MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES
3 4456 0357337 2

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

ORNL-RSIC-16
UC-80 — Reactor Technology

1241

USE OF ICRU-DEFINED QUANTITIES AND UNITS IN SHIELDING

D. K. Trubey

OAK RIDGE NATIONAL LABORATORY
MAIN LIBRARY
4500N
LIBRARY LOAN COPY
DO NOT TRANSFER TO ANOTHER PERSON.
If you wish someone else to see this report, send in name with report and the library will arrange a loan.
ORNL-118 (9-07)

RADIATION SHIELDING INFORMATION CENTER

Printed in the United States of America. Available from Clearinghouse for Federal Scientific and Technical Information, National Bureau of Standards, U.S. Department of Commerce, Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.65

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ORNL-RSIC-16

Contract No. W-7405-eng-26

Neutron Physics Division

USE OF ICRU-DEFINED QUANTITIES AND UNITS IN SHIELDING

D. K. Trubey

OCTOBER 1968

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION

MARTIN MARIETTA ENERGY SYSTEMS LIBRARIES

3 4456 0357337 2

TABLE OF CONTENTS

PAGE

Abstract 1

Acknowledgment 2

Summary. 3

Foreword 5

Introduction 8

The ICRU 9

Types of Radiation Transport Calculation 10

A. Shielding Research 11

B. Shield Design 12

C. Absorbed Dose or Dose Equivalent Calculation. 12

Radiation Field Characterization 13

Dose Equivalent and Multicollision Considerations. 17

Response Functions for the Interior of Shields 21

Discussion ICRU Definitions. 22

Absorbed Dose 22

Dose Equivalent 22

Dose. 23

Flux Density Versus Flux. 25

Kerma 25

Exposure. 27

Conceptual Difficulties with Simple Models 27

Mass Absorption 29

Modified Mass Absorption Coefficient. 29

Buildup Factors. 30

Buildup Factors for Neutrons 31

References 33

PAGE

Appendix I - A Note by H. H. Rossi on Absorbed Dose Versus Kerma 34

Appendix II - A Reprint of: Radiation Quantities and Units International Commission on Radiological Units and Measurements (ICRU). 38

Appendix III - Selected Definitions from USA Standard Glossary of Terms in Nuclear Science and Technology 53

Appendix IV - Selected Definitions of Thorn. 58

ABSTRACT

The International Commission on Radiological Units and Measurements ~~(ICRU)~~ and the USA Standards Institute have offered definitions and made recommendations concerning nomenclature that should be adopted to the greatest extent possible in the shielding community. Because of the nuclear engineering context for shielding calculations and design, special problems have arisen. In order to help alleviate some of these problems, a review is presented along with recommendations. Some of these recommendations are controversial and do not represent a consensus of the many reviewers who have contributed to this work. These cases are identified.

Discussions are given of terms such as flux density, dose, kerma, energy deposition coefficient, response functions, and buildup factors.

ACKNOWLEDGMENT

The author is grateful to F. H. Clark and A. B. Chilton without whose encouragement this report might not have been issued. Their helpful review and those of H. H. Rossi, H. O. Wyckoff, and H. C. Claiborne saved the author from several rash or incorrect statements. The author also would like to thank Miss Patti Callaghan for her careful preparation of the numerous drafts.

SUMMARY

The International Commission on Radiological Units and Measurements (ICRU) and the USA Standards Institute have offered definitions and made recommendations concerning nomenclature which should be adopted to the greatest extent possible in the shielding community. Because of the nuclear engineering context for shielding calculations and design, special problems arise.

A discussion is given of terms such as flux density, dose, kerma, energy deposition coefficient, and buildup factors which leads to the following statements and recommendations.

1. The term "response function" may be applied to any function of energy which is multiplied by the flux density $\phi(\underline{r},E)$ and integrated over energy to produce integral quantities to be used for radiation protection or other purposes.
2. The result of such an integration produces a detector response, kerma, exposure, or a similar quantity. To calculate dose (absorbed dose, dose equivalent), other considerations are necessary.
3. The term dose is a general term denoting the quantity of radiation or energy absorbed (see p.53, Appendix III).
4. The term "first collision" dose should be dropped. Normally, kerma can be substituted but it should be noted that kerma can be calculated for positions in phantoms, mammals, etc. We further recommend that when kerma is actually evaluated this term should be used and should not be called dose.
5. The quantity measured by fast neutron dosimeters, such as the Hurst type, is kerma since such instruments do not respond to secondary gamma rays.

6. Normally, for purposes of determining the magnitude of a gamma-ray field (below about 10 MeV), the quantity evaluated should be exposure.

7. Because of the models often used in gamma-ray transport calculations, the following terms are proposed:

   (a) *Mass Absorption Coefficient* (μ_a/ρ)-
   Consists of the photoelectric and pair-production mass attenuation coefficients and the average energy deposition fraction of the Compton mass attenuation coefficient,

   (b) *Modified Mass Absorption Coefficient* (μ_{am}/ρ) -
   Same as μ_a/ρ except it does not include the fraction of energy carried off by annihilation radiation,

   (c) *Exposure Buildup Factor* -
   Replaces what has been known as dose buildup factor (based on energy absorption in air),

   (d) *Energy-Deposition Buildup Factor* -
   Replaces what has been known as energy-absorption buildup factor,

   (e) *Energy Fluence Buildup Factor* -
   Replaces what has been known as energy buildup factor.

8. The term buildup factor normally should be reserved for electro-magnetic radiation in simple cases where the buildup factor is a function of distance from the source or from plane boundaries.

FOREWORD

The staff of the Radiation Shielding Information Center has become increasingly concerned about the chaotic use of radiation quantities and units in the reporting of calculations and measurements of radiation transport. In an effort to formulate a course of action which might alleviate this problem, a working committee was assembled at Oak Ridge in April of 1966 to discuss the matter. One of the results of the meeting was a recommendation* to publish an RSIC topical report discussing the ICRU recommendations concerning various radiation units as a service to the shielding community. The present report is an effort to fulfill that recommendation. It has been reviewed, but not necessarily wholly endorsed, by those present at that meeting. The attendees were:

N. M. Schaeffer	- Radiation Research Associates
G. H. Anno	- McDonnell-Douglas
W. E. Edwards	- General Electric, Missile and Space Division
S. T. Friedman	- Westinghouse Astronuclear
H. H. Rossi	- Columbia University
A. B. Chilton	- University of Illinois
H. O. Wyckoff	- Armed Forces Radiobiology Research Institute
D. K. Trubey	- Oak Ridge National Laboratory (RSIC)
F. H. Clark	- Oak Ridge National Laboratory
S. K. Penny	- Oak Ridge National Laboratory
F. C. Maienschein	- Oak Ridge National Laboratory

*F. H. Clark, Report of a Meeting of a Working Committee on Shielding Convention, ORNL-TM-1604, (Nov. 8, 1966).

This report has also been reviewed by members of the American Nuclear Society Standards Subcommittee on Shielding (ANS-6), N. M. Schaeffer, chairman. The 1967-1968 members include:

S. T. Friedman	- Westinghouse Astronuclear
H. J. Donnert	- Kansas State University
R. J. Klotz	- Combustion Engineering
A. E. Profio	- Gulf General Atomic
H. C. Claiborne	- Oak Ridge National Laboratory
D. K. Trubey	- Oak Ridge National Laboratory (RSIC)

Other reviewers who have made contributions are:

F. H. Attix	- U. S. Naval Research Laboratory
L. G. Haggmark	- U. S. Naval Radiological Defense Laboratory
C. M. Huddleston	- U. S. Naval Radiological Defense Laboratory
W. R. Johnson	- University of Virginia
K. O'Brien	- USAEC Health and Safety Laboratory
J. A. Auxier	- Oak Ridge National Laboratory

The editorial "we" is used throughout but the author only is responsible for the recommendations and statements made. In many cases there was a consensus; where there was none, we have tried to indicate this fact.

There are two possible philosophies one can adopt in making recommendations. One is to take the majority position which may not be the best usage logically. The other is to adopt what seems logical even if it is a sharp break with tradition. We have chosen sometimes one philosophy and sometimes the other.

We recognize that this is an exercise in social action but feel that

RSIC should take leadership in promoting clear communication. We find difficulties enough in communication without those due to the use of non-standard word usage and ill-defined concepts. We sincerely hope this report will be a contribution to better communication. If not, perhaps a useful purpose will be served if the readers are stimulated to think about these matters once again.

USE OF ICRU DEFINED QUANTITIES AND UNITS IN SHIELDING

D. K. Trubey

INTRODUCTION

To quote from H. H. Rossi[1] - "It is now about (2)* years since the recommendations on quantities and units by the International Commission on Radiological Units and Measurements (ICRU) have been published. Although there appears to have been little overt criticism, it is evident that these recommendations have often been ignored or that their implications have not been appreciated." In the time period since Rossi made the statement very little increase in the use of the defined units has been noted, especially in the shielding community. This community, somewhat distinct from the health physics community, is mostly composed of physicists and engineers primarily concerned with radiation transport through shields. They are concerned only incidentally with dosimetry and hence are generally only vaguely aware of the ICRU recommendations. It is apparent, however, that an understanding of these recommendations and a consistency in the use of dosimetry terms are long overdue. Consequently, this report was prepared and issued in an effort to make the shielding community more aware of the ICRU recommendations and, in the interest of improving communication, to make some recommendations concerning special terms.

---

*This number should be updated to 5 to reflect the interval of time which has passed. It appears that the note is as relevant now as it was in 1965.

THE ICRU

As background concerning the ICRU we quote the following:

The International Commission on Radiological Units and Measurements is one of two permanent commissions operating under the auspices of the International Congress of Radiology. This Commission was first organized in 1925 and has been continuously active ever since. General meetings have been held during each Congress in which recommendations concerning radiological units and standards have been established...

...The International Commission on Radiological Units and Measurements (ICRU), since its inception in 1925, has had the responsibility for developing (1) the basic principles of units, standards, and measurements needed in radiation dosimetry, and (2) the specification of radiation treatment. It has based its activities on a recognition of the need for worldwide uniformity in those fields.

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

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 for current use...

...The ICRU feels it is the responsibility of the 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 units and concepts.

The Commission feels its responsibility lies in developing a system of 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 expedience is inadvisable from a long-term viewpoint; it endeavors to base its decisions on the long-range advantages to be expected...

...The Commission has entered into an official relationship with the World Health Organization (WHO). In this relationship, the ICRU will be looked to for primary guidance in matters of radiation units and measurement, and in turn WHO will undertake the worldwide dissemination of the Commission's recommendations. This relationship does not affect the affiliation of the Commission with the International Congress of Radiology.

Information relations have been developed with the International Organization for Standardization (ISO).

The ICRU invites and welcomes constructive comments and suggestions regarding its recommendations and reports. These may be transmitted to the Chairman.[2]

TYPES OF RADIATION TRANSPORT CALCULATIONS

There are several possible purposes for shield research or design calculations and naturally the final quantity computed will depend on the purpose. Nearly always the ultimate purpose is to estimate biological hazard, radiation damage, or heating, however, and therefore it is likely that something called "dose"* will be evaluated. This is not a basic quantity which characterizes the radiation but rather is a measure of the energy-deposition

---

*The quotation marks indicate the generic nature of this term as generally used. It is not necessarily identical with absorbed dose. (See page 23.)

characteristics in a particular material. There are difficulties in estimating the "dose" but there are advantages. That is, the "dose" from ionizing radiation due to a particular source is generally a simple function, compared to differential quantities, yet has a behavior similar to the dominating components and is, at least roughly, a measure of hazard.

Several of the possible purposes in making a calculation of radiation transport are considered below. (Refer to Appendix II for definitions of terms used here.)

A. Shielding Research

The goals in a research project are generally one or more of the following: development of method, generation of attenuation or similar kinds of data, or comparison of materials and/or specific geometric configurations. Comparison with other calculations or with experiment is desirable and so one must determine the same quantity as that evaluated by others. This will generally include the very basic quantity flux density* as a function of time and position and, optionally, energy and direction. But, because of the large number of variables, very likely most comparisons will be based on integral quantities. The simplest integration is over time and yields the fluence.** Other integrations may yield the fluence integrated over energy or direction. Comparisons between theory and experiment may require integration over the "response function" of an instrument (i.e., the degree of indication per unit fluence or unit flux density as a function of radiation energy.)

---

*We prefer the term "flux density" rather than flux. (See page 25)

**We will tend to use fluence and flux density almost interchangeably since most of our statements will apply equally well to either the time integrated quantities or to the rates.

B. Shield Design

The goal in a shield design calculation is the rapid selection of an economical, "safe" configuration from several possible configurations. When available, data generated by research calculations or measurements are often used. A "safe" shield is one which produces radiation levels below certain prescribed "maximum permissible" levels defined on the basis of hazard. The quantities subject to maximum permissible levels force the design engineer to evaluate attenuation results in certain ways. For example, if the maximum total dose rate specified by a single composite number is the only criterion, the engineer must add the various components and types of radiation. This requires the consideration of biological effect since this is the quantity to be summed. Sometimes, in addition, the maximum dose due to neutrons might be specified. Usually the requirements are based on the maximum permissible dose recommendations of the International Commission on Radiological Protection (ICRP). A discussion of these has recently been published.[3]

C. Obtaining Absorbed-Dose or Dose-Equivalent from the Fluence

It is theoretically possible to incorporate a phantom into the shield configuration but this introduces such great complexity as to generally be impractical. Consequently, for detailed dose determinations it is generally more convenient to utilize the results of separate calculations in which the energy deposition in a man-like phantom has been evaluated in great detail.* A general interpretation can then be made once the relationship between fluence and absorbed dose or dose equivalent for a phantom is established. This relationship is sometimes called a "multi-collision" flux-to-dose rate

*There have been cases where a tissue phantom has been included but the geometry treated has been simple (slabs)[4].

function since repeated collisions in the phantom are considered. Such a function for dose equivalent is based on information concerning the secondary particle tracks and the presumed resulting biological effect. A more complete discussion is given below.

Multi-collision flux-to-dose rate functions for gamma rays have not generally been utilized possibly because multi-collision effects and the energy deposition are less complex. Some depth-dose data for various organs per unit exposure of the surface have been published,[5] however.

RADIATION FIELD CHARACTERIZATION

A physical variable closely related to the potential hazard of a radiation field, and yet relatively easily evaluated once the fluence as a function of energy is determined, is the rate of energy deposition obtained by a multiplication of the energy fluence and the corresponding mass energy absorption coefficient.* Let us consider a small probe in which the primary particles are likely to make no more than one collision but large enough so that the secondary charged particles resulting from primary-particle interactions, are nearly all absorbed. This probe, then, measures the energy-deposition character of the radiation for the particular probe material. Even if the probe is tissue the energy deposition alone is still not sufficient information to predict any particular effect on a biological organism but we will postpone a discussion of these considerations until a later section.

---

*Note that while this term is commonly applied to gamma rays, the NBS Handbook-84 (reproduced as Appendix II) definition applies to all indirectly ionizing particles.

