamma rays

r = radius

R = dosimeter response

R_G = dosimeter response to gamma rays

R_N = dosimeter response to neutrons

R_w = dosimeter response in radiation field w

R_T' = quotient of dosimeter response by its sensitivity to the radiation used for calibration, for a dosimeter having approximately the same sensitivity to neutrons and photons

R_U' = quotient of dosimeter response by its sensitivity to the radiation used for calibration, for a dosimeter having lower sensitivity to neutrons than to photons

s = distance from radiation source to surface of phantom

$s_{m,g}$ = mass stopping power ratio for a medium m and a gas g

$(s_{m,g})_c$ = mass stopping power ratio for a medium m and a gas g for calibration field c

S = surface area

$(\bar{S}/\rho)_{col,g}$ = average collision mass stopping power for a gas g

$(\bar{S}/\rho)_{col,m}$ = average collision mass stopping power for a medium m

t = time

$t(L)$ = differential distribution of track length in linear energy transfer L

T = absolute temperature

T_c = absolute temperature in calibration field c

$T(L)$ = integral distribution of the quotient of the length of charged particle tracks with linear energy transfer up to L by the total track length

$T_{1/2}$ = half-life

v = velocity

v_0 = velocity of thermal neutrons

V = volume

$\bar{W}$ = average energy expended to create an ion pair

$\bar{W}_c$ = average energy expended to create an ion pair in calibration field c

$\bar{W}_G$ = average energy expended to create an ion pair by charged particles produced by gamma rays

$\bar{W}_N$ = average energy expended to create an ion pair by charged particles produced by neutrons

x = length or distance
y = lineal energy
$\bar{y}_D$ = mean of absorbed dose distribution of lineal energy y
$\bar{y}_F$ = mean of frequency distribution of lineal energy y
z = specific energy
$\bar{z}_D$ = mean of absorbed dose distribution of specific energy z
$\bar{z}_F$ = mean of frequency distribution of specific energy z
α = response function
α_c = response function for a neutron calibration spectrum c
β = response function
δ_G = thermal defect for gamma rays
δ_N = thermal defect for neutrons
Δ = cut-off energy
ϵ_m = molar extinction coefficient
ϵ = energy imparted to the matter in a volume
ϵ_{ex} = energy of particles which have left the matter in a volume
ϵ_{in} = energy of particles which have entered the matter in a volume
$\bar{\epsilon}$ = mean energy imparted to the matter in a volume
θ = differential distribution of energy fluence rate in energy for neutrons of intermediate energy
λ = decay constant
μ = linear attenuation coefficient or macroscopic cross section
μ_{tr} = energy transfer coefficient
μ/ρ = mass attenuation coefficient
μ_{tr}/ρ = mass energy transfer coefficient
ρ = density
σ = microscopic cross section
σ_E = microscopic cross section at energy E
σ_0 = microscopic cross section for thermal neutrons
$\bar{\sigma}$ = average microscopic cross section
Σ = linear attenuation coefficient or macroscopic cross section
φ = fluence rate
φ_E = differential distribution of fluence rate in energy E
$\varphi_{E,int}$ = differential distribution of fluence rate in energy E for neutrons of intermediate energy
φ_{int} = fluence rate for neutrons of intermediate energy
Φ = fluence
Φ_E = differential distribution of fluence in energy E
$\Phi(E)$ = integral distribution of fluence for particles up to energy E
ψ = energy fluence rate
Ψ = energy fluence
Ψ_E = differential distribution of energy fluence in energy E
$\Psi(E)$ = integral distribution of energy fluence for particles up to energy E
$\dot{\Psi}_{E,int}$ = differential distribution of energy fluence rate in energy for neutrons of intermediate energy
Ω = angle

CONTENTS

Neutron Dosimetry for Biology and Medicine

1. Introduction

1.1 Applications of Neutrons in Biology and Medicine

The objective of neutron absorbed dose determinations is to describe the energy deposition in irradiated material in such detail that workers in biology and medicine may make unambiguous correlations with observed responses, or predictions of responses, of irradiated biological systems. This implies that the selection of methods employed for this dosimetry and of the numerical data to be registered, will depend to a large extent on the object irradiated and on the endpoint observed (Barendsen and Broerse, 1972).