The probe we have described is a "nonperturbing" detector since its small size limits the effect on the primary radiation to negligible proportions. This type of detector, quite practicable in many experimental situations, can be (and generally is) completely nonperturbing in a calculation since the fluence can be calculated first and then the energy absorption estimated in the probe. This type of calculation has often been called a "first-collision" calculation, since multiple collisions of the primary particles in the probe are assumed to be negligible. The term "first-collision dose" can have one of several meanings[6] and consequently we recommend against the use of this term since other better-defined terms are available. For use in calculations, the quantity designed to replace "first collision dose" is kerma--"kinetic energy released in matter." We might note that the kerma can be evaluated anywhere, e.g. in a phantom. Kerma is the energy transferred to charged particles by primary-particle interaction. Under conditions approximating "charged particle equilibrium absorbed dose and dose and kerma have nearly the same magnitude, and in simple geometries (not near boundaries) are proportional and nearly equal. Two effects are generally present to prevent them from being equal. The charged particles from scattering interactions will tend to move "downstream", i.e., in the same direction as the primary particles, and bremsstrahlung losses prevent all the initial kinetic energy from being deposited locally.

To calculate kerma, one simply has to integrate over energy as follows:

$$K(\underline{r}) = \int F(\underline{r},E) \frac{\Sigma_t(E)}{\rho} P_k(E)\, dE \qquad (1)$$

where

$F(\underline{r},E)$ = Unperturbed energy fluence per unit energy at energy E and space point $\underline{r}$;

$\Sigma_t(E)$ = Total cross section of probe material;

$P_k(E)$ = Fraction of energy transferred to charged particles averaged over all reactions and directions of emergent particles in probe material;

ρ = Density of probe material.

Here we extend the ICRU recommendations (NBS Handbook 84 reproduced as Appendix II) and suggest that we can have any response function $P_x(E)$ which can be used to calculate quantities somewhat similar to kerma by defining:

$$R_x(\underline{r}) = \int F(\underline{r},E)\, P_x(E)\, dE \qquad (2)$$

The function $P_x(E)$ can be a function proportional to the response of an instrument or it can be an arbitrary weighting function. It is our opinion that there is a need to speak of quantities R_x (where $P_x(E)$ is not necessarily precisely defined) in a loose and conversational sense and also in a more precise sense when $P_x(E)$ is defined in a more exact way. R_x might be termed the "response" or "conversion quantity". An example of such a quantity would be the activation of certain nuclides employed as detectors. The word dose is often used in this somewhat general way to mean energy absorbed (see USA Standards Institute definition in Appendix III) and we feel that such usage is acceptable but there is no consensus here.

An important point we would like to emphasize is that kerma and R_x

can be calculated if one knows the fluence as a function of energy at point r regardless where r is located. No other information is needed, not even the angular distribution or geometry data. This is not true for absorbed dose. In a bulk shield or in a large slab of tissue, kerma and absorbed dose may be nearly proportional or even nearly equal for practical shield design purposes. However, in thin layers or near boundaries, such as in clothing or skin, they can be quite different. To calculate absorbed dose, one must know where the secondary radiation deposits its energy; for kerma, one doesn't need such information. This means there is a real difference in concept. One can calculate kerma by Eq. (1), but strictly speaking, one cannot do so for dose. The kerma is a measure of the radiation field for indirectly ionizing radiation. In analagous fashion the absorbed dose is a measure of the charged particle field.* The difference between the numerical value of absorbed dose and that of kerma is due to the lack of "equilibrium" between the charged particle and the uncharged patricle fields.

On the other hand, strictly speaking, one can never measure kerma directly. One can argue that when one wants to make a direct measurement, it must be a measurement of absorbed dose. The "single collision" or "first collision" concept in the past included what is the kerma concept as well as that of *absorbed dose in a mass of tissue sufficiently small that the probability of the indirectly ionizing radiation undergoing more than one interaction is negligible, but sufficiently large that ionic equilibrium exists*. For this second case, the ICRU did not recommend a new unit.** Rigorously speaking, the responses of ionization chambers, proportional counters, etc., are related directly to absorbed energy, and

*This was pointed out to the author by H. O. Wyckoff.

**This was pointed out to the author by J. A. Auxier.

the imparted or released energy can be infered only under the assumption that secondary equilibrium exists. One can also argue that this is an indirect measurement of kerma. We take the view that the Hurst-type fast neutron dosimeter, often used in shielding research, measures kerma since the secondary gamma rays are not included in the response. (See pp 22, and 37). We recommend that the quantity reported be called "tissue kerma for fast neutrons." As suggested above, one could also call the quantity "dose in the counter" but this is not the basic quantity of interest.

DOSE EQUIVALENT AND MULTICOLLISION CONSIDERATIONS

The biological effects of a given absorbed dose is assumed to depend solely on the linear energy transfer* (LET) of the charged particles that deliver it. The LET is the average energy locally imparted to the medium per unit length of particle travel. The dose equivalent (in rem) is the product of absorbed dose and the LET dependent quality factor (QF) and, perhaps other factors. The most satisfactory determination of the dose equivalent would involve a complete computation of fluence, absorbed dose, and dose equivalent in a system containing source, shield, and human phantom. As mentioned earlier, this is an ideal with which we must usually compromise. The most common approach is to use a "multi-collision" flux-to-dose equivalent rate function $P_x(E)$, which was obtained from a detailed calculation of energy deposition due to monoenergetic primary particles impinging on a standard phantom such as a slab or cylinder of tissue. Here we use the term "multi-collision" to differentiate the case where the phantom is present from the "free field" or "small probe in air" case. The values of $P_x(E)$ are selected on the basis of some criteria such as

*See NBS Handbook 84, p. 51 in Appendix II.

peak dose equivalent in the phantom. This is what was done to produce the much used Snyder-Neufeld neutron flux-to-dose rate factors.[7] The rationale in simply applying the Snyder-Neufeld factors for any impinging angular distribution is that the dose rate inside a thick slab of tissue but near the surface, is proportional to the impinging flux density. However, this is true of kerma only, since kerma is proportional to the flux density but in-so-far as dose equivalent and kerma are proportional it will also be true of dose equivalent. One can approximately account for multi-collision effects by applying the ratio shown in Fig. 1 taken from Ref. 7. The quantity shown is the ratio of the maximum absorbed dose in a tissue slab to the free-field kerma. The ratio of the dose equivalent to the kerma multiplied by quality factor is also shown. The ratios are defined on the figure in the older nomenclature. The departure of the ratio from 1 is due mainly to the capture gamma rays originating throughout the tissue slab. It can be seen that the ratio assumes large values for neutrons of low energy.

If one is interested in depth-dose, and one must be if he is interested in the dose to interior organs, he must take incident particle angular distribution into account. If one looks at the average dose in a 30-cm thick slab, one finds this is approximately proportional to the impinging current since there is very little leakage, that is, reflection or transmission. Thus a recent study[8] reports neutron "current-to-dose rate" factors. These were determined by Monte Carlo calculations for a slab of tissue and may be applied as follows.

Let:

$D_f(E,\mu,z)$ = absorbed dose or dose equivalent at a depth z in a slab of tissue per incident neutron of energy E_o and incident angle $\cos^{-1}\mu$ per unit area normal to the direction of travel.

ORNL DWG. 68-6589

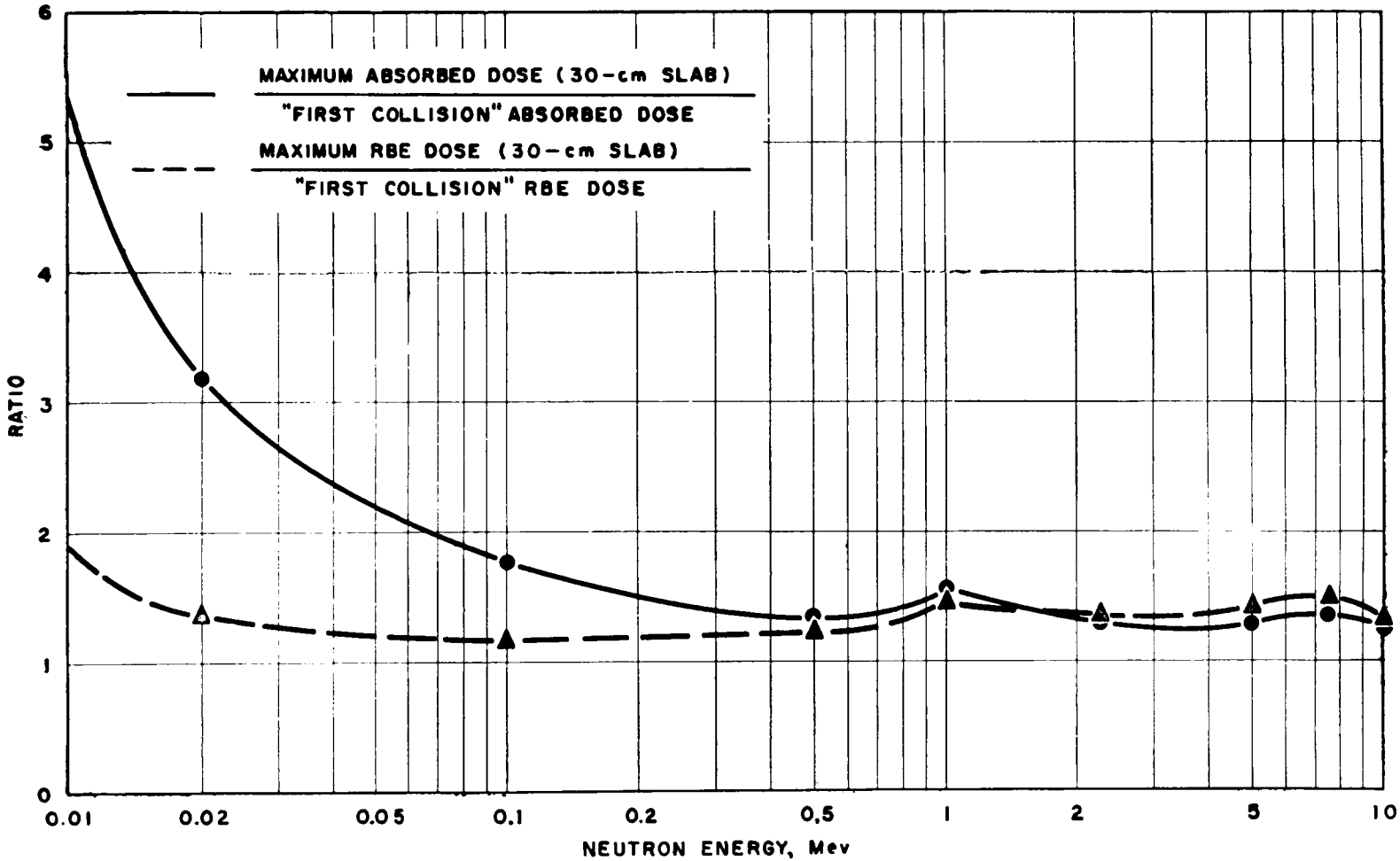

Fig. 1. Dose in Tissue Slab to Free Field Tissue Kerma as a Function of Energy

$D_j(E_o,\mu, z)$ = same quantity as $D_f(E_o,\mu,z)$ except normalized per unit area on the surface

$= D_f(E_o,\mu,z)/\mu$

We will call $D_f(E_o,\mu,z)$ fluence-to-dose or flux density-to-dose rate factors and $D_j(E_o,\mu,z)$ current-to-dose-rate factors. The dose at a depth z is then given by an integration over incident energy and direction:

$$D(z) = \int_{E_o} \int_0^1 F(E_o,\mu)\, D_f(E_o,\mu,z)\, 2\pi\, d\mu\, dE_o \tag{3a}$$

$$= \int_{E_o} \int_0^1 F(E_o,\mu)\mu\, D_j(E_o,\mu,z)\, 2\pi\, d\mu\, dE_o \tag{3b}$$

Note that $F(E_o,\mu)\mu$ is the incident current. The integration is over 2π since it is assumed that the neutrons are incident on the slab from only one side.

These fluence-to-dose factors are different in several significant ways from the simple function of particle energy needed to evaluate kerma.

(1) The functions $D_f(E_o,\mu,z)$ and $D_j(E_o,\mu,z)$ are really kernels for the specification of dose at z due to incident neutrons of a certain initial energy and direction. The transport of the incident particles is accounted for and the final energy deposition is evaluated. These functions are useful for evaluation of depth dose and can include all types of secondary reactions and secondary particle transport. On the other hand, the kerma function evaluates energy transferred to the secondary charged particles at any space point from primary particles of a given energy regardless of their initial energy and space point.

(2) The functions $D_f(E_0,\mu,z)$ and $D_j(E_o,\mu,z)$ are evaluated for a particular phantom composition and geometry.

(3) These functions depend on incident neutron direction while kerma depends on the flux density at the particular space point integrated over all directions.

Such functions have also been reported for high energy electrons and photons for normal incidence.[9]

In addition to the kinetic energy from the primary particles, the multi-collision function $P_x(E)$ includes dose due to secondary radiation such as neutron capture gamma rays and protons from nitrogen (n,p) reactions in tissue. These contributions overshadow the kinetic energy deposition of low-energy neutrons and introduces a geometry effect dependent on gamma-ray transport properties. Thus the fact that a man is not a slab may be significant and affect the values of the function $P_x(E)$.

Additional discussion of dosimetry principles is given in Ref. 10.

RESPONSE FUNCTIONS FOR THE INTERIOR OF SHIELDS

A multi-collision function based on dose in a phantom is not the appropriate one to characterize neutron penetration throughout a shield as is sometimes done for intercomparison of transport calculations. That is, the multi-collision energy deposition in tissue, with its large (n,γ) and (n,p) components is not of great relevance at, for example, a pressure vessel-shield interface. There have been cases in which there was a puzzling discrepancy between results of calculations which was finally traced to differences in the low energy cutoff. The low energy cutoff can markedly affect the contribution of the reactions mentioned above. This might mask differences

in the high energy components which are likely to be the quantities of interest when comparing shielding calculational methods. Probably kerma is the quantity of choice in this application.

DISCUSSION OF ICRU DEFINITIONS

The reader is urged to refer to NBS Handbook 84 (Appendix II) as necessary since our discussion is meant to supplement the definitions given there, with the hope of clarifying some of them in the shield-design context. We do not intend to modify them but enlarge somewhat on the discussion in Handbook 84. Further discussions are also given in Appendix II and Appendix I

Absorbed Dose

The absorbed dose for a material is almost equal to the heat appearing in the material. The usual unit is rad or erg/g.

To calculate this quantity one must determine where the secondary radiation deposits its energy. For example, in a thin layer most of the charged particles escape and so the absorbed dose is not a function of the primary particle flux density only. The total absorbed dose in a phantom includes the energy due to secondary gamma rays and hence is dependent on the phantom geometry and composition.

Dose Equivalent

This quantity, based on secondary particle LET dependent factors, is needed for radiation protection limits. The unit is the rem and it is presumed that the dose equivalent for different kinds of radiation is additive. The dose equivalent (DE) is the product of absorbed dose, D_a, quality factor (QF), dose distribution factor (DF), and other necessary modifying factors. These factors replace the Relative Biological Effectiveness (RBE) factor, used formerly in this context.

---

*For a discussion of RBE see Report of the RBE Committee to the International Commissions on Radiological Protection and on Radiological Units and Measurements, Health Physics, 9, 357-386 (1963).

The values of the neutron and proton quality factors, as a function of energy, as recommended by the ICRP, are shown in Fig. 2 which was taken from Ref. 3 based on data presented in Ref. 7. When the lens of the eye is irradiated, an additional modifying factor is needed. The recommended value of this factor is 3 when the value of QF is 10 or more, and 1 when the value of QF is 1. Where doses in an organ are non-uniform, some further modifying factors are needed. These factors cannot yet be established with certainty and only one is in current use. This is the relative damage factor, applied in calculating the dose equivalent in bone from internal radiation.

All the quality factors and other modifying factors are intended solely for radiation protection at levels of dose limits recommended by ICRP. In practice, they can often be usefully applied to substantially higher doses. but the resultant dose equivalent will not necessarily be an appropriate indication of the likelihood of subsequent biological response. High accidental exposures, therefore, have to be assessed on their merits. ICRP gives no recommendation, but in many cases the absorbed dose will give a better indication of the biological risks than will the dose equivalent based on Fig. 2 In other words, for acute effects due to massive exposures, the QF is approximately 1.

Dose

This term is meant by the ICRU to mean absorbed dose only. However, usage is so ingrained we recommend that it may be used in a generic sense to include exposure and the quantities incorporating the word dose in Appendix III. There is a need for such a generic term in informal speech and when it is desired to be non-specific. Needless to say, such usage is controversial. The USA Standards Institute definition is given in Appendix III.

ORNL DWG. 68-6506

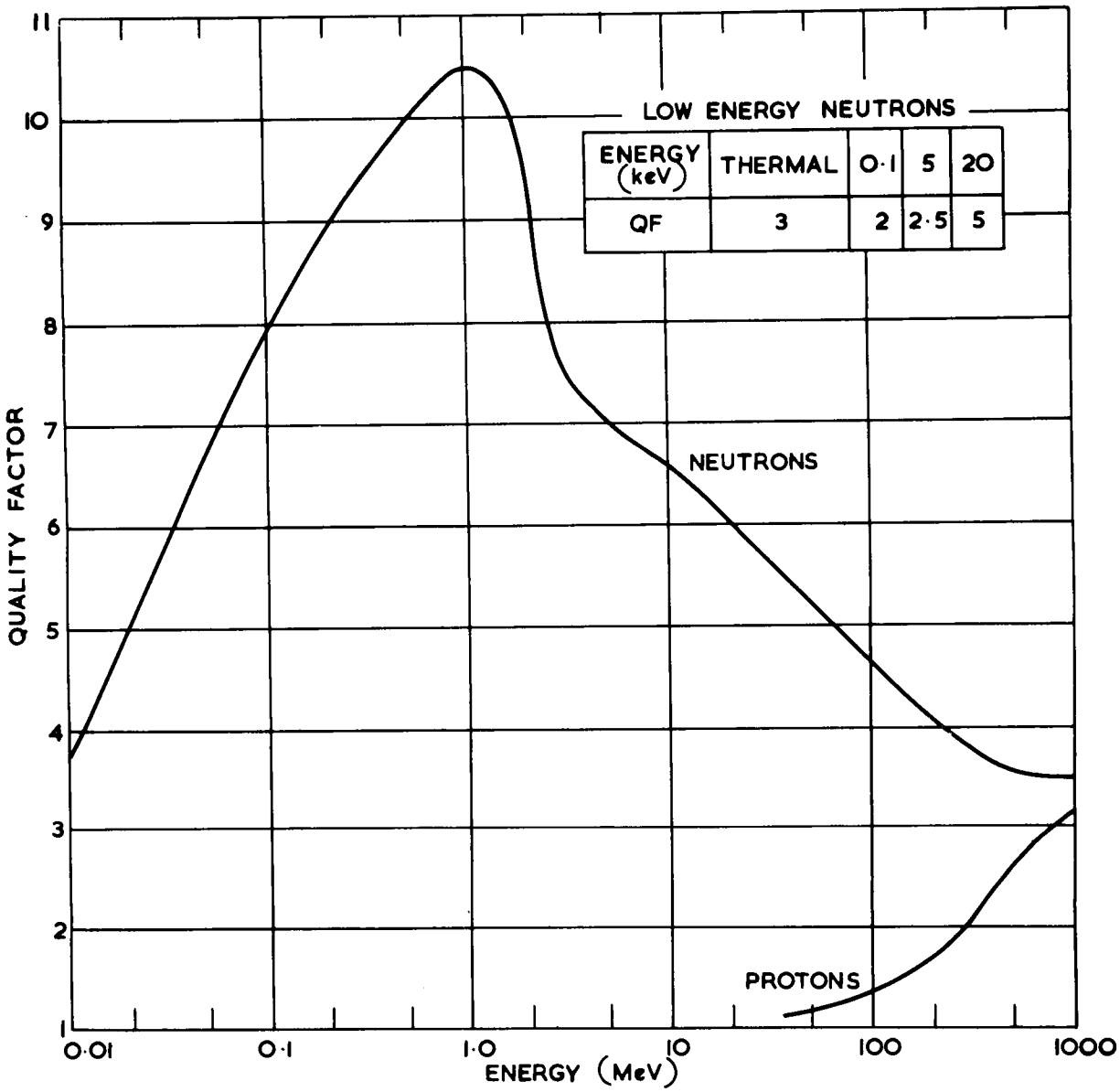

Fig. 2. Quality Factor for Neutrons and Protons

Flux Density versus Flux

We think it is an improvement in nomenclature to acknowledge that this quantity is a density. It is a point function and may be defined as the limit of the total particle track length per unit time per unit volume as the volume approaches zero. The flux density may also be given as a differential in energy or solid angle. There is no consensus on this usage so we are not making a strong recommendation.

Kerma

This is probably the least ambiguous and most easily calculated quantity and so merits use as an index to compare shielding calculational methods for fast neutrons in the low MeV range or gamma rays below a few MeV where a photonuclear cascade is not possible. However, neutron kerma alone may not be adequate to evaluate safety of shields since the neutron hazard is not only dependent on the kinetic energy imparted to the secondary charged particles but also on (n,γ) reactions which are important after the neutron has slowed to the point where the kinetic energy has become insignificant. We might note that, as we proceed from high energies downward, the kerma in tissue decreases but reaches a minimum at about 40 ev. It then increases as we proceed further which is due mostly to (n,p) reactions (see Fig. 3 taken from Ref. 11).

For most reactor and other low MeV radiation shield design purposes, the total neutron and gamma-ray kerma may be considered equivalent to heating since the final deposition of the energy is not far from the site of the energy transfer to the charged particles. The integral of the kerma over all space will equal the primary particle energy only if the secondary particle bremsstrahlung production is neglected which is the case in many calculations.

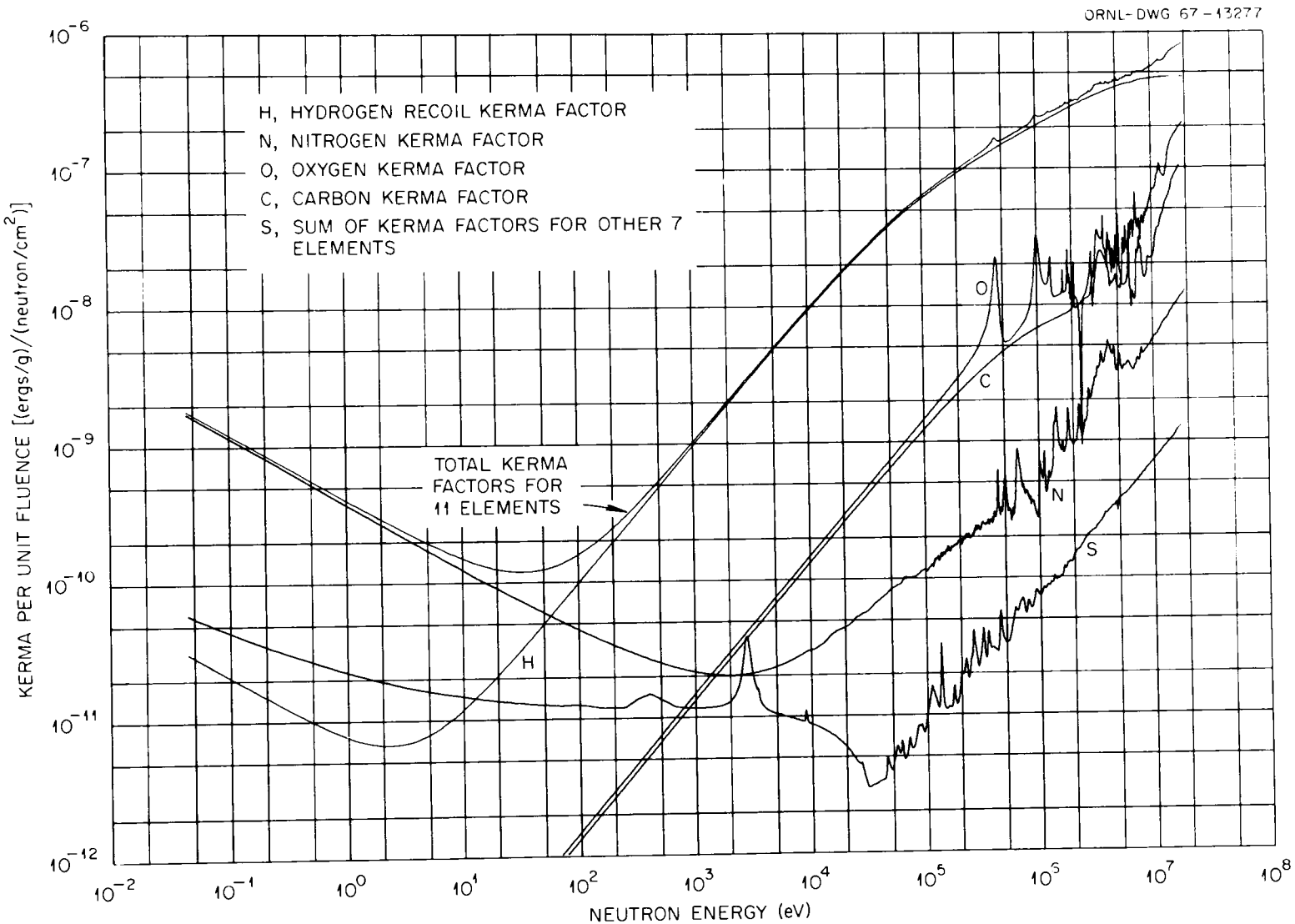

Fig. 3. Tissue Kerma Per Unit Neutron Fluence as a Function of Energy

Exposure (Gamma or X Radiation Only)

This definition appears to be a concession to historical measurements of ionization in air but is probably the best integral quantity to specify the magnitude of a gamma-ray field at least below about 10 MeV. Exposure is a different kind of quantity from those discussed above and has a different unit: charge per unit mass. The term is limited to use with electromagnetic radiation and the usual unit is roentgen (R). Under conditions of charged particle equilibrium, absorbed dose in air is readily converted to exposure. The conversion from energy deposition to exposure can be taken to be 87.7 erg/g per R up to several MeV.

No similar quantity has been defined for neutrons. For specifiying a neutron field in the reactor and weapons context, at least three components must be evaluated: fast, intermediate, and thermal.

Conceptual Difficulties with Simple Models

In proceeding to calculate quantities such as kerma or exposure, one generally first computes the fluence or flux density and then applies Eq. (2) using an appropriate response function, $P_x(E)$. For exposure, the appropriate function is μ_{en} (see p. 51 in Appendix II) and for kerma, μ_k. Handbook 84 states that the difference between μ_{en} and μ_k is the factor 1-G where G is the proportion of the energy of secondary charged particles that is lost to bremsstrahlung in the material. But suppose, as in the usual gamma-ray shielding calculation, that the secondary-produced bremmstrahlung, and the physical processes underlying it, are ignored in the computational model used. Is the quantity calculated to be called kerma or exposure? With such a model, the distinction between μ_{en} and μ_k has little relevance.

Let us consider the simple model used by Goldstein and Wilkins[12] in their calculation of gamma-ray penetration. The only processes considered were (1) the photoelectric effect (taken to be absorption), (2) pair-production (also taken to be absorption), and (3) Compton scattering. Consequently, the calculated flux density did not contain several elements present in more complicated models such as fluorescent radiation, annihilation radiation, and secondary bremmstrahlung. As a result, the flux density is lower than it would be with these components present. The effective absorption cross section is higher, however, since any collision but Compton scattering results in absorption.

It is always satisfying to write equations showing conservation of energy and so for the point source, infinite homogeneous medium, let us set the energy of a unit source equal to that absorbed (integrated over all space),

$$1 = \int_{\underline{r}} \int_{E} F(\underline{r},E) \; P_a(E) \; dE \; d\underline{r} \tag{4}$$

where $P_a(E)$ = absorption cross section. We would write a similar equation for the more complicated model but the values of $F(\underline{r},E)$ and $P_a(E)$ would be different. However, their <u>product</u> should be nearly the same. The point we wish to make is that the rational choice of $P_a(E)$ depends on the model used to calculate $F(\underline{r},E)$. That is, one should not take the values of $F(\underline{r},E)$ from NYO-3075 and use $\mu_{en}(E)$ for $P_a(E)$. Therefore, we propose the following quantities,* in addition to μ_{en} and μ_k. All of them are considered to be *energy-deposition coefficients*.

---

* These terms were suggested by A. B. Chilton

A. Mass Absorption Coefficient (μ_a/p)

This is defined for gamma rays by

$$\mu_a/\rho = \mu/\rho - \mu_s/\rho - \mu_{coh}/\rho \tag{5a}$$

$$= \tau/\rho + \sigma_a/\rho + \gamma/\rho \tag{5b}$$

where

μ = Total absorption coefficient;

μ_s = Compton cross section multiplied by the average fraction of energy retained by the scattered photon;

σ_a = Compton cross section multiplied by the average fraction of energy acquired by the electron;

τ/ρ = photoelectric mass attenuation coefficient;

γ/ρ = pair-production mass attenuation coefficient;

μ_{coh}/ρ = coherent scattering mass attenuation coefficient.

The factor ρ (density) is carried along to indicate that each of the quantities desired are normalized per unit mass. This is the absorption cross section associated with the model which assumes the photoelectric effect and pair-production process are purely absorption processes.

B. Modified Mass Absorption Coefficient (μ_{am}/ρ)

This is defined for gamma rays by

$$\mu_{am}/\rho = \tau/\rho + \sigma_a/\rho + \gamma_a/\rho \tag{6}$$

where

$$\gamma_a/\rho = \frac{\gamma}{\rho} \left(1 - \frac{2m_o c^2}{h\nu}\right)$$

This is the absorption cross section associated with the model which assumes the photoelectric process is absorption but pair production is a scattering process.

BUILDUP FACTORS

The paper by Goldstein and Wilkins[12] is the primary source of data concerning gamma-ray penetration. They reported energy fluence and dose functions for monoenergetic, point isotropic, and monodirectional plane sources. The attenuation functions were reported as buildup factors as follows:

$$B(r) = \frac{\int F(r,E)\ P_{\chi}(E)\ dE}{\int F_o(r,E)\ P_{\chi}(E)\ dE} \tag{7}$$

where

$F(r,E)$ is the energy fluence at a distance r from the source and in dE about energy E.

$P_{\chi}(E) = \mu_a/\rho$

and

$F_o(r,E)$ is the uncollided energy fluence.