In practice the set of physical quantities should include the neutron absorbed dose and the absorbed dose of accompanying photons, and their distribution in time. In addition, it is essential to assess the radiation quality, which can be related to the neutron energy spectrum, lineal energy (y) spectra, or the linear energy transfer (LET) spectrum, in view of the dependence of the relative biological effectiveness (RBE) on these interrelated parameters.

Since neutrons dissipate their energy in tissue through different nuclear interactions with the various constituents of the material, the energy deposition is characterized by a complex spectrum of secondary charged particles. At the microscopic level the physical effects of these particles at a point in the medium can be related to y spectra or to LET distributions (ICRU, 1970a; ICRU, 1971a). For the distribution of the fast neutron absorbed dose in LET two important regions can be distinguished, corresponding to the energy dissipated by protons and by interactions with C, O and N nuclei.

The types of interaction processes depend strongly on the energy of the neutrons; this has led to a rough classification of neutrons according to their energy. In this report the following energy ranges are distinguished: thermal neutrons with energies below the cadmium cut-off energy (approximately 0.5 eV), intermediate neutrons with energies above the cadmium cut-off energy to approximately 10 keV, and fast neutrons with energies above approximately 10 keV.

Due to the complexity of the secondary particle spectra, neutrons are less suitable than are charged particles for fundamental investigations of the mechanisms by which effects of ionizing radiation on living cells are initiated. However, fast neutrons are of great interest for radiobiology since they offer the practical possibility of exposing relatively large objects to high-LET radiation with a relatively uniform absorbed dose distribution throughout the subject. In addition, neutrons from certain sources, e.g., fission reactors, provide large fields for the irradiation of many biological objects at a time, an essential feature for the study of effects in animal populations.

Recent interest in the use of fast neutrons for radiotherapy has posed a number of questions connected with the absorbed dose distribution in patients. Collimators used to provide well-defined beams affect the primary neutron energy spectrum and the contribution from gamma radiation. Additional changes in radiation quality occur due to absorption and scattering processes in the patient and in the environment. Detailed examination of these features is essential for the evaluation of fast neutron radiotherapy.

1.2 Purpose of the Report

This report deals with the currently available methods for determining absorbed dose and kerma of fast neutrons employed for radiobiological and medical applications. Since neutrons are always accompanied by photons, it is necessary to select a dosimetry system which makes possible the quantitative separation of these two radiation components. However, it should be realized that for neutrons the quantities absorbed dose and kerma are of limited value without

information on radiation quality. The concept of quality relates to the microscopic distribution of locally absorbed energy within the irradiated material. The energy spectrum of the incident neutrons determines the microdosimetric quantities and may therefore be considered an implicit characterization of radiation quality (Rossi, 1971). Neutron dosimetry methods will be adequate only if they quantify all the essential parameters: kerma or absorbed dose, quality, contribution of photon radiation and the spatial and temporal variation of these parameters in the irradiated object.

The present report is intended to cover the field of neutron dosimetry in biology and medicine in a comprehensive way. This implies a substantial updating of NBS Handbook 75 on Measurements of Absorbed Dose of Neutrons, and of Mixtures of Neutrons and Gamma Rays (NCRP, 1961), and also some sections of ICRU Report 10b on Physical Aspects of Irradiation (ICRU, 1964). Basic concepts and definitions employed in neutron dosimetry have been discussed in ICRU Report 19 (ICRU, 1971a) and a summary of concepts and units is given in Section 2 of this report. In addition, Section 2 deals with principles of the experimental techniques employed and provides background for subsequent sections. Descriptions of methods and instrumentation are given in Section 3; characteristics of instruments are discussed with reference to random and systematic uncertainties,[1] energy dependence and sensitivity to gamma radiation. During dosimetry measurements or irradiations of biological objects it is usually essential to monitor the irradiation conditions; appropriate techniques are reviewed in Section 4. The uniformity and reproducibility of the absorbed dose patterns in biological specimens and in patients are determined to a large extent by the inherent characteristics of the neutron sources. Specific features of different neutron sources are discussed in Section 5. Specific problems of neutron dosimetry in radiobiology and radiotherapy, are discussed in Section 6 and 7, respectively. Quantitative information on mass energy transfer coefficients and atomic compositions of compounds and mixtures, which is essential for the interpretation of measurements with specific dosimetry systems, is presented in Appendices A and B. In the general discussion of characteristics of measuring devices, presented in Section 3, it was inappropriate to include detailed information on two types of dosimetry systems, i.e., ionization chambers and activation and fission detectors. Three separate appendices give information on properties and fabrication techniques for tissue-equivalent plastics (Appendix C), ionization chamber construction and measurements (Appendix D) and cross section data for selected threshold reactions (Appendix E).

In this report the main emphasis has been placed on the interaction of fast neutrons with biological material; for specific techniques employed to characterize neutron fields in free air the reader is also referred to ICRU Report 13 on Neutron Fluence, Neutron Spectra and Kerma (ICRU, 1969a). Applications of neutron dosimetry in radiation protection are not included in the present report since this information can be found in ICRU Report 20 on Radiation Protection Instrumentation and Its Application (ICRU, 1971b).

The scope of this report has not been confined to a specific neutron energy range. The clinical use of neutrons generated with accelerators implies that dosimetry may have to be carried out for neutron energies up to and beyond 50 MeV. Attention has been given to dosimetry of the intermediate and thermal neutrons which accompany fast neutrons in extended media. Since the use of primary beams of thermal and intermediate neutrons for biology and medicine is limited, however, the dosimetry for neutrons with energies below 10 keV is reviewed only briefly. Specific information on neutron capture therapy and the medical applications of neutron activation analysis is not included in the present report.

1.3 Neutron Interactions with Biological Matter

Neutrons entering a biological medium release energy by different types of interactions; namely, elas-

TABLE 1-I—*Summary of the most important neutron interactions in biological tissues for energies up to approximately 100 MeV*

Element	Interaction
Hydrogen	Elastic scattering Neutron capture
Carbon	Elastic scattering Inelastic scattering (n,n'3α) and (n,α) reactions
Nitrogen	Elastic scattering Inelastic scattering (n,p), (n,d), (n,t), (n,α), (n,2α) and (n,2n) reactions
Oxygen	Elastic scattering Inelastic scattering (n,α) and (n,p) reactions

[1] This report uses the terminology "random uncertainty", "systematic uncertainty" and "overall uncertainty" rather than "precision" and "accuracy". Although the mathematical meaning of a random uncertainty can usually be stated formally, the systematic uncertainty and the overall uncertainty are not amenable to such formalism. All three of these terms generally are used in this report without rigorously qualifying their derivation since this information was either unavailable or the uncertainty was based on a subjective assessment of the approximate maximum possible deviation.

tic scattering, inelastic scattering (interactions with the bombarded nucleus in which the neutron is promptly reemitted and is generally accompanied by the emission of a nuclear deexcitation gamma ray), nonelastic scattering (interactions with the bombarded nucleus which result in the emission of particles other than a single neutron), capture processes and spallation reactions. Table 1-I summarizes the most important neutron interactions in biological tissues for energies up to approximately 100 MeV. The characteristics of the different types of nuclear interactions have been reviewed by Auxier *et al.* (1968).

Compilation of neutron cross sections has been performed by various centers; reference can be made to the ENDF cross section evaluations (Evaluated Nuclear Data Files) prepared by the National Neutron Cross Section Center at Brookhaven National Laboratory, the INDC reports (International Nuclear Data Committee) prepared by the Nuclear Data Section of the International Atomic Energy Agency, the United Kingdom Nuclear Data Library (UKNDL) and the KEDAK data file (Kernenergetische Datenanalysen Kompilation).

For the various calculations of the energy transfer coefficients of fast neutrons, e.g., those carried out by Randolph (1957), Bach and Caswell (1968), Ritts *et al.* (1970) and Dennis (1973), existing neutron cross section data have been employed or extrapolation and interpolation procedures have been applied. Values of kerma per unit neutron fluence reported in the literature for biological tissues show appreciable differences. These are due to the omission of the effects of inelastic scattering and nuclear reactions in some calculations; or to the restriction of the total number of elements included in the calculations to the four elements of greatest abundance, i.e., C, H, O and N. Appendix A presents new tables of mass energy transfer coefficients prepared by Caswell *et al.* (1977) using the most recent cross section evaluations (ENDF/B Version IV). Additional information on the quotient of kerma by fluence and on mass energy transfer coefficients can be found in Section 2.