The model used to calculate the fluence was the one described above in the section defining μ_a/ρ. Consequently, Goldstein and Wilkins used μ_a/ρ (air) for $P_{\chi}(E)$ in computing what they called the "dose" build-up factor and μ_a/ρ for the particular medium in computing what they called the "energy absorption" buildup factor. Note that the buildup factor calculated using μ_a/ρ for air applies equally well for the calculation of exposure or kerma in air since bremsstrahlung is not considered in the model and constants of

proportionality cancel in Eq. (7). They also set $P_\chi = 1$ to compute an "energy" buildup factor.

In connection with the choice of $P_\chi(E)$ in Eq. (7), care must be taken in multiplying the uncollided fluence in the application of published buildup factors. The final result must cancel the denominator to give the correct result.

In light of the above, we feel that these buildup factors should be assigned new names. This is particularly true of the "dose" buildup factor. We recommend that "exposure" buildup factor be used to identify "dose" buildup factor. Likewise, we recommend "energy-deposition" buildup factor instead of "energy-absorption" buildup factor. This associates it with μ_a instead of strictly μ_{en}. The "energy" buildup factor naturally becomes "energy fluence" buildup factor.

The buildup factor has normally been used to represent the point-to-point infinite medium kernel and strictly applies only in these circumstances. The data are often used in other, but reasonably similar, geometries to give quick and sufficiently accurate results. However, the term has also been used in other very complicated situations where the concept breaks down. We recommend that the term buildup factor be used only when the uncollided flux density tends to dominate the penetration; that is, the quantity desired can be obtained by multiplying the uncollided flux density by a simple function of distance from the source or from a plane boundary such as an air-ground interface.

BUILDUP FACTORS FOR NEUTRONS

Sporadic references to buildup factors for neutrons have been made in

the literature but the practice has not become common. This is probably due to the fact that the neutron total cross section (and consequently the build-up factor) varies so radically with energy and nuclide. We believe a better way of reporting data is to give the attenuation function averaged over a source energy band (such as the fission spectrum). See, for example, the report of Krumbein.[13] Consequently, we recommend the use of the term buildup factor in the reactor shielding energy range be reserved for gamma rays. There is no consensus here but we will say to those who must use a "buildup factor" for neutrons, please adequately define what is meant.

In the space and accelerator shielding energy range, the secondary reactions and cascades are so complex and the transport of secondaries so important that the use of buildup factors for particulate radiation is more common. Here, again, however, we recommend explicit definition whenever used.

REFERENCES

1. H. H. Rossi, *Health Phys.*, 11, 779, (1965), reprinted as Appendix I.

2. *NBS Handbook 62*, "Report of the International Commission on Radiological Units and Measurements", (1956).

3. H. J. Dunster, *The Application and Interpretation of ICRP Recommendations in the United Kingdom Atomic Energy Authority*, AHSB (RP), R 78, (June, 1967).

4. W. W. Scott and R. G. Alsmiller, Jr., *Comparisons of Results Obtained with Several Proton Penetration Codes*, ORNL-RSIC-17, (July, 1967).

5. A. R. Jones, "Proposed Calibration Factors for Various Dosimeters at Different Energies", *Health Phys.*, 12, 663-671, (1966).

6. Frank H. Attix, "The Meaning of 'First Collision Dose'", *Health Phys.*, 12, (6), 793-799, (1966).

7. *NBS Handbook 63*, "Protection Against Neutron Radiation up to 30 Million Electron Volts", p. 62, (1967).

8. D. C. Irving, R. G. Alsmiller, Jr., *Tissue Current-to-Dose Conversion Factors for Neutrons with Energies from 0.5 to 60 MeV*, ORNL-4032 and Nucl. Inst. and Methods, 51, 129, (1967).

9. R. G. Alsmiller, Jr. and H. S. Moran, "Dose Rate from High-Energy Electrons and Photons", *Nucl. Inst. and Methods*, 58, 343-344, (1968).

10. F. H. Attix and W. C. Roesch (Editors), *Radiation Dosimetry*, Sec. Ed., Vol. 1, Fundamentals, Academic Press, (1968).

11. J. J. Ritts, M. Solomito, and P. N. Stevens, *Calculations of Neutron Fluence-to-Kerma Factors for the Human Body*, ORNL-TM-2079, (January, 1968).

12. H. Goldstein and J. E. Wilkins, Jr., *Calculations of Penetration of Gamma Rays*, NYO-3075, (1954).

13. A. D. Krumbein, *Summary of NDA Unclassified Results of Moments Calculations for the Penetration of Neutrons through Various Materials*, NDA-92-2, (Rev), (April, 1958).

APPENDIX I

A NOTE BY H. H. ROSSI ON ABSORBED DOSE VERSUS KERMA*

It is now about 2 years since the recommendations on quantities and units by the International Commission on Radiological Units and Measurements (ICRU) have been published. Although there appears to have been little overt criticism it is evident that these recommendations have often been ignored or that their implications have not been appreciated. Thus, terms such as "rad dose" and "tissue rad" are still being employed despite the fact that they are as redundant or meaningless as are "ampere current" or "lead pound". Although such usage was incorrect even before the release of the most recent ICRU recommendations, it is now particularly inappropriate since a coherent set of quantities and units is now available which permits rather concise terminology. Adherence to such a system is not a mere formality and it is in fact apparent that inaccurate terminology has lead to an imperfect understanding of the proper objectives of radiation measurement. In particular there appears to be considerable confusion concerning the quantities "absorbed dose" and "kerma".

Omitting a number of qualifications (which although important are not germane to this discussion) the absorbed dose is the energy imparted by charged particles per unit mass of material at the point of interest. If the point is in free space in the environs of the radiation source, the dose is the energy deposited per unit mass of air. If the point of interest is the skin of a radiation worker,it is the energy imparted by

---

*Reprinted with permission from Dr. H. H. Rossi, Health Physics, 11, 779, (1965), Pergamon Press.

charged particles per unit mass of skin tissue. However, the particles that deliver the dose will usually originate at a location other than the point of interest. Thus if the radiation worker is exposed to gamma radiation the dose at some point in the skin may depend more on the interaction of gamma radiation with the clothing that overlies the skin rather than on its interaction with the skin itself. In general the absorbed dose at any point in or near a region traversed by indirectly ionizing particles (neutrons, photons, etc.) depends not only on the incident radiation but also on the atomic number and distribution of all materials that are within a maximum charged particle range from the point of interest. It is, therefore, meaningless to speak of a dose without specifying the point at which this dose figure is applicable, and when the point is specified, reference to the material at that point becomes redundant. Efforts to measure a "tissue dose" at some point in air as delivered by a radiation source can be likened to a procedure in which a thermometer is placed near a source of heat in a measurement designed to measure "tissue temperature in air."

What is usually meant by the expression "tissue dose" of indirectly ionizing particles when applied to some region in air is what has often been termed "first collision dose" (sometimes the "single collision dose"). The concept has been variously defined but a common interpretation is that it represents the "dose received by a small mass of tissue under conditions of radiation equilibrium." Radiation equilibrium exists in an irradiated mass of material when the energy imparted per unit mass is equal to the energy liberated per unit mass in the form of charged particles. In general this requires that the mass have linear dimensions in excess

of two maximum charged particle ranges--a requirement which can lead to significant absorption or scattering of the indirectly ionizing particles that produce the direct ionizing particles. Under such conditions it is impossible to establish true radiation equilibrium. In fact exact radiation equilibrium never occurs except in an infinite medium containing a uniform concentration of an emitter of indirectly ionizing particles. Because of this and because of the fact that in free air measurements it is the intent to characterize the radiation field in terms of its potential interaction with a specified material a better and less ambiguous quantity is the kerma. This has in essence been defined as the sum of the initial kinetic energies of all charged particles liberated by indirectly ionizing particles per unit mass of a specified material. It is evident that one may define the kerma for any material at any point in a radiation field regardless of the ambient medium and that the concept of the kerma for a given material has a real meaning even for a point located in a vacuum.

Once this concept is adopted and the ICRU recommendation that the unit rad is not to be used for the kerma is also accepted it follows that the results of area monitoring which used to be expressed as tissue dose (or first collision tissue dose) in rads should now be given in terms of kerma for tissue in multiples of ergs per gram. The difference in physical quantity and unit serves to underline the fact that while the kerma (or kerma rate) may be useful for the specification of the output of a radiation source, it is not necessarily equal to the dose that would be received in tissue at the location at which the kerma measurement is carried out. Depending on absorption, scattering and build-up this dose could be considerably smaller or considerably larger than kerma.

The question as to what is being "measured" by a radiation instrument has an obvious answer once these concepts are clarified. Thus, a tissue-equivalent ionization chamber located inside a phantom may be used to derive the absorbed dose provided it fulfills the requirements for a Bragg-Gray cavity. On the other hand, if the radiation is such that one may establish a good approximation to radiation equilibrium by the employment of a wall that is just thicker than the maximum range of secondary charged particles, one may utilize a chamber having such a wall to determine the kerma for tissue in any medium (including free air). However, if it is impossible to obtain resonable approximation to radiation equilibrium or if the chamber is too bulky it may not be employed for a determination of the kerma and all that one can derive from its response is the mean dose to the tissue equivalent gas inside the chamber.

It is possible to obtain from the activation levels of threshold detectors a histogram of neutron fluence vs. neutron energy. This in turn may sometimes be used to calculate the kerma in tissue at the location at which these detectors were placed. Such detectors cannot be used to determine tissue dose unless they are located in tissue at a point where radiation equilibrium exists and unless their introduction does not appreciably disturb the radiation field. Analogous considerations apply to other instruments or devices.

APPENDIX II

UNITED STATES DEPARTMENT OF COMMERCE • Luther H. Hodges, *Secretary*

NATIONAL BUREAU OF STANDARDS • A. V. Astin, *Director*

Radiation Quantities and Units

International Commission on Radiological Units and Measurements (ICRU) Report 10a

1962

National Bureau of Standards Handbook 84

Issued November 14, 1962

Reprinted with corrections, October 1964

(This publication supersedes parts of H78. Handbooks 84 through 89 extend and largely replace H78. For an explanation, see the Foreword. Also, for a list of these titles, see rage 3 of cover.)

For sale by the Superintendent of Documents, U.S. Government Printing Office, Washington 25, D.C.
Price 20 cents

Foreword

The reports of The International Commission on Radiological Units and Measurements for a number of years have been published by the National Bureau of Standards in the Handbook series. In the past, each of the triennial reports of the ICRU represented a complete restatement of the recommendations of the Commission. Because of the increasing scope of its activities, however, the Commission in 1962 decided to modify the previous practice. It will issue a series of reports presenting the current recommendations of the Commission. Each report will cover a particular portion of the area of interest to the ICRU. This procedure will facilitate revision of ICRU recommendations and also spread out in time the workload of the Commission. This Handbook is one of the new series presenting the recommendations of the Commission on one aspect of the field with which the Commission is concerned. It presents recommendations agreed upon at the meeting of the Commission held in Montreux, Switzerland, in April 1962.

The National Bureau of Standards is pleased with its continuing opportunity of increasing the usefulness of these important reports by providing the publication outlet.

A. V. Astin, *Director*.

Preface

A. Scope

The International Commission on Radiological 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 recommendations on radiation quantities, units and measurements in the field of radiation protection. In this connection, its work is carried out in close cooperation with the International Commission on Radiological Protection.

B. 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 for current use.

Recognizing the confusion that exists in the evaluation of different radiological equipment and materials, the ICRU is studying standard methods of determination of characteristic data for the equipment and materials used in diagnostic and therapeutic radiology. This activity is confined to methods of measurement and does not include the standardization of radiological equipment or parts thereof.

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 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 wishes to encourage radiologists to use the quantity "absorbed dose" more widely, since this is the physical quantity which can be most closely correlated with biological effects.

For x and gamma radiation it is convenient to retain the quantity which was called in the 1956 report "exposure dose". At that time the term "exposure" was preferred by the ICRU but the word "dose" was included at the request of ICRP. This request stemmed from the fact that many legal radiation protection documents contained the word dose together with the unit roentgen. In the 1956 report the term "exposure dose" was introduced for this purpose in spite of the possibility of confusion with the term absorbed dose. As the principal reason for the inclusion of dose in the name seems to have disappeared, the ICRU returns to its 1956 preference, exposure, as the name for the quantity of which the roentgen is the unit. The Commission having reviewed the whole question of quantities and units now recommends that the word dose be eliminated and the term "exposure" be used alone to designate the quantity of which the roentgen is the unit. (See following sub-section 4.D.(13)).

In this report the ICRU recommends with considerable reluctance and some misgivings, the use of the symbol *R* instead of *r* for roentgen. Several recognized international groups working in the field of symbols and nomenclature including the International Council of Scientific Unions have agreed upon the convention that the first letter of abbreviations of units named after individuals should be capitalized. At least one country has already officially adopted the symbol *R* for roentgen. There are many indications that, however unnecessary, this trend will continue and hence the Commission has acceded to the pressure for change. As far as medical radiology is concerned, this change will result more in annoyance than confusion.

Hitherto the definitions and recommendations of the ICRU have been made with little explanation of the philosophy on which they were based. It is recognized that this neglect has given rise to confusion in the past. In the present report the ICRU has tried to repair this omission while continuing to develop the necessary technical information to facilitate the interpretation and application of the recommendations. These sections of the 1962 reports represent a further enlargement over the 1959 report.

The aim of much of the work of the ICRU is to improve the accuracy of the evaluation of absorbed dose in all places of interest in a patient or other objects. For comparisons of biological effects, the absorbed dose should be known as accurately as possible and the limits of accuracy should be estimated.

In 1955 the Commission entered into an official relationship with the World Health Organization (WHO). In this relationship, the ICRU will be

looked to for primary guidance in matters of radiation units and measurements, and in turn WHO will undertake the worldwide dissemination of the Commission's recommendations. In 1960 the ICRU entered into consultative status with the International Atomic Energy Agency (IAEA).

The above relations with other international bodies do not affect the basic affiliation of the Commission with the International Society of Radiology.

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

C. Current Program

A two week meeting of the ICRU was held in Montreux, Switzerland, April 2nd to April 14th 1962. This meeting included the Main Commission and all the Committees that had reports prepared for final approval. Some 70 persons attended. An additional meeting of the Commission and Committee Officers was held in Ottawa from August 21 to August 23, 1962 for the principal purposes of the preparation of the status report for the Xth International Congress of Radiology and the outlining of program objectives for the next several years.

The 1956 report called attention to some basic problems in the establishment of radiation quantities and units. These were the subject of major discussions during and following the 1958 meetings. One of the difficulties is in reconciling the differences between radiation quantities and units as they have evolved over the past thirty years, and the system of quantities and units as used in the physical and engineering sciences. There are obvious divergencies between the two sets. It appeared possible that a complete resolution of this difficulty might necessitate abandonment of the quantities and units as used by the medical profession. This would accomplish a reconciliation between the two systems but it did not appear that it would introduce any substantial improvement in the understanding and utilization of the quantities by the medical profession. Naturally there was considerable hesitancy about taking such a step.

As indicated in the 1959 report, an *ad hoc* committee was set up for the purpose of examining the whole problem of radiation quantities and units. This committee has met three times (Munich, 1959; Geneva, 1960; Copenhagen, 1961). The study represented a moderate amount of compromise of viewpoints but it is believed that the final report now provides a consistent and satisfactory set of definitions for use in the field of radiation measurements. The report as approved by the Main Commission makes up the body of this document.