Table 1-I shows that the energy dissipation of fast neutrons in biological materials occurs from recoil protons, heavy recoil nuclei and products of nuclear reactions. The relative contributions to the kerma or absorbed dose in soft tissue from these interactions depend on the neutron energy. Calculations of mass energy transfer coefficients show that the contribution from hydrogen decreases from 97 percent of the total kerma at a neutron energy of 10 keV to 69 percent at an energy of 18 MeV (see Figure 1.1). Due to the large contribution from recoil protons to the neutron kerma, energy deposition in various biological materials is determined mainly by the hydrogen content. Thus, in soft tissue containing about 10 percent by weight of hydrogen, a 1 percent deviation in the total hydrogen content will change the kerma in the tissue by approximately 10 percent.

For biological and medical applications of fast neutrons, energy deposition is generally determined in soft tissue. Biological material varies in its composition and the dosimetric consequences are discussed in Section 6.1. Appendix B gives the atomic compositions for soft tissue and for a number of compounds and mixtures utilized for phantoms and dosimetric

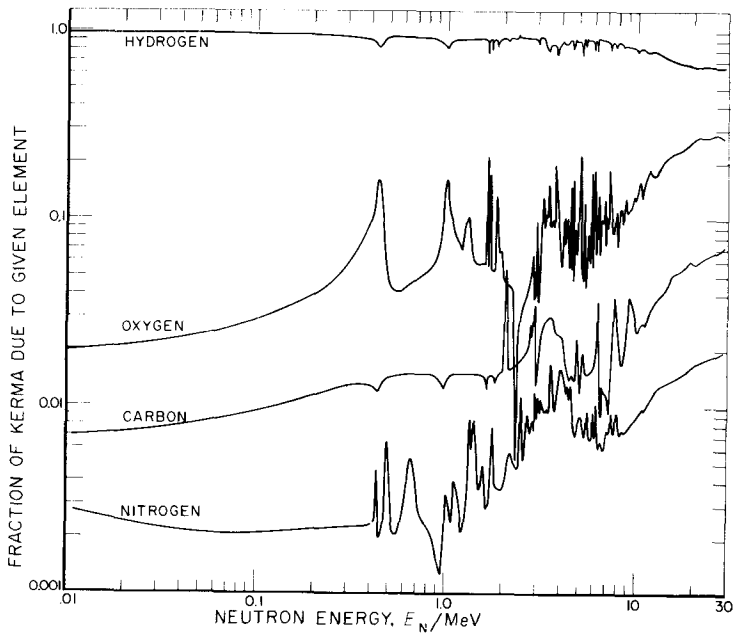

Fig. 1.1 Relative contribution from interaction processes with different elements to kerma in soft tissue (a four-element tissue model has been used for this figure: 10.2% H, 12.3% C, 3.5% N, 74.0% O by weight) (Caswell, *et al.* 1977). [Used with permission of the authors.]

devices. Although endeavors have been made to match practical mixtures to the atomic composition of soft tissue as closely as possible, the composition of materials usually employed show deviations, especially with regard to the relative carbon and oxygen contents. The values of kerma and absorbed dose measured in a detector material can be converted to the corresponding values in tissue by applying the ratio of mass energy transfer coefficients for the two materials. This can be derived for a particular neutron energy spectrum from the quantitative data presented in Appendix A.

2. Concepts and Principles

This section deals with the principal quantities relevant to neutron dosimetry and their interrelation. A more complete compilation of definitions is given in ICRU Report 19 (ICRU, 1971a).

Although the principal object of dosimetry is the determination of absorbed dose and its temporal and spatial variations in the irradiated objects, quantities other than absorbed dose must often be considered. There are several reasons for this. One is that one may deal with a certain radiation field or an arrangement of radiation sources and that a description of this situation is desired which is valid regardless of the objects which may be exposed to the radiation field. In such a case quantities are necessary which relate to the radiation field and permit ready determination or at least approximate estimates of the absorbed dose when different objects are introduced into the radiation field. Sections 2.1 and 2.2 deal with such quantities.

A second and equally important reason for the necessity of additional dosimetric quantities is the inherent limitation of the concept of absorbed dose. All ionizing radiations deposit energy essentially by the same atomic and molecular mechanisms; this is the reason why one single quantity, absorbed dose, is applicable to all types of ionizing radiation. However, the ratio of energy imparted and the mass of the irradiated object, although a meaningful parameter, does not uniquely determine the effects of irradiation; the effect is also determined by the microscopic distribution of the imparted energy. This distribution varies considerably with different radiation types, and quantities which determine these distributions must therefore be considered in addition to absorbed dose. Because neutrons release densely ionizing nuclei of different energies, the fluctuations on a microscopic scale of the energy imparted by such nuclei are considerable. The biological effectiveness of neutrons can for this reason substantially exceed that of sparsely ionizing radiations, but it depends strongly on the energy of the neutrons. It is therefore essential to consider the neutron energy spectrum. Quantities which afford a direct description of the microscopic distribution of energy absorption are described in Section 2.4 which deals with the concept of LET and with microdosimetry.

The basic principles of interface dosimetry, calibration, mixed field dosimetry and effects of finite size of the dosimeter are described in Sections 2.5 to 2.8.

2.1 Fluence and Energy Fluence

The most elementary characterization of a radiation field is one in terms of the type, energy, direction and number of particles. The fluence of a radiation field is defined in terms of the number of particles entering a spherical volume.

Definition: The **fluence,** Φ, of particles is the quotient of dN by da, where dN is the number of particles which enter a sphere of cross-sectional area da.

$$\Phi = \frac{dN}{da} \tag{2.1-1}$$

For simplicity dN is termed a number of particles and da a cross-sectional area. However, both quantities are written as differentials because the definition must apply also to non-uniform fields where the fluence may vary from point to point and where accordingly one has to consider the limiting case of an infinitesimally small sphere in order to obtain the fluence at a specified point. This remark applies also to the definitions which follow.

Furthermore, it must be noted that dN is the differential of a mean, or expectation, value. Due to the statistical nature of the radiation field and of its interaction with matter one always deals with random variables in radiation measurements. The definitions must therefore, at least in principle, be based on mean values which result as limit values when repeated measurements are averaged.

In most of the following definitions this twofold limit process will be understood; only in the case of absorbed dose will the underlying stochastic quantities be considered explicitly.

Instead of a finite period of time one can consider a particular instant of time; then one deals with the quantity fluence rate.

Definition: The **fluence rate,** φ, of particles is the quotient of $d\Phi$ by dt, where $d\Phi$ is the increment of particle fluence in the time interval dt.

$$\varphi = \frac{d\Phi}{dt} \tag{2.1-2}$$

The fluence of a neutron field and the fluence of the associated gamma-ray component give only incomplete information about the field. In general, one deals with radiations which are neither monoenergetic nor isotropic. One must then consider the spec-

tral distributions of the fluence or fluence rate in energy or their angular distributions. These distributions will be considered later in this section.

For a unidirectional field, the fluence, Φ, represents the number of particles traversing a unit surface area normal to the radiation field. In an isotropic radiation field the number of particles traversing a surface of unit area is numerically equal to $\Phi/2$. This follows from the fact that the surface of a sphere is 4 times its cross-sectional area and that the surface is twice traversed by each particle traversing the sphere. Furthermore, one can show that the total mean track length of the particles per unit volume is numerically equal to Φ regardless of whether one deals with an isotropic or nonisotropic field (see e.g. Kellerer, 1971a).

In some situations one is more interested in the total energy transported through certain boundaries than in the number of particles traversing these boundaries. Under these circumstances one can use the quantity energy fluence which is defined in terms of the total kinetic energy of particles entering a spherical volume element within a certain time.