A meeting of Committee III–B (Clinical Dosimetry) was jointly sponsored and financed by the ICRU, WHO, and the IAEA in Geneva in April 1961. For this meeting a number of additional consultants were invited to attend. Following the general meetings, a closed meeting of the committee membership was held for a period of two days to prepare a draft report.

In addition there were meetings of various tasks groups of the committee on Standards and Measurement of Radiological Exposure in Paris in January 1961 and in London in April and September 1961. The committee on Radiological Dosimetry also held a meeting in April 1961. The ICRU was also represented at a meeting of the Consultative Committee on Ionizing Radiation of the International Committee of Weights and Measures at Sèvres in October 1961.

As noted in the last report, two joint committees had been established between the ICRU and the ICRP. The Joint Committee on RBE has met twice with ICRU participation. The Committee on Methods and Instruments for Radiation Protection has not met.

Upon the request from the United Nations Scientific Committee on the Effects of Atomic Radiations, the ICRU and the ICRP agreed to undertake a second study dealing with the Medical and Physical Parameters in Clinical Dosimetry. This committee met in New York for one week in September 1959 and for a week in Stockholm in June 1960. A report of this study entitled "Exposure of Man to Ionizing Radiation Arising from Medical Procedures with Special Reference to Radiation Induced Diseases, An Inquiry into Methods of Evaluation", was published in Physics in Medicine and Biology, **6,** No. 2, 199 (Taylor and Francis, Ltd., London, England, Oct. 1961)

Reports and recommendations of the ICRU originally designed for medical applications, have come into common use in other fields of science, particularly where "dosimetric" considerations are involved. For this reason the committees have included in their membership some scientists having competence outside of the medical radiology field. Material in the reports is designed to meet physical, biological, and medical requirements wherever possible.

This has introduced a small problem in terminology. The name of the Commission includes the term "radiological". In many European countries the term "radiological" is taken as inclusive of both the physical and biological sciences. In other countries, the United States, for example, "radiological" appears to carry the primary connotation of relationship to medicine. It therefore may be desirable to change the name of the Commission from "Radiological" to "Radiation". It is believed that this would be properly understood by all concerned. The question has been debated by the Commission, but final action is being delayed for future consideration.

D. The Current Series of Reports

Hitherto, the triennial reports of the ICRU have been published in single volumes. However the reports are now becoming too extensive, and in some cases too specialized, to make a single publication practicable. Beginning with this 1962 series, the ICRU reports will be issued in smaller entities, each dealing with a limited range of topics. The 1959 report will not be reprinted. Revisions of the 1962 series will be undertaken individually as circumstances warrant. A full listing of ICRU recommendations, including the present series, is given on page iii of the cover of this report.

The current report series includes revisions of much of the material that appeared in the 1959 report in addition to a number of new topics. The following summary indicates some of the highlights of the current report series.

Radiation Quantities and Units (Report 10a)—One of the most important changes is the revision of the section on quantities and units. This revision resulted from the thorough study by the *Ad Hoc* Committee on Quantities and Units mentioned above. It includes new names for certain quantities and clarified definitions for others. It presents a system of concepts and a set of definitions which is internally consistent and yet of sufficient generality to cover present requirements and such future requirements as can be foreseen.

Physical Aspects of Irradiation (Report 10b)—This report deals broadly with the physical aspects of irradiation with a considerable amount of new material added since the 1959 report. It includes an extensive discussion of the various techniques for the measurement of absorbed dose as well as exposure. Characteristics of radiation instrumentation are covered in some detail including the more sophisticated work on standards. The section on spectra has been up-dated and a new section added on neutron measurements and standards. Available data for stopping power ratios and the average energy (W) required to produce an ion pair in a gas have been reviewed. On the basis of this review it has been necessary to modify the previous ICRU tables for these factors. This modification amounts to about 1 or 2 percent change in stopping power ratios and up to 1 percent in *W*.

Radioactivity (Report 10c)—The portions of the report dealing with direct and relative measurements of radioactivity and the availability and requirements for radioactivity standards, and the parts dealing with the techniques and measurements of radioactivity in hospitals and biological laboratories are revisions of the 1959 report, embracing a review of the developments that have occurred since that report and bringing up to date the material included. In addition, a new section on low level radioactivity in materials as related to the problems of radiological measurements has been added. This topic is important because of the problems arising from the contamination, or possible contamination, in the last decade of a great many of the materials used in the construction of counting equipment, shields, and in the reagent chemicals employed in radioactivity measurements.

Clinical Dosimetry (Report 10d)—Much of the Commission's work on clinical dosimetry is brought together in this report. Included is an extensive discussion of practical calibration procedures and the determination of dose along the central ray. Depth dose data relative to stationary and moving-field therapy have been extended as have the conversion data necessary to relate ionization measurements to absorbed dose.

The principal effort has been toward the definition of nomenclature and the indication of methods. While some examples are given and data are provided for these, in general the reader is referred to other published data. The report considers ways of increasing the accuracy and comparability in clinical dosimetry. The discussion includes not only the physical aspects of dose measurement but also the wider subject of planning treatment in such a way as to deliver the prescribed absorbed dose to a defined "target volume". It also includes comments upon the common sources of error in clinical dosimetry and discusses the information which should be recorded during treatment and that which should be reported about any new treatment technique. Appendices to this report include pertinent material taken from other reports in this series.

Radiobiological Dosimetry (Report 10e)—This report deals primarily with radiobiological dosimetry, and considers methods of improving the accuracy and intercomparability of absorbed dose measurements in radiobiology. It is in effect a handbook for the experimental radiobiologist. It emphasizes the great importance of planning the experimental work in a way which makes the dosimetry easier and more accurate and it illustrates how this can be done.

Methods of Evaluating Radiological Equipment and Materials (Report 10f)—This is the first of a new group of ICRU reports dealing with methods of evaluating radiological equipment and materials. It includes a revised discussion on the measurement of focal spots and new sections on grids, image intensifiers and body section equipment.

E. Operating Funds

Throughout most of its existence, the ICRU has 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 available from the ISR.) Recognizing the impracticability of continuing this mode of operation on an indefinite basis, operating funds were sought from various sources in addition to those supplied by the International Society of Radiology.

Prior to 1959, the principal financial assistance to the ICRU had been provided by the Rockefeller Foundation which supplied some $11,000 to make possible various meetings. In 1959 the International Society of Radiology increased its contribution to the Commission to $3,000 to cover the period until the Xth Congress. In 1960 the Rockefeller Foundation supplied an additional sum of some $4,000 making possible a meeting of the Quantity and Units Committee in 1960.

In 1960 and 1961 the World Health Organization contributed the sum of $3,000 each year to the Commission for carrying forward its work. This was increased to $4,000 in 1962. It is expected that this sum will be allocated annually, at least for the next several years. In addition the WHO has provided substantial assistance to the Commission in providing meeting space, secretarial services, etc., for the meetings held in Geneva and Montreux.

In connection with the Commission's Joint Study with the ICRP, the United Nations allocated the sum of $10,000 for the joint use of the two Commissions for the purpose of carrying out their second study. This fund has been administered by the ICRP.

The most substantial contribution to the work of the ICRU has come from the Ford Foundation through the particular efforts of Dr. Paul Pearson. Effective in December 1960, the Ford Foundation made available to the Commission the sum of $37,000 per year for a period of five years. This money is to be used for such items as travel expenses to meetings, for secretarial services and other operating expenses. To a large extent, it is because of this grant that the Commission has been able to hold the several meetings considered to be necessary to move forward actively with its program.

The International Atomic Energy Agency has allocated the sum of $6,000 per year for use by the ICRU. It is expected that this sum will be allocated annually at least for the next several years.

A valuable indirect contribution has been made by the U.S. National Bureau of Standards where the Secretariat has resided. The Bureau has provided substantial secretarial services, reproduction services and travel costs in the amount of several thousands of dollars.

The Commission wishes to express its deep appreciation to all of these and other organizations that have contributed so importantly to its work.

F. Rules

The International Commission on Radiologica Units and Measurements (ICRU) functions under the auspices of the International Congress of Radiology. The Commission was established in 1925 by the First International Congress to define the physical units required in the field of radiology and to make recommendations on the standards required to realize proper measurements in terms of the units defined.

The following rules, amended in 1956, govern the selection and work of the ICRU.

1. (a) The International Commission on Radiological Units and Measurements (ICRU) shall be composed of a chairman and not more than 12 other members. The selection of the members shall be made by the ICRU from nominations submitted to it by the National Delegations to the International Congress of Radiology and by the ICRU itself. The selections shall be subject to approval by the International Executive Committee (IEC) of the Congress. Members of the ICRU shall be chosen on the basis of their recognized activity in the field of radiological units, standards, and measurements, without regard to nationality.

(b) The ICRU shall include at least three medical radiologists and three physicists.

(c) The members of the ICRU shall be approved during each International Congress to serve through the succeeding Congress. Not less than 2 but not more than 4 membership changes shall be made for any one Congress. In the intervening period a vacancy may be filled by the ICRU.

(d) In the event of a member of the ICRU being unable to attend the ICRU meetings, a substitute may be selected by the ICRU as a temporary replacement. Such a substitute member shall not have voting privileges at the meetings unless specifically authorized by the ICRU.

(e) The ICRU shall be permitted to invite individuals to attend its meetings to give special technical advice. Such persons shall not have voting privileges, but may ask permission to have their opinions recorded in the minutes.

2. The Chairman shall be elected by the ICRU from among its regular members to serve for a term corresponding to the interval between Congresses. The choice shall not be limited to the country in which it is proposed to hold the succeeding Congress. The Chairman shall be responsible for reporting the proceedings and recommendations of the ICRU at the next Congress.

3. The ICRU may elect from among its members a Vice-chairman who will serve in the capacity of chairman in the event that the Chairman is unable to perform his duties.

4. Minutes of meetings and records of the ICRU shall be made by a technical secretary selected by the Chairman of the ICRU subject to the approval of its regular members. The technical secretary need not be a regular member of the ICRU. The records of the ICRU shall be passed on to the succeeding secretary.

5. The Chairman, in consultation with the Vice-chairman, shall prepare a program to be submitted to the Commission for discussion at its meetings. Proposals to be considered shall be submitted to the Chairman and circularized to all members of the ICRU and other specially qualified individuals at least two months before any meeting of the ICRU.

6. Decisions of the ICRU shall be made by a majority vote of the members. A minority opinion may be appended to the minutes of a meeting if so desired by any member and upon his submission of same in writing to the secretary.

7. The ICRU may establish such committees as it may deem necessary to perform its functions.

G. Organization of the ICRU Committees

In line with the Commission's policy of rotation of members and Chairmen of Committees whenever feasible, a number of changes have been introduced for the period 1959 to 1962.

H. Composition of the ICRU

(a) It is of interest to note that the membership of the Commission and its committees for the period 1959–62 totals 139 persons drawn from 18 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 members.

(b) The membership of the Main Commission during the preparation of this report was as follows:

LAURISTON S. TAYLOR, Chairman	United States.
L. H. GRAY, Vice-chairman	United Kingdom.
H. O. WYCKOFF, Secretary	United States.
K. K. AGLINTZEV	U.S.S.R.
A. ALLISY	France.
R. H. CHAMBERLAIN	United States.
F. ELLIS	United Kingdom.
H. FRÄNZ	Federal Republic of Germany.
H. E. JOHNS	Canada.
W. J. OOSTERKAMP	Netherlands.
B. RAJEWSKY	Federal Republic of Germany.
H. H. ROSSI	United States.
M. TUBIANA	France.

I. Composition of Committee Preparing Initial Draft of Present Report

H. O. WYCKOFF, Chairman
A. ALLISY
J. W. BOAG
H. FRÄNZ
W. C. ROESCH
H. H. ROSSI
M. TUBIANA

Radiation Quantities and Units

International Commission on Radiological Units and Measurements (ICRU) Report 10a 1962

1. Introduction

There has recently been much discussion of the fundamental concepts and quantities employed in radiation dosimetry. This has arisen partly from the rapid increase in the number of individuals using these concepts in the expanding field of nuclear science and technology, partly because of the need for extending the concepts so that they would be of use at higher photon energies and for particulate as well as for photon radiation, but chiefly because of certain obscurities in the existing formulation of the quantities and units themselves.

The roentgen, for example, was originally defined to provide the best quantitative measure of exposure to medium-energy x radiation which the measuring techniques of that day (1928) permitted. The choice of air as a standard substance was not only convenient but, also appropriate for a physical quantity which was to be correlated with the biological effect of x rays, since the effective atomic number of air is not very different from that of tissue. Thus a given biological response could be reproduced approximately by an equal exposure in roentgens for x-ray energies available at that time. Since 1928 the definition of the roentgen has been changed several times, and this has reflected some feeling of dissatisfaction with the lack of clarity of the concept.

The most serious source of confusion was the failure to define adequately the radiation quantity of which the roentgen was said to be the unit.[1] As a consequence of this omission the roentgen had gradually acquired a double role. The use of this name for the unit had become recognized as a way of specifying not only the magnitude but also the nature of the quantity measured. This practice conflicts with the general usage in physics, which permits, within the same field, the use of a particular unit for all quantities having the same dimensions.

Even before this, the need for accurate dosimetry of neutrons and of charged particles from accelerators or from radionuclides had compelled the International Commission on Radiological Units and Measurements (ICRU) to extend the number of concepts. It was also desired to introduce a new quantity which could be more directly correlated with the local biological and chemical effects of radiation. This quantity, *absorbed dose*, has a generality and simplicity which greatly facilitated its acceptance, and in a very few years it has become widely used in every branch of radiation dosimetry.

The introduction of absorbed dose into the medical and biological field was further assisted by defining a special unit—the *rad*. One rad is approximately equal to the absorbed dose delivered when soft tissue is exposed to one roentgen of medium voltage x radiation. Thus in many situations of interest to medical radiology, but not in all, the numbers of roentgens and rads associated with a particular medical or biological effect are approximately equal and experience with the earlier unit could be readily transferred to the new one. Although the *rad* is merely a convenient multiple of the fundamental unit, erg/g, it has already acquired, at least in some circles, the additional connotation that the only quantity which can be measured in rads is absorbed dose. On the other hand, the rad has been used by some authors as a unit for a quantity called by them *first collision dose;* this practice is deprecated by the Commission.

Being aware of the need for preventing the emergence of different interpretations of the same quantity, or the introduction of undesirable, unrelated quantities or units in this or similar fields of measurement, the ICRU set up, during its meeting in Geneva in September 1958, an *Ad Hoc* Committee. The task of this committee was to review the fundamental concepts, quantities, and units which are required in radiation dosimetry and to recommend a system of concepts and a set of definitions which would be, as far as possible, internally consistent and of sufficient generality to cover present requirements and such future requirements as can be foreseen. The committee was instructed to pay more attention to consistency and rigor than to the historical development of the subject and was authorized to reject any existing quantities or units which seemed to hinder a consistent and unified formulation of the concepts.

---

[1] Fränz, H. and Hübner, W. Concepts and Measurement of Dose, Proceedings of Second International Conference on the Peaceful Uses of Atomic Energy, Geneva 1958, P/971 21, 101, United Nations, Geneva (1958).