Definition: The **energy fluence**, Ψ, of particles is the quotient $\mathrm{d}E_{\mathrm{fl}}$ by $\mathrm{d}a$, where $\mathrm{d}E_{\mathrm{fl}}$ is the sum of the energies, exclusive of rest energies, of all the particles which enter a sphere of cross-sectional area $\mathrm{d}a$.

$$\Psi = \frac{\mathrm{d}E_{\mathrm{fl}}}{\mathrm{d}a} \tag{2.1-3}$$

As in the case of the fluence one may consider the quantity which refers to a particular instant of time.

Definition: The **energy fluence rate**, ψ, is the quotient of $\mathrm{d}\Psi$ by $\mathrm{d}t$, where $\mathrm{d}\Psi$ is the increment of energy fluence in the time interval $\mathrm{d}t$.

$$\psi = \frac{\mathrm{d}\Psi}{\mathrm{d}t} \tag{2.1-4}$$

Spectral distributions in energy are considered below.

In a unidirectional and uniform field, Ψ is the mean energy transported through a unit cross-sectional area perpendicular to the direction of the field. In an isotropic field the energy transported through a unit surface element is numerically equal to $\Psi/2$.

The terms "flux density" and "energy flux density" are occasionally used instead of "fluence rate" or "energy fluence rate."

The integral spectra $\Phi(E)$ and $\Psi(E)$ of fluence and of energy fluence are defined in the same way as fluence and energy fluence, but with the additional condition that only particles up to a kinetic energy, E, are considered. $\Phi(E)$ and $\Psi(E)$ are the fluence and energy fluence due to particles with energies up to E. These two quantities are therefore increasing functions of E, and the limit values of $\Phi(E)$ and $\Psi(E)$ for increasing E are Φ and Ψ.

The **differential spectra of fluence** and **energy fluence** with regard to energy are defined as the derivatives of $\Phi(E)$ and $\Psi(E)$ with respect to E.

$$\Phi_E = \frac{\mathrm{d}\Phi(E)}{\mathrm{d}E} \tag{2.1-5}$$

$$\Psi_E = \frac{\mathrm{d}\Psi(E)}{\mathrm{d}E} \tag{2.1-6}$$

$\Phi_E\,\mathrm{d}E\,\mathrm{d}a$ is equal to the mean number of particles with kinetic energy between E and $E + \mathrm{d}E$ which enter a spherical volume element of cross-sectional area $\mathrm{d}a$.

$\Psi_E\,\mathrm{d}E\,\mathrm{d}a$ is the mean total kinetic energy transported by particles with energy between E and $E + \mathrm{d}E$ entering a spherical volume element of cross-sectional area $\mathrm{d}a$.

The following relations result from the definition of the quantities $\Phi(E)$, $\Psi(E)$, Φ_E, and Ψ_E:

$$\Psi_E = E\ \Phi_E \tag{2.1-7}$$

$$\Phi(E) = \int_0^E \Phi_E\,\mathrm{d}E \tag{2.1-8}$$

$$\Psi(E) = \int_0^E \Psi_E\,\mathrm{d}E = \int_0^E E\ \Phi_E\,\mathrm{d}E \tag{2.1-9}$$

It is sometimes practical to speak of a mean neutron energy in a given field. However, one must note that such a mean can either be defined as the **mean of the fluence spectrum,**

$$\bar{E}_\Phi = \int_0^\infty E\ \Phi_E\,\mathrm{d}E/\Phi = \Psi/\Phi \tag{2.1-10}$$

or as the **mean of the energy fluence spectrum,**

$$\bar{E}_\Psi = \int_0^\infty E\ \Psi_E\,\mathrm{d}E/\Psi = \int_0^\infty E^2\ \Phi_E\,\mathrm{d}E \Big/ \int_0^\infty E\ \Phi_E\,\mathrm{d}E \tag{2.1-11}$$

The two mean values are different whenever one deals with a neutron field which is not monoenergetic; the mean of the energy fluence spectrum always exceeds the mean of the fluence spectrum.

The kinetic energy of the particles is not the only variable necessary to describe the radiation field. In most cases one deals also with an angular distribution of the particle flow. Frequently, for example, if one deals with the irradiation of microorganisms which are randomly oriented in space, one need not account for the angular distribution in the field. In other cases, for example, in the determination of

depth-dose distributions in extended absorbers, the angular distribution of fluence has to be considered. Since the angular distribution of fluence or energy fluence is relevant to specific calculations rather than to those considerations which generally occur in neutron dosimetry, explicit definitions are not given in the present context.

2.2 Interaction Coefficients and Kerma

In the preceding section quantities have been considered which characterize a radiation field. If such information on the radiation field is to be used to determine the energy deposition in an exposed object, further quantities are needed; namely, the material constants which describe the interaction of radiation and matter. First, one deals with those quantities which express the probabilities of interaction of indirectly ionizing radiation with matter, and second, one is concerned with the interaction cross-sections between the charged secondaries and the irradiated medium.

In this section the first class of interaction coefficients will be discussed. They permit the derivation of the changes in fluence which occur when an object is introduced into the radiation field; furthermore, they permit the derivation of kerma, a quantity which is in many cases a suitable approximation to absorbed dose. Kerma will also be discussed in the present section. In Section 2.3 the interrelation of kerma and absorbed dose will be considered.

In the following the term "interaction of neutrons with the irradiated medium" designates events in which the energy or momentum of the neutrons is changed.

Definition: The **mass attenuation coefficient,** μ/ρ, of a material for indirectly ionizing particles of specified energy is the quotient of dN/N by ρdl, where dN/N is the fraction of particles that experience interactions while traversing a distance dl in a medium of density ρ.

$$\frac{\mu}{\rho} = \frac{1}{\rho N}\frac{dN}{dl} \tag{2.2-1}$$

According to this definition the number $N(x)$ of neutrons which have not undergone an interaction after traveling a distance x is:

$$N(x) = N(0)\, e^{-\mu x} \tag{2.2-2}$$

where $N(0)$ is the number of neutrons at $x = 0$. The inverse of μ is the mean free path of the neutrons in the material (see for example Beckurts and Wirtz, 1964).

The quantity μ is frequently called the macroscopic cross section of the material with regard to neutrons of the specified energy; sometimes the symbol Σ is used instead of μ.

A **microscopic cross section,** σ, i.e. an interaction probability per atom and unit fluence, is defined as:

$$\sigma = \frac{M}{N_A}\frac{\mu}{\rho} \tag{2.2-3}$$

where ρ is the density of the material, M is the molar mass of the medium, and N_A is the Avogadro constant.

The mass attenuation coefficient, μ/ρ, or the cross sections μ or σ can be expressed by sums of partial cross sections for the various neutron reactions such as elastic scattering, inelastic scattering, and the other possible interactions of neutrons with the nuclei of the irradiated material.

The quantities μ/ρ, μ, or σ determine the change of the fluence of primary neutrons in a medium exposed to a field of neutrons. In order to derive the fluence of scattered neutrons one must know the differential spectra of the cross sections for various energy transfers and momentum transfers of the neutrons to the nuclei of the irradiated medium (see for example, BNL 325 (1973) and BNL 400 (1970)).

Definition: The **mass energy transfer coefficient,** μ_{tr}/ρ, of a material for indirectly ionizing particles of specified energy is the quotient of dE_{tr}/E by ρdl where dE_{tr}/E is the fraction of incident particle energy (excluding rest energies) that is transferred to kinetic energy of charged particles by interactions in traversing a distance dl in a medium of density ρ.

$$\frac{\mu_{tr}}{\rho} = \frac{1}{\rho E}\frac{dE_{tr}}{dl} \tag{2.2-4}$$

As in the case of the mass attenuation coefficient, the quantity μ_{tr}/ρ can be represented as a sum of partial mass energy transfer coefficients for the various possible neutron reactions at the specified energy in the specified medium. For detailed discussions see ICRU Report 13 (ICRU, 1969).

In the interactions of neutrons and the irradiated medium charged particles with sufficient energy to ionize are liberated. The range of these charged particles is not always small as compared to the dimensions of the irradiated object. The distribution of absorbed dose in the exposed object is therefore determined by the interaction cross sections of the neutrons as well as by the interaction cross sections of the charged secondaries. The derivation of the absorbed dose and its spatial variations can be complicated in such cases, and it is often more practical to treat the problem in an approximation which does