Bertrand Russell [2] in commenting on the use and abuse of the concept of infinitesimals by mathematicians, remarks: "But mathematicians did not at first pay heed to (these) warnings. They went ahead and developed their science, and it is well that they should have done so. It is a peculiar fact about the genesis and growth of new disciplines that too much rigor too early imposed stifles the imagination and stultifies invention. A certain freedom from the strictures of sustained formality tends to promote the development of a subject in its early stages, even if this means the risk of a certain amount of error. Nonetheless, there comes a time in the development of any field when standards of rigor have to be tightened."

The purpose of the present reexamination of the concepts to be employed in radiation dosimetry was primarily "to tighten standards of rigor". If, in the process, some increased formality is required in the definitions in order to eliminate any foreseeable ambiguities, this must be accepted.

2. General Considerations

The development of the more unified presentations of quantities and units which is here proposed was stimulated and greatly assisted by mathematical models of the dosimetric field which had been proposed by some members of the committee in an effort to clarify the concepts. It appeared, however, that the essential features of the mathematical models had been incorporated into the definitions and hence the need for their exposition in this report largely disappeared. The mathematical approach is published elsewhere.[3]

As far as possible, the definitions of the various fundamental quantities given here conform to a common pattern. Complex quantities are defined in terms of the simpler quantities of which they are comprised.

The passage to a "macroscopic limit" which has to be used in defining point quantities in other fields of physics can be adapted to radiation quantities and a special discussion of this is included in the section headed "limiting procedures".

The general pattern adopted is to give a short definition and to indicate the precise meaning of any special phrase or term used by means of an explanatory note following the definition. There has been no attempt to make the list of quantities which are defined here comprehensive. Rather, the Commission has striven to clarify the fundamental dosimetric quantities and a few others (such as activity) which were specifically referred to it for discussion.

It is recognized that certain terms for which definitions are proposed here are of interest in other fields of science and that they are already variously defined elsewhere. The precise wording of the definition and even the name and symbol given to any such quantity, may at some future date require alteration if discussions with representatives of the other interested groups of scientists should lead to agreement on a common definition or symbol. Although the definitions presented here represent some degree of compromise, they are believed to meet the requirements in the field of radiation dosimetry.

3. Quantities, Units, and Their Names

The Commission is of the opinion that the definition of concepts and quantities is a fundamental matter and that the choice of units is of less importance. Ambiguity can best be avoided if the defined quantity which is being measured is specified. Nevertheless, the special units do exist in this as in many other fields. For example, the hertz is restricted, by established convention, to the measurement of vibrational frequency, and the curie, in the present recommendations, to the measurement of the activity of a quantity of a nuclide. One does not measure activity in hertz nor frequency in curies although these quantities have the same dimensions.

It was necessary to decide whether or not to extend the use of the special dosimetric units to other more recently defined quantities having the same dimensions, to retain the existing restriction on their use to one quantity each, or to abandon the special units altogether. The Commission considers that the addition of further special units in the field of radiation dosimetry is undesirable, but continues to recognize the existing special units. It sees no objection, however, to the expression of any defined quantity in the appropriate units of a coherent physical system. Thus, to express absorbed dose in ergs per gram or joules per kilogram, exposure in coulombs per kilogram or activity in reciprocal seconds, are entirely acceptable alternatives to the use of the special units which, for historical reasons, are usually associated with these quantities.

The ICRU recommends that the use of each special unit be restricted to one quantity as follows:

The rad—solely for absorbed dose
The roentgen—solely for exposure
The curie—solely for activity.

It recommends further that those who prefer to express quantities such as absorbed dose and kerma (see below) in the same units should use units of an internationally agreed coherent system.

Several new names are proposed in the present report. When the absorbed dose concept was adopted in 1953, the Commission recognized the need for a term to distinguish it from the quantity of which the roentgen is the unit. In 1956 the Commission proposed the term *exposure* for this latter quantity. To meet objections by the ICRP, a compromise term, "exposure dose" was agreed upon.[4] While this term has come into some use

[2] Russell, B., Wisdom of the West, p. 280 (Doubleday and Co., Inc., New York, 1959).
[3] Rossi, H. H. and Roesch, W. C., Field Equations in Dosimetry, Radiation Res. 16, 783 (1962).
[4] For details see ICRU, 1956 Report, NBS Handb. 62, p. 2 (1957).

since then, it has never been considered as completely satisfactory. In the meantime, the basic cause of the ICRP objection has largely disappeared since most legal codes use either the units rad or rem.

Since in this report the whole system of radiological quantities and units has come under critical review, it seemed appropriate to reconsider the 1956 decision. Numerous names were examined as a replacement for exposure dose, but there were serious objections to any which included the word dose. There appeared to be a minimum of objection to the name *exposure* and hence this term has been adopted by the Commission with the hope that the question has been permanently settled. It involves a minimum change from the older name exposure dose. Furthermore, the elimination of the term "dose" accomplishes the long-felt desire of the Commission to retain the term dose for one quantity only—the absorbed dose.

The term "*RBE* dose" has in past publications of the Commission not been included in the list of definitions but was merely presented as a "recognized symbol." In its 1959 report the Commission also expressed misgivings over the utilization of the same term, "*RBE*", in both radiobiology and radiation protection. It now recommends that the term *RBE* be used in radiobiology only and that another name be used for the linear-energy-transfer-dependent factor by which absorbed doses are to be multiplied to obtain for purposes of radiation protection a quantity that expresses on a common scale for all ionizing radiations the irradiation incurred by exposed persons. The name recommended for this factor is the *quality factor*, (*QF*). Provisions for other factors are also made. Thus a *distribution factor*, (*DF*), may be used to express the modification of biological effect due to non-uniform distribution of internally deposited isotopes. The product of absorbed dose and modifying factors is termed the *dose equivalent*, (*DE*). As a result of discussions between ICRU and ICRP the following formulation has been agreed upon:

The Dose Equivalent

1. For protection purposes it is useful to define a quantity which will be termed the "dose equivalent", (*DE*).
2. (*DE*) is defined as the product of absorbed dose, *D*, quality factor, (*QF*), dose distribution factor, (*DF*), and other necessary modifying factors.

$$(DE)=D\,(QF)\,(DF)\ldots.$$

3. The unit of dose equivalent is the "rem". The dose equivalent is numerically equal to the dose in rads multiplied by the appropriate modifying factors.

Although this statement does not cover a number of theoretical aspects (in particular the physical dimensions of some of the quantities) it fulfills the immediate requirement for an unequivocal specification of a scale that may be used for numerical expression in radiation protection.

Another new name is that for the quantity which represents the kinetic energy transferred to charged particles by the uncharged particles per unit mass of the irradiated medium. This is the same as one of the common interpretations of a concept "first collision dose," that has proved to be of great value in the dosimetry of fast neutrons. The concept is also closely related to the energy equivalent of exposure in an x-ray beam. The name proposed, *kerma*, is based on the initials of *k*inetic *e*nergy *r*eleased in *m*aterial.

Still another new name is the *energy fluence* which is here attached to the quantity in the 1953 ICRU report called *quantity of radiation*. The latter term was dropped in the 1956 ICRU report but the concept—time integral of intensity—remains a useful one and the proposed term appears to be acceptable in other languages as well as English. A related quantity, *particle fluence*, which is equivalent to the quantity *nvt* used in neutron physics, is included to round out the system of radiation quantities.

The quantity for which the curie is the unit was referred to the committee for a name and definition. Hitherto the curie has been defined as a *quantity of the radioactive nuclide* such that 3.7×10^{10} disintegrations per second occur in it. However, it has never been specified what was meant by quantity of a nuclide, whether it be a number, mass, volume, etc. Meanwhile the custom has grown of identifying the number of curies of a radionuclide with its transformation rate. Because of the vagueness of the original concept, because of the custom of identifying curies with transformation rate and because it appeared not to interfere with any other use of the curie, the Commission recommends that the term *activity* be used for the transformation rate, and that the curie be made its unit. It is recognized that the definition of the curie is of interest to other bodies in addition to the ICRU, but by this report we recommend that steps be taken to redefine it as $3.7\times10^{10}s^{-1}$, i.e., as a unit of activity and not of quantity of a radioactive nuclide.

It is also recommended that the term *specific gamma-ray constant* be used instead of *specific gamma-ray emission* for the quotient of the exposure rate at a given distance by the activity. The former term focuses attention on the *constancy* of this quotient for a given radionuclide rather than the *emission* of the source.

4. Detailed Considerations

A. Limiting Procedures

Except in the case of a uniform distribution of sources throughout a large region, radiation fields are in general non-uniform in space. They may also be variable in time. Many of the quantities defined in this report have to be specified as functions of space or time, and in principle they must therefore be determined for sufficiently small regions of space or intervals of time by some limiting procedure. There are conceptual difficulties in taking such limits for quantities which depend upon the discrete interactions between radiations and atoms. Similar difficulties arise with other macroscopic physical quantities such as density or temperature and they must be overcome by means of an appropriate averaging procedure.

To illustrate this procedure we may consider the measurement of the macroscopic quantity "absorbed dose" in a non-uniform radiation field. In measuring this dose the quotient of energy by mass must be taken in an elementary volume in the medium which on the one hand is so small that a further reduction in its size would not appreciably change the measured value of the quotient energy by mass and on the other hand is still large enough to contain many interactions and be traversed by many particles.[5] If it is impossible to find a mass such that both these conditions are met, the dose cannot be established directly in a single measurement. It can only be deduced from multiple measurements that involve extrapolation or averaging procedures. Similar considerations apply to some of the other concepts defined below. The symbol Δ precedes the symbols for quantities that may be concerned in such averaging procedures.

In the measurement of certain material constants such as stopping power, absorption coefficient, etc., the limiting procedure can be specified more rigorously. Such constants can be determined for a given material with any desired accuracy without difficulties from statistical fluctuations. In these cases the formulae quoted in the definitions are presented as differential quotients.

B. Spectral Distributions and Mean Values

In practice many of the quantities defined below to characterize a radiation field and its interaction with matter are used for radiations having a complex energy spectrum. An important general concept in this connection is the *spectral concentration* of one quantity with respect to another. The spectral concentration is the ordinate of the distribution function of the first quantity with respect to the second. The independent quantity need not always be energy or frequency; one can speak of the spectral concentration of flux density with respect to quantum energy or of the absorbed dose with respect to linear energy transfer. The interaction constants (such as μ, S and W) referred to in this report are often mean values taken over the appropriate spectral distributions of the corresponding quantities.

[5] In interpreting radiation effects the macroscopic concept of absorbed dose may not be sufficient. Whenever the statistical fluctuations around the mean value are important, additional parameters describing the distribution of absorbed energy on a microscopic scale are necessary.

C. Units

For any of the quantities defined below the appropriate unit of an internationally agreed coherent system can be used. In addition certain special units are reserved for special quantities:

the rad for absorbed dose
the roentgen for exposure
the curie for activity.

D. Definitions

(1) *Directly ionizing particles* are charged particles (electrons, protons, α-particles, etc.) having sufficient kinetic energy to produce ionization by collision.

(2) *Indirectly ionizing particles* are uncharged particles (neutrons, photons, etc.) which can liberate directly ionizing particles or can initiate a nuclear transformation.

(3) *Ionizing radiation* is any radiation consisting of directly or indirectly ionizing particles or a mixture of both.

(4) The *energy imparted* by ionizing radiation to the matter in a volume is the difference between the sum of the energies of all the directly and indirectly ionizing particles which have entered the volume and the sum of the energies of all those which have left it, minus the energy equivalent of any increase in rest mass that took place in nuclear or elementary particle reactions within the volume.

NOTES: (a) The above definition is intended to be exactly equivalent to the previous meanings given by the ICRU to "energy retained by matter and made locally available" or "energy which appears as ionization, excitation, or changes of chemical bond energies". The present formulation specifies what energy is to be included without requiring a lengthy, and possibly incomplete, catalogue of the different types of energy transfer.

(b) Ultimately, most of the energy imparted will be degraded and appear as heat. Some of it, however, may appear as a change in interatomic bond energies. Moreover, during the degradation process the energy will diffuse and the distribution of heat produced may be different from the distribution of imparted energy. For these reasons the energy imparted cannot always be equated with the heat produced.

(c) The quantity *energy imparted to matter* in a given volume is identical with the quantity often called *integral absorbed dose* in that volume.

(5) The *absorbed dose* (D) is the quotient of ΔE_D by Δm, where ΔE_D is the energy imparted by ionizing radiation to the matter in a volume ele-

ment, Δm is the mass of the matter in that volume element and Δ has the meaning indicated in section 4.A.

$$D=\frac{\Delta E_D}{\Delta m}$$

The special unit of absorbed dose is the *rad*.

$$1 \text{ rad}=100 \text{ erg/g}=\frac{1}{100} \text{ J/kg}$$

NOTE: *J* is the abbreviation for Joule

(6) The *absorbed dose rate* is the quotient of ΔD by Δt, where ΔD is the increment in absorbed dose in time Δt and Δ has the meaning indicated in section 4.A.

$$\text{Absorbed dose rate}=\frac{\Delta D}{\Delta t}$$

A special unit of absorbed dose rate is any quotient of the rad by a suitable unit of time (rad/*d*, rad/min, rad/*h*, etc.).

(7) The *particle fluence* [6] or *fluence* (Φ) of particles is the quotient of ΔN by Δa, where ΔN is the number of particles which enter a sphere [7] of cross-sectional area Δa and Δ has the meaning indicated in section 4.A.

$$\Phi=\frac{\Delta N}{\Delta a}$$

(8) The *particle flux density* or *flux density* (φ) of particles is the quotient of $\Delta\Phi$ by Δt where $\Delta\Phi$ is the particle fluence in time Δt and Δ has the meaning indicated in section 4.A.

$$\varphi=\frac{\Delta\Phi}{\Delta t}$$

NOTE: This quantity may also be referred to as particle fluence rate.

(9) The *energy fluence* (F) of particles is the quotient of ΔE_F by Δa, where ΔE_F is the sum of the energies, exclusive of rest energies, of all the particles which enter a sphere [8] of cross-sectional area Δa and Δ has the meaning indicated in section 4.A.

$$F=\frac{\Delta E_F}{\Delta a}$$

(10) The *energy flux density* or *intensity* (I) is the quotient of ΔF by Δt where ΔF is the energy fluence in the time Δt and Δ has the meaning indicated in section 4.A.

$$I=\frac{\Delta F}{\Delta t}$$

[6] This quantity is the same as the quantity, *nvt*, commonly used in neutron physics.

[7] This quantity is sometimes defined with reference to a plane of area Δa, instead of a sphere of cross-sectional area Δa. The plane quantity is less useful for the present purposes and it will not be defined. The two quantities are equal for a unidirectional beam of particles perpendicularly incident upon the plane area.

NOTE: This quantity may also be referred to as energy fluence rate.

(11) The *kerma* [9] (K) is the quotient of ΔE_K by Δm, where ΔE_K is the sum of the initial kinetic energies of all the charged particles liberated by indirectly ionizing particles in a volume element of the specified material, Δm is the mass of the matter in that volume element and Δ has the meaning indicated in section 4.A.

$$K=\frac{\Delta E_K}{\Delta m}$$

NOTES: (a) Since ΔE_K is the sum of the initial kinetic energies of the charged particles liberated by the indirectly ionizing particles, it includes not only the kinetic energy these charged particles expend in collisions but also the energy they radiate in bremsstrahlung. The energy of any charged particles is also included when these are produced in secondary processes occurring within the volume element. Thus the energy of Auger electrons is part of ΔE_K.

(b) In actual measurements Δm should be so small that its introduction does not appreciably disturb the radiation field. This is particularly necessary if the medium for which kerma is determined is different from the ambient medium; if the disturbance is appreciable an appropriate correction must be applied.

(c) It may often be convenient to refer to a value of kerma or of kerma rate for a specified material in free space or at a point inside a different material. In such a case the value will be that which would be obtained if a small quantity of the specified material were placed at the point of interest. It is, however, permissible to make a statement such as: "The kerma for air at the point P inside a water phantom is . . ." recognizing that this is a shorthand version of the fuller description given above.

(d) A fundamental physical description of a radiation field is the intensity (energy flux density) at all relevant points. For the purpose of dosimetry, however, it may be convenient to describe the field of indirectly ionizing particles in terms of the kerma rate for a specified material. A suitable material would be air for electromagnetic radiation of moderate energies, tissue for all radiations in medicine or biology, or any relevant material for studies of radiation effects.

Kerma can also be a useful quantity in dosimetry when charged particle equilibrium exists at the position and in the material of interest, and bremsstrahlung losses are negligible. It is then equal to the absorbed dose at that point. In beams of x or gamma rays or neutrons, whose energies are moderately high, transient charged-particle equilibrium can occur; in this condition

[8] See footnote 7.

[9] Various other methods of specifying a radiation field have been used, e.g., for a neutron source the "first collision dose" in a standard material at a specified point (see Introduction).

the kerma is just slightly less than the absorbed dose. At very high energies the difference becomes appreciable. In general, if the range of directly ionizing particles becomes comparable with the mean free path of the indirectly ionizing particles, no equilibrium will exist.

(12) The *kerma rate* is the quotient of ΔK by Δt, where ΔK is the increment in kerma in time Δt and Δ has the meaning indicated in section 4.A.

(13) The *exposure* (X) is the quotient of ΔQ by Δm, where ΔQ is the sum of the electrical charges on all the ions of one sign produced in air when all the electrons (negatrons and positrons), liberated by photons in a volume element of air whose mass is Δm, are completely stopped in air and Δ has the meaning indicated in section 4.A.

$$X=\frac{\Delta Q}{\Delta m}$$

The special unit of exposure is the roentgen (R).

$$1\text{R}=2.58\times10^{-4}C/\text{kg}$$ [10]

NOTES: (a) The words "charges on all the ions of one sign" should be interpreted in the mathematically absolute sense.

(b) The ionization arising from the absorption of bremsstrahlung emitted by the secondary electrons is not to be included in ΔQ. Except for this small difference, significant only at high energies, the exposure as defined above is the ionization equivalent of the kerma in air.

(c) With present techniques it is difficult to measure exposure when the photon energies involved lie above a few Mev or below a few kev.

(d) As in the case of kerma (4D(11), note (c)), it may often be convenient to refer to a value of exposure or of exposure rate in free space or at a point inside a material different from air. In such a case the value will be that which would be determined for a small quantity of air placed at the point of interest. It is, however, permissible to make a statement such as: "The exposure at the point P inside a water phantom is . . . ."

(14) The *exposure rate* is the quotient of ΔX by Δt, where ΔX is the increment in exposure in time Δt and Δ has the meaning indicated in section 4.A,

$$\text{Exposure rate}=\frac{\Delta X}{\Delta t}$$

A special unit of exposure rate is any quotient of the roentgen by a suitable unit of time (R/s, R/min, R/h, etc.).

[10] This unit is numerically identical with the old one defined as 1 e.s.u. of charge per .001293 gram of air. C is the abbreviation for coulomb.

(15) The *mass attenuation coefficient* $\left(\frac{\mu}{\rho}\right)$ of a material for indirectly ionizing particles is the quotient of dN by the product of ρ, N, and dl where N is the number of particles incident normally upon a layer of thickness dl and density ρ, and dN is the number of particles that experience interactions in this layer.

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

NOTES: (a) The term "interactions" refers to processes whereby the energy or direction of the indirectly ionizing particles is altered.

(b) For x or gamma radiations

$$\frac{\mu}{\rho}=\frac{\tau}{\rho}+\frac{\sigma}{\rho}+\frac{\sigma_{\text{coh}}}{\rho}+\frac{\kappa}{\rho}^{*}$$

where $\frac{\tau}{\rho}$ is the mass photoelectric attenuation coefficient, $\frac{\sigma}{\rho}$ is the total Compton mass attenuation coefficient, $\frac{\sigma_{\text{coh}}}{\rho}$ is the mass attenuation coefficient for coherent scattering, and $\frac{\kappa}{\rho}$ is the pair-production mass attenuation coefficient.

(16) The *mass energy transfer coefficient* $\left(\frac{\mu_K}{\rho}\right)$ of a material for indirectly ionizing particles is the quotient of dE_K by the product of E, ρ and dl where E is the sum of the energies (excluding rest energies) of the indirectly ionizing particles incident normally upon a layer of thickness dl and density ρ, and dE_K is the sum of the kinetic energies of all the charged particles liberated in this layer.

$$\frac{\mu_K}{\rho}=\frac{1}{E\rho}\frac{dE_K}{dl}$$

NOTES: (a) The relation between energy fluence and kerma may be written as

$$K=F\frac{\mu_K}{\rho}$$

(b) For x or gamma rays of energy $h\nu$

$$\frac{\mu_K}{\rho}=\frac{\tau_a}{\rho}+\frac{\sigma_a}{\rho}+\frac{\kappa}{\rho}^{**}$$

where

$$\frac{\tau_a}{\rho}=\frac{\tau}{\rho}\left(1-\frac{\delta}{h\nu}\right)$$

*This equation for $\frac{\mu}{\rho}$ assumes that the nuclear interactions are not important. An extra term for such interactions may be required for x- or gamma-ray energies in excess of a few MeV.

**This equation for $\frac{\mu_K}{\rho}$ assumes that the nuclear interactions are not important. An extra term for such interactions may be required for x- or gamma-ray energies in excess of a few MeV.

$\left(\frac{\tau}{\rho}\right.$=the photoelectric mass attenuation coefficient, δ=average energy emitted as fluorescent radiation per photon absorbed.$\left.\right)$

and $$\frac{\sigma_a}{\rho}=\frac{\sigma}{\rho}\frac{E_e}{h\nu}$$

$\left(\frac{\sigma}{\rho}\right.$=total Compton mass attenuation coefficient, E_e=average energy of the Compton electrons per scattered photon.$\left.\right)$

and $$\frac{\kappa_a}{\rho}=\frac{\kappa}{\rho}\left(1-\frac{2mc^2}{h\nu}\right)$$

$\left(\frac{\kappa}{\rho}\right.$=mass attenuation coefficient for pair production, mc^2=rest energy of the electron$\left.\right)$

(17) The *mass energy-absorption coefficient* $\left(\frac{\mu_{en}}{\rho}\right)$ of a material for indirectly ionizing particles is $\frac{\mu_K}{\rho}(1-G)$ where G is the proportion of the energy of secondary charged particles that is lost to bremsstrahlung in the material.

NOTES: (a) When the material is air, $\frac{\mu_{en}}{\rho}$ is proportional to the quotient of exposure by fluence.

(b) $\frac{\mu_K}{\rho}$ and $\frac{\mu_{en}}{\rho}$ do not differ appreciably unless the kinetic energies of the secondary particles are comparable with or larger than their rest energy.

(18) The *mass stopping power* $\left(\frac{S}{\rho}\right)$ of a material for charged particles is the quotient of dE_s by the product of dl and ρ, where dE_s is the average energy lost by a charged particle of specified energy in traversing a path length dl, and ρ is the density of the medium.

$$\frac{S}{\rho}=\frac{1}{\rho}\frac{dE_s}{dl}$$

NOTE: dE_s denotes energy lost due to ionization, electronic excitation and radiation. For some purposes it is desirable to consider stopping power with the exclusion of bremsstrahlung losses. In this case $\frac{S}{\rho}$ must be multiplied by an appropriate factor that is less than unity.

(19) The *linear energy transfer* (L) of charged particles in a medium is the quotient of dE_L by dl where dE_L is the average energy locally imparted to the medium by a charged particle of specified energy in traversing a distance of dl.

$$L=\frac{dE_L}{dl}$$

NOTES: (a) The term "locally imparted" may refer either to a maximum distance from the track or to a maximum value of discrete energy loss by the particle beyond which losses are no longer considered as local. In either case the limits chosen should be specified.

(b) The concept of linear energy transfer is different from that of stopping power. The former refers to energy imparted within a limited volume, the latter to loss of energy regardless of where this energy is absorbed.

(20) The *average energy* (W) *expended in a gas per ion pair formed* is the quotient of E by N_W, where N_W is the average number of ion pairs formed when a charged particle of initial energy E is completely stopped by the gas.

$$W=\frac{E}{N_W}$$

NOTES: (a) The ions arising from the absorption of bremsstrahlung emitted by the charged particles are not to be counted in N_W.

(b) In certain cases it may be necessary to consider the variation in W along the path of the particle, and a differential concept is then required, but is not specifically defined here.

(21) A *nuclide* is a species of atom having specified numbers of neutrons and protons in its nucleus.

(22) The *activity* (A) of a quantity of a radioactive nuclide is the quotient of ΔN by Δt where ΔN is the number of nuclear transformations which occur in this quantity in time Δt and Δ has the meaning indicated in section 4.A.

$$A=\frac{\Delta N}{\Delta t}$$

The special unit of activity is the curie (Ci).

$$1\text{Ci}=3.7\times10^{10}\text{s}^{-1}\text{ (exactly)}$$

NOTE: In accordance with the former definition of the curie as a unit of quantity of a radioactive nuclide, it was customary and correct to say: "Y curies of P-32 were administered . . . . ." It is still permissible to make such statements rather than use the longer form which is now correct: "A quantity of P-32 was administered whose activity was Y curies."

(23) The *specific gamma-ray constant* (Γ) of a gamma-emitting nuclide is the quotient of $l^2 \frac{\Delta X}{\Delta t}$ by A, where $\frac{\Delta X}{\Delta t}$ is the exposure rate at a distance l from a point source of this nuclide having an activity A and Δ has the meaning indicated in section 4.A.

$$\Gamma = \frac{l^2 \Delta X}{A \Delta t}$$

Special units of specific gamma ray constant are $Rm^2h^{-1}Ci^{-1}$ or any convenient multiple of this.

NOTE: It is assumed that the attenuation in the source and along l is negligible. However, in the case of radium the value of Γ is determined for a filter thickness of 0.5 mm of platinum and in this case the special units are $Rm^2h^{-1}g^{-1}$ or any convenient multiple of this.

TABLE 4.1. *Table of quantities and units*

No.	Name	Symbol	Dimensions [a]	Units		
				MKSA	cgs	Special
4	Energy imparted (integral absorbed dose).	.....	E.....	J.....	erg.....	g. rad.
5	Absorbed dose.....	D	EM^{-1}....	J kg^{-1}....	erg g^{-1}.....	rad.
6	Absorbed dose rate.	.....	$EM^{-1}T^{-1}$.	J $kg^{-1}s^{-1}$..	erg $g^{-1}s^{-1}$..	rad s^{-1}, etc.
7	Particle fluence or fluence.	Φ	L^{-2}.....	m^{-2}.....	cm^{-2}.....	
8	Particle flux density.	φ	$L^{-2}T^{-1}$...	$m^{-2}s^{-1}$....	$cm^{-2}s^{-1}$....	
9	Energy fluence....	F	EL^{-2}.....	J m^{-2}.....	erg cm^{-2}...	
10	Energy flux density or intensity.	I	$EL^{-2}T^{-1}$.	J $m^{-2}s^{-1}$..	erg $cm^{-2}s^{-1}$	
11	Kerma.....	K	EM^{-1}....	J kg^{-1}....	erg g^{-1}.....	
12	Kerma rate.....	.....	$EM^{-1}T^{-1}$	J $kg^{-1}s^{-1}$..	erg $g^{-1}s^{-1}$..	
13	Exposure.....	X	QM^{-1}....	C kg^{-1}...	esu g^{-1}....	R (roentgen).
14	Exposure rate.....	.....	$QM^{-1}T^{-1}$.	C $kg^{-1}s^{-1}$.	esu $g^{-1}s^{-1}$..	Rs^{-1}, etc.
15	Mass attenuation coefficient.	$\frac{\mu}{\rho}$	L^2M^{-1}....	m^2kg^{-1}...	cm^2g^{-1}.....	
16	Mass energy transfer coefficient.	$\frac{\mu_K}{\rho}$	L^2M^{-1}....	m^2kg^{-1}...	cm^2g^{-1}.....	
17	Mass energy absorption coefficient.	$\frac{\mu_{en}}{\rho}$	L^2M^{-1}....	m^2kg^{-1}...	cm^2g^{-1}.....	
18	Mass stopping power.	$\frac{S}{\rho}$	EL^2M^{-1}..	J m^2kg^{-1}.	erg cm^2g^{-1}.	
19	Linear energy transfer.	L	EL^{-1}.....	J m^{-1}.....	erg cm^{-1}...	kev$(\mu m)^{-1}$.
20	Average energy per ion pair.	W	E.....	J.....	erg.....	ev.
22	Activity.....	A	T^{-1}.....	s^{-1}.....	s^{-1}.....	Ci(curie).
23	Specific gamma-ray constant.	Γ	QL^2M^{-1}..	Cm^2kg^{-1}.	esu cm^2g^{-1}.	Rm^2h^{-1} Ci^{-1}, etc.
	Dose equivalent...	DE	.....	.....	.....	rem

[a] It was desired to present only one set of dimensions for each quantity, a set that would be suitable in both the MKSA and electrostatic-cgs systems. To do this it was necessary to use a dimension Q, for the electrical charge, that is not a fundamental dimension in either system. In the MKSA system (fundamental dimensions M, L, T, I) Q represents the product IT; in the electrostatic-cgs system (M, L, T) it represents $M^{1/2}L^{3/2}T^{-1}$.

☆ U.S. GOVERNMENT PRINTING OFFICE: 1965 O—749-083

APPENDIX III

Selected Definitions from:
USA Standard, Glossary of Terms in Nuclear Science
and Technology (USAS N1.1-1967), published by
USA Standards Institute, 10 E. 40th St.,
New York, New York 10016

DOSE

A general term denoting the quantity of radiation or energy absorbed in a specified mass. For special purposes, its meaning should be appropriately stated, e.g., *absorbed dose*.

DOSE, ABSORBED

The *energy imparted to matter* in a volume element by *ionizing radiation* divided by the mass of irradiated material in that volume element. The special unit of absorbed dose is the *rad*. One rad equals 100 ergs per gram. (Also commonly called *dose*.)

DOSE, EMERGENCY

The *absorbed dose*, when the *maximum permissible dose equivalent* is exceeded, knowingly incurred in the performance of an unusual task to protect individuals or valuable property.

NOTE: More detailed information regarding radiation emergencies may be found in International Commission on Radiological Protection Report No. 9 or National Committee on Radiation Protection Report No. 29.

DOSE, INTEGRAL ABSORBED

The integral of the *absorbed dose* over the mass of irradiated matter in the volume under consideration. It is identical with the *energy*

imparted to matter in that volume. The special unit of integral absorbed dose is the *gram-rad.*

DOSE, MEDIAN LETHAL (LD 50)

The *absorbed dose* which will kill, within a specified time period, 50 percent of the individuals of a large population of organisms of a given species.

DOSE, PERCENTAGE DEPTH

The ratio, expressed as a percentage, of the *absorbed dose* at a given depth within a body to the absorbed dose at some reference point of the body along the central ray. For *x* or *gamma radiation,* the location of the reference point depends on the energy of the incident radiation. It is at the surface for low energies or at or beyond the position of peak absorbed dose for high energies.

DOSE, PERMISSIBLE

(See *dose equivalent, maximum permissible.)*

DOSE, THRESHOLD

The minimum *absorbed dose* that will produce a specified effect.

DOSE, TOLERANCE

Obsolete term for *dose equivalent, maximum permissible.*

DOSE, VOLUME

The product of *absorbed dose* and the volume of the absorbing mass.

NOTE: This term is often confused with *integral absorbed dose.*

DOSE EQUIVALENT (Radiation Protection)

The product of *absorbed dose, quality factor, dose distribution factor,* and other modifying factors necessary to express on a common scale, for all *ionizing radiations,* the irradiation incurred by exposed persons. The special unit of dose equivalent is the *rem.*

NOTE: It is recommended that the term "RBE dose" be used in the field of radiobiology exclusively.

DOSE EQUIVALENT, MAXIMUM PERMISSIBLE (MPD) (Radiation Protection)

The largest *dose equivalent* received within a specified period which is permitted by a regulatory agency or other authoritative group on the assumption that receipt of such dose equivalent creates no appreciable somatic or genetic injury. Different levels of MPD may be set for different groups within a population. (By popular usage, *dose, maximum permissible* is an accepted synonym.)

EXPOSURE

A measure of the ionization produced in air by *x* or *gamma radiation.* It is the sum of the electrical charges on all of the ions of one sign produced in air when all electrons liberated by photons in a volume element of air are completely stopped in the air, divided by the mass of the air in the volume element. The special unit of exposure is the *roentgen.*

FLUENCE, (See fluence, particle)

FLUENCE, ENERGY

At a given point in space, the sum of energies, exclusive of rest energies, of all the particles or photons incident during a given time interval on a small sphere centered at that point, divided by the cross-

sectional area of that sphere. It is identical with the time integral of the *energy flux density*.

FLUENCE, PARTICLE

At a given point in space, the number of particles or photons incident during a given time interval on a small sphere centered at that point divided by the cross-sectional area of that sphere. It is identical with the time integral of the *particle flux density*.

FLUX (See *flux density, particle*)

FLUX DENSITY, DIFFERENTIAL ENERGY

That part of the *energy flux density* resulting from particles or photons having a specified direction, energy, or both per unit interval of solid angle, energy, or both.

FLUX DENSITY, DIFFERENTIAL PARTICLE

That part of the *particle flux density* resulting from particles or photons having a specified direction, energy, or both per unit interval of solid angle, energy, or both.

FLUX DENSITY, ENERGY

At a given point, the sum of energies, exclusive of rest energy, of all the particles or photons incident per unit time on a small sphere centered at that point divided by the cross-sectional area of that sphere. It is identical with the product of the *particle flux density* and the average energy of the particles.

NOTE: This quantity may also be referred to as energy fluence rate.

FLUX DENSITY , PARTICLE

At a given point in space, the number of particles or photons incident per unit time on a small sphere centered at that point divided by the cross-sectional area of that sphere. It is identical with the product of the particle density and the average speed. The term is commonly called *flux*.

KERMA (Kinetic Energy Released in Material)

The ratio of the sum of the initial kinetic energies of all the charged particles liberated by *indirectly ionizing particles* in a volume element to the mass of the matter in the volume element.

APPENDIX IV

Selected Definitions of Thorn*

Absorbed Dose

When ionizing radiation passes through matter, some of its energy is imparted to the matter. The amount absorbed per unit mass of irradiated material at the place of interest is called the absorbed dose and is measured in rads, where

$$1 \text{ rad} = 100 \text{ erg/gm} = 1/100 \text{ joule/kg}$$

The rad unit is applicable to any type of ionizing radiation, but in reporting dose, the type, as well as the irradiated material (for instance, tissue), and the place of interest must be specified. Without the above three factors, a statement of absorbed dose received is incomplete and probably useless, since the same dose of different kinds of radiation, even delivered to the same place, can produce entirely different effects. The roentgen unit has been used as a dose unit for x - and gamma rays for many years, but does not, in fact, measure energy imparted at higher photon energies. This unit and other obsolete units such as the rep should be replaced by the rad for absorbed dose. If ΔE_D is the energy imparted by ionizing radiation to matter in a volume element and Δm is the mass of the matter in that volume element, then the absorbed dose D_a is given by

$$D_a = \frac{\Delta E_D}{\Delta m}$$

---

*Eva M. Thorn. Definitions of Nuclear Terms-Including Formulas and Tables, NWEF Report 10001, Vol. 1. (1967). (AD-822091).

The symbol Δ is defined under limiting procedures in Handbook 84. In most cases, ΔE_D is that energy which is eventually deposited in the medium as heat and is measurable calorimetrically.

Dose Equivalent

The term "RBE dose" has been used in the past in both radiobiology and radiation protection. It is now recommended that the term RBE be used in radiobiology only and that another term be used for purposes of radiation protection. The linear-energy-transfer-dependent factor is multiplied by the absorbed dose, D_a, to obtain a quantity that expresses on a common scale the irradiation received by persons exposed to all ionizing radiations. The name recommended for the linear-energy-transfer-dependent factor is quality factor, QF. Other factors must also be considered for the purpose of radiation protection. A distribution factor, DF, is used to express the modification of the biological effect of radiation due to a non-uniform distribution of isotopes in the body. The distribution factor like the quality factor also affects the absorbed dose when radiation protection is being considered. It is recommended by the International Commission on Radiological Units and Measurements that the final calculated dose received by an individual after the absorbed dose is modified by the above-mentioned factors, plus any other factors that may affect the incoming radiation, be called the dose equivalent, DE. If the only apparent modifying factors are QF and DF, then

$$DE = D_a (QF)(DF).$$

If other factors must be considered and are defined, then

$$DE = D_a (QF) (DF) \ldots$$

where the dots take into account the product of these other factors. The unit of dose equivalent, DE, is the rem. The unit of absorbed dose, D_a, is the rad. Although the above definition of dose equivalent does not cover a number of theoretical aspects (in particular the physical dimensions of some of the quantities) it fulfills the immediate requirement for an unequivocal specification of a scale that may be used for numerical expression in radiation protection.

Exposure

Exposure is a term adopted by the International Commission on Radiological Units and Measurements in 1962 to replace the term "exposure dose" introduced in their 1956 report. The quantity is used for x and gamma radiation. Exposure is the measure at a certain place of radiation which has the ability to produce ionization. The unit of exposure is the roentgen, R, where 1 R equals 2.58×10^{-4} coulombs/kilogram. The definition thus corresponds to the terms roentgen dose (long in use in radiology) and air dose. The wording was made purposely unspecific, thus leaving open the possibility of later defining exposure for radiations other than x- and gamma rays. If ΔQ is the sum of the electric charges on all the ions of one sign produced in air when all the electrons (negatrons and positrons), liberated by photons in a volume element of air, are completely stopped in air, and Δm is the mass of the air, then exposure X is given by

$$X = \frac{\Delta Q}{\Delta m}$$

Δ is defined under limiting procedures in Handbook 84. The words "charges on all ions of one sign" should be interpreted in the mathematically absolute sense. The ionization arising from the absorption of bremsstrahlung emitted by the secondary electrons is not to be included in ΔQ. Except for this small difference, significant only at high energies, the exposure when the photon energies involved lie above a few million electron volts (MeV) or below a few thousand electron volts (keV). As in the case of kerma it may often be convenient to refer to a value of exposure or of exposure rate in free space or at a point inside a material different from air. In such a case the value will be that which would be determined for a small quantity of air placed at the point of interest.

Kerma

Kerma is the sum of the initial kinetic energies of all the charged particles liberated by indirectly ionizing particles in a volume element of a specified material per unit mass. The sum of the kinetic energies include not only the kinetic energy these charged particles expend in collision, but also the energy they radiate in bremmsstrahlung. The energy of any charged particle is also included when this energy is produced in secondary processes occurring within the volume element. Thus, the energy of Auger electrons is part of this energy. If the sum of the above kinetic energies is given as ΔW_k, and Δm is the mass of the matter in the volume element, then

$$\text{kerma} = \frac{\Delta W_k}{\Delta m}$$

where Δ is defined under limiting procedures in Handbook 84. In actual

measurements Δm should be so small that its introduction does not appreciably disturb the radiation field. This is particularly necessary if the medium for kerma is determined is different from the ambient medium. If the distrubance is appreciable, an appropriate correction must be applied. It may often be convenient to refer to a value of kerma or of kerma rate for a specified material in free space or at a point inside a different material. In such a case the value would be that which would be obtained if a small quantity of the specified material were placed at the point of interest. A fundamental physical description of a radiation field is the intensity (energy flux density) at all relevant points. For the purpose of dosimetry, however, it may be convenient to describe the field of indirectly ionizing particles in terms of the kerma rate for a specified material. A suitable material would be air for electromagnetic radiation of moderate energies, tissue for all radiations in medicine or biology, or any relevant material for studies of radiation effects. Various other methods of specifying a radiation field have been used: e.g. for a neutron source, the "first collision dose" in a standard material at a specified point. Kerma can also be a useful quantity in dosimetry when charged particle equilibrium exists at the position and in the material of interest, and bremsstrahlung losses are negligible. It is then equal to the absorbed dose at that point. In beams of x- or gamma rays or neutrons, whose energies are moderately high, transient charged-particle equilibrium can occur; in this condition the kerma is just slightly less than the absorbed dose. At very high energies the difference becomes appreciable. In general, if the range of directly ionizing particles becomes comparable with the mean free path of the indirectly ionizing particles

no equilibrium will exist. In conclusion, kerma is the quantity which represents the kinetic energy transferred to charged particles by the uncharged particles per unit mass of the irradiated medium. This is the same as one of the common interpretations of a concept "first collision dose", that has proved to be of great value in the dosimetry of fast neutrons. The concept is also closely related to the energy equivalent of exposure in an x-ray beam. "Kerma" is based on the initials of kinetic energy released in material.

Flux

Flux is a term used ambiguously for flux density. (See flux density).

Flux Density

For electromagnetic radiation flux density is the power, or energy per unit time, passing through a unit area of surface normal to a beam (energy flux density). For particles or photons flux density is the number passing through a unit area of surface normal to a beam per unit time. The term flux has largely superseded the term flux density in common usage

Neutron Fluence (or Fluence)

Neutron (or particle) fluence is equivalent to the quantity nvt used in neutron physics, where n is the number of neutrons per volume, v the velocity of the neutrons, and t the time. Dimensionally,

$$nvt = \frac{(N_n)}{L^3} \quad \frac{(L)}{(T)} \quad (T) = (N_n)\ (L^{-2})$$

where N_n is number of neutrons and L is length.

Neutron Flux Density *

Neutron flux density is a term used to express the intensity of neutron radiation. It is the number of neutrons passing through a unit area in unit time. For neutrons of a given energy, the neutron flux density is the product of the neutron density and average speed. It is expressed in neutrons/cm^2-sec. The common symbol is nv, where n is the neutron density in cm^{-3}; and v is the average neutron speed in cm-sec^{-1}. Neutron flux density is also defined as

$$\int_E N(E)\ \underline{v}\ (E)\ dE$$

where N(E) is the number of neutrons per unit energy per unit volume with energy E, $\underline{v}(E)$ is vector velocity of the neutrons with energy E. Both N(E) and $\underline{v}$ may vary with time.

---

*This paragraph has been revised to be more consistent with the rest of this report, the most significant change being the addition of the word density.

INTERNAL DISTRIBUTION

1-5. L. S. Abbott
6. R. G. Alsmiller, Jr.
7. J. Auxier
8. J. R. Buchanan
9. H. C. Claiborne
10. F. H. Clark
11. C. E. Clifford
12. L. Dresner
13. W. H. Jordan
14. F. Kertesz
15. Joanne S. Levy
16-17. F. C. Maienschein
18. R. W. Peelle
19. S. K. Penny
20-28. B. F. Maskewitz
29. D. A. Sundberg
30. D. K. Trubey
31. A. M. Weinberg
32. E. R. Cohen (consultant)
33. B. C. Diven (consultant)
34. W. R. Hess (consultant)
35. M. H. Kalos (consultant)
36. L. V. Spencer (consultant)
37-1237. RSIC Distribution
1238-1240. Central Research Library
1241. Document Reference
1242-1262. Laboratory Records Department
1263. Laboratory Records ORNL R. C.
1264. ORNL Patent Office

EXTERNAL DISTRIBUTION

1265. G. H. Anno, Mc-Donnell-Douglas, 3000 Ocean Park Blvd. Santa Monica, California 90405
1266. F. H. Attix, U. S. Naval Research Lab., Washington, D. C.
1267. A. B. Chilton, University of Illinois, Urbana, Illinois
1268. H. J. Donnert, Kansas State University, Manhattan, Kansas
1269. W. E. Edwards, General Electric, Missile and Space Division, P. O. Box 132, Cincinnati, Ohio 45215
1270. Capt. R. W. Enz, Radiation Physics Branch, DASA, Washington, D. C.
1271. S. T. Friedman, Westinghouse Astronuclear Lab., P. O. Box 10864, Pittsburgh, Pa. 15236
1272. L. G. Haggmark, U. S. Naval Radiological Defense Lab., San Francisco, California, 94135
1273. P. B. Hemmig, Division of Reactor Development and Technology, U. S. Atomic Energy Commission, Washington, D. C.
1274. C. M. Huddleston, U. S. Naval Radiological Defense Lab., San Francisco, California, 94135
1275. W. R. Johnson, University of Virginia, 1701 Burnley Avenue, Charlottesville, Virginia, 22903
1276-1277. J. W. Keller, NASA Headquarters, Washington, D. C.
1278. R. J. Klotz, Combustion Engineering, P. O. Box 500, Windsor, Connecticut 06095

EXTERNAL DISTRIBUTION (Cont.)

1279. K. O'Brien, U. S. Atomic Energy Commission Health and Safety Lab., 30 Dover Terrace, Monsey, New York 10952
1280. A. E. Profio, Gulf General Atomic, P. O. Box 608, San Diego, California 92112
1281. Lt. Col. G. C. Reinhardt, Radiation Physics Branch, DASA, Washington, D. C.
1282. A. Reetz, NASA, Headquarters, Washington, D. C. 20546
1283. H. H. Rossi, Columbia University, Dept. of Radiology, New York, New York, 10032
1284. Robert Roussin, 1213 West Clark, Urbana, Illinois
1285. N. M. Schaeffer, Radiation Research Associates, 1506 W. Terrell Avenue, Fort Worth, Texas 76104
1286. H. O. Wyckoff, Armed Forces Radiobiology Research Institute, Bethesda, Maryland 20014
1287. I. F. Zartman, Division of Reactor Development, U. S. Atomic Energy Commission, Washington, D. C.
1288-1307. Division of Technical Information Extension (DTIE)
1308. Laboratory and University Division, AEC, ORO
1309-1592. Given Distribution as shown in TID-4500 under Reactor Technology Category (UC-80) (50 copies-CFSTI)